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ENGINEERING PHYSICS DIVISION PROGRESS REPORT PERIOD ENDING DECEMBER 31, 1983

F. C. Maienschein, Director

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PREFACE

This progress report strives for complete but necessarily sketchy coverage of the work performed within the Engineering Physics Division during the 19-month period from May 31, 1982 to December 31, 1983. It consists of abstracts of papers and reports published by members of the division during this period plus abstracts of papers presented at various meetings. Thus the account tends to be somewhat historical, the moreso since the inevitable delays between the completion of a project and its documentation means that the report covers some work that was performed before May 31, 1982.

The principal disadvantage of including in the report only the abstracts of those papers that have been published or presented is, of course, that current or future directions of research within the division may not be apparent. In general, changes within the Engineering Physics Division continue to reflect a long, slow trend away from the neutronics studies on which the division was founded in 1955. In particular, a major trend has been away from the development of sophisticated experimental and calculational methods for performing neutronics studies and toward the application of those methods to various projects and even to other disciplines. For example, the reactor core analysis methods and the shielding transport methods developed within the division are being applied almost routinely to many systems both by us and by others. Similarly, the sensitivity analysis techniques originally developed for reactor physics and radiation transport studies are still being used for that purpose, but in addition sensitivity analysis is being applied in other new fields of research we have entered, a typical example being our studies of the impact of CO_2 on the climate.

We have also entered new areas of research requiring entirely different approaches and techniques. As noted in our last progress report, our work for fission reactors has been broadened to include risk analyses and human factors studies to support the safe and reliable operation of reactors. More recently, we have initiated research in intelligent control systems that represents an application of broader physical and mathematical techniques to problems of energy systems. A characteristic of all our new research areas is that they tend to be multidisciplinary and multi-institutional.

To make these research trends more apparent, we have organized this report somewhat differently from our earlier progress reports. In general, the titles of the various sections are selected according to the application of the work reported, and introductions to the sections describe the current status of the work. Since new areas of research are thus introduced in separate sections, an imbalance in the report tends to arise because that work has not yet been represented proportionately by publications. This imbalance will be corrected in time as the new areas grow or, perhaps in some cases, die away.

This format for the report is somewhat arbitrary and, it should be noted, does not imply a reduced emphasis upon integration and interaction at the working level. It has been a long-term division policy to encourage such interaction so that methods and techniques developed for one program can be effectively applied in another. Without this type of exchange, which originally resulted from the concentration of nuclear-data and radiation-shielding activities within the division, individual programs in the various areas would be poorly served.

A glance at the table of contents will reveal that we have not completely followed our stated format. As in previous progress reports of the division, the large first section covers our continuing basic studies of the nucleus, some of which are applied in the sense that they are important to and even supported by specific projects but most of which defy precise classification by project. Following this first section are sections describing predominantly applied research for fission reactors, fusion reactors, and high-energy accelerator shielding and detectors. These, in turn, are followed by sections describing nuclear weapons radiation studies, energy-economics modeling and analysis, CO_2 impact research, and research in intelligent control systems.

The final section of the report, on information analysis and distribution, reflects our long-time commitment to facilitating the worldwide use of the data and methods we and others develop. In spite of the great importance of and emphasis on "proper publication," other approaches to technology transfer are often required, one being the specialized information center — such as the Radiation Shielding Information Center (RSIC) in our division. RSIC has for many years played a crucial role in providing data and analysis methods to the diversity of users and potential users who present widely varying technical capabilities for radiation shield design.

For many applications, another approach — that of direct interaction between division staff and designers of reactors or other facilities — represents the optimum utilization of advanced methods and data and their incorporation into industrial practice. This type of direct interaction may take place within reactor projects, in the analysis of counting rates in experiments at high-energy accelerators, or in similar projects. Such interaction often leads to joint publications, as is evidenced by the inclusion in this report of a number of abstracts of papers authored jointly by division staff members and individuals from other institutions (see especially Section 4).

For completely new areas of application, a demonstration or first use of analysis methods may represent the best approach. If the analysis proves to be practical and instructive, then arrangements can be pursued to allow those in the field to incorporate this type of analysis into their work. A current example of this approach is being provided in the division by the development of a sensitivity analysis method for application to waste isolation studies (see Papers 2.58 and 2.59).

Finally, we must point out that our next progress report will include yet another area of research and a change in our division name. For many years, mathematics and statistics research has been sponsored in Oak Ridge by the Department of Energy's Office of Basic Energy Sciences and has been carried out within the Computer Sciences Division of the UCC Nuclear Division, which also operates all central computers in Oak Ridge. This research has now been transferred (as of February 1, 1984) to our division at ORNL, and the division name has been changed to the Engineering Physics and Mathematics Division.

As described in their last progress report (ORNL/CSD-118, for the period ending June 30, 1983), the activities of the mathematics and statistics group include applied analysis, computational mathematics (especially numerical linear algebra), and statistical methods. In addition to the directly funded research, slightly more than one-half of the work consists of direct applications to problems at ORNL or other parts of UCC-ND. The group is comprised of about 20 full-time equivalent technical staff members. We welcome them to our division, together with the additional new area of research they will add to our program.

Section 1

NUCLEAR DATA

Measurements

Cross-Section Analyses, Evaluations, and Reviews

Research Techniques and Facilities

1.0. INTRODUCTION

A decade ago the research performed at the Oak Ridge Electron Linear Accelerator (ORELA) was split about evenly between the ORNL Physics Division and the Engineering Physics Division, the former group emphasizing studies of nuclear structure and neutron reaction mechanisms and the latter devoting more effort toward obtaining precise values of the cross sections needed for the nuclear energy programs. Over the years the emphasis has shifted toward the practical needs. In the fall of 1982, over one-half of the ORELA research being reported through the Physics Division was transferred to our division. The shifted research includes capture cross-section measurements motivated by astrophysical interests; measurements of neutron capture by fission products needed for fast reactor design; detailed high-resolution studies of resonance structure crucial to the determination of level densities and spacing distributions; fission cross sections of heavy alpha-active actinides; and total cross sections over a very broad incident-neutron energy range and for a number of elemental and isotopic samples to study variations in neutron strength versus energy, atomic mass, and angular momentum. Much of this work has included efforts of many visiting scientists from universities and national laboratories in the U.S. and in foreign countries.

Many aspects of the transferred research bear on the development of files of evaluated nuclear data for applied needs and so fit neatly into the established patterns of the division. Other aspects, however, will require a broadening of our scope to include research designed primarily to enhance understanding of physics.

Considerable leadership effort continues within our nuclear data groups to establish useful and correct techniques for codification of the variance-covariance matrix of experimental as well as "evaluated" results. ORELA experiments typically involve thousands or tens of thousands of data points, so correct representation of the covariances among the data elements is challenging. In several cases, we have not achieved adequate success. As publication of experimental results is accompanied by rational and complete expressions of experimental uncertainties, correct combination of experimental results from various sources to obtain evaluated data files will have a more solid (scientific) basis. Indeed, we have now reached the point in neutron cross-section research where only well conceived, executed, and documented experimental results are justifiable for public support. A careful report of uncertainties and correlations is required.

There has nearly always been a free international sharing of experimental neutron cross-section results. In addition, international collaboration on particular experimental research efforts frequently occurs. Such international cooperation has not been widespread in the generation and use of evaluated cross-section data files. However, the Nuclear Energy Agency Nuclear Data Committee, an international group, set a new pattern last year by appointing select committees to collaborate in depth toward resolving two important areas of discrepancy: (a) parameters of the 1.15-keV resonance in the $(n+^{56}Fe)$ cross section and (b) the overall resonance parameterization of the $(n+^{238}U)$ reaction above 1 keV. Members of our division are taking active leading roles on these committees, and the efforts are expected to be largely successful. This international effort originated in the recognition that, despite the great world knowledge of neutron nuclear data, some of the most important problems have not been resolved.

During recent years about one-third of our nuclear data research has been supported by the DOE fast reactor program. (One-fifth of accelerator costs are included.) With the cancellation of the Clinch River Breeder Reactor and the subsequent continuing program re-evaluations, it is uncertain how much emphasis in that program will be placed on technology improvements that have a several year time scale, such as the effort to establish a fully satisfactory fast reactor data base in the next (sixth) version of the ENDF/B nuclear data file. Much of the fast reactor nuclear data work across the country might be transferred to the DOE Office of Energy Research, which already supports most of the ORNL effort through its Basic Energy Sciences program. In the face of these uncertainties, we have cooperated at length during the past year in a multilaboratory effort to solidify and make more efficient the cross-section evaluation work (data combination and codification) that has been supported by the fast reactor program. A written plan has been developed to focus effort on the most important cross sections, taking into account planned new measurements and the evaluation work and review effort thought to be needed to provide fast reactor engineers with a cross-section data base adequate for the precise design of and eventual use of efficient fast reactors of any of the wide variety of types now being considered. The plan assumes a constant level of effort for a few years. Long-term efforts of the type discussed here are not carried out efficiently with irregular funding, so there is an urgency to avoid hasty budgetary changes that could destroy the effectiveness of the present program.

As was indicated in an earlier progress report, we performed some accelerator facility studies to determine whether accelerator design options now exist that would allow relatively economical replacement of ORELA with a 200- to 300-MeV proton accelerator. Members of the Accelerator Technology Division of Los Alamos National Laboratory helped our staff to perform these studies of linear accelerators. As expected, the results indicated that a proton accelerator could give significantly improved intensity for most of the experiments performed at ORELA, especially for those at neutron energies above several MeV. However, such an accelerator facility would require an investment of about $40M (1986 dollars). We concluded that an economical linac-based proton accelerator would not be an improvement over ORELA in its range of strong performance. No further work in this area is presently planned after the documentation of what has already been completed.

In the areas of ORELA operation and development, we have tried very hard to realize the long-established goal of bringing into routine use the "prebuncher" that was constructed to compress the ORELA charge pulse and thereby increase the intensity available for short-pulse experiments. Monitoring of the beam has been greatly improved and some definite obstacles have been identified and overcome; however, additional problems have recently been recognized and the goal to gain performance through routine prebuncher use has not been reached even though the apparatus has been operated without difficulty for periods longer than a day. The best results so far obtained with the prebuncher are not superior to the best ones logged in prior years before the prebuncher was installed. Goals for the near future are to determine whether the difficulties are fundamental and, if not, to discover how the design needs to be modified. Then the prebuncher will be removed and the identified improvements installed if such action seems warranted.

The abstracts reproduced in this first section, representing papers and reports published or accepted for publication since our last progress report and before January 1, 1984, reveal the diversity of our staff and missions. The first subsection includes abstracts of papers containing new experimental results — mostly from research performed at ORELA. The second subsection includes abstracts covering analyses of other experimental results, theoretical calculations, and various reviews of topics in our areas of expertise. The last subsection includes abstracts of papers dealing with techniques of measurement and analysis and with research facilities.

Measurements

1.1

NEUTRON CAPTURE IN THE 1.15-keV RESONANCE OF IRON*

R. L. Macklin

[Abstract of *Nucl. Sci. Eng.* 83, 309 (1983)]

A direct measurement of the capture area in the 1.1515 ± 0.0007 keV ($^{56}Fe + n$) resonance leads to $g\Gamma_n\Gamma_\gamma/\Gamma = 68.7 \pm 1.5$ MeV, $\sigma_0\Gamma_\gamma = 161 \pm 4$b-eV.

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

1.2

NEUTRON CAPTURE IN s-WAVE RESONANCES OF ^{64}Ni*

K. Wisshak[†] F. Käppeler[†]
R. L. Macklin G. Reffo[‡]
F. Fabbri[‡]

(Abstract of KfK-3582, September 1983)

The neutron capture widths of the s-wave resonances at 13.9 and 33.8 keV in ^{64}Ni have been determined using a setup with extremely low neutron sensitivity completely different from all previous experiments on this isotope. This feature is important because these resonances exhibit a very large scattering to capture ratio. A pulsed 3-MV Van de Graaff accelerator and a kinematically collimated neutron beam, produced via the ^{7}Li(p,n) reaction, was used in the experiments. Capture gamma rays were observed by three Moxon-Rae detectors with graphite, bismuth-graphite, and bismuth converters, respectively. The samples were positioned at a neutron flight path of only 6-8 cm. Thus events due to capture of resonance scattered neutrons in the detectors or in surrounding materials are completely discriminated by their additional time of flight. The short flight path and the high neutron flux at the sample position allowed for a signal-to-background ratio of ~1 even for the broad resonance at 33.8 keV. The data obtained with the individual detectors were corrected for the efficiency of the different converter materials. For that purpose, detailed theoretical calculations of the capture gamma-ray spectra of the measured isotope and of gold, which was used as a standard, were performed. The final radiative widths are Γ_γ (13.9 keV) = 1.01 ± 0.07 eV and Γ_γ (33.8 keV) = 1.16 ± 0.08 eV, considerably smaller than the rough estimates obtained in previous work.

*Research sponsored by Kernforschungszentrum Karlsruhe GmbH, Karlsruhe.
†Institut für Kernphysik, Karlsruhe.
‡Dell'Energia Nucleare e Delle Energie Alternative, Bologna, Italy.

1.3

RESONANCE NEUTRON CAPTURE IN ^{86,87}Sr*

G. C. Hicks[†] B. J. Allen[‡]
A. R. de L. Musgrove[‡] R. L. Macklin

[Abstract of *Aust. J. Phys.* 35, 267 (1982)]

The neutron capture cross sections of ^{86,87}Sr have been measured with high energy resolution from 3 to 200 keV at the 40 m station of the Oak Ridge Electron Linear Accelerator. Individual resonances were analyzed to 37 keV for ^{86}Sr and to 14 keV for ^{86}Sr, and average resonance parameters were deduced on the basis of assumed divisions between s- and p-wave resonances. The average radiative widths obtained on this basis are consistent with a capture mechanism which is predominantly statistical.

*Research sponsored by U.S. DOE Division of Physics.
†James Cook University of North Queensland, Townsville, Queensland, Australia.
‡Lucas Heights Research Laboratories, Sutherland, New South Wales.

1.4

TECHNETIUM-99 NEUTRON CAPTURE CROSS SECTION*

R. L. Macklin

[Abstract of *Nucl. Sci. Eng.* **81**, 520 (1982)]

The ^{99}Tc(n,γ) average cross section was measured at the Oak Ridge Electron Linear Accelerator from 2.65 to 2000 keV with an estimated uncertainty ranging from 4.0 to 4.9%. Individual resonance parameters were fitted by least-squares adjustment to the capture yield data from 2.65 to 5.08 keV. The average cross-section data agree with the ENDF/B-V (Mod 1) evaluation above 900 keV and lie within $\pm$15% of the JENDL-1 evaluation below 700 keV.

*Research sponsored by U.S. DOE Division of Nuclear Sciences.

1.5

NEUTRON CAPTURE CROSS SECTIONS OF THE SILVER ISOTOPES ^{107}Ag AND ^{109}Ag FROM 2.6 TO 2000 keV*

R. L. Macklin

[Abstract of *Nucl. Sci. Eng.* **82**, 400 (1982)]

Prompt neutron capture from highly enriched samples of the stable silver isotopes was measured at the Oak Ridge Electron Linear Accelerator neutron time-of-flight facility. Resonance peaks were parameterized from 2.65 to 7 keV, and average capture cross sections were derived as a function of energy up to 2000 keV. The average values for the ^{109}Ag(n,γ) cross section are a few percent smaller than for ^{107}Ag up to 700 keV, above which energy they drop more rapidly, falling to ~60% of the ^{107}Ag(n,γ) cross section at 2000 keV. Average radiation widths found for spin 1 resonances were 152 $\pm$ 7 meV for ^{107}Ag and 146 $\pm$ 6 meV for ^{109}Ag. Maxwellian average cross sections for kT = 30 keV are 801 mb for ^{107}Ag and 778 mb for ^{109}Ag with estimated uncertainties of 3%.

*Research sponsored by U.S. DOE Division of Nuclear Sciences.

1.6

NEUTRON CAPTURE CROSS SECTIONS AND RESONANCES OF ^{127}I AND ^{129}I*

R. L. Macklin

[Abstract of *Nucl. Sci. Eng.* **85**(4), 350 (1983)]

Neutron capture by ^{127,129}I has been measured using the Oak Ridge Electron Linear Accelerator (ORELA) as a pulsed neutron source. Neutron energies were determined by time-of-flight. Resonance peaks were parametrized for radioactive ^{129}I up to 3400 eV, and for stable ^{127}I from 2660 eV to 4260 eV. Average capture cross sections were derived for ^{129}I from 3 keV to 500 keV and for ^{129}I from 3 keV to 2200 keV. Over the 3 keV to 100 keV range, the ^{129}I cross sections average about 70% of the corresponding ^{127}I cross sections but show much more fluctuation as a function of energy. The greater fluctuation is attributed to the approximately three times wider level spacing.

*Research sponsored by U.S. DOE Division of Nuclear Sciences.

1.7

CESIUM-133 NEUTRON CAPTURE CROSS SECTION*

R. L. Macklin

[Abstract of *Nucl. Sci. Eng.* **81**, 418 (1982)]

The ^{133}Cs(n,γ) cross section was measured at the Oak Ridge Electron Linear Accelerator from 2.66 to 600 keV with 3 to 4% uncertainty. Individual resonance parameters were determined by least-squares adjustment to fit the yield data below 6 keV and the average cross section was derived up to 600 keV.

*Research sponsored by U.S. DOE Division of Nuclear Sciences.

1.8

CROSS SECTIONS OF THE ^{169}Tm (n,γ) REACTION FROM 2.6 keV TO 2 MeV*

R. L. Macklin **D. M. Drake**[†]
J. J. Malanify[†] **E. D. Arthur**[†]
P. G. Young[†]

[Abstract of *Nucl. Sci. Eng.* **82**, 143 (1982)]

Neutron capture cross sections have been measured for ^{169}Tm from 3 to 2000 keV at the Oak Ridge Electron Linear Accelerator 40-m station. The data were analyzed for individual resonance parameters up to 4.2 keV. Average strength functions have been deduced. Compound nucleus calculations, made with deformed optical model parameters, agree with experimental cross sections. Our cross sections for lower neutron energies tend to be somewhat less than those from earlier measurements.

*Research sponsored by U.S. DOE Office of Basic Energy Sciences.
[†]Los Alamos National Laboratory, Los Alamos, NM.

1.9

^{178,179,180}Hf AND ^{180}Ta(n,γ) CROSS SECTIONS AND THEIR CONTRIBUTION TO STELLAR NUCLEOSYNTHESIS*

H. Beer[†] **R. L. Macklin**

[Abstract of *Phys. Rev. C* **26**(4), 1404 (1982); paper presented at the International Conference on Nuclear Data for Science and Technology, Antwerp, Belgium, September 6-10, 1982; Proc. p. 945, K. H. Bockoff, Ed. (1983)]

The neutron capture cross sections of ^{178,179,180}Hf were measured in the energy range 2.6 keV to 2 MeV. The average capture cross sections were calculated and fitted in terms of strength functions. Resonance parameters for the observed resonances below 10 keV were determined by a shape analysis. Maxwellian averaged capture cross sections were computed for thermal energies kT between 5 and 100 keV. The cross sections for $kT = 30$ keV were used to determine the population probability of the 8^- isomeric level in ^{180}Hf by neutron capture as $(1.24 \pm 0.06)\%$ and the r-process abundance of ^{180}Hf as 0.0290 (Si $= 10^6$). These quantities served to analyze s- and r-process nucleosynthesis of ^{180}Ta.

*Research sponsored by U.S. DOE Division of Nuclear Sciences.
[†]Kernforschungszentrum Karlsruhe GmbH, Germany.

1.10

NEUTRON CAPTURE CROSS SECTIONS OF TANTALUM FROM 2.6 TO 1900 keV*

R. L. Macklin

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

Neutron capture by a tantalum sample was measured at the Oak Ridge Electron Linear Accelerator (ORELA) pulsed neutron time-of-flight facility on a 40 m flight path. Average cross sections for ^{181}Ta(n,λ) in the energy range from 2.6 to 1900 keV were derived. The partially resolved region from 2620 to 4000 eV was fitted in terms of resonance parameters by least squares adjustment.

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

1.11

NEUTRON CAPTURE CROSS SECTIONS OF ^{182}W, ^{183}W, ^{184}W, AND ^{186}W FROM 2.6 TO 2000 keV*

R. L. Macklin **D. M. Drake**[†]
E. D. Arthur[†]

[Abstract of *Nucl. Sci. Eng.* **84**, 98 (1983)]

Neutron capture cross sections of four stable tungsten isotopes were measured as a function of energy by time of flight at the Oak Ridge Electron Linear Accelerator. The resolution achieved, $\Delta E/E$ of $\sim 1/750$ at full-width at half-maximum, has allowed the analysis of several hundred resonance peaks at energies a few kiloelectron volts above the neutron binding energy. Strength functions were fitted to the average cross sections up to $\sim$100 keV, and

average cross sections were extended with less precision from 100 to 2000 keV. The capture cross section of natural tungsten was calculated from measurements for individual isotopes. Compound nucleus calculations have been made with deformed optical model parameters for comparison with experimental cross sections.

*Research sponsored by U.S. DOE Division of Nuclear Sciences.

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‡Los Alamos National Laboratory, Theoretical Division, Los Alamos, NM.

1.12

COMPARISON OF MEASURED AND CALCULATED ^{238}U CAPTURE SELF-INDICATION RATIOS FROM 4 TO 10 keV*

R. B. Perez G. de Saussure
J. T. Yang† J. L. Munoz-Cobos‡
J. H. Todd§

[Abstract of *Trans. Am. Nucl. Soc.* 44. 537 (1983); also paper presented at the Second Jackson Hole Colloquium on Fast Reactor Physics: The Doppler Effect in LMFBRs, Teton Village, Wyoming, June 27-29, 1983; Proc. Session III, No. 4 (1983)]

From 4 to 149 keV, the ^{238}U cross sections are represented in ENDF/B-V by unresolved-resonance parameters (URPs). The purpose of this representation is to enable the calculation of resonance self-protection as a function of temperature and dilution. Since the URPs are not defined unambiguously by the cross-section data, it is important that the unresolved representation be tested with appropriate experiments, such as capture self-indication ratio (SIR) measurements. In this paper, we compare ^{238}U capture SIR measurements in the 4- to 10-keV energy range with calculations done with ENDF/B-V and with recently published resolved-resonance parameters.

The above comparisons suggest that:

1. The measured ^{238}U capture SIR has a considerable amount of structure as a function of energy.

2. The ENDF/B-V unresolved-resonance representation fails to reproduce this structure.

3. Calculations done with available resolved-resonance parameters in the 4- to 6-keV interval reproduce the structure fairly well.

4. From 6.5 to 10 keV, ENDF/B-V underestimates resonance self-protection for the thicker samples.

5. Below 5.5 keV, ENDF/B-V overestimates resonance self-protection for the thin samples but is in agreement with the measurements for the thick samples.

6. Calculations based on resolved parameters agree significantly better with the measurements than those based on ENDF/B-V.

These considerations underscore the need to extend the ^{238}U resolved-resonance representation of ENDF/B to higher energies if the 1% accuracies in computed ^{238}U self-shielding factors requested by reactor designers is to be achieved.

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

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‡Universidad Politecnica de Valencia-Spain.

§Instrumentations and Controls Division.

1.13

OVERLAPPING β DECAY AND RESONANCE NEUTRON SPECTROSCOPY OF LEVELS IN ^{87}Kr*

S. Raman† B. Fogelberg‡
J. A. Harvey R. L. Macklin
P. H. Stelson† A. Schröder§
K. L. Kratz§

[Abstract of *Phys. Rev. C* 28(2), 602 (1983); also *Bull. Am. Phys. Soc.* 27(7), 727 (1982)]

Energy levels in ^{87}Kr have been studied, with special emphasis on the unbound region, using two different methods. The first method comprises neutron capture and transmission measurements on an enriched gas target of ^{87}Kr using neutron time-of-flight techniques. In this way, neutron widths were determined for 39 resonances below 400 keV and capture areas for 14 resonances below 90 keV. The second method is a decay study of 56-s ^{87}Br in which a level scheme for ^{87}Kr has been established that shows 126 levels in the bound and 12 levels in the unbound region. A detailed comparison amongst the neutron resonance, the γ-ray decay, and available delayed neutron results has been made.

Almost a one-to-one correspondence exists between the currently observed *p*-wave resonances below 250 keV and levels in ^{87}Kr studied through delayed neutron emission. The overall β-strength distribution derived from the present data shows broad resonance-like structures. However, no marked selectivity is observed in the β decay to individual levels in the unbound region of ^{87}Kr. The neutron capture cross section of ^{87}Kr is found to be about 5 mb for 30-keV neutrons with a Maxwellian energy distribution. The future of delayed neutron spectroscopy as a new tool for obtaining level-density information is discussed.

*Research sponsored by U.S. DOE Division of Nuclear Sciences.

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§Institut für Kernchemie, Universität Mainz, D-65 Mainz, Federal Republic of Germany.

1.14

NEUTRON SPECTROSCOPY AS A HIGH-RESOLUTION PROBE: IDENTIFICATION OF THE MISSING $1/2^+$ STATES IN ^{31}Si*

J. A. Harvey W. M. Good†
R. F. Carlton‡ B. Castel§
J. B. McGrory† S. F. Mughabghab¶

[Abstract of *Phys. Rev. C* **28**(1), 24 (1983)]

The neutron total cross section of ^{30}Si$+n$ was measured from 0.2 to 1400 keV and the data were analyzed to obtain resonance parameters. Two s-wave resonances corresponding to $1/2^+$ states in ^{31}Si predicted from shell model calculations were found. Contrary to earlier *(d,p)* measurements, the present measurement indicates a fairly uniform fragmentation of p-wave strength, in agreement with earlier weak coupling shell model calculations of negative parity states for ^{29}Si and ^{33}S.

*Research sponsored by U.S. DOE Division of Nuclear Sciences.

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1.15

SOLID STATE EFFECTS ON THERMAL NEUTRON CROSS SECTIONS AND ON LOW ENERGY RESONANCES*

J. A. Harvey H. A. Mook†
N. W. Hill‡ O. Shahal§

[Abstract of paper presented at the International Conference on Nuclear Data for Science and Technology, Antwerp, Belgium, September 6-10, 1982; Proc. p. 961, K. H. Bockoff, Ed. (1983)]

The neutron total cross sections of several single crystals (Si, Cu, sapphire), several polycrystalline samples (Cu, Fe, Be, C, Bi, Ta), and a fine powder copper sample have been measured from 0.002 to 5 eV. The Cu powder and polycrystalline Fe, Be, and C data exhibit the expected abrupt changes in cross section. The cross section of the single crystal of Si is smooth with only small broad fluctuations. The data on two "single" Cu crystals, the sapphire crystal, cast Bi, and rolled samples of Ta and Cu have many narrow peaks $\sim 10^{-3}$ eV wide. High resolution (0.3%) transmission measurements were made on the 1.057-eV resonance in ^{240}Pu and the 0.433-eV resonance in ^{180}Ta, both at room and low temperatures to study the effects of crystal binding. Although the changes in Doppler broadening with temperature were apparent, no asymmetries due to a recoilless contribution were observed.

*Research sponsored by U.S. DOE Division of Nuclear Sciences.

†Solid State Division.

‡Instrumentation and Controls Division.

§Nuclear Research Centre-Negev, Beer-Sheva, Israel.

1.16

UNBOUND STATES OF ^{35}S AND DOORWAY STATE CALCULATIONS*

R. F. Carlton† J. A. Harvey
W. M. Good‡ B. Castel§

[Abstract of *Bull. Am. Phys. Soc.* 27(7), 712 (1982)]

Total cross section measurements have been made on an enriched ^{34}S (90.0%) sample ($1/N_t$ = 10.79) from 20 eV to 1.45 MeV using the 80-m flight path at ORELA. The measurements were made using a ^{6}Li glass scintillator up to ~50 keV and using a NE-110 scintillation detector above ~50 keV with an energy resolution of 0.1% below 250 keV and $0.2\%\sqrt{E(in\ MeV)}$ above 250 keV. The data were analyzed with the R-matrix multilevel code SAMMY to obtain resonance parameters for s-, p-, and d-wave neutrons. The s- and p-wave neutron widths were compared with a doorway state calculation using shell model parameters determined by a ^{35}Cl-^{35}S bound state calculation. Agreement with data is satisfactory enough to assign theoretical doorway state configurations to specific groups of resonances.

*Research sponsored by U.S. DOE Division of Nuclear Sciences.

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1.17

APPLICATION OF NEW TECHNIQUES TO ORELA NEUTRON TRANSMISSION MEASUREMENTS AND THEIR UNCERTAINTY ANALYSIS: THE CASE OF NATURAL NICKEL FROM 2 keV TO 20 MeV*

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

(Abstract of ORNL/TM-8203, October 1983)

The neutron transmission through a 2.54-cm sample of natural nickel has been measured for neutron energies between 2 keV and 20 MeV. The Oak Ridge Electron Linear Accelerator (ORELA) was used to provide the neutrons which were detected at the 200-m flight path by a NE110 proton recoil detector. A selective rating system was utilized to minimize background effects due to large light-level events which produce phototube afterpulsing and long decay-constant light emission in the detector. A detailed discussion of the development of this system is given.

Known background sources are described, and the methods used to correct for these backgrounds are presented. An in-depth uncertainty analysis is given for this measurement, with explicit formulas derived for each effect contributing to the cross-section uncertainty. Parameter uncertainties and correlations among the parameters describing the backgrounds, deadtime, and other sources of uncertainty are given. To obtain a covariance matrix for this measurement, the final cross-section results are binned into 15 energy groups, and a covariance matrix is provided for this 15-group set. We find that the largest contributions to the cross-section uncertainty are due to sample properties, beam monitors (used to normalize sample-in and sample-out counts), and ORELA power variations during the run which affect the deadtime correction. Overall uncertainties in the cross section for this measurement are on the order of 2%. The resulting cross sections are compared with the ENDF/B-V file for nickel; any resonances not presently in the file are observed, and energy-scale differences are noted.

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

†UCC-ND Computer Sciences Division.
‡Instrumentation and Controls Division.
§Physics Division.

1.18

HIGH RESOLUTION NEUTRON TOTAL CROSS SECTION IN THE SEPARATED ISOTOPES OF ^{52}Cr AND ^{54}Cr*

H. M. Agarwal† J. B. Garg† J. A. Harvey

[Abstract of *Bull. Am. Phys. Soc.* 27(7), 716 (1982)]

High resolution neutron total cross sections in the energy interval of 10-900 keV have been performed in the enriched isotopes of chromium with masses 52 and 54. These measurements were made with the ORELA facility using time-of-flight techniques. From these data, parameters (E, Γ_n, J^π) of s and p-wave resonances were obtained using an R-matrix multilevel analysis. Moreover we have investigated the statistical properties of level parameters such as neutron reduced width distribution, short and long range

spacing distributions and neutron strength functions. A detailed discussion of these results is presented.

*Research sponsored by U.S. DOE Office of Basic Energy Sciences.
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1.19

HIGH RESOLUTION NEUTRON RESONANCE SPECTROSCOPY IN ^{89}Y*

H. M. Agarwal† J. B. Garg†
J. A. Harvey

[Abstract of *Bull. Am. Phys. Soc.* 28(4), 736 (1983)]

The neutron total cross section in ^{89}Y in the energy range of 10 keV to 800 keV has been measured using ORELA facility. R-matrix analysis of this high resolution data has provided values of E_o, Γ_n, J^π for s- and p-wave resonances. From these results the values of average resonance parameters such as level spacing and strength functions for s- and p-wave neutron interactions have been obtained. The s- and p-wave neutron reduced width distributions have also been studied. Moreover we have investigated $S_o(J^\pi = 0^-)$, $S_o(J^\pi = 1^-)$, $S_1(J^\pi = 0^+)$, $S_1(J^\pi = 1^+)$, $S_1(J^\pi = 2^+)$ vs. energy, respectively, in search of intermediate structures which were theoretically predicted by Ramavataram *et al.* A detailed discussion of these findings is presented.

*Research sponsored by U.S. DOE Office of Basic Energy Sciences.
†State University of New York-Albany.

1.20

^{187}Os + *n* RESONANCE PARAMETERS IN THE INTERVAL 27-500 eV NEUTRON ENERGIES*

R. R. Winters† R. F. Carlton‡
J. A. Harvey N. W. Hill§

[Abstract of paper presented at the International Conference on Nuclear Data for Science and Technology, Antwerp, Belgium, September 6-10, 1982; Proc. p. 943, K. H. Bockoff, Ed. (1983)]

The neutron total cross section for ^{187}Os in the energy range 27 eV to 500 eV has been measured at the ORELA facility by the neutron time-of-flight technique, utilizing a 2.0 g osmium sample (n = 0.008401 Os-nuclei/barn) enriched to 70.38% ^{187}Os. Measurements were performed at a 80 m flight station with an energy resolution, $\Delta E/E$, of 0.1% using a ^{6}Li glass scintillator. Resolved resonances have been analyzed by a Reich-Moore multilevel code (SAMMY) to obtain parameters for 85 resonances up to 500 eV. Preliminary determinations of the level spacing (5 eV) and s-wave strength function (3.9 x 10^{-4}) for ^{187}Os are in agreement with recent analyses of the osmium isotopes, made in connection with the use of the Re/Os chronometer for estimating the duration of stellar nucleosynthesis.

*Research sponsored by U.S. DOE Division of Nuclear Sciences.
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1.21

THE TOTAL NEUTRON CROSS SECTIONS OF ^{249}Bk AND ^{249}Cf BELOW 100 eV*

R. W. Benjamin† J. A. Harvey
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R. F. Carlton‖

[Abstract of *Nucl. Sci. Eng.* 85, 261 (1983)]

The neutron total cross sections of ^{249}Bk and ^{249}Cf have been measured from 0.03 to 100 eV using the Oak Ridge Electron Linear Accelerator as a source of pulsed neutrons. The 1.6-mm-diam cylindrical transmission samples initially contained up to 5.3 mg of 98% ^{249}Bk and 2%

^{249}Cf; 4.5 yr later, when the final measurements were made, the composition of the samples had become 2.5% ^{249}Bk, 96.9% ^{249}Cf, and 0.6% ^{245}Cm. Samples were cooled with liquid nitrogen to reduce Doppler broadening. Thirty-nine resonances were identified in ^{249}Bk and analyzed using a single-level Breit-Wigner formalism. Fifty-five resonances were identified in ^{249}Cf and analyzed using an R-matrix multilevel formalism. The resonance parameters obtained have been used to determine the average level spacings and the s-wave neutron and fission strength functions. Where possible, bound-level parameters were derived to fit the thermal neutron total cross-section data.

*Research sponsored by U.S. DOE Division of Nuclear Sciences.

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1.22

MEASUREMENT OF THE $^{232}Th(n,f)$ SUBTHRESHOLD AND NEAR-SUBTHRESHOLD CROSS SECTION*

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J. H. Todd† J. T. Yang‡
G. F. Auchampaugh§

[Abstract of *Phys. Rev. C* 28(4), 1635 (1983)]

A measurement of the $^{232}Th(n,f)$ cross section for incident neutron energies between 100 eV and 1.6 MeV has been performed at the Oak Ridge Electron Linear Accelerator. The weak subthreshold fission cross section found in this measurement confirms the model of a low first barrier in the triple-humped fission barrier which has been theoretically predicted for the (^{232}Th + n) system. However, the appearance of a series of plateaus in the near-threshold fission cross section region presents a challenge to current barrier calculations in the ^{233}Th compound nucleus.

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

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1.23

HIGH-RESOLUTION MEASUREMENTS AND R-MATRIX ANALYSIS OF THE TOTAL AND FISSION CROSS SECTIONS OF ^{237}Np + n FROM 1 TO 600 eV*

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J. D. Moses† R. O. Nelson†
C. E. Olsen† R. C. Extermann†
N. W. Hill‡ J. A. Harvey

(Abstract of Los Alamos Report LA-9756-MS, May 1983)

High-resolution measurements of the total and fission cross sections of ^{237}Np + n have been made using the pulsed-neutron facilities at the Oak Ridge Electron Linear Accelerator and at the Los Alamos Meson Physics Facility. The samples were cooled to liquid-nitrogen temperature. This report presents the total and fission cross sections from 1 to 600 eV, as well as the parameters obtained from an R-matrix fit of these data.

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

†Los Alamos National Laboratory, Los Alamos, NM.
‡Instrumentation and Controls Division.

1.24

NEUTRON FISSION CROSS SECTIONS OF ^{239}Pu AND ^{240}Pu RELATIVE TO ^{235}U*

L. W. Weston J. H. Todd†

[Abstract of *Nucl. Sci. Eng.* 84, 248 (1983)]

The ratios of the neutron fission cross sections $\sigma_f(^{240}Pu)/\sigma_f(^{239}Pu)$, $\sigma_f(^{240}Pu)/\sigma_f(^{235}U)$, and $\sigma_f(^{239}Pu)/\sigma_f(^{235}U)$ have been measured simultaneously with a multiplate ionization fission chamber using the Oak Ridge Electron Linear Accelerator as a neutron source over the neutron energy range from 5 keV to 20 MeV. The ^{240}Pu ratio data are in overall agreement with ENDF/B-V with exceptions in relatively narrow neutron energy regions. Below 150 keV and from 10 to 20 MeV, the present ^{239}Pu/^{235}U fission ratios indicate significant discrepancies when compared to ENDF/B-V. These ratios are important for thermal and fast reactor applications.

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

†Instrumentation and Controls Division.

1.25

MEASUREMENT OF THE ^{241}Am NEUTRON FISSION CROSS SECTION*

J. W. T. Dabbs C. H. Johnson†
C. E. Bemis, Jr.†

[Abstract of *Nucl. Sci. Eng.* 83, 22 (1983)]

The fission cross section of ^{241}Am has been measured from 0.02 eV to 20 MeV using time-of-flight techniques at the Oak Ridge Electron Linear Accelerator. A "honeycomb" fission ionization chamber that contained six deposits totaling 1.43 mg of ^{241}Am, six deposits totaling 116 mg of ^{235}U, and a single deposit of ^{252}Cf, which served as a monitor for the chamber performance, was used. The ^{235}U fission served as the cross-section standard for energies above 101 keV while ^{6}Li(n,α), normalized to ^{235}U fission in the 7.8- to 11.0-eV interval, served as a shape standard below 101 keV. Approximately 700 h of data were obtained at a flight path distance of 9.1 m, primarily with 40-ns bursts. Because the fission cross section of ^{241}Am is very small in the midrange of neutron energies, particular attention was paid to correction of backgrounds, particularly inscattered neutron-induced events. The fission resonance integral was found to be 14.1 $\pm$ 0.9/ b.

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

†Physics Division.

1.26

MEASUREMENT OF THE ^{242m}Am NEUTRON FISSION CROSS SECTION*

J. W. T. Dabbs C. E. Bemis, Jr.†
S. Raman†

[Abstract of *Nucl. Sci. Eng.* 84, 1 (1983)]

The fission cross section of ^{242m}Am has been measured from 0.005 eV to 20 MeV using time-of-flight techniques at the Oak Ridge Electron Linear Accelerator. A hemispherical plate fission ionization chamber with five pairs of plates contained three deposits totaling 507 μg of ^{242m}Am, one deposit of 168 μg ^{235}U, and a "weightless" deposit of ^{252}Cf, which served as a monitor of chamber performance. The fission of ^{235}U served as the cross-section standard for energies above 101 keV while ^{6}Li(n,α), normalized to ^{235}U fission in the 7.8- to 11.0-eV interval, served as a shape standard below 101 keV. Approximately 360 h of data were obtained at a flight path distance of 9.1 m, primarily with 40-ns bursts. Particular attention was paid to correction of backgrounds, especially inscattered-neutron-induced events. The fission resonance integral was found to be 1800 $\pm$ 65/ b.

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

†Physics Division.

1.27

FISSION CROSS SECTION MEASUREMENTS OF ^{244}Cm, ^{246}Cm AND ^{248}Cm†*

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R. Hoff‖ R. Lougheed‖

[Abstract of paper presented at ANS Topical Meeting on Advances in Reactor Physics and Core Thermal Hydraulics, Kiamesha Lake, NY, September 22-24, 1982; Proc. NUREG/CP-0034, Vol. 1, p. 1090 (1982)]

The RINS (Rensselaer Intense Neutron Spectrometer) system, which includes a 75-ton lead slowing-down-time spectrometer and the RPI

100-MeV electron linac, has been used to measure the neutron-induced fission cross sections of ^{244}Cm, ^{246}Cm and ^{248}Cm. These isotopes have either large spontaneous fission background (e.g., ^{248}Cm) or large alpha decay rates (e.g., ^{244}Cm), and it required the high neutron intensity of the RINS system to carry out these measurements. Preliminary measurements of ^{248}Cm, ^{246}Cm and ^{248}Cm have been reported previously, and this paper includes the earlier results, combined with the results of a recent measurement in which all three isotopes were measured simultaneously. The fission chamber consists of five pairs of hemispherical electrodes contained in 2 atm of methane, and for the latest measurements the electrodes were coated with ^{244}Cm (5.2 μg), ^{246}Cm (17 μg), ^{248}Cm (35 μg), ^{235}U and ^{252}Cf. The ^{235}U chamber was used for normalization, and the ^{252}Cf chamber monitored any effects caused at early slowing-down times by the linac gamma flash or by RF pickup from the pulsing of the accelerator.

These measurements span the energy range from 0.1 eV to 100 keV. The data below ~20 eV are the first obtained in this energy range, and the data above ~20 eV complement the nuclear explosion time-of-flight measurements. Low-energy resonances have been resolved at 7.7, 17, 23 and 35 eV in ^{244}Cm; 4.3 and 15 eV in ^{246}Cm; and 7.3, 27 and 76 eV in ^{248}Cm. Parameters will be determined for these resonances.

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

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1.28

FAST NEUTRON INELASTIC SCATTERING FROM ^{56,57}Fe*

J. K. Dickens

[Abstract of *Bull. Am. Phys. Soc.* 27(7), 716 (1982)]

ORELA produced neutrons were used to measure gamma-ray production cross sections due to neutron reactions with ^{56,57}Fe for incident neutron energies $0.16 \leq E_n \leq 40$ MeV. Data reduction for ^{57}Fe$(n,n'\gamma)$ are compared with compound-nucleus calculations. The agreement is reasonable for $E_n < 3$ MeV.

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

1.29

NEUTRON-INDUCED GAMMA-RAY PRODUCTION IN ^{57}Fe FOR INCIDENT-NEUTRON ENERGIES BETWEEN 0.16 AND 21 MeV*

Z. W. Bell† J. K. Dickens
D. C. Larson J. H. Todd‡

[Abstract of *Nucl. Sci. Eng.* 84, 12 (1983)]

Interactions of neutrons with the iron isotope ^{57}Fe have been studied by measuring gamma-ray production cross sections for incident-neutron energies between 0.16 and 21 MeV. Neutrons produced by the Oak Ridge Electron Linear Accelerator impinged on a metallic iron sample enriched to 93% in the isotope ^{57}Fe. The resulting gamma radiation was detected using a 100-cm^3 Ge(Li) detector placed at 125 deg with respect to the neutron beam line. A complete description of the experiment is given. Absolute gamma-ray production cross sections were measured for gamma rays corresponding to the ^{57}Fe$(n,n'\gamma)^{57}$Fe, ^{57}Fe$(n,\gamma)^{58}$Fe, ^{57}Fe$(n,\alpha)^{54}$Cr, ^{57}Fe$(n,2n)^{56}$Fe, and ^{57}Fe$(n,p)^{57}$Mn reactions. The cross section for the ^{57}Fe$(n,2n)^{56}$Fe reaction exceeds 1 b for E_n ~15 MeV, and the cross section for the ^{57}Fe$(n,p)^{57}$Mn reaction exceeds 0.2 b for E_n ~9 MeV. A new excited state is postulated for ^{57}Mn to account for observed data. Several new transitions are reported for decay of levels in ^{57}Fe. Measured cross sections are compared with data obtained from the current ENDF/B evaluation.

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

†UCC-ND Computer Sciences Division.

‡Instrumentation and Controls Division.

1.30

GAMMA-RAY DECAY OF LEVELS IN ^{63}Cu AND ^{65}Cu*

J. K. Dickens

[Abstract of *Nucl. Phys.* **A401**, 189 (1983)]

Gamma-ray decay of levels in the stable copper isotopes ^{63}Cu and ^{65}Cu has been studied using ^{63,65}Cu($n,n'\gamma$) reactions for incident neutron energies between threshold and 6 MeV. Of the 257 γ-rays or γ-ray groups observed for neutron interactions with ^{63}Cu, 137 have been placed or tentatively placed among 88 levels in ^{63}Cu up to an excitation energy of 5.9 MeV. Similarly, of 335 γ-rays or γ-ray groups observed for neutron interactions with ^{65}Cu, 127 have been placed among 100 levels in ^{65}Cu up to an excitation energy of 5.8 MeV, including new decay data for >70 excited states of ^{65}Cu.

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

1.31

GAMMA RAY PRODUCTION DUE TO NEUTRON INTERACTIONS WITH COPPER FOR NEUTRON ENERGIES BETWEEN 0.7 AND 10.5 MeV*

G. G. Slaughter† J. K. Dickens

[Abstract of *Nucl. Sci. Eng.* **84**, 395 (1983)]

Differential cross sections for 22 gamma rays produced by neutron interactions with elemental copper having energies, E_γ, between 0.36 and 2.5 MeV have been measured for neutron energies, E_n, between 0.7 and 10.5 MeV for $\theta_\gamma = 90°$ using a Ge(Li) detector system. The present data are compared with previously reported measurements; agreement with earlier reported data is generally good for $E_n < 3$ MeV and generally poor for $E_n > 4$ MeV.

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

1.32

CROSS SECTIONS FOR THE Mo(n,xn) REACTIONS BETWEEN 3 AND 21 MeV*

Z. W. Bell†

(Abstract of ORNL/TM-8863, October 1983)

Differential cross sections for Mo(n,xn) reactions have been measured for incident neutron energies between 3 and 21 MeV and for secondary neutron energies between 2 and 21 MeV. The data are compared with current ENDF/B evaluated data and with experimental data of two recent measurements. Discrepancies with evaluation for E_n between 5 and 9 MeV are discussed but not resolved. The present data agree reasonably well with evaluation for $E_n > 10$ MeV and with previous experimental data for $E_n = 4$ MeV.

*Research sponsored by U.S. DOE Office of Basic Energy Sciences.
†UCC-ND Computer Sciences Division.

1.33

NEUTRON SCATTERING FROM Os ISOTOPES AT 60 keV AND Re/Os NUCLEOCHRONOLOGY*

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M. Balakrishnan† N. W. Hill‡
M. T. McEllistrem†

[Abstract of *Phys. Rev. C* **28**(6), 2249 (1983)]

Neutron elastic and inelastic scattering cross sections have been measured for 60.5 keV neutrons incident on ^{187,188}Os. For ^{187}Os the cross section for elastic scattering is 11 b and for inelastic scattering to the 9.75-keV level it is 1.13 b. These results and other scattering properties are represented in phase shift and scattering potential analyses, which then allow the prediction of neutron capture by ^{187}Os nuclei in the 9.75-keV and ground levels. Capture rates thus provided are used in the Re/Os nucleochronology, and lead to a long galactic age, about 10 to 20 Gyr.

*Research sponsored by U.S. DOE Office of Basic Energy Sciences.
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1.34

^{187}Os(*n*,*n*') INELASTIC CROSS SECTION AT 34 keV*

R. L. Macklin R. R. Winters[†]
N. W. Hill[‡] J. A. Harvey

[Abstract of *Astrophysical Journal* 274, 408 (1983)]

A measurement of the ^{187}Os inelastic neutron cross section to the $3/2^-$, 9.75-keV excited state with 34 ± 2 keV incident neutrons gives 1.5 ± 0.2 barns. Pulsed neutron time of flight and a short anisotropic iron-aluminum filter allowed separation of the inelastic yield from the $1/2^-$ ground-state elastic yield at 24 keV. The influence of the result on *r*-process galactic age calculations via the Re-Os decay chronometer is discussed, and the need for further theoretical calculations is emphasized.

*Research sponsored by U.S. DOE Division of Nuclear Sciences.
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[‡]Instrumentation and Controls Division.

1.35

GAMMA-RAY TRANSITIONS AMONG LEVELS OF ^{206}Pb*

J. K. Dickens

[Abstract of *Phy. Rev. C* 28(2), 916 (1983)]

A study of γ-ray data produced by neutron inelastic scattering from a lead sample enriched in the isotope ^{206}Pb has resulted in placements of 106 γ rays as transitions among 88 known or postulated levels of the ^{206}Pb level structure.

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

Cross-Section Analyses, Evaluations, and Reviews

1.36

EXPERIMENTAL AND CALCULATIONAL ANALYSES OF ACTINIDE SAMPLES IRRADIATED IN EBR-II*

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R. L. Walker[‡] S. Raman[§]
P. H. Stelson[§]

(Abstract of ORNL-5791, October 1982)

Higher actinides influence the characteristics of spent and recycled fuel and dominate the long-term hazards of the reactor waste. Reactor irradiation experiments provide useful benchmarks for testing the evaluated nuclear data for these actinides. During 1967-1970, several actinide samples were irradiated in the Idaho EBR-II fast reactor. These samples have now been analyzed, employing mass and alpha spectometry, to determine the heavy element products. A simple spherical model for the EBR-II core and a recent version of the ORIGEN code with ENDF/B-V data were employed to calculate the exposure products. A detailed comparison between the experimental and calculated results has been made. For samples irradiated at locations near the core center, agreement within 10% was obtained for the major isotopes and their first daughters, and within 20% for the nuclides up the chain. A sensitivity analysis showed that the assumed flux should be increased by 10%. The lessons learned from the present study were incorporated into the planning of a future irradiation study to be carried out in the UK Prototype Fast Reactor at Dounreay.

*Research sponsored by U.S. DOE Office of Basic Energy Sciences.
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[‡]Analytical Chemistry Division.
[§]Physics Division.

1.37

RESONANCE PARAMETERS OF ^{60}Ni + *n* FROM MEASUREMENTS OF TRANSMISSION AND CAPTURE YIELDS FROM 1 TO 450 keV*

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R. L. Macklin R. R. Winters†
F. G. Perey

[Abstract of ORNL-5893, November 1982; also *Phys. Rev. C* 27(6), 2556 (1983)]

High-resolution transmission and capture measurements of ^{60}Ni-enriched targets have been made from a few eV to 1800 keV in transmission and from 2.5 keV to 5 MeV in capture. The transmission data from 1 to 452 keV were analyzed with a multilevel $\dot{R}$-matrix code which uses the Bayes' theorem for the fitting process. This code provides the energies and neutron widths of the resonances inside the 1- to 452-keV region as well as a possible parameterization for outside resonances to describe the smooth cross section in this region. The capture data were analyzed from 2.5 to 452 keV with a least-squares fitting code using the Breit-Wigner formula. Average parameters for the 30 observed s-wave resonances were deduced. The average level spacing, D_o, was found to be equal to 15.2 ± 1.5 keV; the strength function, S_o, equal to (2.2 ± 0.6) × 10^{-4}; and the average radiation width, Γ_γ, equal to 1.30 ± 0.07 eV. The staircase plot of the reduced level widths and the plot of the Lorentz-weighted strength function averaged over various energy intervals show possible evidence for doorway states. The level densities calculated with the Fermi-gas model for ℓ = 0 and for $\ell > 0$ resonances were compared with the cumulative number of observed resonances, but the analysis is not conclusive. The correlation coefficient ρ between Γ_n^o and Γ_γ is equal to 0.53 ± 0.18. The average capture cross section as a function of the neutron incident energy is compared to the tail of the giant electric dipole resonance prediction.

*Research sponsored by U.S. DOE Office of Basic Energy Sciences.
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1.38

RESOLVED RESONANCE PARAMETERS FOR ^{238}U FROM 4 TO 6 keV*

D. K. Olsen P. S. Meszaros†

[Abstract of *Nucl. Sci. Eng.* 83, 174 (1983)]

Neutron widths for 145 resonances from 4 to 6 keV are reported from a least-squares shape analysis of the ORELA 150-m, 4-sample ^{238}U transmission data. The resultant s-wave strength function from 4 to 6 keV is found to be substantially smaller than that from 0 to 4 keV.

*Research sponsored by U.S. DOE Office of Reactor Research and Technology.
†UCC-ND Computer Sciences Division.

1.39

OBSERVED ℓ-DEPENDENCE OF NEUTRON OMP REAL WELL DEPTHS FOR ^{30}Si AND ^{34}S*

C. H. Johnson† J. A. Harvey
R. F. Carlton‡

[Abstract of *Bull. Am. Phys. Soc.* 28(4), 648 (1983)]

From the R-matrix parameters required to fit high resolution transmission data for 0.0 to 1.5 MeV neutrons on ^{30}Si and ^{34}S we have deduced average s- and p-wave scattering functions in the same manner as done previously for ^{32}S. When we fit these averages using a spherical optical model potential, we find, within the uncertainties of about ± 1 MeV, the same real well depths for ^{30}Si and ^{34}S as obtained previously for ^{32}S, namely 61.5 MeV for p-waves and 51.5 MeV for s-waves. (The real well had r_o=1.21 f and a=0.66 f.) For ^{32}S this ℓ-dependence has been explained in terms of deformations. It is reasonable that the ℓ-dependence for ^{30}Si and ^{34}S can also be attributed to deformations.

*Research sponsored by U.S. DOE Office of Basic Energy Sciences.
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‡Middle Tennessee State University, Murfreesboro, TN.

1.40

PROBLEMS AND PROGRESS REGARDING RESONANCE PARAMETERIZATION OF ^{235}U AND ^{239}Pu FOR ENDF/B*

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G. R. Smith‡

(Abstract of a paper presented at Symposium on Uranium and Plutonium Isotope Resonance Parameters, Vienna, Austria, September 28 - October 2, 1981)

The procedures used to obtain the resolved and unresolved resonance parameterization of ^{235}U and ^{239}Pu contained in the U.S. Evaluated Nuclear Data File ENDF/B-V are reviewed. For ^{235}U, recommendations are made to improve the representation by including information on resonance spins and fission-channel vector orientations, and some preliminary results are presented. We review evidence that it is the fission channels rather than the spins of the resonances that lead to differences in fission mass distributions, the number of neutrons emitted per fission, and fission kinetic energies. The improved parameterization may thus have physics content that will prove of interest in future applications.

*Research sponsored by U.S. DOE Division of Reactor Research and Technology.
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1.41

IMPLEMENTATION OF AN ADVANCED PAIRING CORRECTION FOR PARTICLE-HOLE STATE DENSITIES IN PRECOMPOUND NUCLEAR REACTION THEORY*

C. Y. Fu

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

An advanced pairing correction for an existing formula of particle-hole state densities, needed in calculation of cross sections with the precompound nuclear reaction theory, is examined. The Pauli correction is derived to be consistent with this pairing correction. The accuracy of the pairing correction plus the Pauli correction is shown to be sufficient for applied calculations. Numerical solutions of the pairing equations, needed for generating the corrections, have been carried out. The relevant numerical results are presented as simple functions of the excitation energy and the exciton number. A relationship between the pairing correction for particle-hole state densities and the pairing correction for the total state densities in the closed-form formulation is developed. Utilization of the existing level-density parameter and data for deducing parameters for the particle-hole state densities is shown.

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

1.42

PAIRING CORRECTION FOR PARTICLE-HOLE STATE DENSITIES*

C. Y. Fu

(Abstract of paper presented at IAEA Advisory Group Meeting on Basic and Applied Problems of Nuclear Level Densities, Upton, New York, April 11-15, 1983)

The pairing correction proposed by Ignatyuk and Sokolov for particle-hole state densities has been examined. It has been found that the accuracy of the correction is sufficient for practical applications only if the system is in its normal state ($\Delta = 0$). In the superfluid state ($\Delta \neq 0$), a consistent pairing-Pauli correction is developed here for improved accuracy. Practical implementations of the pairing correction are given and further developments are outlined.

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

1.43

COMMENT ON "DETERMINATION OF GAMMA-RAY ENERGIES AND ABUNDANCES OF ^{229}Th"*

J. K. Dickens

[Abstract of *Phys. Rev. C* **28**, 1404 (1983)]

It is shown that the absolute normalization of γ-ray yields (I_γ) for transitions among levels in

^{225}Ra following α decay of ^{229}Th reported by Rattan *et al.* should be increased by at least 22%. Disagreements of reported values of I_γ with those in a recent report are discussed but not resolved.

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

1.44

ELECTRON SPECTRA FROM DECAY OF FISSION PRODUCTS*

J. K. Dickens

(Abstract of ORNL/TM-8285, September 1982)

Electron spectra following decay of individual fission products ($72 \leqslant A \leqslant 162$) are obtained from the nuclear data given in the compilation using a listed and documented computer subroutine. Data are given for more than 500 radionuclides created during or after fission. The data include transition energies, absolute intensities, and shape parameters when known. An "average" beta-ray energy is given for fission products lacking experimental information on transition energies and intensities. For fission products having partial or incomplete decay information, the available data are utilized to provide best estimates of otherwise unknown decay schemes. This compilation is completely referenced and includes data available in the reviewed literature up to January 1982.

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

1.45

s-PROCESS STUDIES IN THE LIGHT OF NEW EXPERIMENTAL CROSS SECTIONS: DISTRIBUTION OF NEUTRON FLUENCES AND *r*-PROCESS RESIDUALS*

F. Käppeler[†] **H. Beer**[†]
K. Wisshak[†] **D. D. Clayton**[‡]
R. L. Macklin **R. A. Ward**[§]

[Abstract of *Astrophysical J.* 257, 821 (1982)]

A best set of neutron-capture cross sections has been evaluated for the most important *s*-process isotopes. With this data base, *s*-process studies have been carried out using the traditional model which assumes a steady neutron flux and an exponential distribution of neutron irradiations. The calculated σN curve is in excellent agreement with the empirical σN-values of pure *s*-process nuclei. Simultaneously, good agreement is found between the difference of solar and *s*-process abundances and the abundances of pure *r*-process nuclei. We also discuss the abundance pattern of the iron group elements where our *s*-process results complement the abundances obtained from explosive nuclear burning. The results obtained from the traditional *s*-process model such as seed abundances, mean neutron irradiations, or neutron densities are compared to recent stellar model calculations which assume the He-burning shells of red giant stars as the site for the *s*-process.

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

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1.46

NEW DEVELOPMENTS IN THE UNRESOLVED RANGE*

R. B. Perez **G. de Saussure**

[Abstract of paper presented at the Specialists' Workshop on ^{238}U Capture in Fast Reactors, Teton Village, Wyoming, June 22-24, 1982; Proc. 2B (1982)]

Considerable amounts of effort have gone into improving processing methods in the unresolved region but substantially less into validating them as a representation of the ^{238}U(n,γ) cross section in the unresolved region. The unresolved resonance-region methodology endeavors to: (1) reconstruct the infinitely diluted cross sections, and (2) provide a formalism to calculate self-shielding factors as a function of temperature and dilution.

The purpose of this paper is to examine the sensitivity of self-shielding factor calculations in the unresolved resonance region to both average resonance parameters and calculation methods. We have observed that: (a) presently available

average resonance parameters are not fully consistent for self-shielding factor calculations in the unresolved region, and (b) large uncertainties arise from statistical sampling errors in the Monte Carlo method utilized in the unresolved resonance region. These results lead to the conclusion that whenever resonance self-shielding is important, one should parametrize the gross features of the cross section so that resolved resonance region methods could be utilized, while treating only the smaller resonances by statistical methods or by a File 3 smooth contribution approach. In summary we recommend the extension of the ^{238}U resolved resonance region up to roughly 10 keV.

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

1.47

UNCERTAINTIES IN THE ^{238}U RESOLVED RESONANCE PARAMETERS AND THEIR IMPACT ON CALCULATED GROUP CONSTANTS*

G. de Saussure R. Q. Wright† R. B. Perez

[Abstract of paper presented at the Second Jackson Hole Colloquium on Fast Reactor Physics: The Doppler Effect in LMFBRs, Teton Village, Wyoming, June 27-29, 1983; Proc. Session III, No. 3 (1983)]

In assessing the contribution of the uncertainties in a data set to the uncertainty in a computed performance parameter (such as capture to fission ratio in a benchmark, or Doppler coefficient of reactivity), it is required to assess the uncertainty in the pertinent group constants due to uncertainty in the basic data.

In this presentation we will review the data base for the ENDF/B-V evaluation of the ^{238}U resolved resonance parameters and show how the analysis of the data base leads to the estimated systematic uncertainties in the evaluated parameters. We will then show how these systematic uncertainties propagate to uncertainties in Bondarenko-type group constants as a function of dilution and temperature. The impact on benchmarks is discussed in the presentation of R. Q. Wright.

We will also compare the ENDF/B-V ^{238}U resolved resonance parameters with those of two recent independent evaluations (JENDL-2 and Moxon and Sowerby 1982). The comparison shows that the very different approaches used in the three evaluations lead to very consistent average data.

*Research sponsored by U.S. DOE Division of Reactor Research and Technology.
†UCC-ND Computer Sciences Division.

1.48

IMPACT OF UNCERTAINTIES IN ^{238}U RESONANCE PARAMETERS ON PERFORMANCE PARAMETERS OF THERMAL LATTICES*

R. Q. Wright† G. de Saussure R. B. Perez

[Abstract of *Trans. Am. Nucl. Soc.* 45, 701 (1983)]

The calculated and measured values of integral parameters may be considered discrepant if their differences are significantly greater than the combined uncertainties assigned to the integral experiment, the calculational methods, and the evaluated differential data. This paper presents an assessment of the uncertainties in the calculated values of performance parameters of selected uranium metal and uranium oxide lattices due to uncertainties in ENDF/B-V data, particularly the ^{238}U resolved resonance parameters.

ENDF/B-V provides ^{238}U resolved resonance parameter covariance files to describe the estimated uncertainties in the evaluated data, but these covariance data have not yet been applied in thermal reactor data testing. One difficulty in using resonance parameter uncertainties has been the lack of an adequate methodology to derive resonance parameter sensitivities, particularly where self-shielding effects are important.

Two kinds of uncertainties are assigned to the resonance parameters: statistical uncertainties, which are uncorrelated from one level to the next, and systematic uncertainties correlated over resolved resonances of the energy range 1 to 4000 eV. The contribution of the systematic uncertain-

ties dominates and, therefore, was the only contribution considered in the present work.

The uncertainties in k_{eff}, ρ^{28}, δ^{25}, δ^{28}, and CR of thermal lattices due to systematic uncertainties in the ^{238}U resolved resonance parameters were obtained by recomputing the group constants for several values of the ^{238}U resolved resonance parameters corresponding to the different types of systematic uncertainties.

The uranium metal lattice TRX-1 and the uranium oxide lattice BAPL-1 were selected for this study because they have been widely used in thermal data testing. These lattices have ^{238}U dilution, σ_0, of 21.3 and 45.0 b, respectively.

Calculational methods uncertainties are difficult to assess. Ignoring the methods uncertainties, it can be concluded that (a) the measured and calculated values of the TRX-1 and BAPL-1 parameters are reasonably consistent when uncertainties in the differential data and integral measurements are considered, and (b) the uncertainties due to the ^{238}U resolved resonance parameters are the most significant contribution to the uncertainties in the calculated values of ρ^{28} and CR for thermal lattices.

*Research sponsored by Nuclear Power Division of the Electric Power Research Institute (EPRI).

†UCC-ND Computer Sciences Division.

1.49

IMPACT OF UNCERTAINTIES IN ^{238}U RESONANCE CAPTURE CROSS SECTIONS ON BENCHMARK PERFORMANCE PARAMETERS*

R. Q. Wright† G. de Saussure
R. B. Perez

[Abstract of paper presented at the Second Jackson Hole Colloquium on Fast Reactor Physics: The Doppler Effect in LMFBRs, Teton Village, Wyoming, June 27-29, 1983; Proc. Session IV, No. 5 (1983)]

Calculated uncertainties in CSEWG fast reactor benchmarks using uranium and plutonium evaluations from ENDF/B-V are a major concern to the fast reactor community. As an example, the performance parameters for the plutonium oxide fueled ZPR-6/7 and the uranium oxide fueled ZPR-6/6A benchmarks are reviewed.

In a companion paper, de Saussure et al. describe uncertainties in the ^{238}U Bondarenko-type group constants, as a function of dilution and temperature, prepared from the ENDF/B-V ^{238}U evaluation.[1] Sensitivity calculations were used to investigate the effect of changes in the group constants on the benchmark performance parameters. In this presentation, the impact of the uncertainties in the ^{238}U capture cross sections on ZPR-6/7 and ZPR-6/6A eigenvalues and central reaction rate ratios are discussed. Of special interest are performance parameter uncertainties resulting from uncertainties in resolved resonance Γ_n and Γ_γ and in the unresolved resonance range. The impact of uncertainties in ^{238}U capture will be compared to the impact due to uncertainties in ^{239}Pu fission for the ZPR-6/7 eigenvalue and ^{238}U capture/^{239}Pu fission reaction ratio at the center of the one-dimensional spherical model of the assembly.

Revision 2 to the ENDF/B-V ^{239}Pu has been proposed.[2] The impact of the proposed evaluation on performance parameters of Pu-fueled benchmark assemblies are discussed.

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

†UCC-ND Computer Sciences Division.

1. G. de Saussure, R. Q. Wright, and R. B. Perez, paper 1.46, this report.
2. E. D. Arthur, *Los Alamos Evaluation of ^{239}Pu for Revision 2 of ENDF/B-V*, LANL, January 24, 1983.

1.50

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

R. Q. Wright† W. E. Ford, III†
J. L. Lucius† C. C. Webster†
J. H. Marable

[Abstract of paper presented at ANS Topical Meeting on *Advances in Reactor Physics and Core Thermal Hydraulics*, Kiamesha Lake, NY, September 22-24, 1982; Proc. NUREG/CP-0034, Vol. 2, p. 1135 (1982)]

Data from the ENDF/B-V processed multigroup cross-section libraries VITAMIN-E and

FORSS-V were used in the calculation of twelve CSEWG fast reactor benchmarks. Results obtained include the eigenvalues, central reaction rate ratios, and central reactivity worths. Conclusions and recommendations based on the trends in these results are given. Finally, the ORNL results are compared with those obtained by other data testers.

*Research sponsored by U.S. Department of Energy.
†UCC-ND Computer Sciences Division.

1.51

BENCHMARK DATA TESTING OF ENDF/B-V*

Edited by

C. R. Weisbin R. D. McKnight†
J. Hardy, Jr.‡ R. W. Roussin
R. E. Schenter§ B. A. Magurno¶

[Abstract of BNL-NCS-31531 (ENDF-311), August 1982]

The testing of Cross Section Evaluation Working Group (CSEWG) data is primarily intended to assess the adequacy of ENDF/B nuclear data for use in nuclear design and applications. The purpose of this report is to describe the results of the benchmark testing of ENDF/B-V and to identify areas of basic nuclear data and computational methods and integral experiments that may require additional research. The benchmark testing is classified according to five major applications: (1) fast reactor design, (2) thermal reactor design, (3) radiation shielding design, (4) fission product and actinide analyses, and (5) dosimeter response analyses. In each case the analyses of pertinent integral experiments are presented and the current limitations in the data-testing process are discussed. Finally, areas of agreement and disagreement are pointed out, and recommendations for future investigations are given.

*Research sponsored by U.S. Department of Energy.
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1.52

ORELA CONTRIBUTION TO THORIUM CYCLE NUCLEAR DATA*

D. K. Olsen

(Abstract of paper presented at the U.S.—Japan Joint Seminar on the Thorium Fuel Cycle, Nara, Japan, October 18-22, 1982)

The measurements of direct importance to the ^{232}Th/^{233}U fuel cycle using neutrons from the Oak Ridge Linear Accelerator facility are gathered together and discussed. These measurements were done in response to specific data discrepancies, as part of generic programs, and for basic fission physics studies. In particular, completed transmission and capture work on ^{232}Th has yielded the most accurate parameters for the first four s-wave resonances; the largest average capture width, 25.2 meV; and the largest s-wave strength function of recent measurements. These results allow improved agreement between differential and integral capture rates. Moreover, the ORNL ^{252}Cf $\bar{\nu}$ measurement of unprecedented accuracy and ^{233}U $\bar{\nu}$ ratio measurement give a ^{233}U prompt $\bar{\nu}$ (thermal) of 2.490 ± 0.009 neutrons/fission. This result allows a much more satisfactory understanding of the ^{233}U 2200-m/s constants. In addition, a ^{232}Th γ-ray production measurement provides needed cross sections for shielding, and important data for both application and fission physics were obtained from ^{232}Th fission and both ^{231}Pa and ^{234}U fission and total cross section measurements. Data requirements and discrepancies suggested from this work are discussed.

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

1.53

THE NUCLEAR DATA OF THE MAJOR ACTINIDE FUEL MATERIALS*

W. P. Poenitz† G. de Saussure

(Abstract of chapter in *Cross-Section Data for Nuclear Reactor Analyses*. Progress in Nuclear Energy series, Pergamon Press, Oxford, in press)

Nuclear data for the major actinides (^{232}Th, ^{233}U, ^{235}U, ^{238}U, ^{239}Pu, ^{240}Pu, ^{241}Pu, and ^{242}Pu)

are reviewed with respect to their application in the optimization of fuel cycles and core designs for thermal and fast breeder reactors. The most important measurements are reviewed and the consistency and completeness of contemporary evaluations (ENDF-V, JENDL-2, KEDAK, SOKRATOR, etc.) are compared. The impact of uncertainties or discrepancies in the data on neutronic calculations is assessed.

*Research sponsored by U.S. Department of Energy.
†Argonne National Laboratory, Argonne, IL.

1.54

MICROSCOPIC BETA AND GAMMA DATA FOR DECAY HEAT NEEDS*

J. K. Dickens

(Abstract of paper presented at the OECD/NEA Nuclear Data Committee Specialists Meeting on Yields and Decay Data, Fission Product Nuclides, Brookhaven National Laboratory, Upton, New York, October 23-27, 1983)

Microscopic beta and gamma data for decay heat needs are defined as absolute intensity spectral distributions of beta and gamma rays following radioactive decay of radionuclides created by, or following, the fission process. Four well-known evaluated data files, namely the U.S. ENDF/B-V, the U.K. UKFPDD-2, the French BDN (for fission products), and the Japanese JNDC Nuclear Data Library, are reviewed. Comments regarding the analyses of experimental data (particularly gamma-ray data) are given; the need for complete beta-ray spectral measurements is emphasized. Suggestions on goals for near-term future experimental measurements are presented.

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

1.55

REPORT TO THE DOE NUCLEAR DATA COMMITTEE*

F. G. Perey

(Abstract of BNL-NCS-32614, p. 134, May 1983)

This report was prepared for the DOE Nuclear Data Committee and covers work performed at ORNL since January 1982 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 reproduced the complete abstract of the original publication with only minor editing. In a few cases progress reports were written specifically for this publication. The 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.

Research Techniques and Facilities

1.56

OFFICE OF BASIC ENERGY SCIENCES PROGRAM TO MEET HIGH PRIORITY NUCLEAR DATA NEEDS OF THE OFFICE OF FUSION ENERGY (1983 REVIEW)*

R. C. Haight† D. C. Larson‡ and Other Panel Members

(Abstract of Lawrence Livermore Laboratory Report UCID-19930, November, 1983)

This review was prepared during a coordination meeting held at Oak Ridge National Laboratory on September 28-29, 1983. Participants included research scientists working for this program, a representative from the OFE's Coordination of Magnetic Fusion Energy (MFE) Nuclear Data Needs Activities, and invited specialists.

*Research sponsored by U.S. DOE Office of Basic Energy Sciences.
†Panel chairman.
‡Panel secretary.

1.57

USER'S GUIDE FOR BAYES: A GENERAL-PURPOSE COMPUTER CODE FOR FITTING A FUNCTIONAL FORM TO EXPERIMENTAL DATA*

N. M. Larson†

[Abstract of ORNL/TM-8185 (ENDF-323), August 1982]

This report is intended as a user's manual for a general-purpose computer program BAYES to solve "Bayes' equations" for updating parameter values, uncertainties, and correlations. Bayes' equations are derived from Bayes' theorem, using linearity and normality assumptions. The method of solution is described, and details are given, including problem description and solution method, FORTRAN coding, and sample input and output. A companion code LEAST, which solves the usual least-squares equations rather than Bayes' equations but which encourages non-diagonal data weighting, is also described.

*Research sponsored by U.S. Department of Energy.
†UCC-ND Computer Sciences Division.

1.58

APPLICATION OF GROUP THEORY TO DATA REDUCTION*

F. G. Perey

(Abstract of ORNL-5908, September 1982)

The analysis within the framework of a theory of what was observed in experiments is essential to the testing of theories and is fundamental to physics. It is shown in this report how group theory can be used to provide a general method of data reduction whereby only the laws of a particular theory are used in the analysis of observations. This application of group theory involves introducing a group of transformations of the physical system upon which the observations were made. This group of transformations leaves invariant the entities of the theory corresponding to the observations made but transforms the entities that were not observed from what they are presumed to be in this theory into what they are not, called possibilities for what they are. This group of transformations is called the possibilities-generating group for the entities for which the observations are being reduced. Since possibilities for entities of theories so obtained are functionals of a known group of transformations, the subsequent use of these possibilities in the theories must be made consistent with the theory of representations of groups. There is a well-known invariant associated with functionals of group elements which is invariant with respect to the parametrization of this group. It is the normalized volume measure in the group manifold of the possibilities-generating group. In this report this invariant measure is called the physical probability of the possibilities since it is a probability measure which has a Borel algebra. The above proposed method of data reduction allows us to deal unambiguously with the uncertainties in entities of physical theories obtained from all of the observations made in experiments and the distinction between systematic and statistical uncertainties disappears. Concrete realizations of possibilities-generating groups are given and explanations in group theoretical terms are offered for several important intuitive notions related to probabilities.

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

1.59

COUNTING ANTICOINCIDENCES TO REDUCE STATISTICAL UNCERTAINTY IN THE CALIBRATION OF A MULTIPLICITY DETECTOR*

R. W. Peelle R. R. Spencer

[Abstract of *Nucl. Instrum. Methods* **211**, 167 (1983)]

The observation of anticoincidences reduced statistical uncertainties as well as pileup corrections in the efficiency calibration of a multiplicity detector. The efficiency of the 4π gadolinium-loaded organic scintillation detector was measured for detecting capture of neutrons scattered by the hydrogen in a second detector. In such situations the effect of background upon the

counting fluctuations is reduced below the value that would occur if the average counting rate during coincidence gates were utilized. The applicable formulae were obtained by elementary methods and were verified with experimental data and with computer simulations for a broad range of detector efficiencies and background intensities. The general requirements for use of the technique are indicated.

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

1.60

AN ANNOTATED BIBLIOGRAPHY COVERING GENERATION AND USE OF EVALUATED CROSS-SECTION UNCERTAINTY FILES*

R. W. Peelle T. W. Burrows[†]

[Abstract of BNL-NCS-51684 (ENDF-331), March 1983]

Literature references related to evaluated cross-section uncertainty (variance-covariance) files are listed with comments intended primarily to guide the reader toward materials of immediate interest. Papers are included that cover covariance information from individual experiments and that relate to production and use of multigroup covariance matrices. Titles are divided among several major categories. The comments by the compilers may not reflect the views of the authors of the papers cited.

*Research sponsored by U.S. DOE Office of Reactor Research and Technology.
[†]Brookhaven National Laboratory, Upton, NY.

1.61

NEUTRON FILTERS FOR PRODUCING MONOENERGETIC NEUTRON BEAMS*

J. A. Harvey N. W. Hill[†]
J. R. Harvey[‡]

[Abstract of paper presented at the International Conference on Nuclear Data for Science and Technology, Antwerp, Belgium, September 6-10, 1982; Proc. p. 856, K. H. Bockoff, Ed. (1983)]

Neutron transmission measurements have been made on high-purity, highly enriched samples of ^{58}Ni (99.9%), ^{60}Ni (99.7%), ^{64}Zn (97.9%), and ^{184}W (94.5%) to measure their neutron "windows" and to assess their potential usefulness for producing monoenergetic beams of intermediate energies from a reactor. Transmission measurements on the Los Alamos Sc filter (44.26 cm Sc and 1.0 cm Ti) have been made to determine the characteristics of the transmitted neutron beam and to measure the total cross section of Sc at the 2.0 keV minimum. When corrected for the Ti and impurities, a value of 0.35 ± 0.03 b was obtained for this minimum.

*Research sponsored by U.S. DOE Division of Nuclear Sciences.
[†]Instrumentation and Controls Division.
[‡]CEGB Berkeley Nuclear Laboratories, Berkeley, England.

1.62

DEVELOPMENT OF AN ATOM BUNCHER*

G. S. Hurst[†] **M. G. Payne**[†]
R. C. Phillips[†] **J. W. T. Dabbs**
B. E. Lehmann[‡]

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

An "atom buncher" for controlling the concentration of gaseous samples has been conceptualized, evaluated theoretically, fabricated, and tested with excellent results. In effect, the atom buncher greatly increases the probability that a free atom will be in a small detector volume at a desired time. This was accomplished by using cryogenic techniques to condensate atoms on a small spot and a pulsed laser to momentarily heat the spot to release the atoms at the desired time. Our work on noble gas atom counting by using Resonance Ionization Spectroscopy (RIS) is discussed as one example of the applications of the atom buncher.

*Research sponsored by U.S. DOE Office of Health and Environmental Research and the Swiss Nationale Genossenschaft fur die Lagerung radioaktiver Abfalle (NAGRA).
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[‡]University of Bern, Bern, Switzerland.

1.63

AFTERPULSES AT SEVERAL μsec FOR AN RCA-8854 MULTIPLIER*

C. H. Johnson[†] N. W. Hill[‡]
J. A. Harvey D. J. Horen[†]

[Abstract of *Bull. Am. Phys. Soc.* **28**(7), 992 (1983)]

At the ORELA time-of-flight facility we have extended the energy for neutron detection down to 5 keV for liquid or plastic scintillators by accepting pulses corresponding to a few photoelectrons. At such low levels there is a background of afterpulses which occur in the photomultiplier, a 5-inch RCA-8854 PMT, several microseconds after the earlier detection of a neutron or γ-ray. To study these we placed a hydrogenous filter in the flight path to remove essentially all neutrons and leave only the narrow initial γ-ray burst and its afterpulses. The time spectrum of the afterpulses (after the 1.1 μsec clock dead time) shows sharp peaks at 1.15, 2.15, 2.4, 2.9, 5.65, and 6.3 μsec and much broader peaks at 4.5, 7.5, and 14 μsec. For the γ-ray burst, which includes γ-ray energies mainly below 1.5 MeV, there is a 14% probability that a detector event will have an afterpulse later than 1.1 μsec. The probability increases with initial pulse height. Presently we are investigating the use of two photomultipliers in coincidence for minimizing this background.

*Research sponsored by U.S. DOE Office of Basic Energy Sciences.
†Physics Division.
‡Instrumentation and Controls Division.

1.64

A PARALLEL PLATE ^{10}B NEUTRON DETECTOR*

J. H. Todd[†] L. W. Weston
G. J. Dixon[‡]

(Abstract of *Nucl. Instrum. and Methods*, in press)

Parallel plate ionization chambers with vacuum deposited ^{10}B electrodes have been constructed and tested for ^{10}B plating thicknesses from 24 μg/cm^2 to 61 μg/cm^2. Pulse-height resolutions of the alpha particle and of the ^{7}Li fragment from the ^{10}B(n,α) reaction were measured using low-energy neutrons. The pulse-height resolutions of the chambers were found to be better than a theoretical analysis had indicated.

*Research sponsored by U.S. DOE Division of Reactor Research and Technology.
†Instrumentation and Controls Division.
‡1981 Summer Participant, Oak Ridge Associated Universities, Oak Ridge, TN.

1.65

A DIGITAL PULSE-PAIR DETECTING CIRCUIT*

Z. W. Bell[†] J. W. McConnell[‡]
E. D. Carroll[§]

[Abstract of *Nucl. Instrum. Methods* **211**, 551 (1983)]

A completely digital pulse-pair detecting circuit is described. The inspection interval ranges from approximately 100 ns to 1.3 ms in steps of 20 ns. Tests of the circuit indicate reliable operation at input rates up to 25 MHz.

*Research sponsored by U.S. DOE Office of Basic Energy Sciences.
†UCC-ND Computer Sciences Division.
‡Physics Division.
§Instrumentation and Controls Division.

1.66

NEUTRON FLUX MEASUREMENTS AT THE 22-METER STATION OF THE OAK RIDGE LINEAR ACCELERATOR FLIGHT PATH NO. 8*

Z. W. Bell[†] J. K. Dickens
J. H. Todd[‡] D. C. Larson

(Abstract of ORNL/TM-8514, May 1983)

The measurements of the ORELA flux at neutron energies from 0.16 to 31 MeV are described. The techniques of fitting calculated responses to measured responses and the method of calculating the covariance matrix are detailed. The measured neutron flux is presented.

*Research sponsored by U.S. DOE Office of Basic Energy Sciences.
†UCC-ND Computer Sciences Division.
‡Instrumentation and Controls Division.

1.67

CALCULATION OF THE ORELA NEUTRON MODERATOR SPECTRUM AND RESOLUTION FUNCTION*

C. Coceva† R. Simonini†
D. K. Olsen

[Abstract of *Nucl. Instrum. Methods* 211, 459 (1983)]

The yield, spectrum, and delay-time distributions (resolution function) of escape neutrons from the ORELA water-moderated and water-cooled tantalum target assembly are calculated using time-dependent three-dimensional Monte Carlo techniques. The resulting spectrum and delay-time distributions of the moderated neutrons are compared with measured values.

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

†Dell'Energia Nucleare e Delle Energie Alternative, Bologna, Italy.

1.68

A PROPOSAL FOR REPLACING THE ORELA PDP-10 COMPUTING SYSTEM*

D. C. Larson J. G. Craven†

(Abstract of a proposal submitted to U.S. DOE Basic Energy Sciences Program, May 24, 1983)

This proposal documents the history of the present DEC PDP-10 located at the Oak Ridge Electron Linear Accelerator (ORELA) and outlines problems with the current system, now 14 years old. Three possible solutions to the problems are presented and examined in detail. Finally, a possible system to meet the ORELA computing needs for the next 15 years is presented, along with cost estimates and factors of improvement over the existing system.

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

†UCC-ND Computer Sciences Division.

Section 2

FISSION REACTOR RESEARCH

Reactor Shielding — Integral Experiments and Analyses

Reactor Analyses

Reactor Safety, Reliability, and Human Factors Studies

2.0. INTRODUCTION

As has been noted in the preface to this report, the research trends within the division are away from neutronics studies and toward other types of investigations. Nowhere is this more apparent than in fission reactor research. Until recent years the term *fission reactor research* as used by us referred exclusively to experiments and calculations that determined the neutronics characteristics of reactor cores and shields, together with the development of methods to perform these studies. Such investigations are still being performed, but, in addition, our fission reactor research also now includes reactor safety and reliability studies that focus on non-neutronics aspects of reactor systems. These include a wide range of human factors studies, particularly studies of man-machine interactions that support the "systems approach to training (SAT)" currently being adopted by the nuclear industry.

Reactor Shielding — Integral Experiments and Analyses

Reactor shielding research during this reporting period has largely been associated with High Temperature Graphite Reactors (HTGRs), and to a lesser extent with Liquid Metal Fast Breeder Reactors (LMFBRs). (Note: The LMFBR work is categorized by the Department of Energy as being "applied" and thus cannot be covered in this report.) In addition, a renewed interest in the use of reactors for space applications has prompted us to reassess the status of candidate shielding materials for space reactors with the expectation that we will participate in a structured program in the future.

The bulk of the HTGR shielding work has consisted of studies of the shielding effectiveness of the bottom reflector and core support block of the current HTGR design. Following an analysis of this region of the HTGR, an experiment mocking up the below-core components was designed and the first of two planned experiments was performed at the ORNL Tower Shielding Facility. The results of this experiment and its analysis have now been reported.

Although not covered in this report, the division has also participated in studies of clusters of small modular HTGR power systems. Such systems are envisioned to be semi-portable, easy to construct, easy to replace, and relatively economical. These criteria will place very severe constraints on the radiation shields in that they will have to be highly effective with minimum bulk.

In addition to reactor shielding per se, we have surveyed the shielding data and methods available for designing shields for fuel reprocessing centers. Such centers will be remotely operated and maintained and thus the shielding concerns are primarily the protection of instrumentation and equipment that will have to operate in high-intensity radiation fields.

All of the above analyses have primarily been carried out with highly sophisticated radiation transport methods developed within the division in earlier years. The only major methods development effort currently under way is the TORT system, which is mentioned in the introduction to Section 5. Of course, the maintenance of our computer systems and the updating of the data used in them remain important functions of the division, and in at least one instance we have helped upgrade a major transport system developed elsewhere (France's TRIPOLI-2 Monte Carlo system). The fact that we, and others, have at hand the calculational tools needed to address massive neutronics problems speaks well of our past performance. Moreover, the methods we have developed can be successfully adapted in non-neutronics areas, as is apparent from other sections of this report.

Reactor Analyses

During this reporting period, reactor analysis research has proceeded at reduced funding levels but perhaps in more diverse areas than in the past. Reactor analysis methods developed in the past have been applied recently to several dosimetry and neutronics problems, and investigations in a new area have been initiated. In addition, we have continued to improve and document our analysis methods.

The documentation effort has included the contribution of two chapters to the *CRC Handbook of Nuclear Reactor Calculations*, one on perturbation theory and another on diffusion theory. Many major contributions to both the theoretical development and the practical implementation of these methods to neutronics problems have originated in this division. The chapter on perturbation theory gives a detailed review of the theory and highlights (both theoretically and by means of numerical examples) the usefulness and limitations of the theory as a tool for the analysis of neutronics problems. The chapter on diffusion theory describes problems for which the diffusion theory approximation to neutron transport is appropriate, references major codes based on diffusion theory, and discusses a number of special analysis techniques that can be employed.

In other publications, discussions are presented on the solution of complicated problems by coupling diverse methods under automated procedures, on solving the critical core neutronics problem, and on solving the uncommon nuclear reactor core neutronics problems.

Important applications of code systems developed within the division have included the application of the large VENTURE/DEPTH-CHARGE code system to a sensitivity analysis of the inherent neutron source strength and to a sensitivity analysis of the Doppler coefficient in heterogeneous Liquid Metal Fast Breeder Reactors.

The LEPRICON code system, whose purpose is to obtain best estimates and reduced uncertainties for the spectrum and total fluence in light water reactor pressure vessels, has undergone further validation and development, and specific conclusions regarding the accuracy of the calculational neutron transport methods and nuclear data used in LEPRICON have been drawn from comparisons with measurements performed in several benchmark fields. Also, a series of calculated and measured average cross-section ratios in the ^{252}Cf benchmark field have been analyzed to obtain consistent data sets. In addition, a new, efficient technique has been developed to calculate ex-core dosimeter activities at end of irradiation for an arbitrary spatial and time reactor power distribution.

The interest in high-temperature gas-cooled reactors as inherently safe energy sources prompted two studies for the pebble-bed reactor, one related to fuel temperature peaking and the other to controllability. Pebble management schemes were studied as a means to reduce fuel temperatures, and control-rod worth calculations were performed using Monte Carlo analysis for adequate representation of the void above the core. In addition, the concept of an "accelerator breeder," which we have studied in the past, was reexamined and several innovative design changes were introduced.

Historically, the study of stability and stochastic processes for power reactor surveillance and diagnostics has been based on the linearization of the governing equations. In BWRs, though, the assumption of linearity is inadequate under certain conditions such as low flow/high power. As a result of newly initiated research regarding the influence of nonlinearities on BWR stability, it has been shown that reactors cannot become linearly unstable, but instead reach limit cycles characterized by power oscillations. The amplitude of these oscillations is limited by operating conditions. The onset of these oscillations can be diagnosed by a decrease in stochasticity in power traces, and by the appearance of harmonics in the power spectral densities.

Reactor Safety, Reliability, and Human Factors Studies

The division's reactor safety and reliability research program continues to consist of two major and relatively independent efforts. One effort is oriented around hardware; that is, the safety and/or reliability of various components of the reactor system itself are studied. The other focuses on the interdependency of human performance and hardware performance; that is, the importance of human factors in reactor operation and maintenance is studied. Both programs are performed under the auspices of the Nuclear Regulatory Commission and are supplemented by two data bases maintained by the Department of Energy: CREDO, which consists of engineering, event, and operating data on advanced reactors; and SACRD, which consists of evaluated data for use in fast reactor computer codes.

Human Factors Studies. During this reporting period the safety-related operator action (SROA) program that was introduced in our last progress report was concluded. In this program operator responses to emergency and/or abnormal events on training simulators for both pressurized water reactors and boiling water reactors were calibrated with corresponding actions "in the field." A final report presents a modeling approach to evaluate the acceptability of assigning safety-related actions to nuclear power plant operators.

The SROA program has led into a broader investigation of operator qualifications, training, and performance that is now under way and includes three separate projects. One is a direct extension of the earlier program and again includes the calibration of operator responses on simulators to operator responses in real situations. This project differs from the earlier program in that the simulator experiments are more controlled and the field data collection is more comprehensive. (The field data collection is concentrating on fewer nuclear plants and fewer "events," but more in-depth data are being obtained for each event.)

The second project that has been an outgrowth of the earlier SROA program is one in which methods are being developed to evaluate the entry level qualifications for control room personnel and also the effectiveness of specific training programs for control room personnel. These methods, which follow the systems approach to training (SAT), will eventually be field tested to ensure that they are practical and effective.

The third current project resulting from the SROA program consists of the development of measurement instruments (e.g., written tests, oral examinations, and simulator-based tests) that can be used to quantify the performance of control room operators. These measurement instruments could be used directly in operator licensing and are essential for validation of qualifications and training requirements.

Two other current human factors studies can be categorized as human reliability assessments. One is a human factors review that is part of the Severe Accident Sequence Analysis (SASA) program. The review is for the case of a degraded core accident resulting from a particular sequence [anticipated transient without scram (ATWS)] at a particular reactor (TVA's Browns Ferry). The "front end" review will predict operator responses up to the point that melting of the core cladding begins, and the "back end" review will attempt to predict the responses of control room personnel during core degradation and the release of radionuclides to the environment. The goal of this review is to identify potential human factors problems that will affect a sequence at a particular plant and to make recommendations for improvements in procedures, display equipment, etc.

The second human reliability assessment program is concentrating on nuclear power plant maintenance personnel. Job analyses have been performed for four maintenance positions and a computer model (MAPPS, MAintenance Personnel Performance Simulation) has been developed to quantitatively predict the reliability of maintenance personnel given such input parameters as the maintainer's ability and the environment in which he will be working (temperature, radiation levels, etc.) The model is ready for field testing and should be transferred to users (via instructional courses) in 1985.

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 major assessment in which the division is directly participating is concerned with the current unresolved safety issue of pressurized thermal shock (PTS) to the reactor pressure vessel following a reactor transient condition that leads to rapid cooldown and repressurization. Work is currently under way to identify the detailed sequences that will lead to PTS, together with their probabilities of occurrence, for three nuclear power plants. The program integrates the frequency of the event, the resulting thermal hydraulic transients, the fracture mechanisms, and sensitivity and uncertainty analyses.

Another risk assessment study completed during this reporting period consisted in the development of a methodology (by subcontractor JBF Associates, Inc.) for performing common cause failure analysis for nuclear power plants. This work included developing techniques to assess the susceptibility of the total risk to common cause initiators and the contribution to the total risk of the initiators.

Other on-going nuclear power plant risk studies are addressing (1) decision making under uncertainty, (2) value-impact analysis, (3) fire risks, (4) accident sequence precursor methodology programs, (5) the impact of truncation on risk quantification, (6) in-plant reliability data, (7) application of risk analysis to NRC inspection programs, (8) accidents at UF_6 facilities, (9) recycle of contaminated materials, and (10) probabilistic analysis of containment leakage.

Finally, several division members are participating in the international Society for Risk Analysis as it relates to NRC risk assessments. For example, the society's newsletter is edited by two division members, one of whom also serves on the editorial board of the society's journal, *Risk Analysis*.

Reactor Shielding — Integral Experiments and Analyses

2.1

THE STATUS OF REACTOR SHIELDING RESEARCH IN THE UNITED STATES*

D. E. Bartine

[Abstract of paper presented at the Sixth International Conference on Radiation Shielding, Tokyo, Japan, May 16-20, 1983; Proc. Vol. I, p. 526 (1983)]

Shielding research in the United States continues to place emphasis on (1) the development and refinement of shielding design calculational methods and nuclear data and (2) the performance of confirmation experiments, both to evaluate specific design concepts and to verify specific calculational techniques and input data. The successful prediction of the radiation levels observed within the now-operating Fast Flux Test Facility (FFTF) has demonstrated the validity of this two-pronged approach, which has since been applied to U.S. fast breeder reactor programs and is now being used to determine radiation levels and possible further shielding needs at operating light water reactors, especially under accident conditions. A similar approach is being applied to the back end of the fission fuel cycle to verify that radiation doses at fuel element storage and transportation facilities and within fuel reprocessing plants are kept at acceptable levels without undue economic penalties. Finally, the same approach is being used to develop a sophisticated fusion reactor shielding technology and to study shields for possible future space power reactor systems.

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

2.2

SPACE REACTOR SHIELDING: AN ASSESSMENT OF THE TECHNOLOGY*

D. E. Bartine W. W. Engle, Jr.

[Abstract of paper presented at a Symposium on Advanced Compact Reactor Systems, National Academy of Sciences, Washington, D.C., November 15-17, 19o2; *Proc. of Advanced Compact Reactor Systems*, pp. 364-382 (1983)]

Analytical and experimental techniques for designing and testing space reactor shields were developed at Oak Ridge National Laboratory more than a decade ago. The analytical techniques, which are available for application to current systems, automatically optimize the size, weight, and shape of the shield by careful placement of selected shield materials around the reactor. The design of the shield varies with angle from the reactor center, depending on the dose-rate constraints specified for different points outside the system. The selection of the materials used in the design is aided by experimental determinations of the radiation-attenuating characteristics of various candidate materials at the ORNL Tower Shielding Facility (TSF), which uses a 1-MW(th) reactor as the source (the Tower Shielding Reactor II). Measurements with the TSR-II also confirm basic data (such as cross sections) needed as input in the analytical methods. Prototypic shield designs can also be tested at the TSF with a 10-kW(th) SNAP reactor used as the source. These same techniques can serve as a basis for developing a space reactor shielding technology to meet the demands of future systems. Such a shielding technology program should include generic studies of various shield materials to determine not only their shielding characteristics, but also their material and structural integrity in severe temperature and radiation environments and under launch and accident conditions. It should also include generic studies of radiation streaming through penetrations in shields such as those required to accommodate coolant flow and reactor control. Finally, it should include experimental tests and detailed analyses of prototypic shields. Only with such a technology will designers be assured that the shield weight, which can dominate the overall weight of space systems, is kept to a minimum while at the same time being assured that the dose rates and total doses delivered to the payload, to the reactor electronic components and to the maintenance and operating personnel do not exceed the maximum allowable limits.

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

2.3

TECHNOLOGY STATUS OF CANDIDATE SHIELDING MATERIALS FOR SPACE POWER REACTORS*

W. W. Engle, Jr.   D. E. Bartine

(Abstract of paper presented at 1st Symposium on Space Nuclear Power Systems, Albuquerque, N.M., January 10-13, 1984)

The criteria for selecting shielding materials for space power reactors are: good radiation attenuation characteristics; minimum weight; and stability under operating temperature and radiation environments and under launch and accident conditions. Shielding materials for space power reactors are usually separated into two distinct classes: neutron shielding materials and gamma-ray shielding materials. This paper will focus on lithium hydride and tungsten as the most often mentioned neutron and gamma-ray shielding materials respectively. It will also describe and compare some of the characteristics of other candidate shielding materials as well as point out areas where further shielding technology development is required.

Lithium hydride is attractive as a neutron shielding material because of its low density (approximately 0.8 g/cc) and its high neutron attenuation properties. No other candidate neutron shielding material is as light as lithium hydride. Its disadvantages are that it is brittle, its thermal conductivity is low, and at elevated temperatures, dissociation may become a problem so that active cooling may be necessary. Other candidate neutron shielding materials are calcium hydride, titanium hydride, borated beryllium and beryllium oxide, and boron carbide.

Most of the shielding analyses in the SNAP space reactor program considered tungsten or tungsten-based alloys as the best gamma-ray shield materials. Tungsten alloys such as hevimet (a copper-nickel-tungsten alloy) or W-3.5Ni-1.5Fe were generally used to avoid the difficulties encountered in forming and machining tungsten. Another reason for choosing tungsten in the SNAP program was the feeling that more was known about secondary gamma-ray production in tungsten than in any other candidate materials. Lead is widely used as a gamma-ray shield in many terrestrial applications because of its low secondary gamma-ray production, but it is usually eliminated from space shielding applications because of its low melting point (600 K). Depleted uranium is often mentioned as a candidate material, but it is not normally used near the reactor core because fissions occurring in the uranium could cause heating and possibly swelling in the uranium shield. One of the last studies performed in the SNAP program suggested that a borated stainless steel shield would perform as well as a tungsten shield of the same weight. Other possible candidate materials are zirconium hydride and titanium hydride, both of which can be considered combination gamma-ray and neutron shield materials.

In addition to comparing the physical properties of the candidate shielding materials, this paper will compare the shielding effectiveness of the materials. Since space power reactor shields are usually designed with alternating layers of gamma-ray and neutron shielding materials, any comparison of shielding worth must utilize properly optimized thicknesses and placements of the chosen combination of materials. Examples of several optimized combinations of shield materials will be described for a typical space power reactor configuration.

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

2.4
DESCRIPTION AND SPECIFICATIONS FOR HTGR LOWER REFLECTOR AND CORE SUPPORT NEUTRON STREAMING EXPERIMENT*

C. O. Slater   D. T. Ingersoll

(Abstract of ORNL/GCR-83/2, February 1983)

This report is a formal response to the requirements specified by the General Atomic Company for the HTGR Lower Reflector and Core Support Neutron Streaming Experiment to be performed at the ORNL Tower Shielding Facility. The preanalyses requested by GAC have been performed and are presented. They include comparisons of neutron spectra and spatial distributions of several detector responses calculated for comparable locations within the HTGR design and the experiment. The test requirements have also been examined, and the methods by which they are expected to be met are described, with exceptions noted. In conjunction with the experiment design, several additional calculations have been performed, the results of which are also reported.

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

2.5
PHASE I MEASUREMENTS FOR THE HTGR BOTTOM REFLECTOR AND CORE SUPPORT BLOCK NEUTRON-STREAMING EXPERIMENT*

F. J. Muckenthaler   L. B. Holland†
J. L. Hull†   J. J. Manning

(Abstract of ORNL/TM-8977, in press)

This report presents the Phase I measurements of the High-Temperature Gas-Cooled Reactor Bottom Reflector and Core Support Neutron Streaming Experiment conducted at the ORNL Tower Shielding Facility during FY-1983. In this phase only the first four of eight segments that comprise the full experimental mockup were utilized. These were (1) the upper boron pin layer (graphite matrix) followed by a boral shroud, (2) the graphite reflector layer, (3) a graphite coolant crossover layer, and (4) a graphite follow-on layer. With the Tower Shielding Reactor II used as the neutron source, neutron measurements were made behind each layer as it was added to the configuration and also behind an iron and graphite spectrum modifier that preceded the upper boron pin layer. The measurements included neutron energy spectra behind the spectrum modifier and the boron pin layer and integral neutron fluxes behind the spectrum modifier and all four layers of the mockup. The measurements were divided into two parts, the two parts differing in in the composition of the boron pin layer. During Part I, a reference pin pattern was used which consisted of a combination of graphite and boronated graphite pins. During Part II, a full pattern of graphite pins was used. The experimental data are presented in both tabular and graphical form.

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

2.6
ANALYSIS OF PHASE I OF THE HTGR BOTTOM REFLECTOR AND CORE SUPPORT BLOCK NEUTRON-STREAMING EXPERIMENT*

C. O. Slater

(Abstract of ORNL-6014, in press)

The results of the analysis of Phase I of the HTGR bottom reflector and core support block neutron-streaming experiment performed at the ORNL Tower Shielding Facility are presented. The full experiment has two major objectives: (1) to study thermal-neutron streaming within an array of boron pins, and (2) to study the effects on neutron flux levels in the HTGR lower region of neutrons streaming through the large coolant holes in the bottom reflectors and core support blocks. In Phase I the emphasis was on the streaming through the boron pin layer. Two pin arrangements were used: a reference pattern con-

sisting of graphite and boronated graphite pins; and a full pattern consisting of all-graphite pins.

In the analysis, neutron spectra and detector count rates behind the various sections of the experimental configuration were calculated with the two-dimensional discrete ordinates radiation transport code DOT-IV using a 44-neutron energy group structure (four thermal neutron groups with E < 3.05 eV), a P_3 cross section expansion with upscatter, and a 150-direction biased quadrature. In many cases, the DOT calculation in the void behind a configuration was performed with a source that was obtained by using results from a MORSE Monte Carlo calculation to correct DOT boundary fluxes at the end of the configuration for streaming. The calculated results were compared with the measured results and were often found to be in agreement with the measured results within the target accuracy of ±20%. The calculations through the boron pin layers showed thermal-neutron streaming factors up to 5.0. The full pattern of boron pins was found to be about 2.2 times more effective in shielding against the thermal-neutron flux directly behind the pin layer than the reference pattern, but it was actually only a factor of 1.3 times more effective due to a dominant thermal-neutron source being established by the thermalization of neutrons in the bottom reflectors below the pin layer.

Maximum neutron streaming factors through the last section of the Phase I configuration were estimated for the reference case at about 17 for E > 0.1 MeV; 21 for 9.5 keV < E < 0.1 MeV; 10 for 0.4 eV < E < 9.5 keV; and 3.0 for E < 0.4 eV. It was estimated that streaming factors for sections E-G of the experimental configuration would have to be on the order of 120, 335, and 530 for the fast-, intermediate-, and thermal-neutron energy ranges, respectively, in order for the streaming factors for the experiment to be comparable to those for the design. The good agreement achieved between calculated and measured detector count rates through the largest configurations gives a preliminary verification of the design calculations.

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

2.7

SURVEY OF SHIELDING DATA AND METHODS FOR FUEL REPROCESSING APPLICATIONS*

D. T. Ingersoll

[Abstract of *Trans. Am. Nucl. Soc.* **44**, 476 (1983)]

Radiation shielding R&D in the past few years has been dominated by reactor applications, especially liquid-metal fast breeder reactor and other advanced reactor concepts. As a result, the emphasis has been on improving data and methods for computing neutron transport and secondary gamma-ray production. However, as fuel reprocessing needs are addressed, a new emphasis on shielding data and methods may evolve. Recently, a survey was conducted to determine the present status of nuclear data, computational methods, and integral data applicable to shielding design for fuel reprocessing plants. It was concluded that the general status of the data and methods is good, and that many of the improvements that have resulted from reactor programs serve equally well for reprocessing applications. However, a few additional areas of concern were identified that are generic to reprocessing applications and some that are specific to particular design concepts.

*Research sponsored by U.S. DOE Office of Spent Fuel Management and Reprocessing Systems.

2.8

ALTERNATE METHODS OF UTILIZING CROSS-SECTION SENSITIVITY COEFFICIENTS IN RADIATION SHIELDING PROBLEMS*

S. I. Bhuiyan[†] R. W. Roussin
J. L. Lucius[‡] J. H. Marable
D. E. Bartine

[Abstract of *Nucl. Sci. Eng.* (in press); also paper presented at the Sixth International Conference on Radiation Shielding, Tokyo, Japan, May 16-20, 1983; Proc. Vol. I, p. 54 (1983)]

Attempts to devise techniques for rapidly calculating radiation transport in relatively simple shields have led to the development of two calculational models that are based on the use of cross-section sensitivity coefficients and are possible improvements over the traditional linear model. The two models, one an exponential model and the other a power model, were tested, along with the linear model, by applying them to 1- and 2-m-thick concrete slab problems in which the water content, reinforcing steel content, and composition of the concrete were varied. Comparing the results obtained with the three models with those obtained from an exact one-dimensional discrete ordinates transport calculation indicated that the exponential model, named the BEST model (for *b*asic *e*xponential *s*hielding *t*rend), is a particularly promising predictive tool for shielding problems dominated by exponential attenuation. When applied to a deep-penetration sodium problem, the BEST model also yielded better results than did calculations based on second-order sensitivity theory.

*Research sponsored by U.S. Department of Energy.
†Bangladesh Atomic Energy Commission, Institute of Nuclear *Science and Technology, Dacca, Bangladesh.*
‡UCC-ND Computer Sciences Division.

2.9
THE MORSE MONTE CARLO RADIATION TRANSPORT CODE SYSTEM*

M. B. Emmett†

(Abstract of ORNL-4972/R1, February 1983)

This report is an addendum to the MORSE report, ORNL-4972, originally published in 1975. This addendum contains descriptions of several modifications to the MORSE Monte Carlo Code, replacement pages containing corrections, Part II of the report which was previously unpublished, and a new Table of Contents.

The modifications include a Klein-Nishina estimator for gamma rays. Use of such an estimator required changing the cross-section routines to process pair production and Compton scattering cross sections directly from ENDF tapes and writing a new version of subroutine RELCOL.

Another modification is the use of free-form input for the SAMBO analysis data. This required changing subroutines SCORIN and adding new subroutine RFRE.

Part II of the MORSE report is a guide on using MORSE. It attempts to help the user solve some of the more common problems incurred in running the code. It is by no means a complete list of troubles; rather, it is an attempt to help the user do his own troubleshooting.

MORSE-CG is now available for the CRAY computer. It has been tested on the Lawrence Livermore Laboratories MFE CRAY and is now available from the Radiation Shielding Information Center (RSIC) at ORNL.

*Research sponsored by Defense Nuclea[r A]gency.
†UCC-ND Computer Sciences Division.

2.10
TRIPOLI-2: NEUTRON GAMMA COUPLING — APPLICATIONS TO SHIELDING BENCHMARKS AND DESIGNS*

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[Abstract of paper presented at the Sixth International Conference on Radiation Shielding, Tokyo, Japan, May 16-20, 1983; Proc. Vol. I, p. 161 (1983)]

Recent additions in the on-going development of the TRIPOLI Monte Carlo code system include conversion to the ENDF/B data format and an automated coupling scheme for neutron and secondary gamma-ray calculations. Two shielding calculations are presented here which feature these two new developments.

*Research sponsored by French Atomic Energy Commission, Saclay, France.
†CEA - DEMT/SERMA/LEPF, France.

2.11
A USER'S MANUAL FOR THE FERDO AND FERD UNFOLDING CODES*

B. W. Rust† **D. T. Ingersoll**
W. R. Burrus‡

(Abstract of ORNL/TM-8720, September 1983)

FERDO and FERD are unfolding codes which can be used to correct observed pulse-height distributions for the nonideal response of a pulse-height spectrometer or to solve poorly conditioned linear equations. It is assumed that the response of the spectrometer is given by $\underline{\underline{A}}\underline{x} = \underline{b}$, where $\underline{\underline{A}}$ is the spectrometer response function matrix, $\underline{x}$ is the unknown spectrum, and $\underline{b}$ is the pulse-height distribution. FERDO does not solve directly for x, but instead solves for $\underline{p} = \underline{\underline{W}}\underline{x}$, where $\underline{\underline{W}}$ is a "window function matrix." Typically, $\underline{\underline{W}}$ is the resolution function of an ideal spectrometer which has a single Gaussian response. The effective resolution of the unfolding solution may be varied by choice of $\underline{\underline{W}}$. Confidence intervals ($p^{lo}$, p^{up}) are found for each element of the solution p. FERDO and FERD are written in IBM 360/370 Fortran (Level H). This manual describes and derives the mathematical procedures used by the codes, tells how to write input data for them, and gives solutions to some sample problems to illustrate the output.

*Research sponsored by U.S. Department of Energy and the National Bureau of Standards.

†U.S. Department of Commerce, National Bureau of Standards, Washington, D.C.

‡Science Applications, Inc., Oak Ridge, TN.

Reactor Analyses

2.12
GENERALIZED PERTURBATION THEORY WITH DERIVATIVE OPERATORS FOR POWER DENSITY INVESTIGATIONS IN NUCLEAR REACTORS*

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D. G. Cacuci

[Abstract of *Nucl. Sci. Eng.* **84**, 206 (1983)]

This work presents an efficient method to analyze variations that nuclear data perturbations induce in one-dimensional power density distributions. This method is called the Taylor-generalized perturbation theory (Taylor-GPT) method since it is based on (a) use of a Taylor series expansion of the response variation and (b) use of generalized perturbation theory (GPT) to evaluate the derivative operators that appear as coefficients in this Taylor series. Equations satisfied by the importance functions for the derivatives of the response variations are derived and solved with existing GPT codes. The characteristics of these functions are highlighted analytically.

Particular attention is focused on the numerical value and location of the maximum power density. This is because perturbations in system parameters affect not only the value at the maximum, but also the location of this maximum. The Taylor-GPT method can efficiently assess such efforts.

The practical usefulness of the Taylor-GPT method is illustrated by considering test cases involving a simplified heterogeneous liquid-metal fast breeder reactor model. The results indicate that this method is as accurate as the GPT method, yet requires fewer calculations when investigating space-dependent power density variations.

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

†Georgia Institute of Technology, Atlanta, GA.

2.13
PERTURBATION THEORY FOR REACTOR ANALYSIS*

M. L. Williams

[Abstract of ORNL/TM-8768, in press; also a chapter in the *CRC Handbook of Nuclear Reactor Calculations*, Y. Ronen, Ed., CRC Press, Boca Raton, FL (in press)]

This report was prepared as a tutorial discourse on applications of perturbation theory to reactor physics and shielding problems. Although the mathematical development of perturbation theory is presented in detail, the emphasis of the text is on a physical description of the method, and on useful applications which arise in reactor analysis. Chapters II and III discuss generic aspects of perturbation theory, while most of the remaining chapters treat the development of perturbation methodologies for specific types of

problems encountered in neutronics calculations. The intent is to present perturbation theory from a unified viewpoint, while at the same time to discuss the details and special considerations which arise in the different applications to eigenvalue problems, fixed source calculations and time-dependent problems. Several numerical examples are used to illustrate the power, as well as the limitations, of perturbation theory as a tool for nuclear reactor analysis. A short chapter on higher order perturbation is also given.

*Research sponsored by the Engineering Technology Division of the U.S. Nuclear Regulatory Commission.

2.14

SENSITIVITY ANALYSIS OF THE INHERENT NEUTRON SOURCE STRENGTH*

J. R. White[†] J. H. Marable
R. E. Schenter[‡]

[Abstract of *Trans. Am. Nucl. Soc.* 43, 723 (1982)]

Developing monitoring techniques for a shutdown reactor or designing shipping casks for irradiated fuel requires a knowledge of the inherent neutron source strengths of the core and fuel bundles, respectively. Accurate prediction of the source strengths due to spontaneous fission and α,n reactions versus irradiation time requires that the buildup of trace quantities of several heavy actinides be precisely determined. However, the relatively large uncertainties associated with the methods and nuclear data utilized in typical actinide buildup and depletion calculations constrain the accuracy to which the source can be predicted. The ultimate goal of this study is to quantitatively estimate the accuracy limitation and to identify the major sources of the overall uncertainty in the calculated neutron source strength.

As a first step, a detailed calculation and depletion sensitivity analysis of the end-of-cycle-four (EOC-4) neutron source strength in a representative 300-MW(e) fast reactor design was performed to identify the nuclear data significant to the calculation of the source strength. The core was modeled as a two-dimensional hexagonal model, and the neutronics and fuel depletion calculations were performed with the VENTURE/BURNER system using nine-group ENDF/B-V cross sections derived from the VITAMIN-E library. The adjoint calculation was performed with the DEPTH-CHARGE/VENTURE system, with the adjoint source for the problem derived from the EOC-4 total neutron source strength response. The calculated adjoint functions were then utilized to generate nuclide and data sensitivity coefficients for the EOC-4 neutron source strength. Analysis of the results identified the ^{242}Cm inventory and corresponding source density as the major contribution (~90%) to the total source strength at shutdown. In addition, the sensitivities required for future uncertainty analysis of the source strength were obtained, which will help to identify the major nuclear data and initial nuclide concentrations that significantly affect the source buildup versus irradiation.

*Research sponsored by U.S. DOE Division of Reactor Research and Technology.
†UCC-ND Computer Sciences Division.
‡Hanford Engineering Development Laboratory, Richland, WA.

2.15

SENSITIVITY ANALYSIS OF THE DOPPLER COEFFICIENT IN HETEROGENEOUS LMFBRs*

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J. H. Marable J. L. Lucius[†]

[Abstract of *Trans. Am. Nucl. Soc.* 45, 769 (1983)]

A comprehensive data sensitivity analysis of the fuel Doppler reactivity in LMFBRs has been completed. The analysis was performed with the VENTURE/DEPTH-CHARGE computational system using nuclear data based on the 174-group ORACLE adjusted cross section library.

A strong energy dependence of the sensitivities is observed, including changes in the sign of many sensitivity coefficients near the ^{238}U capture resonance region. In addition, the data sensitivities required for future uncertainty analysis of the Doppler coefficient have been quantitatively determined and have helped to identify the major nuclear data that significantly affect the Doppler reactivity.

*Research sponsored by U.S. DOE Department of Breeder Reactor Systems.
†University of Lowell, Lowell, MA.
‡UCC-ND Computer Sciences Division.

2.16

VALIDATION OF NEUTRON TRANSPORT CALCULATIONS IN BENCHMARK FACILITIES FOR IMPROVED DAMAGE FLUENCE PREDICTIONS*

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F. B. K. Kam† F. W. Stallmann†

(Abstract of paper presented at the 11th Water Reactor Safety Information Meeting, National Bureau of Standards, Washington, D.C., October 24-28, 1983)

A review is presented of the needs and requirements for benchmark experiments, of the status of current dosimetry benchmark experiments and the results and conclusions obtained so far, and of the areas where further benchmarking is still needed. Specific conclusions regarding the accuracy of existing calculational transport methods and nuclear data used in the LEPRICON code system are drawn from comparisons with measurements performed in the ^{252}Cf, ISNF, PCA, PSF, and SDMF benchmark fields.

*Research sponsored by the Engineering Technology Division of the U.S. Nuclear Regulatory Commission.
†Operations Division.

2.17

UNCERTAINTY ANALYSIS OF BENCHMARK DOSIMETRY MEASUREMENTS*

J. J. Wagschal† R. E. Maerker
B. L. Broadhead‡

[Abstract of paper presented at the International Conference on Nuclear Data for Science and Technology, Antwerp, Belgium, September 6-10, 1982; Proc. p. 436, K. H. Bockoff, Ed. (1983)]

The joint consistency of a series of measured quantities and their calculated coun‹›parts is defined. Ten measurements of average cross sections or average cross-section ratios in benchmark ^{252}Cf spontaneous fission neutron fields are analyzed. The consistency of calculations based on a pure Maxwellian fission spectrum and ENDF/B-V dosimetry cross sections and the corresponding measured values is being tested. Dosimetry cross-section covariance matrices based on ENDF/B-V and covariances of the measured values are used in the analysis. Clear evidence of an inconsistency is presented and discussed. Removing one measurement results in a consistent data set.

*Research sponsored by the Electric Power Research Institute.
†Racah Institute of Physics, Hebrew University of Jerusalem, Israel.
‡UCC-ND Computer Sciences Division.

2.18

ACCOUNTING FOR TIME-DEPENDENT SOURCE VARIATIONS IN SURVEILLANCE DOSIMETRY ANALYSIS*

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B. L. Broadhead†

[Abstract of *Trans. Am. Nucl. Soc.* 45, 591 (1983)]

A technique has been developed that allows one to calculate ex-core dosimeter activities at the end of irradiation for an arbitrary spatial and time reactor power distribution. If it is assumed that a calculation of the saturated activities exists for some nominal spatial power distribution, then only one additional transport calculation is required — an adjoint based on any one of the dosimeter responses with the source placed anywhere beyond the downcomer. Use of this one flexible adjoint calculation in conjunction with the aforementioned forward calculation then allows an accurate calculation of all the dosimeter activities for any near-midplane location beyond the downcomer for any arbitrary source distribution. This simplification is a direct result of the establishment of a spectral equilibrium beyond the water which is determined by the higher source energies ($\gtrsim$ 8 MeV) in the core periphery but not on their distribution.

If the core power is represented by

$$P(\bar{r},t) = F_i(\Delta t_i)P_j(\bar{r},\Delta t_j),$$

where t lies in both Δt_i, the fine time interval over which the "instantaneous" nominal core power fraction is F_i, and Δt_j, the coarse time interval over which the quasi-static spatial power distribution P_j occurs, $\bar{r}$ lies in assembly m, and P_j can be calculated from the in-core self-powered neutron detectors and a pin-by-pin PDQ7 or similar calculation, then the expression relating the activity of dosimeter d' at the end of irradiation to F_i and P_j is the following, in dps per atom:

$$A_{d'}(T) = \exp(-\lambda_{d'}T)\left(\frac{R_{d'}}{R_d}\right)$$

$$\times \sum_{j=1}^{J} C_j\left[\int_{core} P_j(\bar{r}) \sum_{g=1}^{G} \chi_g \Omega^*_{g,d}(\bar{r})d\bar{r}\right]$$

$$\times \sum_{i=1}^{I(j)} F_i\left[\exp(\lambda_{d'}t_{i+1}) - \exp(\lambda_{d'}t_i)\right],$$

where

$\lambda_{d'}$ – decay constant of dosimeter d', in days^{-1},

T – irradiation period, in days,

$(R_{d'}/R_d)$ – ratio of reaction rates of dosimeter d' to dosimeter d calculated using some reference source in a forward DOT4 run (dosimeter d is the response cross section used as the source in the adjoint DOT4 run),

J – number of distinct relative spatial distributions used over T,

C_j – conversion factor relating neutron source and assembly power, in units of neutrons per second per megawatt thermal,

χ_g – relative number of neutrons in group g produced per fission event, normalized to unity,

$\phi^*_{g,d}$ – adjoint flux in group g (i.e., $g = 1$ corresponds to lowest energy group) calculated in the adjoint DOT4 run using the response of dosimeter d as a source, in disintegrations per nucleus per neutron per cubic centimeter,

P_j – power density distribution in the peripheral core based on SPND data and supplemented by PDQ7 calculations, in megawatts thermal per cubic centimeter,

$I(j)$ – number of subintervals describing "instantaneous" total power history for each interval j,

F_i – fraction of SPND-determined total power, characteristic of interval j, due to "instantaneous" power levels i,

t_i, t_i+1 – beginning and end of time subinterval i relative to beginning of irradiation, in days.

*Research sponsored by Electric Power Research Institute.
†UCC-ND Computer Sciences Division.

2.19

DIFFUSION THEORY*

D. R. Vondy

[Abstract of Chapter in *CRC Handbook of Nuclear Reactor Calculations*, Y. Ronen, Ed., CRC Press, Boca Raton, FL (in press)]

The computational challenge of the problems that are of interest in nuclear reactor core analysis efforts force the use of the diffusion theory approximation to neutron transport, and this approximation is discussed. The problems of interest include the usual eigenvalue problem to produce k, the search, the fixed source and importance types. A variety of mathematical formulations are in use to apply the techniques of separability, synthesis, finite-difference, finite element and response, alone or in combinations. Solution methods are discussed with emphasis on iteration and acceleration of the rate of solution. A number of analysis techniques that find use are

discussed, such as buckling loss, perturbation and importance. Many of the major core neutronics computer codes are referenced and application considerations are discussed.

Special attention is paid to the global problems that arise in analysis of a nuclear reactor core that involve the neutronics problem as a subset and require advanced computer calculations to produce useful information.

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

2.20

IMPLEMENTING A MODULAR SYSTEM OF COMPUTER CODES*

D. R. Vondy T. B. Fowler

(Abstract of ORNL/TM-8736, July 1983)

A modular computation system has been developed for nuclear reactor core analysis. The codes can be applied repeatedly in blocks without extensive user input data, as needed for reactor history calculations. The primary control options over the calculational paths and task assignments within the codes are blocked separately from other instructions, admitting ready access by user input instruction or directions from automated procedures and promoting flexible and diverse applications at minimum application cost. Data interfacing is done under formal specifications with data files manipulated by an informed manager. This report emphasizes the system aspects and the development of useful capability, hopefully informative and useful to anyone developing a modular code system of much sophistication.

*Research sponsored by U.S. DOE Office of Converter Reactor Deployment.

2.21

ON SOLVING THE CRITICAL CORE NEUTRONICS PROBLEM*

D. R. Vondy

[Abstract of *Ann. Nucl. Energy* **10**, 1 (1983)]

This is a discussion about solving the nuclear reactor-core neutronics problem for the critical state. Methods are described that have evolved for routine application in the assessment of core performance. Of prime interest are the conditions associated with the critical state, and supporting methods development is challenged to solve these problems. Special procedures have been found to be necessary to generate reliable information at a reasonable computation cost. Techniques that have proven useful are discussed, as well as the difficulties that have been experienced in applying multidimensional diffusion theory.

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

2.22

SOLVING THE UNCOMMON NUCLEAR REACTOR CORE NEUTRONICS PROBLEMS*

D. R. Vondy T. B. Fowler

[Abstract of *Nucl. Sci. Eng.* **83**, 100 (1983)]

Calculational procedures have been implemented for solving importance and higher harmonic neutronics problems. Solutions are obtained routinely to support analysis of reactor core performance, treating up to three space coordinates with the multigroup diffusion theory approximation to neutron transport. The techniques used and some of the calculational difficulties are discussed.

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

2.23

THE EFFECT OF PEBBLE THROUGHPUT STRATEGIES ON PEBBLE-BED REACTOR FUEL TEMPERATURES*

B. A. Worley

[Abstract of *Trans. Am. Nucl. Soc.* **43**, 769 (1982)]

An interesting option in using a pebble-bed reactor is the number of times a pebble can be passed through the core for a given fuel residence time. Thus, for a residence time T and n number of passes, the pebbles traverse the core in a time

of Γ/n and are discharged after n passes. The number of passes affects the neutron flux and temperature distributions. As the number of pebble passes increases, the peak power decreases and moves toward the axial center of the core. For a small modular reactor size of 250 MW(t) with coolant upflow, a second pass reduces the peak temperature difference between the minimum achievable and the one-pass upflow case by 40%. A third pass reduces the difference by another 25%. Only small fuel temperature reductions are realized by more than three passes. However, the decrease in peak fuel temperature is accompanied by a dramatic increase in the core-average fuel temperature.

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

2.24

MONTE CARLO CALCULATIONS OF CONTROL-ROD WORTH OF A MEDIUM-SIZE PEBBLE-BED REACTOR*

J. S. Tang† B. A. Worley

(Abstract of ORNL/TM-8111, September 1982)

Three-dimensional full-core control-rod calculations by the Monte Carlo method are performed for a 1640-MW_{th} pebble-bed reactor. The worths of all control rods at several insertions are investigated. The reactivity provided by the control rods in the 1-meter void above the core is about 3% and a 20% negative reactivity can be obtained by axially inserting all control rods 3/4 into the reactor core.

The multigroup KENO Monte Carlo code with a six-group neutron cross-section library was used in the calculations.

*Research sponsored by U.S. DOE Office of Reactor Research and Technology.
†UCC-ND Computer Sciences Division.

2.25

PRE-CONCEPTUAL DESIGN STUDY OF THE ORNL TERNARY METAL FUELED ELECTRONUCLEAR FUEL PRODUCER (TMF-ENFP)*

J. O. Johnson† D. E. Bartine
T. A. Gabriel

(Abstract of paper presented at Workshop on Accelerator Breeders, Chalk River Nuclear Laboratories, Ontario, Canada, September 19-20, 1983)

The original TMF-ENFP target/blanket design proposed by T. J. Burns et al.[1] represented a promising accelerator-breeder concept for the production of the fissile isotopes U-233 and Pu-239. Recent innovative design changes in the accelerator beam, however, have relaxed certain design parameters which affect performance. A preliminary assessment of the neutronic characteristics of a modified design incorporating the recent advances in accelerator beam technology is described here.

The design configuration utilized was evaluated from the standpoints of fissile production, heat generation, and target interaction. A 300-mA beam of 1.1-GeV protons was assumed as the source from the accelerator. A calculational model of the target/blanket assembly was developed which incorporated the new spread beam design and certain other modifications introduced as a result of the spread beam (i.e., axial symmetry, fuel pebbles, etc.). A preliminary assessment of the nuclear performance and thermal-hydraulics parameters of the system was made. Also, parameter studies were performed to optimize the design with respect to cycle length, reprocessing mode, and fissile enrichment.

Although the recent changes in the accelerator beam distribution have relaxed certain design parameters, the basic accelerator-driven blanket concept still requires a tradeoff between fissile production and the heat-removal capability. The spread beam coupled with the fuel pebble geometry allowed the ternary metal fuel to reside in the proton beam path and greatly enhanced the number of neutrons per source proton produced. The fissile production rate of the final design considered was 9.3 kg/day of U-233 with a residence

time of 90 FPD. The overall design concept evaluated here approaches technical feasibility, but some real engineering problems remain (such as heat dissipation and first wall replacement).

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

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1. T. J. Burns, D. E. Bartine, and J. P. Renier, "Concept Evaluation of a Nuclear Design for Electronuclear Fuel Production: Evaluation of ORNL's Proposed TMF-ENFT, ORNL/TM-6828, May 1979.

2.26

SENSOR FAULT ANALYSIS USING DECISION THEORY AND DATA-DRIVEN MODELING OF PRESSURIZED WATER REACTOR SUBSYSTEMS*

B. R. Upadhyaya† M. Skorska

[Astract of *Nucl. Tech.* 64, 70 (1984)]

Instrument fault detection and estimation is important for process surveillance, control, and safety functions of a power plant. The method incorporates the dual-hypotheses decision procedure and system characterization using data-driven time-domain models of signals representing the system. The multivariate models can be developed on-line and can be adapted to changing system conditions. For the method to be effective, specific subsystems of pressurized water reactors were considered, and signal selection was made such that a strong causal relationship exists among the measured variables. The technique is applied to the reactor core subsystem of the loss-of-fluid test reactor using in-core neutron detector and core-exit thermocouple signals. Thermocouple anomalies such as bias error, noise error, and slow drift in the sensor are detected and estimated using appropriate measurement models.

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2.27

REMARKS ON THE INADEQUACY OF ONE-GROUP HETEROGENEOUS THEORIES FOR THE INTERPRETATION OF INCORE NEUTRON NOISE IN BWR'S*

F. C. Difilippo

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

In a recent paper, Analytis (1983) stated that my work in this field (Difilippo, 1982) is contradictory because either the spatial relaxation length of the local component will be overestimated, or the spatial relaxation length of the global component will be underestimated. As it will be indicated below, I partially agree with the statement between quotations but I believe that, taking into consideration the main purpose of my work, there was not a contradiction.

1. It is known (Quddus et al., 1969) that a Feinberg-Galanin model with an explicit treatment of the slowing-down process predicts several relaxation lengths. Quddus demonstrated that, irrespective of the details in the moderation process, one of the relaxation lengths is equal to the *thermal* diffusion length of the moderator. Indeed this relaxation length is preserved when we formally make the Fermi age vanish. This suggests that, in order to calculate the local relaxation length, it is appropriate to use, as in my model, thermal moderator constants in the one-group heterogeneous balance equation.

2. Many homogeneous models used to interpret noise in boiling water reactors (BWRs) utilized a detailed description of the balance equation in the energy domain but they made the implicit assumption that the homogenizing process preserves the relaxation lengths. It is well known that a homogeneous one-group reactor model predicts only one relaxation length of global character. In order to emphasize the importance of heterogeneous effects, I (Difilippo, 1980, 1982) deliberately used a one-group heterogeneous model to demonstrate that both local and global relaxation lengths naturally appear in an

explicit heterogeneous model. Despite the well-known limitation of the one-group theory, including an underestimation of the global relaxation length, it was clearly demonstrated that the homogenizing process does not preserve the relaxation lengths. This demonstration was the main purpose of my work.

3. I never claimed, implicitly or explicitly, that a one-group heterogeneous approach would probably be sufficient as Analytis (1983) wrote. Quite the contrary, I wrote that the theoretical model utilized in this work has obvious pitfalls.

Finally, I would like to mention that the Analytis (1983) paper did not contain reference to my paper published in the *Transactions of the American Nuclear Society* (1980), which was the first analytical demonstration of the importance of heterogeneous effects in the modeling of noise in BWR. In the Technical Note published by Sweeney and myself (1982) in this journal, we list some of the important references in this area which reflect also the chronology of the work and its evolution in time.

*Research sponsored by the Instrumentation and Control Branch, Division of Facility Operations of the Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission.

1. Analytis, G. Th., *Annals of Nuclear Energy* **10**, 379 (1983).
2. Difilippo, F. C., *Trans. Am. Nucl. Soc.* **35**, 592 (1980).
3. Difilippo, F. C., *Nucl. Sci. Eng.* **80**, 211 (1982).
4. Quddus, M. A., R. G. Cochran and E. E. Emon, *Nucl. Sci. Eng.* **35**, 342 (196?).
5. Sweeney, F. J. and F. C. Difilippo, *Annals of Nuclear Energy* **9**, 225 (1982).

2.28

NUCLEAR POWER PLANT SURVEILLANCE BY HEURISTIC LEARNING PARAMETER IDENTIFICATION*

E. L. Machado[†] R. B. Perez

[Abstract of *Trans. Am. Nucl. Soc.* **44**, 550 (1983)]

Continuous surveillance of large dynamic systems such as nuclear power plants can improve system availability by early detection of incipient failure and by avoiding unnecessary periodic maintenance. Noise analysis has been used for surveillance because it does not interfere with plant operation and has been proven effective for the detection of several types of anomalies. When models for the measured noise descriptors are available, the identification of plant parameters can improve the assessment of the plant operational condition. For fast anomaly detection, the parameter identification must be repeated very often, in which case, automatic, efficient, and reliable parameter identification methods are necessary.

Parameter identification typically involves finding the set of parameters that minimizes a functional of the residuals between a set of measured descriptors and the corresponding model-calculated values.

The purpose of this paper is to present a new identification method based on some ideas of heuristic learning control that (a) incorporates previous knowledge in reactor diagnostics and (b) is able to learn from past experience, with the ultimate goal of performing automated, on-line parameter identification for nuclear reactor surveillance and diagnostics.

This method was applied for parameter identification of a U-tube pressurized water reactor steam generator. Three parameters — the overall heat transfer coefficient, the primary water mass flow rate, and the magnitude of the fluctuations of the steam mass flow rate — were identified by using the steam pressure power spectral density (PSD) and the primary water exit temperature PSD (over the frequency range of 0.001 to 0.1 Hz) as the plant descriptors. Comparison with the pattern search method showed that this method, after 20 identifications, is more efficient by a factor of 8 in the number of performed functional calculations.

In conclusion, the heuristic learning parameter identification method can use past experience to improve future identifications, making it an efficient methodology in situations where the same class of identifications must be repeated several times, as in the case of continuous surveillance of nuclear power plants.

*Research sponsored by U.S. DOE Office of Reactor Research and Technology.
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2.29

A PHYSICAL MODEL OF NONLINEAR NOISE WITH APPLICATION TO BWR STABILITY*

J. March-Leuba† R. B. Perez

[Abstract of *Trans. Am. Nucl. Soc.* 44, 523 (1983)]

Historically, the study of stochastic processes for power reactor surveillance and diagnostics has been based on the linearization of the governing equations. This approach has proved to be adequate for many problems; for some conceivable situations, however, such as boiling water reactors (BWRs) close to the boundary of instability, the linear assumption may be violated due to possible large power oscillations. The purpose of this paper is to study the influence of nonlinearities on BWR neutron noise and its effects on stability.

The physical model was described by the following equations:

$$\frac{\partial n}{\partial t} = \frac{\rho-\beta}{\Lambda} n + \lambda c + \frac{\rho}{\Lambda}, \qquad (1)$$

$$\frac{\partial c}{\partial t} = \frac{\beta}{\Lambda} n - \lambda c, \qquad (2)$$

$$\frac{\partial T}{\partial t} = Q - a_3 T, \qquad (3)$$

$$\frac{\partial^2 \rho}{\partial t^2} + a_2 \frac{\partial \rho}{\partial t} + a_1 \rho = T, \qquad (4)$$

$$Q = K[n + f(t)], \qquad (5)$$

which includes (a) one delayed-group point kinetics, Eqs. (1) and (2), (b) the fuel, in a single-node approximation, Eq. (3), and (c) the thermal-hydraulic loop, which has been reduced to a second-order equation for the feedback reactivity, Eq. (4). This last equation has been derived from a two-node approximation of the continuity and energy equations. The power generation, Q, is assumed to be proportional to the neutron density plus a band-limited Gaussian white noise, $f(t)$, acting as the driving source, Eq. (5); the constant K is an adjustable feedback gain that determines the stability of the linear system. The critical value at which the linearized system becomes unstable is $Kc = 0.08912$.

The above set of equations was solved numerically using an A-stable routine.

The results show that for stable systems, $K < Kc$, the Power Spectral Densities (PSDs) exhibit a single peak at the characteristic frequency as predicted by the linearized studies: as K approaches Kc while maintaining the driving source variance constant, however, the PSDs develop peaks at the harmonics of this fundamental frequency. For $K > Kc$, the power oscillations increase in time and eventually reach a limit cycle, with an enhancement of the harmonic components of the PSD.

In conclusion, within the framework of the present model, it is shown that the reactor cannot be unstable in the linear sense, but rather executes limited power oscillations of a magnitude that depends on the operating conditions. The onset of these oscillations can be diagnosed by the decrease in stochasticity in the power traces and by the appearance of harmonics in the PSD.

*Research sponsored by U.S. DOE Division of Reactor Research and Technology.
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2.30

UNIVERSALITY AND APERIODIC BEHAVIOR OF NUCLEAR REACTORS*

J. March-Leuba† D. G. Cacuci
R. B. Perez

(Abstract of *Phys. Rev. Lett.*, in press)

This work reports that a lumped-parameter model of a nuclear reactor can undergo period-doubling pitchfork bifurcations. Until aperiodicity commences, the model behaves in the universal manner predicted by M. J. Feigenbaum. In the aperiodic region, though, the model displays hysteresis. At all times, the model's dynamic evolution remains bounded.

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2.31

NONLINEAR DYNAMICS OF BOILING WATER REACTORS*

J. March-Leuba† D. G. Cacuci
R. B. Perez

[Abstract of *Trans. Am. Nucl. Soc.* 45, 725 (1983)]

This work highlights the qualitative nonlinear behavior of a lumped-parameter representation of a BWR when the reactor's neutron population is increased beyond that at steady-state criticality. The appearance of limit cycle oscillations such as those predicted in this work has already been observed experimentally. This work also shows that a continued reactivity increase will destabilize these limit cycles, leading to aperiodic reactor behavior. Finally, this work shows that the reactor's dynamical evolution remains bounded at all times. An increased understanding of the reactor behavior in the limit cycle and aperiodic regions may open the possibility to practical operation of reactors beyond their current operating conditions.

*Research sponsored by the Instrumentation and Control Branch, Division of Facility Operations of the Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission.

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Reactor Safety, Reliability, and Human Factors Studies

2.32

THE SAFETY-RELATED OPERATOR ACTIONS PROGRAM*

P. M. Haas

(Abstract of paper presented at the 10th Water Reactor Safety Information Meeting, National Bureau of Standards, Gaithersburg, Maryland, October 12, 1982)

The Safety-Related Operator Actions (SROA) Program is intended to provide information and data for use by NRC in assessing the performance of nuclear power plant (NPP) control room operators in responding to abnormal/emergency events. The primary effort has involved collection and assessment of data from simulator "experiments" (actually recorded observations of training exercises) and from historical records of abnormal/emergency events that have occurred in operating plants (field data). These data are to be used to develop criteria for acceptability of the use of manual operator action for safety-related functions. The program also has included studies of training simulator capabilities, of procedures and data for specifying and verifying simulator performance, and of methods and applications of task analysis. This program is scheduled to be completed in FY 1983. This paper summarizes the major results of the program to date, as well as the plans for completion of the program and the general plans for two related programs which have been initiated.

Data on performance of qualified operators in simulator exercises and during actual operating conditions show highly variable "response times", with time corresponding more to operational characteristics of the event than to event severity. Comparison of simulator to field data suggests qualified operators respond much more quickly to simulators (by as much as a factor of six or seven) than in actual accident events.

Studies of simulator design specification and verification in nuclear and non-nuclear applica-

tions suggest the need for a systems approach to training and training media designs.

In a related task, methods for performing a task analysis of NPP control room crew tasks were developed, and a structure for a computerized data base for storage, searching and analysis of task analytic task was suggested.

The interrelated tasks being carried out in this program are providing an improved technical basis for NRC to evaluate control room operator performance and related issues of simulator training and simulator capabilities.

*Research sponsored by U.S. Nuclear Regulatory Commission Office of Nuclear Regulatory Research.

2.33

CRITERIA FOR SAFETY-RELATED NUCLEAR POWER PLANT OPERATOR ACTIONS: INITIAL BOILING WATER REACTOR (BWR) SIMULATOR EXERCISES*

A. N. Beare[†] **D. S. Crowe**[†]
E. J. Kozinsky[†] **D. B. Barks**[†]
P. M. Haas

[Abstract of ORNL/TM-8195 (NUREG/CR-2534), November 1982]

The primary objective of the Safety-Related Operator Action Program at Oak Ridge National Laboratory is to provide a data base to support development of criteria for safety-related action by nuclear power plant operators. This report presents initial data obtained from ten exercises conducted in a boiling water reactor power plant control room simulator. The ten exercises were performed by 24 groups of operators from three utilities. Operator performance was recorded automatically by a program called the Performance Measurement System run on the simulator's computer. Data tapes were subsequently analyzed to extract operator response time (RT) and error rate information. In addition, demographic and subjective data were collected and analyzed in an attempt to identify and evaluate the possible effects of selected performance-shaping factors on operator performance. Operator RTs to the simulated events generally occurred within the intervals allowed in the draft ANSI-N660 design standard; however, they did not appear to be systematically related to the severity of the event, which was the basis for allocation of time margins in the standard. More collective experience in power plant operations was weakly correlated with faster responses. Limited data on omission errors yielded an error rate of greater than five percent.

The data collected will be compared to field data being collected on similar malfunctions. That comparison will provide a basis for extrapolation of simulator data to actual operating conditions. A base of operator performance data developed from simulator experiments can then be used to establish criteria and standards, evaluate the effects of 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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2.34

CRITERIA FOR SAFETY-RELATED NUCLEAR POWER PLANT OPERATOR ACTIONS: INITIAL SIMULATOR TO FIELD DATA CALIBRATION*

A. N. Beare[†] **R. E. Dorris**[†]
E. J. Kozinsky[†] **J. J. Manning**
P. M. Haas

[Abstract of ORNL/TM-8599 (NUREG/CR-3092), February 1983]

This report presents preliminary comparisons of field and simulator operator performance data collected in an NRC-funded research program directed by Oak Ridge National Laboratory. The primary objective of the program is to develop an empirical data base on operator performance to support development of criteria and standards for safety-related operator actions. The comparison of simulator and field data is intended to provide a "calibration" of simulator results so that they can be more confidently extrapolated to field conditions. Collection of PWR/BWR field data was

performed by the Memphis State University Center for Nuclear Studies. The collection of PWR/BWR simulator data was performed by General Physics Corporation, using the Electric Power Research Institute's Performance Measurement System.

The performance measure used in this study was the time required for operators to initiate the first correct manual action in response to an abnormal or emergency event. Techniques for data collection as well as the problems and limitations of field data are reported, along with the initial results.

Response times for experienced BWR operators in the simulator were generally shorter (by a factor of six to seven) and less variable than in the field data. Very limited data from PWR trainees suggest that response times for relatively inexperienced operators in the simulator may be as great as or greater than typical response times in the field.

Two classes of events were distinguished, step events which occur suddenly and ramp events which develop more slowly. For all events, the range of response times (RTs) was very large, with the 95th percentile RT averaging five times the 50th percentile RT. For the BWR events examined, both the 50th percentile RT and the range were much larger for ramp events than for step events in the field, but not in the simulator.

To date, simulator events have not modeled the wide variety of circumstances in which field events are embedded, and which are thought to be responsible for the extreme variability of RTs for field events. Thus, the simulator-to-field calibration is expected to be very crude initially, and it may be some time before a predictive mathematical model can be produced.

*Research sponsored by U.S. Nuclear Regulatory Commission Office of Nuclear Regulatory Research.

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2.35

CRITERIA FOR SAFETY-RELATED NUCLEAR POWER PLANT OPERATOR ACTIONS: 1982 PRESSURIZED WATER REACTOR (PWR) SIMULATOR EXERCISES*

D. S. Crowe† A. N. Beare†
E. J. Kozinsky† P. M. Haas

[Abstract of ORNL/TM-8626 (NUREG/CR-3123), June 1983]

The primary objective of the Safety-Related Operator Action (SROA) Program at Oak Ridge National Laboratory is to provide a data base to support development of criteria for safety-related actions by nuclear power plant operators. When compared to field data collected on similar events, a base of operator performance data developed from the simulator experiments can then be used to establish safety-related operator action design evaluation criteria, evaluate the effects of performance shaping factors, and support safety/risk assessment analyses.

This report presents data obtained from refresher training exercises conducted in a pressurized water reactor (PWR) power plant control room simulator. The 14 exercises were performed by 24 teams of licensed operators from one utility, and operator performance was recorded by an automatic Performance Measurement System. Data tapes were analyzed to extract operator response time (RTs) and error rate information. Demographic and subjective data were collected by means of brief questionnaires and analyzed in an attempt to evaluate the effects of selected performance shaping factors on operator performance.

Operator RTs appeared to be log-normally distributed. The RT for the initial response to a casualty was not related to the severity (potential consequences) of the simulated event as reflected

in the Plant Process Condition classification of the events. There was no statistically significant relation between several measures of operator experience and RT.

Failure to operate a control specified by the appropriate written procedure was defined as an omission error. The overall omission rate for process-related controls was 5%. No relation between error rate and RT was found.

Rating scales completed by the operators after each exercise indicated significant differences among the simulated casualties in terms of the difficulty experienced in diagnosing the nature of the problem and deciding how to respond to it.

*Research sponsored by U.S. Nuclear Regulatory Commission Office of Nuclear Regulatory Research.
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2.36

CRITERIA FOR SAFETY-RELATED OPERATOR ACTIONS*

L. H. Gray P. M. Haas

(Abstract of paper presented at 11th Water Reactor Safety Information Meeting, National Bureau of Standards, Washington, D.C., October 24-28, 1983)

The Safety-Related Operator Actions (SROA) Program, completed in FY-1983, was designed to provide information and data for use by NRC in assessing the performance of nuclear power plant (NPP) control room operators in responding to abnormal/emergency events. The program has included a number of separate but related studies concerned with NPP operator performance, task analysis techniques, and the use of simulators in operator training. The program is one of the earlier NRC research programs in the human factors area, having begun prior to TMI-2, and has in some ways "evolved" with the NRC research effort. The central task — development of criteria for safety-related operator actions based on simulator and field data — was completed in FY-1983 and this terminated the program as scheduled. An initial but substantial base of performance data has been accumulated, and a model of operator performance has been developed and tested which is proposed as a tool to help evaluate the acceptability of assignment of safety-related actions to the operator.

*Research sponsored by U.S. Nuclear Regulatory Commission Office of Nuclear Regulatory Research.

2.37

CRITERIA FOR SAFETY-RELATED OPERATOR ACTIONS: FINAL REPORT*

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[Abstract of ORNL/TM-8942 (NUREG/CR-3515), in press]

This report presents a design evaluation methodology for safety-related actions by nuclear power plant reactor operators and identifies a supporting data base. It is the eleventh and final *NUREG/CR* report on the Safety-Related Operator Actions Program, conducted by Oak Ridge National Laboratory for the U.S. Nuclear Regulatory Commission. The operator performance data were developed from training simulator experiments involving operator responses to simulated scenarios of plant disturbances, from field data on events with similar scenarios, and from task analytic data. A conceptual model to integrate the data was developed and a computer simulation of the model was run, using the SAINT modeling language. Proposed is a quantitative predictive model of operator performance, the "Operator Personnel Performance Simulation (OPPS) Model," driven by task requirements, information presentation, and system dynamics. The model output, a probability distribution of predicted time to correctly complete safety-related operator actions, provides data for objective evaluation of quantitative design criteria.

*Research sponsored by U.S. Nuclear Regulatory Commission Office of Nuclear Regulatory Research.
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2.38

EVALUATION OF TRAINING PROGRAMS AND ENTRY LEVEL QUALIFICATIONS FOR NUCLEAR POWER PLANT CONTROL ROOM PERSONNEL BASED ON THE SYSTEMS APPROACH TO TRAINING*

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M. J. Hanley† R. T. Mercer†

[Abstract of ORNL/TM-8848 (NUREG/CR-3414), September 1983; also *Trans. Am. Nucl. Soc.* 43, 233 (1982)]

This report summarizes results of research sponsored by the U.S. Nuclear Regulatory Commission (NRC) Office of Nuclear Regulatory Research to initiate the use of the Systems Approach to Training in the evaluation of training programs and entry level qualifications for nuclear power plant (NPP) personnel. Variables (performance shaping factors) of potential importance to personnel selection and training are identified, and research to more rigorously define an operationally useful taxonomy of those variables is recommended. A high-level "model" of the Systems Approach to Training for use in the nuclear industry, which could serve as a model for NRC evaluation of industry programs, is presented. The model is consistent with current publicly stated NRC policy, with the approach being followed by the Institute for Nuclear Power Operations, and with current training technology. Checklists to be used by NRC evaluators to assess training programs for NPP control-room personnel are proposed which are based on this model. In an appendix, a "typical" media selection model is illustrated which might be used in the design of training systems for NPP control-room personnel. Further assessment of the proposed checklists to assure practicality, utility and acceptability is recommended. In addition, other issues related to training-effectiveness evaluation are identified, and a comprehensive research approach to address them is outlined.

*Research sponsored by U.S. Nuclear Regulatory Commission Office of Nuclear Regulatory Research.
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2.39

NUCLEAR POWER PLANT CONTROL ROOM TASK ANALYSIS: PILOT STUDY FOR BOILING WATER REACTORS*

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[Abstract of ORNL/SUB/79-40432/1 (NUREG/CR-3415), September 1983]

This report presents the results of the second phase of a two-phase project undertaken to develop a more detailed understanding of the manner in which control room crews perform assigned tasks when mitigating plant malfunctions. Phase II had these objectives: (1) apply the task analysis (TA) methodology developed in Phase I for assessing pressurized water reactor (PWR) crews to create a data base for boiling water reactor (BWR) crews; (2) show how objective data collected during high-fidelity simulation runs can be used to augment TA data collected concurrently; (3) examine additional objective measures of operator performance; and (4) begin to apply the TA data to the development of predictive models of operator/system performance.

Ten plant malfunctions were presented to two-man crews on a Boiling Water Reactor training simulator. Videotaped records of all simulation runs, written operating procedures, interviews, and talk-throughs were data sources for the entry of task analysis information into a computer-searchable data base. When conducting the TA, we assumed that the performance of the operators was goal-directed, with observable facets of behavior mediated by the information available in the control room and by the associated thought processes. By sorting the TA data according to a modified Berliner classification of behaviors, we were able to estimate the amount of time each operator spends perceiving, thinking, communicating, and manipulating controls when responding to the casualty events. We found that by measuring shifts in the times calculated for

these behavioral categories, we might specify how behavior patterns vary across different accident scenarios. In terms of the objective data collected during the simulation runs, we examined two potential performance measures: (1) the timing used by crews in completing multiple actions in a required sequence, and (2) two variants of crew precision in process control — the extent to which a parameter (e.g., reactor water level) deviates from acceptable operating limits and the percentage of time a parameter is out of tolerance.

Predictive modeling was discussed as a means of capturing the dynamic processes by which the operator and the hardware and software components of a system interact. We showed that the development of a dynamic model of nuclear power plant performance would be aided by, and in fact its structure dependent upon, a detailed understanding of the operator's tasks derived from TA data. While development of a full-scale systems model exceeded the scope of this project, we implemented two simple operator models using SAINT networking techniques to demonstrate how data from TA may be incorporated into models which will yield quantitative predictions of one aspect of operator performance (initial response times).

*Research sponsored by U.S. Nuclear Regulatory Commission Office of Nuclear Regulatory Research.

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2.40

NUCLEAR POWER PLANT PERSONNEL ENTRY LEVEL QUALIFICATIONS AND TRAINING*

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(Abstract of paper presented at 11th Water Reactor Safety Information Meeting, National Bureau of Standards, Washington, D.C., October 24-28, 1983)

This paper summarizes the results and current status of a research program at Oak Ridge National Laboratory (ORNL) which initiates the use of the Systems Approach to Training (SAT) in the evaluation of training programs and entry level qualifications for nuclear power plant (NPP) control room personnel. The program is to some extent an outgrowth of previous work at ORNL under the Safety-Related Operator Actions Program. The FY-1982 funded effort (March 1982-July 1983) was focused on adaptation of the SAT process to provide a practical structure for NRC to evaluate training systems and entry level qualifications. Variables (performance shaping factors) of potential importance to entry level qualifications and training were identified, and research to more rigorously define an operationally useful taxonomy of those variables was recommended. A high-level "model" of the SAT for use in the nuclear industry, which could serve as a model for NRC evaluation of industry programs, was constructed. Checklists to be used by NRC evaluators to assess training programs for NPP control room personnel were proposed based on this model. An additional task accomplished was the development of a technique that can be used on an interim basis (until full implementation of the SAT in the nuclear industry) to rank possible plant malfunctions for their importance to training, especially simulator training.

In the FY-1983 funded work, which was recently initiated, the emphasis is on development of specific methodologies (evaluation tools with user guides) to operationalize the evaluation requirements developed in the earlier work. Four tasks are planned: (1) the development of an analytical framework or "structuring model" for describing skilled task performance, (2) further development of the interim malfunction selection technique to develop a task selection methodology to identify which safety-related tasks should be included in training programs, (3) the analysis of factors contributing to precision in task performance measurement, and (4) the development of a training simulator evaluation procedure.

*Research sponsored by U.S. Nuclear Regulatory Commission Office of Nuclear Regulatory Research.

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2.41
A RANKING SCHEME FOR MAKING DECISIONS ON THE RELATIVE TRAINING IMPORTANCE OF POTENTIAL NUCLEAR POWER PLANT MALFUNCTIONS*

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[Abstract of ORNL/TM-8950 (NUREG/CR-3523), in press]

The research summarized in this report was conducted as part of a program entitled "Nuclear Power Plant Entry Level Qualification and Training." A process is developed to assist in the selection of plant malfunctions which should be specifically addressed as part of the training program, and further guidance is given for determining which of those malfunctions should be included in simulator training. Consequences (C), difficulty (D), and frequency (F) rating forms are developed to determine the relative importance of training for any system malfunction. Plant malfunctions were categorized for 46 plant systems. Thirteen of these malfunction categories were then used to demonstrate the C-D-F rating forms.

*Research sponsored by U.S. Nuclear Regulatory Commission Office of Nuclear Regulatory Research.

2.42
PROCEEDINGS OF WORKSHOP ON COGNITIVE MODELING OF NUCLEAR PLANT CONTROL ROOM OPERATORS*

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R. A. Kisner[§] L. S. Abbott
Editors

[Abstract of ORNL/TM-8614 (NUREG/CR-3114), December, 1982]

This document presents 11 invited papers and the deliberations of four working groups at a Workshop on Cognitive Modeling of Nuclear Plant Control Room Operators that was held in Dedham, Massachusetts, under the sponsorship of the U.S. Nuclear Regulatory Commission. The purpose of the workshop was to review the status of "cognitive modeling" and to recommend to the NRC whether it should support research directed toward the development of a cognitive model of a reactor operator that could be useful by itself or as a part of a larger model of the human-machine system. It was the consensus of the invited papers and the working groups that some cognitive models developed for other types of systems can be adapted to the reactor operator under limited and precisely defined conditions (and, indeed, some already are being used). However, the development of a comprehensive model for the reactor operator should be preceded by an improved understanding of the task. In the meantime, the need for further applied research in operator cognition is apparent, as is the need for supporting data collection. Work in these areas could begin immediately.

*Research sponsored by U.S. Nuclear Regulatory Commission Office of Nuclear Regulatory Research.
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[‡]U.S. Nuclear Regulatory Commission, Washington, DC.
[§]Instrumentation and Controls Division.

2.43
JOB ANALYSIS OF THE MAINTENANCE SUPERVISOR AND INSTRUMENT AND CONTROL SUPERVISOR POSITIONS FOR THE NUCLEAR POWER PLANT MAINTENANCE PERSONNEL RELIABILITY MODEL*

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P. J. Federman[†]

[Abstract of ORNL/TM-8299 (NUREG/CR-2668), November 1982]

The job analysis of the maintenance supervisor and instrument and control supervisor positions is part of the work being done within a program that is developing and will evaluate a computer simulation model that will generate nuclear power plant (NPP) maintenance performance reliability data. This report is a second in a series of four job analysis studies which characterize maintenance positions in NPP's. This

characterization takes the form of detailed information about: (1) frequency of task performance, (2) time required for task performance, (3) the training required for adequate task performance and (4) the perceived consequences of inadequate performance. Additionally, information is also presented about the cognitive and perceptual-motor loading imposed by various supervisory tasks.

A list of 51 supervisory tasks was compiled and verified through a number of visits to NPP's. A formal questionnaire concerning these tasks was distributed to 27 NPP's and resulted in an overall 63% return rate. Results of the analysis of the data received formed the basis of this job analysis.

Adequate performance of supervisory tasks was found to require some on-the-job or classroom training and the perceived consequences of inadequate performance were always felt to be greater than no negative consequences. Supervisory tasks were found to require substantial cognitive abilities and because supervisors generally are promoted from the technical level which emphasizes psycho-motor abilities, special training may be required for newly promoted supervisors.

Information from the job analysis will have direct influence on the development of the computer simulation model.

*Research sponsored by U.S. Nuclear Regulatory Commission Office of Nuclear Regulatory Research.

†Applied Psychological Services, Inc., Wayne, PA.

2.44

FRONT-END ANALYSIS FOR THE NUCLEAR POWER PLANT MAINTENANCE PERSONNEL RELIABILITY MODEL*

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J. J. Wolf† H. E. Knee
P. M. Haas

[Abstract of ORNL/TM-8300 (NUREG/CR-2669), August 1983]

The front-end analysis performed for the Nuclear Power Plant Maintenance Personnel Reliability Modeling Program consisted of three primary tasks which are addressed within this report. The first of these was a front-end user survey which investigated the need for and potential content of a structured methodology for nuclear power plant maintenance. For this survey, personal interviews and mail surveys were completed to collect the needed information. A large percentage of the interviewees (80%) said that a structured methodology would be "extremely useful" or "very useful." The classes of information rated most highly by potential model users were: training, undetected errors, radiation exposure, time required for task completion, and proficiency.

The second task of the front-end analysis assessed available human behavioral methodologies for applicability and adaptability potential for nuclear power plant maintenance activities. Although a large number of human behavioral methodologies were found to exist, many were very narrow in scope, had little applicability to maintenance tasks, and did not supply the types of information identified through the front-end user survey. Evaluation of the existing human behavioral methodologies coupled with the needs identified through the front-end user survey led to the choice of simulation modeling as the type of methodology to be developed.

The third task of the front-end analysis was the development of a comprehensive program plan for the simulation model. The plan that was developed spans a 38-month period, and includes three phases: the model development phase, which will result in the release of a debugged/sensitivity tested/calibrated version of the model; the validation phase, which will result in the release of a validated version of the model; and the dissemination phase.

*Research sponsored by U.S. Nuclear Regulatory Commission Office of Nuclear Regulatory Research.

†Applied Psychological Services, Inc., Wayne, PA.

2.45

JOB ANALYSIS OF THE INSTRUMENT AND CONTROL TECHNICIAN POSITION FOR THE NUCLEAR POWER PLANT MAINTENANCE PERSONNEL RELIABILITY MODEL*

A. I. Siegel[†] W. D. Bartter[†]
P. J. Federman[†]

[Abstract of ORNL/TM-8754 (NUREG/CR-3274), August 1983]

The job analysis of the Instrument and Control (I&C) Technician is part of the work being done within a program that is developing and will validate a computer simulation model that will generate maintenance performance reliability data. This report is the third in a series of job analysis studies which characterize maintenance positions in nuclear power plants (NPPs). This characterization takes the form of detailed information about: (1) frequency of task performance, (2) time required for task performance, (3) the training required for adequate task performance, and (4) the perceived consequences of inadequate performance. Information is also presented about the mental and perceptual-motor loading imposed by various work functions.

A list of 252 tasks were compiled and verified through a number of visits to NPPs. Two formal questionnaires concerning these tasks were distributed to 27 NPPs and resulted in a 58% return rate. Results from the data received from the questionnaires formed the basis of this job analysis. Key aspects of the I&C technician position as identified by the job analysis include the high training and ability requirements and perceived high consequences of inadequate performance.

Information from this job analysis report will have direct influence on the development of the computer simulation model.

*Research sponsored by U.S. Nuclear Regulatory Commission Office of Nuclear Regulatory Research.

[†]Applied Psychological Services, Inc., Wayne, PA.

2.46

JOB ANALYSIS OF THE ELECTRICIAN POSITION FOR THE NUCLEAR POWER PLANT MAINTENANCE PERSONNEL RELIABILITY MODEL*

P. J. Federman[†] W. D. Bartter[†]
A. I. Siegel[†]

[Abstract of ORNL/TM-8755 (NUREG/CR-3275), in press]

The job analysis of the electrician position is part of the work being done within a program that is developing and will evaluate a computer simulation model that will generate maintenance performance reliability data. This report is the fourth and last in a series of job analysis studies which characterize maintenance positions in nuclear power plants (NPPs). This characterization takes the form of detailed information about: (1) frequency of task performance, (2) time required for task performance, (3) the training required for adequate task performance, and (4) the perceived consequences of inadequate performance. Information is also presented about the mental and perceptual-motor loading imposed by various work functions.

A list of 199 tasks were compiled and verified through a number of visits to NPPs. Two formal questionnaires concerning these tasks were distributed to 24 NPPs and resulted in a 61% return rate. Results from the data received from the questionnaires formed the basis of this job analysis.

A statistically significant positive correlation was found between electrician training requirements and the perceived severity of adverse consequences following inadequate task performance. This and other information from this job analysis report will have direct influence on the development of the computer simulation model.

*Research sponsored by U.S. Nuclear Regulatory Commission Office of Nuclear Regulatory Research.

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2.47

MAINTENANCE PERSONNEL PERFORMANCE SIMULATION (MAPPS) — A MODEL FOR PREDICTING MAINTENANCE PERFORMANCE RELIABILITY IN NUCLEAR POWER PLANTS*

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(Abstract of paper presented at 11th Water Reactor Safety Information Meeting, National Bureau of Standards, Washington, D.C., October 24-28, 1983)

This pape summarizes the status and results to date of a program designed to develop, validate, and disseminate a practical, acceptable, and useful methodology for the quantitative assessment of NPP maintenance crew reliability. The program has four phases, and is currently near the end of phase two.

Phase one was a scoping study consisting of: (1) a user survey to identify the reliability data needs of potential users; (2) a literature survey of human behavioral methodologies; (3) job analyses of four maintenance positions; and (4) the formulation of a comprehensive program plan for model development, validation, and dissemination.

Phase two involves the formulation and development of the Maintenance Personnel Performance Simulation model (MAPPS). Existing human behavioral methodologies and theories, as well as applicable reliability and probability theories, were operationalized into a simulation framework. The model focuses on the differences between the abilities required for adequate task performance and the current ability levels of the maintenance team. It also addresses a number of performance shaping factors including stress, radiation level, effects of protective clothing, etc., that are used to modify the ability levels of the maintenance team. The model includes a decision-making module and a separate troubleshooting module. In addition, the model is being developed to be minimally dependent upon the availability of data base information.

Phase three is model validation. Validation of the MAPPS model will seek to demonstrate content, construct, and criterion validity. Content validity is supported by the inclusion of critical variables identified in the front-end user survey. Construct validity will be assessed through an examination of correlations among module and model outputs. Criterion validity will involve comparisons of model predictions with criterion performance data collected through acceptable and feasible data sources. If necessary, the model will undergo any needed recalibration to improve simulation accuracy prior to its general release, which is tentatively scheduled for January, 1985.

Phase four involves the dissemination of the MAPPS model to potential users. For this effort, appropriate instructional training materials will be developed and workshops will be conducted to assist users in both the proper applications of the model and the effective interpretation of model results.

The development and validation of the MAPPS model is endeavoring to supply an acceptable, practical, and useful source of maintainer performance reliability data for a number of applications, including PRA. When properly utilized, the human reliability data generated by MAPPS should prove to be an important contribution to efforts addressing the improvement of the nuclear power plant maintenance structure, as well as to studies that are aimed at minimizing the risks associated with nuclear power plant operation.

*Research sponsored by U.S. Nuclear Regulatory Commission Office of Nuclear Regulatory Research.
†Applied Psychological Services, Inc., Wayne, PA.
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2.48

MAPPS: A MODEL FOR ESTIMATING NUCLEAR POWER PLANT MAINTENANCE PERSONNEL RELIABILITY*

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A. I. Siegel†

[Abstract of *Trans. Am. Nucl. Soc.* 45, 210 (1983)]

Experience with nuclear power plant (NPP) operation suggests that human errors committed during maintenance functions can contribute significantly to the overall risk associated with plant operation. As part of the U.S. Nuclear Regulatory Commission's (NRC's) goals of assessing and reducing the risks associated with the operation of NPPs, the potential contribution of maintenance errors to overall NPP risk has been recognized and has provided the motivation for the current program.

The primary objective of the program is to develop, validate, and disseminate a simulation model for the quantitative assessment of NPP maintenance personnel reliability for input into NRC's probabilistic risk assessment (PRA) studies.

Behavioral simulation models attempt to represent logically and mathematically some category of human behavior. They are concerned with person-machine, person-person, and person-environment relationships and with their interactions; i.e., with the performance of individuals and groups under varying conditions due to the environment, training, operational doctrines, individual capabilities, and characteristics of equipment and systems. Behavioral simulation models are effectively dynamic representations of behavior and behavioral influences implemented on a digital computer to allow control or prediction of an event or set of events. MAPPS addresses a number of maintainer, task, and environmental parameters that may affect maintenance performance. These include intellective ability, perceptual-motor ability, stress, motivation, radiation doses, fatigue, quality of procedures, time limits, temperature, communication, use of protective clothing, etc. The model will also include a decision-making module based on Simon-Newell problem-solving theories and a module that will address troubleshooting during maintenance tasks.

The model is being developed to minimize its data base dependency and is rich in output parameters that are available at different levels such as by iteration, by work shift, by subtask, or by task.

Development of the MAPPS model will be completed during the first quarter of FY-1984, and the model will be validated during calendar year 1984. The MAPPS model will be a valuable tool in supplying much needed maintenance reliability data to the PRA studies being carried out by the NRC.

*Research sponsored by U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research.

†Applied Psychological Services, Inc., Wayne, PA.

2.49

THE CENTRALIZED RELIABILITY DATA ORGANIZATION (CREDO): THE SYSTEM AND ITS STATUS*

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[Abstract of *Trans. Am. Nucl. Soc.* 43, 437 (1982)]

The Centralized Reliability Data Organization (CREDO) is an advanced reactor reliability/availability data base and data analysis center. It was established at Oak Ridge National Laboratory (ORNL) in 1978 by the Reactor Research and Technology Division of the U.S. Department of Energy. Since its establishment, a comprehensive data management system has been developed, and significant progress has been achieved in compiling quantitative and qualitative engineering, event, and operating data that are specific to advanced reactors. This paper focuses on the efforts of the first two years of initial operation of the CREDO system and highlights the advances that have been experienced with respect to data collection, system development, and data analysis services.

Continued development of the CREDO system has resulted in a large increase in the amount of data that have been compiled. In addition, emphasis has been placed on the common-cause and data-pooling areas, so that the data in CREDO can be utilized by a larger spectrum of data analysts. As the system moves toward more routine-type operation, CREDO will provide a valuable, comprehensive, and much needed source of breeder reactor component data that currently exists nowhere else in the world.

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

†UCC-ND Computer Sciences Division.

2.50

STATUS OF SACRD: A DATA BASE FOR FAST REACTOR SAFETY COMPUTER CODES*

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[Abstract of *Trans. Am. Nucl. Soc.* 43, 430 (1982)]

The second version of SACRD, a computerized collection of evaluated data for use in fast reactor safety computer codes, was released in October 1981 as Version 81. SACRD includes data for more than 70 materials (mixtures, isotopes, etc.) and more than 150 property categories (thermal conductivity, heat capacity, etc.) of the types needed for breeder reactor safety analyses. The collection is updated on a continuing basis; however, new or revised data are not available except by special request until the next official version of SACRD is released.

The SACRD effort is directed by a Safety Analysis Data Coordinating Group, which coordinates the activities of nine subcommittees overseeing the data collection of thermophysical properties, structural mechanical properties, fuel mechanical properties, aerosol transport properties, radiological data, neutronics data, concrete properties, external code interfaces and formats, and uncertainty data. The data are stored on-line at Oak Ridge National Laboratory and are maintained with the data base management facilities of the JOSHUA system. The data can be supplied in several formats, and since the collection is computerized, interactive access is available through remote terminals with dial-up capabilities. Data can also be obtained as hard copy listings or on tape, along with a simple editing and searching program. Graphical output of tabular data is also available, and the entire collection can be obtained in microfiche format. In addition, several collections of information in SACRD are available as informal handbooks.

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

[†]UCC-ND Computer Sciences Division.
[‡]U.S. Department of Energy.

2.51

IMPACT OF CONTAINMENT BUILDING LEAKAGE ON LWR ACCIDENT RISK*

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[Abstract of ORNL/TM-8964 (NUREG/CR-3539), in press]

The consequences, or risks, from LWR accidents were evaluated as a function of containment building leakage rates. The analysis used the set of generic source terms and frequencies of occurrence developed as representative of the range of postulated types of accidents currently applied in reactor safety research. The variable, M_{sp}, termed the accident spectrum weighted impact fraction rate from containment building leakage, was computed. Explicitly, M_{sp} was formulated as the sum of fractional increases in consequences due to the building leakage for each type of accident weighted by its frequency of occurrence. The base case common to similar types of analyses was applied. The computed result was $M_{sp} \leq 1.5 \cdot 10^{-3}$ fractional increase in the accident spectrum risk per %/day containment building leakage rate.

*Research sponsored by U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research.

[†]UCC-ND Computer Sciences Division.

2.52

DEALING WITH UNCERTAINTY ARISING OUT OF PROBABILISTIC RISK ASSESSMENT*

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(Abstract of ORNL/TM-9076, in press; previously published as Rand Technical Note R-3045-ORNL)

In addressing the area of safety goal implementation, the question of uncertainty arises. This report suggests that the Nuclear Regulatory Commission (NRC) should examine how other regulatory organizations have addressed the issue. Several examples are given from the chemical industry, and comparisons are made to nuclear power risks. Recommendations are made as to

various considerations that the NRC should require in probabilistic risk assessments in order to properly treat uncertainties in the implementation of the safety goal policy.

*Research sponsored by U.S. Nuclear Regulatory Commission Office of Nuclear Regulatory Research.
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2.53

COMMON CAUSE EVALUATIONS IN APPLIED RISK ANALYSIS OF NUCLEAR POWER PLANTS*

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(Abstract of ORNL/TM-8297, April 1983)

Qualitative and quantitative approaches were developed for the evaluation of common cause failures (CCFs) in nuclear power plants and were applied to the analysis of the auxiliary feedwater systems of several pressurized water reactors (PWRs). Key CCF variables were identified through a survey of experts in the field and a review of failure experience in operating PWRs. These variables were classified into categories of high, medium, and low defense against a CCF. Based on the results, a checklist was developed for analyzing CCFs of systems.

Several known techniques for quantifying CCFs were also reviewed. The information provided valuable insights in the development of a new model for estimating CCF probabilities, which is an extension of and improvement over the Beta Factor method. As applied to the analysis of the PWR auxiliary feedwater systems, the method yielded much more realistic values than the original Beta Factor method for a one-out-of-three system.

*Research sponsored by U.S. Nuclear Regulatory Commission Office of Nuclear Regulatory Research, Division of Risk Analysis.
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2.54

STANDARD SETTING STANDARDS: A SYSTEMATIC APPROACH TO MANAGING PUBLIC HEALTH AND SAFETY RISKS*

Baruch Fischhoff†

[Abstract of NUREG/CR-3508 (ORNL/Sub-7576/3), February 1984]

Standards will be an effective means for managing hazardous technologies only if three conditions are satisfied: (a) setting general standards is preferable to case-by-case decision making; (b) some general safety philosophy, balancing risk and other factors, can be justified on normative grounds; (c) that philosophy is faithfully translated into operational terms. In practice, standards are rarely developed and enforced in an integrated systematic way. As a result, they often miss their mark. This guide presents a general framework for the design, development, and implementation of safety standards. That framework is derived from the logical character of the standard setters' task and from experience with actual standards. First, it identifies the conditions under which standards are an appropriate management tool. Second, it presents four generic methods that may be used to develop safety policy. Third, it characterizes the design issues that arise in making that policy operational. At each step, it suggests particular strategies, along with their inherent strengths and weaknesses. In particular, it shows the sensitivity of a standard's effective policy to seemingly technical aspects of the way it is drafted.

*Research sponsored by U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research.
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2.55

SAFETY GOALS FOR NUCLEAR POWER*

Baruch Fischhoff[†]

[Abstract of NUREG/CR-3507 (ORNL/Sub-7576/2), February 1984]

The key policy question in managing hazardous technologies is often some variant of "How safe is safe enough?" The U.S. Nuclear Regulatory Commission has recently broached this topic by adopting "safety goals" defining acceptable risk levels for nuclear power plants. These goals are analyzed here with a general theory of standard setting (Fischhoff, 1983) which asks: (a) Are standards an appropriate policy tool in this case? (b) Can the Commission's safety philosophy be defended? (c) Do the operational goals capture that philosophy? The analysis shows the safety goals proposal to be sophisticated in some respects, incomplete in others. More generally, it points to difficulties with the concept of "acceptable risk" and any attempt to build policy instruments around it. Although focused on the NRC's safety goals, the present analysis is a prototype of what can be learned by similarly detailed consideration of other standards, not only for nuclear power but also for other hazardous technologies, as well as for issues unrelated to safety.

*Research sponsored by U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research.
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2.56

RISK ANALYSIS: UNDERSTANDING "HOW SAFE IS SAFE ENOUGH?"*

S. L. Derby[†] **R. L. Keeney**[‡]

[Abstract of *Risk Analysis* 1(3), 217 (1981)]

The basic characteristics of determining acceptable risk are discussed. Technical, political, and social aspects of the problem add much complexity. The appropriate manner to reach responsible decisions regarding acceptable risk is suggested. This explicitly addresses the alternatives, the objectives, the uncertainty, and the values which constitute the information necessary to arrive at any solution. The inappropriateness of many "solutions" currently in use or suggested is exposed.

*Research sponsored by U.S. Nuclear Regulatory Commission Office of Nuclear Regulatory Research.
[†]Department of Engineering-Economic Systems, Stanford University, Stanford, CA.
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2.57

EVALUATION OF MORTALITY RISKS FOR INSTITUTIONAL DECISIONS*

R. L. Keeney[†]

[Abstract of a chapter to be published in the book *Analyzing and Aiding Decision Processes*, P. C. Humphreys, O. Svenson, and A. Vari, Eds., North-Holland Publishers, New York (in press)]

This chapter investigates the implications of various value judgments on the evaluation of public mortality risks. Using utility functions to quantify values indicates the mutual inconsistency of three reasonable goals: minimize the expected number of fatalities, promote equitable distribution of public risk, and a preference for catastrophe avoidance. Utility analyses and some other approaches for assisting with decision making involving public risks are appraised from an institutional or governmental perspective.

*Research sponsored by U.S. Nuclear Regulatory Commission Office of Nuclear Regulatory Research.
[†]Woodward-Clyde Consultants, San Francisco, CA.

2.58

GRESS — GRADIENT-ENHANCED SOFTWARE SYSTEM — VERSION B USER'S GUIDE*

E. M. Oblow

(Abstract of ORNL/TM-8339, April 1983)

An automated procedure for performing sensitivity analyses has been developed. The procedure uses a new FORTRAN compiler with computer calculus capabilities to generate the analytic derivative equations needed to set up sensitivity equations. The new compiler is called

GRESS — Gradient Enhanced Software System. It reads standard FORTRAN programs with no additional modifications and generates a new derivative-enhanced FORTRAN version of the code suitable for sensitivity studies. This new approach was found to preserve the traditional advantages of adjoint theory while removing the tedious human effort previously needed to apply this theoretical methodology. The automated procedure should be applicable in numerical analysis and large-scale modeling sensitivity studies.

*Research sponsored by Office of Nuclear Waste and Isolation, BATTELLE Project Management Division, Columbus, OH.

2.59

AN AUTOMATED PROCEDURE FOR SENSITIVITY ANALYSIS USING COMPUTER CALCULUS*

E. M. Oblow

[Abstract of ORNL/TM-8776, May 1983; also *Nucl. Sci. Eng.* (in press)]

An automated procedure for performing sensitivity analyses has been developed. The procedure uses a new FORTRAN compiler with computer calculus capabilities to generate the derivatives needed to set up sensitivity equations. The new compiler is called GRESS — Gradient Enhanced Software System. Application of the automated procedure with "direct" and "adjoint" sensitivity theory for the analysis of non-linear, iterative systems of equations is discussed. Calculational efficiency consideration and techniques for adjoint sensitivity analysis are emphasized. The new approach was found to preserve the traditional advantages of adjoint theory while removing the tedious human effort previously needed to apply this theoretical methodology. Conclusions are drawn about the applicability of the automated procedure in numerical analysis and large-scale modelling sensitivity studies.

*Research sponsored by Office of Nuclear Waste and Isolation, BATTELLE Project Management Division, Columbus, OH.

2.60

COMPBRN — A COMPUTER CODE FOR MODELING COMPARTMENT FIRES*

N. O. Siu†

[Abstract of NUREG/CR-3239 (UCLA-ENG-8257), May 1983]

The computer code COMPBRN deterministically models the behavior of fire in a compartment. This manual presents information necessary to run COMPBRN, including descriptions of required input and resulting output. Also included are a sample problem and a listing of the code (written in FORTRAN for an IBM 3033 computer). This manual is to be used in conjunction with NUREG/CR-2269, "Probabilistic Models for the Behavior of Compartment Fires," August 1981, which describes the fire models employed by COMPBRN.

*Research sponsored by the U.S. Nuclear Regulatory Commission Office of Nuclear Regulatory Research.

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Section 3

FUSION REACTOR RESEARCH

3.0. INTRODUCTION

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 principal 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 stainless steel 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 Hevimet. 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.

3.1

STREAMING OF 14-MeV NEUTRONS THROUGH AN IRON DUCT - COMPARISON OF MEASURED NEUTRON AND GAMMA-RAY ENERGY SPECTRA WITH RESULTS CALCULATED USING THE MONTE CARLO MCNP CODE*

R. T. Santoro J. M. Barnes[†]
P. D. Soran[‡] R. G. Alsmiller, Jr.

[Abstract of ORNL/TM-8504, November 1982; also *Nucl. Sci. Eng.* **84**, 260 (1983)]

Neutron and gamma-ray energy spectra resulting from the streaming of 14-MeV neutrons through a 0.30-m-diam duct (length-to-diameter ratio = 2.83) have been calculated using the Monte Carlo MCNP code. The calculated spectra are compared with measured data and data calculated previously using a combination of discrete ordinates and Monte Carlo methods. Comparisons are made at 12 detector locations on and off the duct axis for neutrons with energies above 850 keV and for gamma rays with energies above 750 keV. The neutron spectra calculated using MCNP agree with the measured data within ~5 to ~50%, depending on detector location and neutron energy. Agreement with the measured gamma-ray spectra is also within ~5 to ~50%. The spectra obtained with MCNP are also in favorable agreement with the previously calculated data and were obtained with less calculational effort.

*Research sponsored by U.S. DOE Office of Fusion Energy.
[†]UCC-ND Computer Sciences Division.
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3.2

MONTE CARLO CALCULATIONS OF NEUTRON AND GAMMA-RAY ENERGY SPECTRA FOR FUSION REACTOR SHIELD DESIGN: COMPARISON WITH EXPERIMENT*

R. T. Santoro J. M. Barnes[†]

[Abstract of ORNL/TM-8707, August 1983; also Fifth Topical Meeting on the Technology of Fusion Energy, Knoxville, Tennessee, April 26-28, 1983; Proc. *Nucl. Tech./Fusion* **4**(2), Part 2, 367 (1983)]

Neutron and gamma-ray energy spectra resulting from the interactions of ~14 MeV neutrons in laminated slabs of stainless steel type-304 and borated polyethylene have been calculated using the Monte Carlo code MCNP. The calculated spectra are compared with measured data as a function of slab thickness and material composition and as a function of detector location behind the slabs. Comparisons of the differential energy spectra are made for neutrons with energies above 850 keV and for gamma rays with energies above 750 keV. The measured neutron spectra and those calculated using Monte Carlo methods agree within 5% to 50%, depending on the slab thickness and composition and neutron energy. The agreement between the measured and calculated gamma-ray energy spectra are also within this range. The MCNP data are also in favorable agreement with attenuated data calculated previously by discrete ordinates transport methods and the Monte Carlo code SAM-CE.

*Research sponsored by U.S. DOE Office of Fusion Energy.
[†]UCC-ND Computer Sciences Division.

3.3

INTEGRAL EXPERIMENTS FOR FUSION REACTOR SHIELD DESIGN — SUMMARY OF PROGRESS*

R. T. Santoro R. G. Alsmiller, Jr.
J. M. Barnes† G. T. Chapman

[Abstract of paper presented at the Sixth International Conference on Radiation Shielding, Tokyo, Japan, May 16-20, 1983; Proc. Vol. II, p. 627 (1983)]

Neutron and gamma-ray energy spectra from the reaction of ~14-MeV neutrons in blanket and shield materials and from the streaming of these neutrons through a cylindrical duct (L/D ~2) have been measured and calculated. These data are being obtained in a series of integral experiments to verify the radiation transport methods and nuclear data that are being used in nuclear design calculations for fusion reactors. The experimental procedures and analytic methods used to obtain the calculated data are reviewed. Comparisons between measured and calculated data for the experiments that have been performed to date are summarized.

*Research sponsored by U.S. DOE Office of Fusion Energy.
†UCC-ND Computer Sciences Division.

3.4

THE ORNL INTEGRAL EXPERIMENT TO PROVIDE DATA FOR EVALUATING MAGNETIC-FUSION-ENERGY SHIELDING CONCEPTS. PART I: ATTENUATION MEASUREMENTS*

G. T. Chapman G. L. Morgan†
J. W. McConnell‡

(Abstract of ORNL/TM-7356, August 1982)

Integral experiments to measure the energy spectra of neutrons and gamma rays due to the transport of approximately 14-MeV T(*d*,*n*)^{4}He neutrons through laminated stainless-steel and borated-polyethylene shield configurations have been performed at the Oak Ridge National Laboratory. An NE-213 detector and conventional pulse-shape-discrimination circuitry were used to record the pulse-height distributions from which the energy spectra were derived. Descriptions of the facility and experimental techniques are given in this paper along with tables and curves showing the results of the measurements.

*Research sponsored by U.S. DOE Office of Fusion Energy.
†Los Alamos National Laboratory, Los Alamos, NM.
‡Physics Division.

3.5

DOSE RATES FROM INDUCED ACTIVITY IN THE ELMO BUMPY TORUS PROOF-OF-PRINCIPLE DEVICE*

R. G. Alsmiller, Jr. R. T. Santoro
J. Barish† J. M. Barnes†

[Abstract of ORNL/TM-8505, October 1982; also *Nucl. Tech.* **4**, 491 (1983)]

Calculated results of the dose rates from induced activity in the enclosure of the Elmo Bumpy Torus proof-of-principle device (EBT-P) are presented. A cylindrical model of EBT-P is used. EBT-P will have a hydrogen plasma and thus the plasma will not produce neutrons, but substantial numbers of photoneutrons will be produced and it is the induced activity from these photoneutrons that is considered. The activation dose rates are presented for a variety of operating times and times after shutdown.

*Research sponsored by U.S. DOE Office of Fusion Energy.
†UCC-ND Computer Sciences Division.

3.6

MICROWAVE TRANSPORT IN EBT DISTRIBUTION MANIFOLDS USING MONTE CARLO RAY TRACING TECHNIQUES*

R. A. Lillie T. L. White†
T. A. Gabriel R. G. Alsmiller, Jr.

[Abstract of paper presented at the Fifth Topical Meeting on the Technology of Fusion Energy, Knoxville, Tennessee, April 26-28, 1983; Proc. *Nucl. Tech./Fusion* **4**(2), Part 3, 1436 (1983)]

Ray-tracing Monte Carlo calculations have been carried out using an existing Monte Carlo

radiation transport code to obtain estimates of the microwave power exiting the torus coupling links in EBT microwave manifolds. The microwave power loss and polarization at surface reflections were accounted for by treating the microwaves as plane waves reflecting off plane surfaces. Agreement on the order of 10% was obtained between the measured and calculated output power distribution for an existing EBT-S toroidal manifold. A cost-effective iterative procedure utilizing the Monte Carlo history data was implemented to predict design changes which could produce increased manifold efficiency and improved output power uniformity.

*Research sponsored by U.S. DOE Office of Fusion Energy.
†Fusion Energy Division.

3.7

NEUTRONICS — TRITIUM BREEDING*

R. T. Santoro

[Abstract of *Res. Mechanica* **8**, 1 (1983)]

The neutronics of tritium breeding are reviewed. The ramifications of breeding are delineated with respect to overall blanket design. A brief commentary on the one-dimensional discrete ordinates method for estimating tritium production is presented. Some examples of the tritium breeding capability of prototypic blanket designs are summarized along with examples of the influence on calculated tritium breeding ratios of cross-section uncertainties.

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

3.8

CONTROL OF ACTIVATION LEVELS TO SIMPLIFY WASTE MANAGEMENT OF FUSION REACTOR FERRITIC STEEL COMPONENTS*

F. W. Wiffen† R. T. Santoro

(Abstract of paper presented at the Topical Conference on Ferritic Alloys for Use in Nuclear Energy Technologies, Snowbird, Utah, June 19-23, 1983)

Activation characteristics of a material for service in the neutron flux of a fusion reactor first wall fall into three areas: waste management, reactor maintenance and repair, and safety. Of these, the waste management area is the most likely to impact the public acceptance of fusion reactors for power generation. The decay of the activity in steels within tens of years could lead to simplified waste disposal or possibly even to materials recycled. Whether or not these can be achieved will be controlled by (1) selection of alloying elements, (2) control of critical impurity elements, and (3) control of cross contamination from other reactor components. Several criteria can be used to judge the acceptability of potential alloying elements in iron, and to define the limits on content of critical impurity elements. One approach is to select and limit alloying additions on the basis of the activity. If material recycle is a goal, N, Al, Ni, Cu, Nb, and Mo must be excluded. If simplified waste storage by shallow land burial is the goal, regulations limit the concentration of only a few isotopes. For first-wall material that will be exposed to 9 MW-y/m^2 service, allowable initial concentration limits include (in at. ppm) Ni < 20,000; Mo < 3,650; N < 3,650, Cu < 2,400; and Nb < 1.0. The other constituent elements of ferritic steels will not be limited. Possible substitutes for the molybdenum normally used to strengthen the steels include W, Ta, Ti, and V.

*Research sponsored by U.S. DOE Office of Fusion Energy.
†Metals and Ceramics Division.

3.9

VANADIUM ALLOYS AND MODIFIED STEELS FOR LOW-ACTIVATION FUSION REACTOR DESIGN*

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R. T. Santoro F. W. Wiffen†

[Abstract of *Trans. Am. Nucl. Soc.* **43**, 305 (1982)]

The motivations for low-activation designs of fusion reactors fall into three categories: reactor maintenance, safety, and waste management and storage. Reducing activity levels to achieve contact maintenance of the blanket and plasma chamber of fusion power reactor is probably not

achievable. However, any reductions of induced activity will make maintenance easier in that it will reduce shielding requirements. Likewise, reductions in biological hazard potential (BHP) and the production of radioactive isotopes with long half-lives can have a significant impact on safety and waste management. One approach to the activation issue is the use of ultra-high-purity SiC and aluminum alloys as structural materials. An alternative approach is the selection of alloys that are specifically tailored for reduced activity. A powerful advantage of the latter approach is that it is fully compatible with an established and proven design methodology.

There are two major considerations that must be addressed with regard to the broad utilization of vanadium-base alloys for fusion reactors:

1. The development of credible and reliable means to ensure the exclusion of air or other reactive gases from contact where vanadium alloys temperatures might exceed ~350°C. (This is a basic design requirement for vanadium-base alloys irrespective of concerns over low levels of induced radioactivity.)

2. The identification and control of residual impurities, since it is the activation of these impurities that will dominate the radioactivity levels.

*Research sponsored by U.S. DOE Office of Fusion Energy.
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3.10

EBT REACTOR ANALYSIS*

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R. T. Santoro D. A. Spong†
T. Uckan† L. W. Owen‡
J. M. Barnes‡ J. B. McBride†

(Abstract of ORNL/TM-8712, August 1983)

This report summarizes the results of a recent ELMO Bumpy Torus (EBT) reactor study that includes ring and core plasma properties with consistent treatment of coupled ring-core stability criteria and power balance requirements. The principal finding is that constraints imposed by these coupling and other physics and technology considerations permit a broad operating window for reactor design optimization. Within this operating window, physics and engineering systems analysis and cost sensitivity studies indicate that reactors with < β_{core} > ~6-10%, P ~1200-1700 MW(e), wall loading ~1.0-2.5 MW/m^2, and recirculating power fraction (including ring-sustaining power and all other reactor auxiliaries) ~10-15% are possible. A number of concept improvements are also proposed that are found to offer the potential for further improvement of the reactor size and paramete. These include, but are not limited to, the use of (1) supplementary coils or noncircular mirror coils to improve magnetic geometry and reduce size, (2) energetic ion rings to improve ring power requirements, (3) positive potential to enhance confinement and reduce size, and (4) profile control to improve stability and overall fusion power density.

*Research sponsored by U.S. DOE Office of Fusion Energy.
†Fusion Energy Division.
‡UCC-ND Computer Sciences Division.

3.11

PLASMA ENGINEERING ANALYSIS OF AN EBT OPERATING WINDOW*

R. T. Santoro N. A. Uckan†
J. M. Barnes‡

[Abstract of paper presented at the Fifth Topical Meeting on the Technology of Fusion Energy, Knoxville, TN, April 26-28, 1983; Proc. *Nucl. Tech./Fusion* 4(2), Part 2, 491 (1983)]

The operating space for EBT reactors is calculated using a newly developed systems code that incorporates recent advances in EBT physics. The calculation includes a self-consistent treatment of coupled ring-core stability and power balance requirements. The essential elements of the systems code are reviewed, including magnetics, stability, ring power, power balance, confinement time, and cost calculations. Finally, a typical

reactor systems analysis is summarized for a family of EBT reactors that fall within the allowed operating space.

*Research sponsored by U.S. DOE Office of Fusion Energy.
†Fusion Energy Division.
‡UCC-ND Computer Sciences Division.

3.12

EBT REACTOR CHARACTERISTICS CONSISTENT WITH STABILITY AND POWER BALANCE REQUIREMENTS*

N. A. Uckan† R. T. Santoro

[Abstract of paper presented at the Fifth Topical Meeting on the Technology of Fusion Energy, Knoxville, TN, April 26-28, 1983; Proc. *Nucl. Tech./Fusion* 4(2), Part 3, 1326 (1983)]

This paper summarizes the results of a recent EBT reactor study that includes both ring and core plasma properties and consistent treatment of coupled ring-core stability criteria and power balance requirements. The principal finding is that constraints imposed by these coupling and other physics and technology considerations permit a broad operating window for reactor design optimization. A number of concept improvements are also proposed that are found to offer the potential for further improvement of the reactor size and parameters.

*Research sponsored by U.S. DOE Office of Fusion Energy.
†Fusion Energy Division.

3.13

PLASMA ENGINEERING ANALYSIS OF THE TENNESSEE TOKAMAK*

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J. B. Miller† W. E. Bryan‡
P. W. King§ R. T. Santoro
T. E. Shannon‡ N. A. Uckan§

(*Abstract of paper presented at the 10th Symposium on* Fusion Engineering, Philadelphia, PA, December 5-9, 1983)

This paper summarizes the results of the plasma engineering and systems analysis studies for the Tennessee Tokamak (TENTOK) fusion power reactor. TENTOK is a 3000-MW(t) central station power plant that uses deuterium-tritium fuel in a D-shaped tokamak plasma configuration with a double-null poloidal divertor. The major parameters are R_o = 6.4 m, a = 1.6 m, σ (elongation) = 1.65, $\langle n \rangle$ = 1.5 × 10^{20} m^{-3}, $\langle T \rangle$ = 15 keV, $\langle \beta \rangle$ = 6%, B_T (on-axis) = 5.6 T, I_p = 8.5 MA, and wall loading = 3 MW/m^2. Detailed analyses are performed in the areas of (1) transport simulation using the one-and-one-half-dimensional (1-1/2-D) WHIST transport code, (2) equilibrium/poloidal field coil systems, (3) neutral beam and radiofrequency (rf) heating, and (4) pellet fueling. In addition, impurity control systems, diagnostics and controls, and possible microwave plasma preheating and steady-state current drive options are also considered. Some of the major features of TENTOK include rf heating in the ion cyclotron range of frequencies, superconducting equilibrium field coils outside the superconducting toroidal field coils, a double-null poloidal divertor for impurity control and alpha ash removal, and rf-assisted plasma preheating and current startup.

*Research sponsored by U.S. DOE Office of Fusion Energy.
†University of Tennessee, Knoxville, TN.
‡Engineering Division.
§Fusion Energy Division.

3.14

NEUTRONIC EVALUATION OF THE FISSION SUPPRESSED TANDEM-MIRROR HYBRID REACTOR (TMHR)*

J. O. Johnson† T. J. Burns

(Abstract of ORNL/TM-8976, in press)

A computational study was performed on the Lawrence Livermore National Laboratory (LLNL) Tandem Mirror Hybrid Reactor (TMHR) blanket design to qualify the methods and data bases available at Oak Ridge National Laboratory (ORNL) for use in analyzing the neutronic performance of fissile fuel breeding blankets. The eventual goal of the study is to establish a capability for analysis and optimiza-

tion of advanced fissile fuel production blanket designs.

Discrete ordinates radiation transport calculations were performed in one-dimensional cylindrical geometry to obtain the blanket spatial distribution and energy spectra of the neutron and gamma-ray fluxes resulting from the monoenergetic (14.1-MeV) fusion first-wall source. Key macroscopic cross sections of the blanket materials were then folded with the flux spectra to obtain reaction rates critical to evaluating blanket feasibility. Finally, a time-dependent depletion analysis was performed to evaluate the blanket performance during equilibrium cycle conditions. The results of the study are presented both as graphs and tables.

The primary objective aimed at qualifying the computational methods and data bases used at ORNL for analyzing various fissile fuel breeding blanket designs has been realized. In general, the nuclear performance characteristics calculated by ORNL agreed to within 10% with the LLNL results. There were, however, a few discrepancies in key reaction rates which were attributed to the cross-section processing method and the data base used. Because of these discrepancies, a more detailed comparison of the basic data base and cross-section processing methodologies utilized by ORNL and LLNL is needed before a capability that can be used for analysis and optimization of advanced fuel production blankets can be offered.

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

†University of Tennessee, Knoxville, TN.

Section 4

HIGH-ENERGY ACCELERATOR SHIELDING AND DETECTOR RESEARCH

4.0. INTRODUCTION

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.

4.1

PRELIMINARY MONTE CARLO CALCULATIONS OF THE RESPONSE OF THE GONDOLA COUNTERS OF UA1 TO $e^{\pm}, \pi^{\pm}, p$*

T. A. Gabriel R. Wilson[†]

(Abstract of CERN-UA1/TN82-23, May 1982)

For the update of the UA1 experiment (renumbered UA7) at CERN we propose to build a uranium/tetramethyl silane calorimeter to replace the existing gondolar counters and bouchon counters. This report is a calculational analysis, using the Oak Ridge Monte Carlo programs for development of hadron showers, and the EGS program for electromagnetic showers of the performance of such a system.

*Research sponsored by U.S. DOE Office of High Energy and Nuclear Physics.
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4.2

NEUTRON AND GAMMA-RAY SHIELDING REQUIREMENTS FOR A BELOW-GROUND NEUTRINO DETECTOR SYSTEM AT THE RUTHERFORD LABORATORY SPALLATION NEUTRON SOURCE*

T. A. Gabriel R. A. Lillie
R. L. Childs[†] J. Wilczynski[‡]
B. Zeitnitz[‡]

(Abstract of ORNL/TM-8355, March 1983)

The neutron and gamma-ray shielding requirements for a proposed neutrino detector system below the target station at the Rutherford Laboratory Spallation Neutron Source (SNS) are studied. The present shield below the station consists of 2 meters of iron and 1 meter of concrete, below which is chalk ($CaCO_3$). An underground bunker housing the neutrino detector system would require additional shielding consisting of 6 meters of the chalk plus ~3 meters of iron to reduce the number of high-energy (~7 MeV) neutrons and gamma rays entering the detector system to an acceptable level of ~1 per day.

*Research sponsored by U.S. DOE Office of High Energy and Nuclear Physics.
[†]UCC-ND Computer Sciences Division.
[‡]Nuclear Research Center, Karlsruhe, West Germany.

4.3

THE USE OF Gd-LOADED SCINTILLATION DETECTOR SYSTEMS FOR INVERSE BETA DECAY REACTIONS*

T. A. Gabriel R. A. Lillie
R. L. Childs[†]

[Abstract of ORNL/TM-8711, May 1983; also *Nucl. Instrum. Methods* (in press); also Workshop on Neutrino Research, Nuclear Research Center, Karlsruhe, W. Germany, July 5, 1983]

The results of calculations carried out to determine the time frame and spatial separation of neutron captures in Gd resulting from inverse beta decay reactions are presented for a modular neutrino detector system.

*Research sponsored by U.S. DOE Office of High Energy and Nuclear Physics.
[†]UCC-ND Computer Sciences Division.

4.4

A SENSITIVE SEARCH FOR NEUTRON-ANTINEUTRON TRANSITIONS*

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R. A. Lillie P. D. Miller[‡]
F. E. Obenshain[‡] R. R. Spencer
G. R. Young[‡] J. Brau[§] W. M. Bugg[§]
G. T. Condo[§] T. Handler[§]
J. L. Hargis[§] E. L. Hart[§]

(Abstract of ORNL/PHYS-82/1, July 1982)

We proposed to perform a sensitive search for the phenomenon of baryon number non-conservation ($\Delta B = 2$). We propose to search for neutron → 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 the order of $10^{7\pm1}$ seconds. In a run of 120 days, we will observe the transitions if the period is less than 2×10^8 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. We show that it enables us to perform a more sensitive measurement of the transition time than does any other neutron source in the world. Due to the large number of neutrons which must be released in the experiment, we have taken advantage of the extensive expertise at ORNL in reactor physics, neutron physics and shielding design. 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 exists the field-free region will annihilate on a thin 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 with a multiplicity of three or larger.

*Research sponsored by U.S. DOE Office of High Energy and Nuclear Physics.
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4.5

NEUTRON OSCILLATIONS AND THE STABILITY OF MATTER*

G. R. Young† **T. A. Gabriel**

(Abstract of Final Report on Seed Money Project 3203-0239, August 5, 1982)

This report describes studies, carried out under the sponsorship of the ORNL Seed Money Program, directed towards designing a workable detector to be used in a reactor experiment searching for the spontaneous transition of a neutron into its antimatter partner, an antineutron. A summary of results is given in this section. Detailed results concerning detector design and response to the sought-for events, behavior of the detector in ambient reactor backgrounds, and background from cosmic ray muons are given in the subsequent three sections.

*Research sponsored by U.S. Department of Energy.
†Physics Division.

4.6

NANO — THE HARVARD-OAK RIDGE NATIONAL LABORATORY-UNIVERSITY OF TENNESSEE NEUTRON-ANTINEUTRON OSCILLATION SEARCH*

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R. A. Lillie **P. D. Miller**†
F. E. Obenshain† **R. R. Spencer**
G. R. Young† **M. S. Goodman**‡
R. Wilson‡ **W. M. Bugg**§
G. T. Condo§ **T. Handler**§
E. L. Hart§

[Abstract of paper presented at the Informal Workshop on Neutron-Antineutron Oscillations, Harvard University, Cambridge, MA, April 30 — May 1, 1982; Proc., p. 123, M. S. Goodman, M. Machacek, and P. D. Miller, Eds. (1982)]

An experiment to search for ($\Delta B = 2$) neutron-antineutron oscillation is described. The facility is based on the use of a large beam port at the Oak Ridge Research Reactor. The proposed experiment should be capable of attaining values of $Nt^2 > 10^9$ sec. Calculations of detector response and background rates are presented along with a detailed discussion of the results of tests which we performed at the Tower Shielding Reactor Facility in Oak Ridge. We conclude that a detector capable of working at the ORR reactor can be constructed which can efficiently detect antineutron annihilation events while rejecting reactor-related background and producing a high level of cosmic-ray rejection.

*Research sponsored by U.S. DOE Office of High Energy and Nuclear Physics.
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4.7

NEUTRONICS CALCULATIONS FOR A NEUTRON-ANTINEUTRON OSCILLATION EXPERIMENT[*]

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F. E. Obenshain[†] R. R. Spencer
G. R. Young[†] M. S. Goodman[‡]
R. Wilson[‡] W. M. Bugg[§] G. T. Condo[§]
J. L. Hargis[§] E. L. Hart[§]

[Abstract of *Bull. Am. Phys. Soc.* 28(4), 709 (1983)]

Calculations have been made of the thermal- and fast-neutron leakage from a reactor to be used as the neutron source in a neutron-antineutron oscillation experiment. Calculations of the flux at the exit of the oscillation region, of scattered fast-neutron and gamma-ray fluxes reaching the detection system and cosmic-ray veto system, and of the rates of radiation damage to detector elements and drift tube elements have been made. Expected residual dose rates in the detector area after one year of experiment operation have also been calculated. The results show background rates are sufficiently small and that radiation damage rates and radioactivity buildup rates are acceptable.

[*]Research sponsored by U.S. DOE Office of High Energy and Nuclear Physics.

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4.8

REACTOR BEAM TESTS OF DETECTOR ELEMENTS FOR A NEUTRON-ANTINEUTRON OSCILLATION EXPERIMENT[*]

R. R. Spencer H. O. Cohn[†]
T. A. Gabriel R. A. Lillie
P. D. Miller[†] F. E. Obenshain[†]
G. R. Young[†] W. M. Bugg[‡]
G. T. Condo[‡] E. L. Hart[‡]
M. S. Goodman[§] R. Wilson[§]

[Abstract of *Bull. Am. Phys. Soc.* 28(4), 709 (1983)]

Tests have been made of the response of SF5-type lead glass blocks and cast acrylic scintillators to the background radiation resulting from a reactor neutron source of the type to be used in a neutron-antineutron oscillation experiment. Various neutron beam filters were used to isolate the response to thermal neutrons, fast neutrons, and reactor gamma-ray background. The final shielding and beam collimation design, which is still quite transparent to the products of the antineutron annihilation, will be presented, along with expected false trigger rates due to reactor background.

[*]Research sponsored by U.S. DOE Office of High Energy and Nuclear Physics.

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4.9

EXPERIMENTAL CONSIDERATIONS FOR A SENSITIVE NEUTRON-ANTINEUTRON OSCILLATION SEARCH[*]

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R. A. Lillie P. D. Miller[‡]
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G. T. Condo[§] T. Handler[§] E. L. Hart[§]

[Abstract of *Indian J. Phys.* 209 (1983)]

The experimental considerations for the design of an experiment to search for ($\Delta B = 2$) neutron-antineutron mixing are presented. A comparison of free-neutron sources leads to the choice of a reactor facility based on the ORR reactor at the Oak Ridge National Laboratory as the facility with the largest Nt^2. A proposed experiment to measure the antineutron annihilation signature (~2 GeV into pions) is described. Features of the design of the experiment, including a 20-m evacuated magnetically shielded flight path, a detector sensitive to energy and particle multiplicity, and an active cosmic-ray shield and anticoincidence, are presented. A brief status report of studies currently in progress, including both detector tests at a reactor facility and com-

putational modelling, is given. The proposed experiment should be capable of attaining values of $Nt^2 > 3 \times 10^9$ sec.

*Research sponsored by U.S. DOE Office of High Energy and Nuclear Physics.
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4.10

MONTE CARLO CALCULATIONS OF DETECTOR RESPONSE FOR A NEUTRON-ANTINEUTRON OSCILLATION EXPERIMENT*

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J. L. Hargis[§] E. L. Hart[§]

[Abstract of *Bull. Am. Phys. Soc.* **28**(4), 709 (1983)]

Calculations have been made of the response of a large array of lead glass blocks to thermal antineutron annihilation events and to cosmic-ray background events. This array is to be used as the central detector element in a neutron-antineutron oscillation experiment. Studies of the effect of various threshold level settings, detector energy resolution, and necessary level of segmentation for multiplicity information have been made. Cuts allowing efficient cosmic-ray muon rejection were studied and will be discussed.

*Research sponsored by U.S. DOE Office of High Energy and Nuclear Physics.
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4.11

NUCLEON-MESON TRANSPORT CAPABILITY FOR ACCELERATOR BREEDER TARGET DESIGN*

T. A. Gabriel R. G. Alsmiller, Jr.

(Abstract of a paper presented at the EPRI Conference on Accelerator Breeding, Palo Alto, CA, June 9-10, 1982)

A state-of-the-art code system for nucleon-meson-lepton transport which has direct applicability to accelerator breeders is presented. Some pertinent data that have been obtained using this system are discussed and compared with experimental data.

*Research sponsored by U.S. DOE Office of High Energy and Nuclear Physics.

4.12

SHIELDING CONSIDERATIONS FOR MULTI-GeV/NUCLEON HEAVY ION ACCELERATORS: THE INTRODUCTION OF A NEW HEAVY ION TRANSPORT CODE, HIT*

T. A. Gabriel B. L. Bishop[†]
R. A. Lillie

(Abstract of ORNL/TM-8952, January 1984)

A preliminary heavy ion transport code, HIT, has been developed and used to generate basic shielding information for multi-GeV/nucleon heavy ion accelerators. The data presented are the number of neutrons emitted with energies >100 MeV and the amount of energy carried off by these particles external to a central shielding block composed of iron. These data are compared to similar data generated by the high-energy hadronic transport code, HETC. These data will allow preliminary estimates for shielding of heavy ion accelerators based on current shielding requirements around proton accelerator facilities.

*Research sponsored by U.S. DOE Office of High Energy and Nuclear Physics.
†UCC-ND Computer Sciences Division.

4.13

INVESTIGATION OF BUILDUP DOSE FROM ELECTRON CONTAMINATION OF CLINICAL PHOTON BEAMS*

P. L. Petti[†] M. S. Goodman[†]
T. A. Gabriel R. Mohan[‡]

[Abstract of *Med. Phys.* 10(1), 18 (1983); also paper presented at the 24th Meeting of the American Association of Physicists in Medicine, New Orleans, LA, August 1-5, 1982]

The contribution made by contaminating electrons present in a clinical photon beam to the buildup dose in a polystyrene phantom has been calculated and compared to measurements. A Monte Carlo technique was employed. The calculation was divided into two parts. First, the accelerator treatment head was simulated in detail using the EGS-PEGS electromagnetic shower code. Then, information obtained from these calculations was used to compute dose curves in a polystyrene phantom. Two cases were considered, one in which both electrons and photons were incident upon the phantom, and another in which electrons were eliminated from the incident beam. Results of these calculations agree with recent experimental findings. A decrease in buildup dose as well as a shift in d_{max} was observed when electrons were eliminated from the beam.

*Research sponsored by U.S. DOE Office of High Energy and Nuclear Physics.

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4.14

THE EFFECTS AND SOURCES OF ELECTRON CONTAMINATION OF CLINICAL PHOTON BEAMS*

P. L. Petti[†] M. S. Goodman[†]
T. A. Gabriel R. Mohan[‡]

(Abstract of paper presented at the 1982 Nuclear Science Symposium, Washington, D.C., October 20-22, 1982)

The contribution made by contaminating electrons present in Varian's Clinac-35 photon beam to the buildup dose in a polystyrene phantom has been calculated in detail using the EGS-PEGS electromagnetic shower code. The results support experimental findings indicating that, for the Clinac-35, electrons are responsible for enhanced buildup doses. The sources of these contaminating electrons have also been investigated with the intention of devising a reasonable means for their elimination.

*Research sponsored by U.S. DOE Office of High Energy and Nuclear Physics.

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4.15

SOURCES OF ELECTRON CONTAMINATION FOR THE CLINAC-35 25-MeV PHOTON BEAM*

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J. M. Sisterson[†] P. J. Biggs[‡]
T. A. Gabriel R. Mohan[§]

(Abstract of *Med. Phys.*, in press)

A detailed Monte Carlo approach has been employed to investigate the sources of electron contamination for the 25-MV photon beam generated by Varian's Clinac-35. Three sources of contamination were examined, a) the flattening filter and beam monitor chamber, b) the fixed primary collimators downstream from the monitor chamber and the adjustable photon jaws, and c) the air volume separating the treatment head from the observation point. Five source-to-surface distances (SSD) were considered for a single field size. It was found that for small SSDs (80 cm -- 100 cm), the dominant sources of electron contamination were the flattening filter and the beam monitor chamber which accounted for 70% of the unwanted electrons. Thirteen percent of the remaining electrons originated in the downstream primary collimators and the photon jaws, and 17% were produced in air. At larger SSDs, the fraction of unwanted electrons originating in air increased. At 400 cm SSD, 61% of the contaminating electrons present in the beam were produced in air, 34% originated

in the flattening filter and beam monitor chamber, and 5% were due to interactions in the fixed collimators downstream from the monitor chamber and the adjustable photon jaws. These calculated results are substantiated by recent experiments.

*Research sponsored by U.S. DOE Office of High Energy and Nuclear Physics.

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Section 5

STUDIES OF NUCLEAR WEAPONS EFFECTS

5.0. INTRODUCTION

It was noted in our last progress report that the division's interest in studying the transport of radiation through the atmosphere had been revived because of the concern in recent years that the neutron doses received by victims at Hiroshima and Nagasaki had not been accurately defined. That concern led to a renewed emphasis within the U.S. and Japan in reassessing the doses. In the U.S. the program, called the U.S. Dose Reassessment Program, involves several national laboratories, consultant firms, and universities, as well as a committee from the National Academy of Science. Similarly, in Japan the program is being conducted under the auspices of the Radiation Effects Research Foundation and also involves several national laboratories and universities. Groups from the two nations meet regularly to exchange information on the progress of the reevaluation.

The division's participation in this program continues to center on calculations of the transport of neutrons and gamma rays in an air-over-ground environment, using the best calculational techniques and source descriptions available. A recently initiated extension of this work has as its goal the development of techniques for calculating the doses received by survivors of the blasts who were in concrete buildings relatively near the blast centers. In order to estimate these doses, a computer code is needed that can economically model three-dimensional geometries, which has prompted work within the division to develop a three-dimensional discrete ordinates code called TORT (Three-dimensional Oak Ridge Transport) for operation on machines using vectorized processors such as the CRAY computer.

A new dimension to the reassessment program will be provided by the newly initiated, comprehensive sensitivity/uncertainty analysis within the division. This systematic analysis will lead not only to improved estimates of survivor doses, but also to defensible and reduced uncertainties in these doses.

Division efforts in the reassessment program have also included measurements of the integral fluxes and energy spectra of the neutrons given off by a replica of the Little Boy (Hiroshima) bomb with a controlled source. In a related effort, calculations have been performed to define the radiation fields in which experimental animals have been exposed at the Armed Forces Radiobiology Research Institute (AFRRI). And in another related effort, work is being initiated to more accurately define the response to radiation of the detectors (ionization chambers) used by health physicists. If successful, this work should lead not only to greater confidence in the reported measurements, but also to improved designs of the instruments themselves.

Both the reassessment program and our participation in it are expected to continue for some time.

5.1

TRANSPORT IN AN AIR-OVER-GROUND ENVIRONMENT OF PROMPT NEUTRONS AND GAMMAS FROM THE HIROSHIMA AND NAGASAKI WEAPONS*

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

[Abstract of paper presented at Symposium on Reevaluations of Dosimetric Factors [for] Hiroshima and Nagasaki, Germantown, Maryland, September 15-16, 1981; also Proc. CONF-810928 (DE81026279), p. 131 (1982)]

Much of the work on radiation shielding in the last two decades has been aimed at developing adequate data on transport methods and cross sections to describe the numerous prompt-neutron and the prompt and secondary gamma-ray interactions through the various materials. When adequate experimental data are available, the calculational results can be benchmarked. In the absence of such test data, however, one must rely on results obtained from the particle-transport calculations. The two most accurate methods for these calculations are the discrete-ordinates S_n method and the Monte Carlo method.

This paper is concerned with the application of the S_n method for approximating a solution to the Boltzmann transport equation in an air-over-ground two-dimensional, cylindrical geometry as applied to the Hiroshima and Nagasaki environments. The calculational sequence used to determine any response that depends on the transported particle fluence is as follows:

> Determination of the proper energy and angular source distribution, the air and ground cross sections to be used, and the proper material compositions.
>
> Determination of the suitable response functions, which may require adjoint transport calculations, inputting data to the appropriate transport codes, mitigating any ray effects, and calculating the desired free-field dose responses. For complicated geometries, coupled transport calculations may be required.

Additional tasks that should be completed before acceptance of any reevaluation of dosimetric effects include the best possible air-over-ground transport calculations using the most reliable weapon source data together with a more realistic ground composition; an air-over-ground sensitivity analysis to indicate the relative importance of source description components; cross sections for individual elements as a function of energy and reaction type, source height, and ground range; and transport calculations and comparisons of several Nevada Test Site weapons shots and the Ichiban, BREN, and Burlington AEC Plant bomb experiments for benchmark purposes.

*Research sponsored by Defense Nuclear Agency.
†UCC-ND Computer Sciences Division.

5.2

TISSUE KERMA VS DISTANCE RELATIONSHIPS FOR INITIAL NUCLEAR RADIATION FROM THE ATOMIC DEVICES DETONATED OVER HIROSHIMA AND NAGASAKI*

G. D. Kerr† J. V. Pace, III‡
W. H. Scott, Jr.§

[Abstract of ORNL/TM-8727, June 1983; also paper presented at the U.S.—Japan Joint Workshop for Reassessment of Atomic Bomb Radiation Dosimetry in Hiroshima and Nagasaki, Nagasaki, Japan, February 16-17, 1983; Proc. p. 57 (1983)]

Initial nuclear radiation is comprised of prompt neutrons and prompt primary gammas from an exploding nuclear device, prompt secondary gammas produced by neutron interactions in the environment, and delayed neutrons and delayed fission-product gammas from the fireball formed after the nuclear device explodes. These various components must all be considered in establishing tissue kerma vs distance relationships which describe the decrease of initial nuclear radiation with distance in Hiroshima and in Nagasaki.

An interest in initial nuclear radiation at distances of as much as two kilometers demands the economical use of discrete ordinates transport (DOT) techniques. The two-dimensional DOT-IV code developed at the Oak Ridge National Laboratory was used to calculate the tissue kerma in an air-over-ground geometry from prompt neutrons and prompt primary gammas and from prompt secondary gammas produced in air and ground. Data from the Los Alamos National Laboratory were used as the source terms.

The tissue kerma at ground level from delayed fission-product gammas and delayed neutrons was investigated using the NUIDEA code developed by Science Applications, Inc. This code incorporates very detailed models which can take into account such features as the rise of the fireball, the rapid radioactive decay of fission products in it, and the perturbation of the atmosphere by the explosion.

Tissue kerma vs distance relationships obtained by summing results of these current state-of-the-art calculations will be discussed. Our results clearly show that the prompt secondary gammas and delayed fission-product gammas are the dominant components of total tissue kerma from initial nuclear radiation in the cases of the atomic (or pure-fission) devices detonated over Hiroshima and Nagasaki.

*Research sponsored by Defense Nuclear Agency.
†Health and Safety Research Division.
‡UCC-ND Computer Sciences Division.
§Science Applications, Inc., La Jolla, CA.

5.3

INTEGRAL MEASUREMENTS OF NEUTRON AND GAMMA-RAY LEAKAGE FLUXES FROM THE LITTLE BOY REPLICA*

F. J. Muckenthaler

(Abstract of ORNL/TM-9005, in press)

This report presents integral measurements of neutron and gamma-ray leakage fluxes from a critical mockup of the Hiroshima bomb Little Boy at Los Alamos National Laboratory with detector systems developed by Oak Ridge National Laboratory. Bonner ball detectors were used to map the neutron fluxes in the horizontal midplane at various distances from the mockup and for selected polar angles keeping the source-detector separation constant. Gamma-ray energy deposition measurements were made with thermoluminescent detectors at several locations on the iron shell of the source mockup. The measurements were performed as part of a larger program to provide benchmark data for testing the methods used to calculate the radiation released from the Little Boy bomb over Hiroshima.

*Research sponsored by U.S. DOE Office of Health and Environmental Research.

5.4

CALCULATIONS OF RADIATION FIELDS AND MONKEY MID-HEAD AND MID-THORAX RESPONSES IN AFRRI-TRIGA REACTOR FACILITY EXPERIMENTS*

J. O. Johnson† M. B. Emmett‡
J. V. Pace, III‡

(Abstract of ORNL/TM-8807, July 1983)

A computational study was performed to characterize the radiation exposure fields and the mid-head and mid-thorax response functions for monkeys irradiated in the Armed Forces Radiobiological Research Institute (AFRRI) reactor exposure facilities. Discrete ordinates radiation transport calculations were performed in one-dimensional spherical geometry to obtain the energy spectra of the neutrons and gamma rays entering the room through various spectrum modifiers and reaching the irradiation position. Adjoint calculations performed in two-dimensional cylindrical geometry yielded the mid-head and mid-thorax response functions, which were then folded with flux spectra to obtain the monkey mid-head and mid-thorax doses (kerma rates) received at the irradiation position. The results of the study are presented both as graphs and as tables. The resulting spectral shapes compared favorably with previous work; however, the magnitudes of the fluxes did not. The differences in the magnitudes may be due to the normalization factor used.

*Research sponsored by Defense Nuclear Agency.
†University of Tennessee, Knoxville, TN.
‡UCC-ND Computer Sciences Division.

Section 6

ENERGY ECONOMICS MODELING AND ANALYSIS

6.0. INTRODUCTION

The division's continuing program in the area of energy-economics modeling and analysis is now being performed under the auspices of the Department of Energy's Office of Fossil Energy (DOE/FE), which has a strategic goal of highest priority of reducing the country's vulnerability to the economic and security disruptions that would accompany a large reduction in the supply of petroleum and/or a sharp increase in its cost. The specific objectives of DOE/FE are to develop a knowledge base for alternative supply technologies that can provide new sources of liquid and gaseous fuels, including coal-derived liquids and gas. In support of these objectives, the division, together with two subcontractors, Decision Focus, Inc. and Lewin and Associates, Inc., is conducting a research program that has the following principal elements:

1. Developing computerized models for U.S. liquid and gaseous fuel supplies (LFS, the Liquid Fuels Supply model; and GFS, the Gaseous Fuels Supply model) to enable quantitative assessment of competitive supply technologies; and providing DOE/FE with operational support in the use of the models.
2. Developing an automated capability for LFS/GFS sensitivity and uncertainty analyses.
3. Developing a decision-making methodology based on sensitivity theory to examine critically the potential benefits of various R&D programs.
4. Planning for contingencies in the event of supply interruptions (supply vulnerability analysis).
5. Providing 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, unconventional gas, shale oil, synthetics, and conventional onshore and offshore production in meeting future U.S. needs for liquid and gaseous fuels.

6.1

EXISTENCE AND UNIQUENESS OF SOLUTIONS FROM THE LEAP EQUILIBRIUM ENERGY-ECONOMY MODEL*

E. M. Oblow

(Abstract of ORNL/TM-8177, October 1982)

A study was made of the existence and uniqueness of solutions to the long-range, energy-economy model LEAP. The code is a large-scale, long-range (50-year) equilibrium model of energy supply and demand in the U.S. economy used for government and industrial forecasting. The study focused on the two features which distinguish LEAP from other equilibrium models - the treatment of product allocation and basic conversion of materials into an energy end product. Both allocation and conversion processes are modeled in a behavioral fashion which differs from classical economic paradigms. The results of the study indicate that while LEAP contains desirable behavioral features, these same features can give rise to non-uniqueness in the solution of allocation and conversion process equations. Conditions under which existence and uniqueness of solutions might not occur are developed in detail and their impact in practical applications are discussed.

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

6.2

EVALUATION OF THE MATHEMATICAL AND ECONOMIC BASIS FOR CONVERSION PROCESSES IN THE LEAP ENERGY-ECONOMY MODEL*

E. M. Oblow

[Abstract of ORNL/TM-8178, October 1982; also *J. Applied Math. Modeling* 7, 405 (1983)]

An evaluation was made of the mathematical and economic basis for conversion processes in the LEAP energy-economy model. Conversion processes are the main modeling subunit in LEAP used to represent energy conversion industries and are supposedly based on the classical economic theory of the firm. The study arose out of questions about uniqueness and existence of LEAP solutions and their relation to classical equilibrium economic theory. An analysis of classical theory and LEAP model equations was made to determine their exact relationship. The conclusions drawn from this analysis were that LEAP theory is not consistent with the classical theory of the firm. Specifically, the capacity factor formalism used by LEAP does not support a classical interpretation in terms of a technological production function for energy conversion processes. The economic implications of this inconsistency are suboptimal process operation should be terminated. A new capacity factor formalism, which retains the behavioral features of the original model, is proposed to resolve these discrepancies.

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

6.3

THE APPLICATION OF ADJOINT SENSITIVITY THEORY TO A LIQUID FUELS SUPPLY MODEL*

R. G. Alsmiller, Jr. J. Barhen
J. E. Horwedel[†] J. L. Lucius[†]
J. D. Drischler

[Abstract of ORNL/TM-8850, September 1983; also *Energy* (*in press*)]

Adjoint sensitivity theory has been applied to a liquid fuels supply model to determine the sensitivity of calculated results to the parameters in the model. The power of the adjoint sensitivity methodology is that it provides, in principle, an efficient means of calculating the sensitivity, dR/dx, of any calculated result R to every parameter x in the model. The liquid fuel supply model considered is one of a class of models that may be constructed from the Generalized Equilibrium Modeling System (GEMS) developed by Decision Focus, Inc., Los Altos, California. It is

shown that for this class of models the lengthy development effort that is usually associated with applying the adjoint sensitivity methodology can be avoided by using the GEMS program to evaluate the large number of partial derivatives that are needed to apply the methodology.

*Research sponsored by U.S. DOE Office of Fossil Energy.
†UCC-ND Computer Sciences Division.

6.4

AN INVESTIGATION OF THE COMPONENTS OF DOMESTIC FUEL SUPPLY WITH EMPHASIS ON RESOURCE BASE TECHNOLOGY*

F. Morra† V. A. Kuuskraa†
R. G. Alsmiller, Jr. J. Barhen
J. R. Einstein C. R. Weisbin
D. M. Nesbitt‡

(Abstract of Invited Paper of Operation Research Society of America TIMMS Conference, Orlando, Florida, November 7-8, 1983)

This paper discusses the impacts of several factors on the future costs of domestic extracted oi., including:

- Uncertainty about the size of the undiscovered resources base
- Availability of frontier resources areas
- Degree of advancement of enhanced oil recovery (EOR) technology.

The study examines the marginal costs of extracting *success in future* increments of the undiscovered resource, thereby producing estimates of the price required to replace current production with new reserves. The United States is analyzed in terms of eight onshore regions, four offshore regions, and the contribution of EOR. Assuming that the lowest cost resource is produced first, an algorithm is used to project the costs of meeting a fixed production target (e.g., 8 million barrels per day) into the future. Although this assessment is based exclusively on domestic resources, it has implications for the world oil price in that the domestic replacement cost is an effective ceiling on the price of imports. The assessment also can help the reader in examining the likely penetration of synthetics into the market. The principal value of the assessment is to provide some quantitative perspective on alternative scenarios, such as:

- Increases in imports versus increase in domestic oil prices.
- The role of frontier resources or improved EOR technology in maintaining domestic production.
- The market lead time required for the initiation of synthetic fuel production.

*Research sponsored by U.S. DOE Office of Fossil Energy.
†Lewin and Associates, Inc., Washington, DC.
‡Decision Focus, Ir.., Palo Alto, CA.

6.5

LIQUID FUELS SUPPLY MODEL DATA BASE: UNCONVENTIONAL RECOVERY AND COAL LIQUEFACTION*

V. A. Kuuskraa† F. Morra†
R. G. Alsmiller, Jr. J. Barhen
J. E. Horwedel‡

(Abstract of ORNL/TM-8345, in press)

An overview of the unconventional recovery and coal liquefaction components of the Liquid Fuels Supply (LFS) model data base is presented. The LFS model, developed under the sponsorship of the Office of Planning and Environment (DOE/FE), enables the quantitative assessment of competitive supply technologies. Thus, it contributes 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. This report presents the data used in the LFS model for modeling enhanced oil recovery and oil production from shale, tar sands, and coal liquefaction.

*Research sponsored by U.S. DOE Office of Fossil Energy.
†Lewin and Associates, Inc., Washington, D.C.
‡UCC-ND Computer Sciences Division.

6.6 SCREENING SENSITIVITY THEORY*

E. M. Oblow F. G. Perey

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

A comprehensive rigorous theory is developed for screening sensitivity coefficients in large-scale modeling applications. The theory uses Bayesian inference and group theory to establish a probabilistic framework for solving an underdetermined system of linear equations. The underdetermined problem is directly related to statistical screening sensitivity theory as developed in recent years. Several examples of the new approach to screening are worked out in detail and comparisons are made with statistical approaches to the problem. The drawbacks of these latter methods are discussed at some length.

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

Section 7

ANALYSES OF CO_2 IMPACT ON CLIMATE

7.0. INTRODUCTION

The division's CO_2 research program, which was introduced in our last progress report by a paper on the application of the adjoint method of sensitivity analysis to a radiative-convective model, has now been expanded into one of our major areas of research. While sensitivity analysis is still being emphasized within the program, new areas of research in climatic effects are also being addressed.

One of the major questions in CO_2 research is how idealizations in climate models could affect the predictions of CO_2 induced climate change. To help answer this question, current research aims at demonstrating the adjoint method of sensitivity analysis to the Oregon State University (OSU) atmospheric general circulation model (GCM). To obtain sensitivities for such a model by the conventional method of rerunning is prohibitively expensive.

Several journal articles now document the efficiency and accuracy of the adjoint method for a radiative convective climate model (RCM). Although the radiation calculations for this RCM and the OSU GCM are the same, the horizontal resolution of the GCM requires more complicated numerical procedures to implement the adjoint method. In particular, the spatial and temporal resolution of the adjoint calculations must be chosen to obtain a balance between efficiency and accuracy.

A successful demonstration of the adjoint method for the OSU GCM will provide the basis for application of the method to other GCM's, including updated versions of the OSU GCM. Detailed sensitivity information for GCM's not only will indicate the potential effects of model idealizations on the predicted CO_2 climate change, but also will provide a means of accounting for the different predictions given by different models. Furthermore, the sensitivity information will alert the modeling community to the most important measurements or parameterization improvements that need to be made to reduce uncertainties in model results.

7.1

APPLICATION OF SASGRAPH IN CARBON DIOXIDE AND CLIMATE INFORMATION ANALYSIS RESEARCH*

J. A. Watts† J. D. Drischler
W. E. Ford,III‡ J. E. Horwedel‡

(Abstract of paper presented at the SAS Users' Group International Conference, New Orleans, LA, January 16-19, 1983)

The Carbon Dioxide Information Center (CDIC) has been established by the Department of Energy Carbon Dioxide Research Division to develop an "evaluated" data base of carbon dioxide and climatic data files for preparation of state-of-the-art assessment reports and a statement-of-findings report in 1985. Data are being inventoried and placed in CDIC files on the carbon cycle, climate and climatic change, vegetation effects, first detection, and indirect effects. Examples of these data include fossil fuel production and CO_2 emissions, vegetation and land use change, long-term climatological records, physical and chemical oceanographic records, sea surface temperatures, tree ring chronologies, crop yields, volcanic chronologies, etc.

The Statistical Analysis System (SAS) is being used to edit, graphically display, and format CDIC data files for on-line access. Graphic display is an especially useful tool both for preliminary analyses and as a mechanism for showing complex information in an easy to access and grasp manner.

SASGRAPH has been applied extensively to numeric data pertinent to the carbon dioxide issues by the CIDC staff. In this paper, we present several examples of the application of SASGRAPH to the complex CO_2 issue and demonstrate the value of graphical display in effectively presenting large volumes of data in a concise, easy-to-use format. In particular, long-term temperature trends and anomalies are contrasted on both spatial and temporal scales to illustrate the regional vs. global aspects of temperature changes and related implications on land use and vegetation changes.

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

†Environmental Sciences Division.
‡UCC-ND Computer Sciences Division.

7.2

PHYSICAL INTERPRETATION OF THE ADJOINT FUNCTIONS FOR SENSITIVITY ANALYSIS OF ATMOSPHERIC MODELS*

M. C. G. Hall D. G. Cacuci

[Abstract of *Journal of Atmospheric Sciences* 40(10), 2537 (1983)]

The adjoint functions for an atmospheric model are the solution to a system of equations derived from a differential form of the model's equations. The adjoint functions can be used to calculate efficiently the sensitivity of one of the model's results to variations in any of the model's parameters. This paper shows that the adjoint functions themselves can be interpreted as the sensitivity of a result to instantaneous perturbations of the model's dependent variables. This interpretation is illustrated for a radiative convective model, although the interpretation holds equally well for general circulation models. The adjoint functions are used to reveal the three time scales associated with (1) convective adjustment, (2) heat transfer between the atmosphere and space, and (3) heat transfer between the ground and atmosphere. Calculating the eigenvalues and eigenvectors of the matrix of derivatives occurring in the set of adjoint equations reveals similar physical information without actually solving for the adjoint functions.

The sensitivities given by the adjoint functions are verified by comparison with sensitivities obtained directly from recalculations. Despite sharp changes in the adjoint functions arising from convective adjustment switching on and off during a diurnal cycle, a first order numerical scheme to solve the adjoint equations gives agreement with direct recalculations to three significant figures. For a model with N time steps, this comparison has also shown that the adjoint method is at least $N/2$ times more efficient than recalculation. Such an efficient method of calculating the sensitivity of a simulated synoptic state to all previous synoptic states is valuable not only to identify the time scales of various physical processes, but also to assimilate data for initialization.

*Research sponsored by U.S. DOE Division of Carbon Dioxide Research and Office of Basic Energy Sciences.

7.3 SYSTEMATIC ANALYSIS OF CLIMATIC MODEL SENSITIVITY TO PARAMETERS AND PROCESSES*

M. C. G. Hall D. G. Cacuci

[Abstract of chapter in *Climate Processes and Climate Sensitivity*, Maurice Ewing Series V, J. E. Hansen and T. Takahashi, Eds., American Geophysical Union, Washington, D.C. (in press)]

This paper demonstrates the adjoint method of sensitivity analysis on a radiative-convective climate model with a diurnal cycle. A single adjoint calculation, which requires about the same computation time as the original model, suffices to calculate sensitivities of average surface air temperature to all 312 model parameters. The sensitivities accurately predict the effect on average surface air temperature of small variations in the model parameters. Relative sensitivities rank the importance of all the parameters, the most important parameters being those that characterize solar radiation and determine transmission functions. The uncertainties in the model results are expressed formally in terms of all the sensitivities and parameter covariances. For this model, a 10% standard deviation in surface albedo causes the increase in average surface air temperature after doubling CO_2 to have a standard deviation of 0.24 K about the expected value of 1.66 K. For results that cannot readily be compared with observation (for example, the results of a CO_2 doubling experiment), this method of uncertainty analysis is the only systematic way to estimate the reliability of model results. This estimate of the reliability, though, will only be realistic if all important physical processes and feedback mechanisms have been included, however crudely, in the model.

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.

*Research sponsored by U.S. DOE Division of Carbon Dioxide Research.

Section 8

INTELLIGENT CONTROL SYSTEM RESEARCH

8.0. INTRODUCTION

A Center for Engineering Systems Advanced Research (CESAR) has recently been established within the division to address long-range, energy-related research in intelligent control systems. These systems are intended to plan and perform a variety of tasks in unstructured environments, given only qualitatively specified goals. Building upon extensive experience in remote operations and human engineering, the Center will provide a framework for merging concepts from the fields of artificial and machine intelligence with advanced control theory. Emphasis is on interdisciplinary research for large-scale distributed processes applicable to many energy-related technologies. Research objectives stress the optimization of energy efficiency and the minimization of associated risks in its production and utilization. Potential applications include emergency situations, remote operations, resource exploration, transportation systems, and large-scale power generation systems.

CESAR is intended to be a national resource, and a major objective will be to disseminate its accomplishments widely and comprehensively. Accordingly, its results and technology will be distributed through refereed journal publications, through the organization of specialists' workshops, and through the development of products which demonstrate concepts. CESAR will cooperate with universities, laboratories, and industry, serving as a user facility to provide guests with access to modern computers, unique equipment, and a stimulating scientific environment.

8.1

REAL-TIME ALGORITHMS FOR ROBOTIC CONTROL OF TELEOPERATORS*

S. M. Babcock† J. Barhen

(Abstract of paper to be presented at Robots-8 Conference, Detroit, Michigan, June 4-7, 1984)

The Department of Energy recently established a Center for Engineering Systems Advanced Research (CESAR) at the Oak Ridge National Laboratory (ORNL). The Center's charter is to address long-range, energy-related research in intelligent control systems operating in unstructured environments. The purpose of this paper is to report initial results in the development of real-time algorithms, incorporating inverse dynamics, for robotic control of a six-degree-of-freedom manipulator with back-drivable actuation designed for teleoperation. The inverse dynamics equations are formulated with the aid of the DINAH computer code under development at ORNL; this code accepts specification for arbitrary robot configurations and generates symbolically reduced equations of motion. The present research is an integral part of a large systems integration effort with complementary tasks in robot strategy planning, sensor fusion, etc.

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

†Instrumentation and Controls Division.

8.2

BASIC RESEARCH ON INTELLIGENT ROBOTIC SYSTEMS OPERATING IN HOSTILE ENVIRONMENTS: NEW DEVELOPMENTS AT ORNL*

J. Barhen S. M. Babcock†
W. R. Hamel† E. M. Oblow
G. N. Saridis‡ G. de Saussure
A. D. Solomon§ C. R. Weisbin

(Abstract of paper to be presented at the ANS/RSTD National Topical Meeting on Robotics and Remote Handling in Hostile Environments, Gatlinburg, Tennessee, April 23-26, 1984)

Robotics and artificial intelligence research carried out within the Center for Engineering Systems Advanced Research (CESAR) is presented. Activities focus on the development and demonstration of a comprehensive methodological framework for intelligent machines operating in unstructured hostile environments. Areas currently being addressed include mathematical modeling of robot dynamics, real-time control, "world" modeling, machine perception and strategy planning.

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

†Instrumentation and Controls Division.

‡Rensselaer Polytechnic Institute, Troy, NY.

§UCC-ND Computer Sciences Division.

8.3

PARALLEL ALGORITHMS FOR ROBOT DYNAMICS*

J. Barhen S. M. Babcock†

(Abstract of paper to be presented at The First World Conference on Robotics Research, Robotics Research: The Next Five Years and Beyond, Lehigh U., PA, August 14-16, 1984)

The Department of Energy recently established a Center for Engineering Systems Advanced Research (CESAR) at the Oak Ridge National Laboratory (ORNL). The Center's charter is to address long-range, energy-related research in intelligent control systems. The purpose of this paper is to report our initial results in developing parallel algorithms for efficiency enhancement in real-time solutions of manipulator dynamics equations. The analytic formulation of such algorithms is carried out with the help of a computer code (DINAH) which is under development at ORNL and uses the powerful symbolic (and numeric) processing capabilities of the VS-FORTRAN language recently released by IBM. This research is an integral part of a large systems integration effort with complementary tasks in strategy planning, sensor fusion, etc.

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

†Instrumentation and Controls Division.

8.4

STRATEGY PLANNING BY AN INTELLIGENT MACHINE*

C. R. Weisbin G. de Saussure
J. Barhen

(Abstract of paper to be presented at The First World Conference on Robotics Research, Robotics Research: The Next Five Years and Beyond, Lehigh U., Pa., August 14-16, 1984)

The Center for Engineering Systems Advanced Research (CESAR) has recently been established at the Oak Ridge National Laboratory (ORNL) to address long-range, energy-related research in intelligent control systems. The purpose of this paper is to describe our initial investigations in developing efficient strategies to be used by an autonomous robot with hierarchical control structure operating in an unstructured real world environment. This research is an integral part of a system with complementary efforts in mathematical modeling, real-time control, sensor integration, etc. Our current work builds upon traditional areas of strength in artificial intelligence (i.e., knowledge representation, inferential reasoning, and parallel planning) through the application requirements of real-time decision making and on-line learning.

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

Section 9

INFORMATION ANALYSIS AND DISTRIBUTION

9.0. INTRODUCTION

Information analysis and distribution, an important activity of the division ever since the Radiation Shielding Information Center (RSIC) was established in FY 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 information 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 computing technology.

During this reporting period, the division hosted the newly established Carbon Dioxide Information Center (CDIC), and gave leadership, technical guidance and other support as needed. EPIC staff members continue to give technical support and consultation to CDIC as needed in FY 1984. CDIC collects, organizes, evaluates, packages and disseminates literature, numeric data and computing technology (algorithms, codes, models) as required for a fully integrated information analysis center program needed in support of a global assessment of the CO_2 problem. The center is sponsored by the Carbon Dioxide Research Division of DOE's Office of Basic Energy Sciences. It is expected that CDIC, which is administered by the ORNL Information Division, will be moved to another site during this fiscal year.

RSIC and TDMC are described in greater detail as follows.

Radiation Shielding Information Center (RSIC). During this reporting period, RSIC observed it's 20th year in operation. It 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-based techniques for solving radiation transport or nuclear data problems; 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 appropriate areas of need.

During this reporting period work has continued in the development of fine- and broad-group cross-section data libraries. The 174 neutron, 38 gamma-ray group VITAMIN-E general-purpose cross-section data library based on ENDF/B-V now contains 70 materials. It has been used with success for fusion neutronics, fast reactor core and shielding applications, and various other radiation transport calculations within the division. The generation of multigroup data for some of ENDV/B-V evaluations, Revision 2, has also been completed and the resulting data will be released as part of VITAMIN-E.

In addition, the ENDF/B photon interaction data have been updated 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 the RSIC director. Other staff members are actively involved with the American Nuclear Society in developing radiation protection and shielding standards and standards for scientific computer programming and its documentation.

In the area of computing technology, RSIC now makes available 655 computer code packages and 107 data packages. The RSIC Newsletter is currently mailed monthly to 1680 people. During FY 1983, more than 3213 letters/telephone calls of requests were processed by RSIC, resulting in 9857 separate activities required to satisfy the requests. A total of 61 persons came for an orientation visit and/or to use the center's facilities.

The RSIC computer-based information system (SARIS), containing more than 13,286 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 statistical record system (ADES), operable on the EPIC minicomputer (ECLIPSE) and used to compile information about the profile of the RSIC user community, was maintained and modified for more versatility.

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 technical 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 enchancement, 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.

Also treated during the report period were five code systems (PAVAN, METD, XOQDOQ, SPIRT, and TACT) for studies of atmospheric dispersion, meteorological evaluation, and data processing tasks related to routine effluent releases at nuclear power stations. The finished packages are disseminated via RSIC.

Subtask B has focused on computing technology in the areas of radiological assessment, meteorological (atmospheric) and other environmental transport. The documentation and packaging of IBM and ECLIPSE MV/8000 versions of the LPGS code system for calculating radiation exposures from accidental releases was completed during the reporting period. Work on an update to the associated data library (GENDOS) and the data management utility RADU continues. The subtask was enlarged to include the implementation of specific code systems on the Data General ECLIPSE MV/8000 at NRC Bethesda offices with responsibility to maintain a one-to-one correspondence between that in the RSIC collection and those available on the NRC inhouse system, e.g., LPGS, LADTAP-II, PAVAN and TACT5. The work included bringing the code, in each case, into conformance with the new FORTRAN 77 language standard.

Subtask B will again be enlarged in FY 1984 to include documentation and maintenance of the computing technology of the NRC Accident Evaluation Branch, e.g., CRAC, CRAC2, CORSOR, TACT5, TRAC, TRAPMELT, and TRAPQUIK.

Subtask C includes calculations to generate data for the proposed standard on Gamma-Ray Attenuation Coefficients and Buildup Factors for Engineering Materials produced in collaboration with the ANS-6 Standards Subcommittee. 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 method buildup factor data for water, iron, and lead. Results of ASFIT and Monte Carlo calculations have been compared with the moments data. Various features have been investigated and revisions have been made where indicated. Additional Monte Carlo calculations are needed where moments data were found to be in error. 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 was published in *Nuclear Science and Engineering*. The behavior of the dose rate at a shield-tissue interface has also been investigated and tables of the finite medium correction factors to the infinite-medium data and tables to correct for bremsstrahlung are being prepared. Fitting function parameters are being obtained. 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. The first draft of the text for the proposed standard has been prepared and is being reviewed by the working group. Work on this project is proceeding cooperatively in the U.S. at Oak Ridge, Los Alamos, and GA Technologies and in India and Japan.

Subtask D — A user's manual for LPGS was prepared and published as ORNL/TDMC-2 (NUREG/CR-2974). Others are in progress.

9.1

RSIC AFTER 20 YEARS — A LOOK BACK AND A LOOK AHEAD*

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

[Abstract of paper presented at the Sixth International Conference on Radiation Shielding, Tokyo, Japan, May 16-20, 1983; Proc. Vol. II, p. 1272 (1983)]

On the occasion of RSIC's 20th anniversary year, this review includes highlights and lessons learned. In June 1963, the first RSIC Newsletter was published, and information analysis procedures and practices were initiated. Evidence indicates that RSIC served for 20 years as the focal point for the exchange and transfer of radiation transport technology and contributed to the advancement of the state of the art. The original concept is found to be sound: operate an information analysis center by collecting, organizing, evaluating, and analyzing all relevant information and making the information available in a form readily useful to scientists and engineers. Computing technology, a computer-based literature information system, and an advisory service remain important elements of the center. Continuing interaction between the center, developers, and users of information products and services has been a key to RSIC success. A look to the future reflects optimism.

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

9.2

BIBLIOGRAPHY, SUBJECT INDEX, AND AUTHOR INDEX OF THE LITERATURE EXAMINED BY THE RADIATION SHIELDING INFORMATION CENTER*

D. K. Trubey R. W. Roussin
A. B. Gustin

(Abstract of ORNL/RSIC-5/V7, August 1983)

An indexed bibliography of literature selected by the Radiation Shielding Information Center since the previous volume was published in 1980 is presented in the area of radiation transport and shielding against radiation from nuclear reactors (fission and fusion), x-ray machines, radioisotopes, nuclear weapons (including fallout), and low energy accelerators (e.g., neutron generators). The bibliography was typeset from computer files constituting the RSIC Storage and Retrieval Information System. In addition to lists of literature titles by subject categories (accessions 6201-10156), an author index is given.

Most of the literature selected for Vol. VII was published in the years 1977-1981.

*Research sponsored by U.S. Department of Energy, U.S. Nuclear Regulatory Commission, and the Defense Nuclear Agency.

9.3

AVAILABLE COMPUTER CODES AND DATA FOR RADIATION TRANSPORT ANALYSIS*

B. L. McGill D. K. Trubey
B. F. Maskewitz R. W. Roussin

(Abstract of paper presented at the American Nuclear Society Topical Meeting Poster Session on Advances in Reactor Computations, Salt Lake City, Utah, March 28-31, 1983)

The Radiation Shielding Information Center (RSIC) is a technical institute serving the international shielding community. It acquires, evaluates, packages, and disseminates radiation protection, radiation transport and shielding information. The major activities include: (1) answering technical inquiries, (2) operating a computer-based bibliographic information system, and (3) collecting, testing, packaging, and distributing computer code systems and evaluated and processed data libraries. The data packages include multigroup coupled neutron-gamma-ray cross sections and kerma coefficients, other nuclear data, and radiation transport benchmark problem results.

As an integral part of its information collection and processing activities, RSIC collects, makes operable, tests, packages, and disseminates computer code packages to nuclear scientists and engineers engaged in radiation transport research or analysis. The various codes are designed for calculations related to radiation from fission and

fusion reactors, radioisotopes, weapons and accelerators and to radiation occurring in space.

The radiation treated by the majority of the code systems is either neutron or gamma radiation or both, but some codes treat charged particles. The types of geometry treated vary widely, with many codes allowing a general three-dimensional geometry.

The numerical methods applied in the radiation transport computing technology are primarily discrete ordinates (1 and 2 dimensions), Monte Carlo (up to 3 dimensions), and kernel integration (generally 3 dimensions). The remaining methods include integral transport, removal-diffusion (Spinney), moments method, spherical harmonics, and invariant embedding. Miscellaneous computer codes are also available to calculate such things as fission-product buildup, release and dose; charged particle stopping power; optimization of shield design parameters; and sensitivity of computed results to input parameters.

In addition, RSIC is involved in many data activities with emphasis being placed on nuclear cross section data. Through cooperation with various agencies, RSIC assists in improving the adequacy of basic evaluated and processed cross-section data and packages and distributes various types of data libraries useful in radiation transport analysis.

*Research sponsored by U.S. Department of Energy, Defense Nuclear Agency, and the U.S. Nuclear Regulatory Commission.

9.4

THE STATUS OF MULTIGROUP CROSS-SECTION DATA FOR SHIELDING APPLICATIONS*

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

[Abstract of paper presented at the Sixth International Conference on Radiation Shielding, Tokyo, Japan, May 16-20, 1983; Proc. Vol. I, p. 89 (1983)]

Multigroup cross-section libraries for shielding applications in formats for direct use in discrete ordinates or Monte Carlo codes have long been a part of the Data Library Collection (DLC) of the Radiation Shielding Information Center (RSIC). In recent years libraries in more flexible and comprehensive formats, which allow the user to derive his own problem-dependent sets, have been added to the collection. The current status of both types is described, as well as projections for adding data libraries based on ENDF/B-V.

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

9.5

DESCRIPTION OF THE DLC-99/HUGO PACKAGE OF PHOTON INTERACTION DATA IN ENDF/B-V FORMAT*

R. W. Roussin J. R. Knight[†]
J. H. Hubbell[‡] R. J. Howerton[§]

[Abstract of ORNL/RSIC-46 (ENDF-335), December 1983]

A new photon interaction data library, DLC-99/HUGO, is described. The library was prepared by incorporating newly evaluated data from the National Bureau of Standards with that from an existing data library, DLC-7F/HPICE, which is the ENDF/B-IV photon interaction data. It contains pair and triplet cross sections, photoelectric cross sections, and atomic form factors and the corresponding coherent scattering cross sections. Evaluated data in ENDF/B-V format are provided for elements Z=1 to 100. The data package, available from the Radiation Shielding Information Center (RSIC) at Oak Ridge National Laboratory, will be submitted to CSEWG for consideration as the ENDF/B-V Photon Interaction Library. Two computer codes, EDPHOT for selectively printing the data and COMP23 for comparing two photon interaction libraries, are also provided.

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

[†]UCC-ND Computer Sciences Division.
[‡]National Bureau of Standards, Washington, D.C.
[§]Lawrence Livermore National Laboratory, Livermore, CA.

9.6

STRUCTURAL, HEAVY COOLANT, AND SHIELDING MATERIAL CROSS-SECTION DATA*

R. W. Roussin

(Abstract of chapter in *Cross-Section Data for Nuclear Reactor Analyses*, Progress in Nuclear Energy series, Pergamon Press, Oxford, in press)

A review is presented of the cross-section data needs for radiation transport problems involving structural, heavy coolant, and shielding materials. Basic evaluated data needs are met through the ENDF/B Evaluated Cross-Section Library which is distributed from the National Nuclear Data Center at Brookhaven National Laboratory. The Cross Section Evaluation Working Group (CSEWG) develops the formats and procedures and evaluates and tests ENDF/B. The CSEWG Shielding Subcommittee has been important in helping assure that ENDF/B provides adequate data for radiation transport applications involving the subject materials. The evaluation and data-testing activities of CSEWG are discussed and the status of the evaluated data in ENDF/B-V is presented. The analyst performing calculations requires cross-section libraries that can be used directly in radiation transport computer codes; that is, the ENDF/B data must first be processed into forms consistent with transport code requirements. The current status of such processed data libraries, which can be obtained from the Radiation Shielding Information Center at Oak Ridge National Laboratory, is also reported.

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

9.7

STANDARD REFERENCE DATA FOR GAMMA-RAY TRANSPORT IN HOMOGENEOUS MEDIA*

D. K. Trubey

[Abstract of paper presented at the Sixth International Conference on Radiation Shielding, Tokyo, Japan, May 16-20, 1983; Proc. Vol. II, p. 1241 (1983)]

An American Nuclear Society Standards Committee Working Group, identified as ANS-6.4.3, is developing a set of evaluated gamma-ray isotropic point-source buildup factors and attenuation coefficients for a standard reference data base. As a first step, a largely unpublished set of buildup factors calculated with the moments method is being evaluated by recalculating key values with Monte Carlo, integral transport, and discrete ordinates methods. Attention is being given to frequently neglected processes such as bremsstrahlung and the effect of introducing a tissue phantom behind the shield. The proposed standard will contain data for a source energy range from 15 keV to 15 MeV and for approximately 12 elements and 3 mixtures (water, air, and concrete).

*Research sponsored by U.S. Nuclear Regulatory Commission and the U.S. Department of Energy.

9.8

USER'S MANUAL FOR LPGS: A COMPUTER PROGRAM FOR CALCULATING RADIATION EXPOSURE RESULTING FROM ACCIDENTAL RADIOACTIVE RELEASES TO THE HYDROSPHERE*

J. E. White K. F. Eckerman[†]

[Abstract of ORNL/TDMC-2 (NUREG/CR-2974), March 1983]

The LPGS computer program was developed to calculate the radiological impacts resulting from radioactive releases to the hydrosphere. The hydrosphere is represented by the following types of water bodies: estuary, small river, well, lake, and one-dimensional (1-D) river. The program is principally designed to calculate radiation dose (individual and population) to body organs as a function of time for the various exposure pathways. The radiological consequences to the aquatic biota is estimated. Several simplified radionuclide transport models are employed with built-in formulations to describe the release rate of the radionuclides. Optionally, a tabulated user-supplied release model can be input. Printer plots of dose versus time for the various exposure pathways are provided.

*Research sponsored by U.S. Nuclear Regulatory Commission Office of Administration [†]Health and Safety Research Division.

APPENDICES

SCIENTIFIC AND PROFESSIONAL ACTIVITIES

June 1, 1982 — December 31, 1983

L. S. Abbott

Editor, *The Acorn*, newsletter published by Oak Ridge/Knoxville Section of the American Nuclear Society
Associate Editor, *Risk Newsletter*, published by Society for Risk Analysis
Designer, American Nuclear Society Exhibit for 1984 World's Fair Exhibit
Member, Public Information Committee, American Nuclear Society, 1983-84

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
Invited lectures, (1) "Alternatives to Breeder Reactors for the Conversion of Fertile to Fissile Fuel," Physics Dept., California Institute of Technology, March 1983; (2) "Nucleon-Meson Transport Capability for Accelerator Breeder Target Design," with T. A. Gabriel, EPRI Conference on Accelerator Breeding, Palo Alto, CA, June 1982
Referee, *Nuclear Science and Engineering*
Referee, *Nuclear Technology Fusion*
Referee, *Operations Research*
Referee, *Radiation Research*
Referee, *Physics in Medicine and Biology*

J. Barhen

Senior member, Robotics International
Member, Applied Machine Vision, Robotics International
Bylaws Chairman, Oak Ridge/Knoxville Section of Robotics International
Invited paper, "An Investigation of the Components of Domestic Fuel Supply with Emphasis on Resource Base and Technology," with F. Morra *et al.*, Operations Research Society of America, ORSA-TIMMS Conference, Orlando, Florida, November 7-9, 1983
Referee, National Science Foundation
Referee, DOE/Office of Basic Energy Sciences

D. E. Bartine

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, Election Inspection Committee for Oak Ridge/Knoxville Section of American Nuclear Society (1982)
Member, Public Policy Committee, American Nuclear Society
Member, Professional Divisions Committee, American Nuclear Society

Member, Reactor Physics Division Executive Committee, American Nuclear Society

Organizer and Chairman, ANS/HPS New Orleans Miniplenary Coordination Committee

Member, Executive Committee, Reactor Physics Division, American Nuclear Society

Chairman, Radiation Protection and Shielding Division, Americ n Nuclear Society, 1982-83

Member, Johns Hopkins Medical Institute Committee to Develop CARER

Member, UCC-ND *Ad Hoc* Committee for Minicomputer and Microcomputer Acquisition and Use

Technical Coordinator, US/Japanese LMFBR Exchange Agreement (1982-83)

Advisor, Nuclear Engineering Department, University of Missouri at Rolla, Evaluation of Long-Term Facility Requirements

Who's Who in Frontier Science and Technology

Certificate of Governance awarded by ANS Board of Directors (1983)

Invited paper, "Technology Status of Candidate Shielding Materials," 1st Symposium on Space Nuclear Power Systems, Albuquerque, New Mexico, January 1984

Invited presentation, "Highlights of Shielding Analysis and Integral Experiments Performed for the U.S. Fast Flux Test Facility," U.S. DOE/Japan PNC Bilateral Exchange Agreement for Shielding Data, Tokyo, Japan, May 1983

Invited paper, "Space Reactor Shielding: An Assessment of the Technology," Symposium on *Advanced Compact Reactor Systems, National Academy of Sciences, Washington,* D.C., November 1982

Invited paper, "Background of Dose Estimates for Hiroshima and Nagasaki," ANS Annual Meeting, Los Angeles, California, June 1982

Invited paper, "The Status of Reactor Shielding Research in the United States," 6th International Conference on Radiation Shielding, Tokyo, Japan, May 1983

T. J. Burns

Member, UCC-ND *Ad Hoc* Committee on Minicomputer and Microcomputer Acquisition and Use

Participant, IEA Workshop on Swedish (ASEA-ATOM) Reactor Proposal (PIUS), June 14-15, 1982

Patent, "Gamma Thermometry Based Reactor Core Liquid Level Detector," September 20, 1983 (U.S. Patent No. 4,406,011)

Invited lecturer, Workshop on Gamma Thermometers as Replacement In-Core Power Detectors for Light-Water Reactors, Technology for Energy Corporation, July 26-27, 1982

D. G. Cacuci

Secretary, National Planning Committee, American Nuclear Society

Associate Editor (Designate), *Nuclear Science and Engineering*, the Journal of the American Nuclear Society

Invited lecture, "The Adjoint Method of Sensitivity Analysis for Nonlinear Systems: Theory and Applications," Princeton University, 1983

Invited paper, "The Adjoint Method of Sensitivity Analysis Applied to Atmospheric Models," with M. C. G. Hall, National Center for Atmospheric Research, Boulder, Colorado, August 5, 1983

Invited paper, "Systematic Analysis of Climatic Model Sensitivity to Parameters and Processes," with M. C. G. Hall at 4th Maurice Ewing Symposium on Climate Processes: Sensitivity to Insultation and CO_2, Columbia University, October 25-27, 1982

Referee, *Nuclear Science and Engineering*

Referee, *Journal of Mathematical Physics*

Referee, *Journal of Computational Physics*

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)

J. W. T. Dabbs

Member, Editorial Board, *Nuclear Instruments and Methods in Physics Research*
Member, ORNL Plant-wide Selection Committee for *Research and Development Magazine* IR-100 Award Entries
Member, ORNL Technology Transfer Selection Committee
Member, ORELA Management Committee
Referee, *Physical Review*

G. de Saussure

Member, Program Committee and Session Organizer, Topical Meeting on Reactor Physics and Shielding, American Nuclear Society, September 17-19, 1984
Member, General Program Advisory Committee, International Conference on Nuclear Data for Basic and Applied Sciences, May 13-17, 1985
Member, Technical Program Committee, Fifth Pacific Basin Nuclear Conference, American Nuclear Society, May 1985
Chairman, International Meeting Subcommittee of National Program Committee, American Nuclear Society (until June 1983)
Member, National Program Committee, American Nuclear Society (until June 1983)
Member, Steering Subcommittee of National Program Committee, American Nuclear Society (until June 1983)
Honorary Professor, Nuclear Engineering Department, University of Tennessee
Member, Thesis Committee, Ph.D. Thesis, University of Tennessee (C. Yang, finished April 1982; and B Broadhead, finished April 1983)
Invited paper, "U-238 Issues Resolved and Unresolved" (with A. B. Smith, Argonne National Laboratory), International Conference on Nuclear Data for Science and Technology, September 6-10, 1982
Invited seminar at Georgia Institute of Technology, Reactor Physics Department
Referee, *Nuclear Science and Engineering*
Referee, *The Physical Review*

J. K. Dickens

Member, American Nuclear Standards Committee 5; Working Group 5.1, Decay Heat Standard; Working Group 5.2, Fission Product Yields
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: "Pauli's Little Neutral One—Fifty Years of Hide and Seek," presented at (1) Harvard University, Department of Physics, October 1982; (2) Department of Physics, University of Georgia, March 1983; and (3) Department of Physics and Astronomy, Denison College, December 1983; "Nuclear Beta Decay and the Elusive Neutrino," Society of Physics Students and Department of Physics, Michigan Technical University, April 1983
Invited speech, "The Demise of Nuclear Power in the U.S.," Denison College, December 1983
Speaker's Bureau Presentations: "Nuclear Safety," Greenville Lion's Club, September 9, 1982; Kingston Lion's Club, January 24, 1983; Faculty Seminar, Tusculum College, March 30, 1983; American Society of Safety Engineers, Kingsport, December 12, 1983
Reviewer, Proposals to the Department of Energy, Division of Nuclear Physics
Referee, *Nuclear Science and Engineering*
Referee, *The Physical Review*

F. C. Difilippo

Referee, *Nuclear Science and Engineering*

G. F. Flanagan

Member, Reactor Safety Division Program Committee, American Nuclear Society
Member, Standards Committee 19.3, American Nuclear Society

Member, Executive Board, Oak Ridge/Knoxville Chapter of the American Nuclear Society
Chairman, Nominating Committee, Oak Ridge/Knoxville Chapter of the American Nuclear Society
General Chairman, 1985 Fast Reactor Safety Topical Organizing Committee, April 21-25, Knoxville, Tennessee
Vice Chairman/Chairman-elect, East Tennessee Chapter, Society for Risk Analysis, 1983-84
Member, Standards Committee 5.4, Institute of Electrical and Electronics Engineers (IEEE)
Member, Standards Committee 19.3, American Nuclear Society
Member, Editorial Board, *Risk Analysis Journal*
Chairman, Oak Ridge/Knoxville Section, American Nuclear Society, 1982-83
Editor, *Risk Newsletter*, published by Society for Risk Analysis
Member, Ph.D. Recruiting Program, ORNL
Member, ORNL Speaker's Bureau
Representative, ORNL Advisory Committee for Office of Risk Analysis
Member, American Nuclear Society Speaker's Bureau
Invited speaker, Iowa State University, "PRA Techniques Used in Solution of the Pressurized Thermal Shock Problem"
Speaker, K-25 Management Luncheon Series, "Are Today's Nuclear Power Plants Safe?"
Speaker, U.T. Elder Hospice Short Course, "Nuclear Power and the Environment"
Speaker, Union college, Kentucky, "Nuclear Power Plant Fundamentals"
Certificate of Governance awarded by ANS Board of Directors (1982)
Award of Appreciation, Oak Ridge/Knoxville Section, American Nuclear Society, 1983
Referee, *Nuclear Safety Journal*
Referee, *Nuclear Technology*
Referee, *Risk Analysis Journal*

C. Y. Fu

Member, Evaluation Methods Subcommittee, Cross Section Evaluation Working Group (CSEWG)
Member, Standards Subcommittee, Cross Section Evaluation Working Group (CSEWG)
Referee, *Nuclear Science and Engineering*

T. A. Gabriel

Member, ASTM Subcommittee E10.08, Procedures for Neutron Radiation Damage Simulation
Member, ASTM Subcommittee *E10.08.02, Simulation of Helium Effects in Irradiated* Spectral Tailoring
Member, Engineering Physics Division Internal Audit Committee
Invited paper, Workshop on Neutrino Research, Nuclear Research Center, Karlsruhe, W. Germany (with R. A. Lillie), "Use of Gd-Loaded Scintillator Detector Systems for Inverse Beta Decay Reactions," July 4-7, 1983
Invited speaker, Department of Energy, Germantown, Maryland, "Calorimeter Design Calculations," November 1983
Invited paper, EPRI Conference on Accelerator Breeding, Palo Alto, CA, June 9-10, 1982 (with R. G. Alsmiller, Jr.), "Nucleon-Meson Transport Capability for Accelerator Breeder Target Design"

L. H. Gray

Invited paper, Eleventh Water Reactor Safety Research Information Meeting, October 1983, "Criteria for Safety-Related Operator Actions" (with P. M. Haas)

C. M. Haaland

Advisor, National Council on Radiation Protection and Measurements, Scientific Committee 63, Radiation Exposure Control in a Nuclear Emergency

Member, Union Carbide Nuclear Division Speakers' Bureau
Invited lecture, "Nuclear Weapons Effects and Defenses," Arizonans for for National Security, Phoenix, Arizona, September 1983
Invited lecture, "Nuclear Weapon Effects and Defenses," The American Civil Defense Association, Washington, D.C., October 1983
Invited lecture, "Radiological Monitoring," Annual Meeting of Radiological Defense Officers, Atlanta, Georgia, July 1982
Invited lecture, "Radiation Safety for Emergency Workers," Annual Meeting of Radiological Defense Officers, Seattle, Washington, July 1983
Panel Member, Kansas City TV Station KMB, following showing of ABC Movie "The Day After"
Referee, *Journal of Civil Defense*

P. M. Haas

Member, Institute of Electrical and Electronics Engineers (IEEE), Nuclear Power Engineering Committee (NPEC) Subcommittee SC-7, Human Factors and Control Facilities
Liaison Member, IEEE NPEC Subcommittee, SC-5, Reliability
Chairman, Centralized Reliability Data Organization (CREDO) Steering Committee
Section Editor, *Nuclear Safety*, Accident Analysis Section
Invited papers to 10th Water Reactor Safety Research Information Meeting and 11th Water Reactor Safety Research Information Meeting
Member, In-Plant Reliability Data System (IPRDS) Steering Committee

M. C. G. Hall

Invited paper, "Systematic Analysis of Climatic Model Sensitivity to Parameters and Processes." with D. G. Cacuci at 4th Maurice Ewing Symposium on Climate Processes: Sensitivity to Insultation and CO_2, Columbia University, October 25-27, 1982
Invited paper, "The Adjoint Method of Sensitivity Analysis Applied to Atmospheric Models," with D. G. Cacuci, National Center for Atmospheric Research, Boulder, Colorado, August 5, 1983

J. A. Harvey

Member, ORNL Accelerators and Radiation Sources Review Committee
Labor Coordinator, Engineering Physics Division
Secretary-Treasurer, American Physical Society, Division of Nuclear Physics, 1967-1984
Member, Advisory Committee of Topical Conference on Neutron-Nucleus Collisions: A Probe of Nuclear Structure
Member, Program Selection Committee, Fifth International Symposium of Capture Gamma-Ray Spectroscopy and Related Topics, Knoxville, Tennessee, September 10-14, 1984
Invited lectures, "Oak Ridge Electron Linear Accelerator," "Basic and Applied Neutron Physics," and "Neutron Detectors," Beijing, China, November 1983
Referee, *Physical Review C*
Referee, *Nuclear Science and Engineering*

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, Executive Committee, Radiation Protection and Shielding Division, American Nuclear Society
Invited paper, "Survey of Shielding Data and Methods for Fuel Reprocessing Applications," Annual Meeting of the American Nuclear Society, Detroit, Michigan, June 12-16, 1983

Best Paper Award, Radiation Protection and Shielding Division of American Nuclear Society, June 12-16, 1983
Referee, *Nuclear Technology*

C. C. Jorgensen

Member, National Academy of Science Special Panel on Human Operator Modeling, 1984
Cochairman, In-Service Human Factors Technical Advisory Group Modeling, 1982-1984
Member, Emerging Technologies Advisory Committee, National Security Industrial Association (NSIA)
Invited paper, premier issue, *Training Technology Journal*, Vol. I, 1983 (with L. O'Brien), "Early Training Estimation System (ETES), An Automated Training Needs Assessment Technique"
Referee, *Training Technology Journal*

H. E. Knee

Member, Centralized Reliability Data Organization (CREDO) Steering Committee
Member, CREDO Data Evaluation Working Group and CREDO Input/Output Working Group
Member, In-Plant Reliability Data System (IPRDS) Steering Committee
Member, IPRDS Executive Working Group and Data Analysis Working Group
Member, Reliability Advisory Working Group
Section Editor, *Nuclear Safety*, Accident Analysis Section

B. L. Kirk

Publication (with R. W. Rust), "The Solar Cycle Effect on Atmospheric Carbon Dioxide Levels," in *Weather and Climate Responses to Solar Variations*, Billy M. McCormac, Editor, Colorado Associated University Press, Boulder, Colorado. pp. 129-236 (1983)

D. C. Larson

Cochairman, General Purpose Evaluations Subcommittee, Cross Section Evaluation Working Group (CSEWG)
Member, Evaluation Methods Subcommittee, CSEWG
Member, Department of Energy Nuclear Data Committee
Member, LMFBR Nuclear Data Evaluation Task Force
Co-Chairman, Engineering Physics Division seminars
Member, Office of Basic Energy Sciences (OBES) Committee on Nuclear Data for Fusion Reactor Applications
Referee, *Nuclear Science and Engineering*
Proposal reviewer, Officer of Basic Energy Sciences
Recipient, 1982 Union Carbide Community Service Award

R. A. Lillie

Member, ORNL Reactor Experiments Review Committee
Chairman, Periodic ORR and BSR Experiments Review Subcommittee
Member, ASTM Subcommittee E10.08, Procedures for Neutron Radiation Damage Simulation
Chairman, ASTM Subcommittee E10.08.02, Simulation of Helium Effects in Irradiated Spectral Tailoring
Lecturer, Tennessee Industries Week, University of Tennessee, "Discrete Ordinates Methods and Applications"
Invited paper, Workshop on Neutrino Research, Nuclear Research Center, Karlsruhe, W. Germany (with T. A. Gabriel), "Use of Gd-Loaded Scintillator Detector Systems for Inverse Beta Decay Reactions," July 4-7, 1983
Referee, *Nuclear Technology*
Referee, *Nuclear Technology/Fusion*

R. L. Macklin

Member, Engineering Physics Division QA Audit Team
Member, Engineering Physics Division Safety Inspection Quarterly Review
Collaborative Experiment, Kernforschungszentrum Karlsruhe, West Germany, October 10-19, 1982
Fellow, American Association for the Advancement of Science
Fellow, American Physical Society
Referee, *Nuclear Science and Engineering*
Referee, *Physical Review*
Referee, *Nuclear Physics*

R. E. Maerker

Member, ASTM E10.05 Subcommittee, 1978-1982
Member, Shielding Subcommittee, Cross Section Evaluation Working Group (CSEWG)
Member, Data Testing Subcommittee, Cross Section Evaluation Working Group (CSEWG)
Invited principal speaker at EPRI Workshop on the LEPRICON expanded program methodology, Palo Alto, CA, April 1983
Invited lecturer, University of Tennessee, Department of Nuclear Engineering Seminar, "Incorporating Babcock and Wilcox Reactor Geometry into the DOT-4 Code," May 1983
Referee, *Nuclear Science and Engineering*

F. C. Maienschein

Member, Nuclear Energy Agency Committee on Reactor Physics (NEACRP)
Member, ORNL Awards Advisory Committee
Member, ORNL Affirmative Action 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

B. F. Maskewitz

Member, American Nuclear Society Board of Directors (1980-1983)
Member, American Nuclear Society Executive Committee (1981-82; 1982-83)
Member, American Nuclear Society Planning Committee (1979-82; 1982-85)
Chairman, American Nuclear Society Planning Committee (1983-84)
Chairman, American Nuclear Society, Radiation Physics and Shielding *Ad Hoc* Committee for the Sixth International Conference on Radiation Shielding (ICRS), Tokyo, Japan, May 1983
Member, ANS-10 Subcommittee of the ANS Standards Committee
Chairman, ANS 10.3. Subcommittee, Software Documentation Standards
Member, Steering Subcommittee of Public Information Committee, American Nuclear Society (1982-83)
Certificate of Governance awarded by American Nuclear Society Board of Directors (1983)
Vice President/President Elect Candidate, American Nuclear Society 1984 Election
Member, Organizing Committee, Symposium on Numerical Data for Energy Systems, Committee on Data for Science and Technology (CODATA) of the International Council of Scientific Unions (ICSU) Ninth International CODATA Conference, to be held in Jerusalem, Israel, June 24-28, 1984
Member, Biogeochemical Cycles and Climate Impact Assessment Working Groups of ICSU/UNEP Scientific Committee on Problems of the Environment (SCOPE)
Member, ORNL Speakers' Bureau
Member, ORNL Honors and Awards Advisory Committee

Invited address, "RSIC After 20 Years — A Look Back and A Look Ahead," 6th ICRS. Tokyo, Japan, May 18, 1983

Invited lecturer, Institute of Atomic Energy and Institute of Nuclear Engineering, Beijing, and the 728 Reactor Design Institute, Shanghai, People's Republic of China, May-June, 1983

Invited paper, "Relationship of Energy Systems to CO_2-Climate Issues Is of Global Concern," 9th International CODA Conference, Jerusalem, Israel, June 24-28, 1984

Received tribute/certificate in honor or RSIC's 20th Anniversary, 6th ICRS, Tokyo, Japan, May 18, 1983

Invited paper, "The Role of the Information Analysis Center in Quality Assurance," American Society of Information Services, Knoxville, Tennessee, June 13, 1982

Referee, Computer Code Abstracts, *Nuclear Science and Engineering*

D. L. Moses

Member, Tower Shielding Reactor Subcommittee, ORNL Reactor Operations Review Committee

Referee, *Nuclear Science and Engineering*

Referee, *Nuclear Technology*

D. K. Olsen

Chairman, Committee on Proton Accelerator for ORELA Replacement

Referee, *Nuclear Science and Engineering*

Proposal Reviewer, National Science Foundation

Invited paper, "ORELA Contribution to Thorium Cycle Nuclear Data," Japan-U.S. Seminar on Thorium Fuel Reactors, Nara, Japan, October 19, 1982

Invited lectures, "Recent ORELA Cross Section Measurements," KURRI Kumatori, Japan, October 25, 1982, and JAERI, Tokai, Japan, October 27, 1982

R. W. Peelle

Chairman, Cross Section Evaluation Working Group (CSEWG) Data Uncertainty and Correlation Subcommittee

Member, CSEWG Executive Committee

Member, General Program Advisory Committee for 1985 International Conference on Nuclear Data for Basic and Applied Science

Referee, *The Physical Review*

Referee, *Nuclear Science and Engineering*

C. M. Perey

Publication cited as a "Citation Classic" in the Field of Physical, Chemical, and Earth Sciences, May 1983

F. G. Perey

U.S. Member, Nuclear Energy Agency Nuclear Data Committee (NEANDC)

Member, Department of Energy Nuclear Data Committee

Member, DOE Review Panel for the TRISTAN facility (BNL)

Chairman, NEANDC Task Force to resolve discrepancies in the Fe 1.15 keV resonance parameters

Invited seminar, Physics Division, Brookhaven National Laboratory, "Group Theoretical Foundations of Probabilities," November 1982

Invited talk to IAEA Vienna, "Results of the Fe Task Force," November 1983

Invited seminar, Physics Division, Harwell, "Treatment of Covariances in Large Data Sets," October 1983

Invited seminar, Physics Division, Bruyeres le Chatel, France, "Group Theoretical Foundation of Probabilities," November 1983

Referee, *Nuclear Science and Engineering*
Referee, *Physical Review*
Referee, *Nuclear Physics*

R. B. Perez

Professor, Nuclear Engineering Department, University of Tennessee
Referee, *Nuclear Science and Engineering*
Referee, National Science Foundation

W. A. Rhoades

Member, Technical Program Committee, 1985 Mathematics and Computation Division Topical Meeting, American Nuclear Society
Member, Vector Processing Benchmark Committee, ORNL

R. W. Roussin

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)
Chairman, Speakers' Bureau, Oak Ridge/Knoxville Section of the American Nuclear Society, 1982-83
Invited contribution, "Structural, Heavy Coolant, and Shielding Material Cross Section Data," a chapter in "Cross-Section Data for Nuclear Reactor Analyses," Vol. 2/3 of *Progress in Nuclear Energy*, Pergamon Press, Oxford
Invited paper, "VITAMIN-E 174-n, 38g Cross Section Library for Fusion Neutronics Calculations," American Nuclear Society Meeting, June 1984

R. T. Santoro

Member, Executive Committee, Radiation Protection and Shielding Division, American Nuclear Society
Member, Honors and Awards Committee, Fusion Energy Division, American Nuclear Society
Member, Triennial Audit Review Committee for Activities of Reactor Experiments Review Committee and Reactor Operations Review Committee
Salaried Employee Complaint Counselor, ORNL, 1982-83
Invited paper, "Cross-Section Needs Determination and Status" (with C. R. Weisbin *et al.*), ANS Topical Meeting, Kiamesha Lake, New York, September 22-24, 1982
Invited paper, "Vanadium Alloys and Modified Steels for Low Activation Fusion Design" (with E. E. Bloom *et al.)*, Winter Meeting, American Nuclear Society, Washington, D. C., November 14-18, 1982
Referee, *Nuclear Science and Engineering*
Referee, *Nuclear Technology*
Referee, *Nuclear Technology/Fusion*

D. L. Selby

Member, Executive Board, Oak Ridge/Knoxville Section American Nuclear Society
Member, ANS Speakers' Bureau
Invited talk, "Nuclear Power Plants — How Do They Work and What Is Their Risk?" to Knox County schools science teachers during in-service session
Invited presentation to University of Tennessee Communications classes, "Nuclear, Coal, Hydro, Solar, and Synfuel Plants — How Do They Work and What Is Their Risk?"
Outstanding Member of the Year Award, Oak Ridge/Knoxville Section of the American Nuclear Society, 1982-83

C. O. Slater

Member, NEED Committee of the American Nuclear Society
Member, Program Committee for the 1984 Reactor Physics and Shielding Topical Meeting, American Nuclear Society
Member, Program Committee, American Nuclear Society, Radiation Protection and Shielding Division

R. R. Spencer

Radiation Control and Safety Officer, Engineering Physics Division
Member, ^{252}Cf 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
Member, ANS-6.5 Shielding Glossary, American Nuclear Society
Member, Book Publishing Committee, American Nuclear Society
Member, Telecommunications Subcommittee of Planning Committee, American Nuclear Society
Member, Nominations Committee, Radiation Protection and Shielding Division, American Nuclear Society
Citation for Outstanding Service and Technical Achievement, American Nuclear Society, Radiation Protection and Shielding Division, June 1983

D. R. Vondy

Member, Benchmark Committee, Mathematics and Computation Division, American Nuclear Society
Member, ANS-10 Standards Committee, American Nuclear Society
Alternate Member, ANSI Standards Committee, X-3
Alternate Member, Fortran Committee, X3J3
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 CO_2 and Climate
Director, Carbon Dioxide Information Center (June 1983)
Chairman, Data Testing and Applications Committee, Cross Section Evaluation Working Group (CSEWG), March 1983
Director, Center for Engineering Systems Advanced Research (CESAR)
Member, Editorial Board, *Nuclear Science and Engineering*

L. W. Weston

Cochairman, General Purpose Evaluation Subcommittee, Cross Section Evaluation Working Group (CSEWG)
Member, LMFBR Nuclear Data Evaluation Task Force
Referee, *Nuclear Science and Engineering*

B. A. Worley

Member, Program Committee, Mathematics and Computation Division, American Nuclear Society

Member, Executive Board, Oak Ridge/Knoxville Section, American Nuclear Society
United Way Coordinator, Engineering Physics Division, ORNL
Referee, *Nuclear Science and Engineering*
Referee, *Nuclear Technology*
Proposal Reviewer, National Science Foundation

ENGINEERING PHYSICS DIVISION SEMINARS AT ORNL

Current seminar coordinators are D. C. Larson and M. C. G. Hall. B. A. Worley and R. R. Spencer were seminar coordinators during the first half of the period covered by this report. The following seminars were held during 1982-1983.

R. G. Alsmiller, Jr., J. Barhen, and J. R. Einstein, Engineering Physics Division, "Development of Strategic Planning Capabilities for Liquid and Gaseous Fuel Supplies"

J. E. Beavers, Civil and Architectural Engineering, Y-12, "Earthquakes in Tennessee"

B. J. Bell, Sandia National Laboratories, "Human Reliability Analysis in Nuclear Power Plants"

K. B. Cady, Cornell University, Ithaca, NY, "The Theory of Response"

R. J. Carter, U.S. Army Research Institute for the Behavioral and Social Sciences, Ft. Bliss, TX, "Training Effectiveness Research at Army Research Institute"

R. Chawla, EIR, Switzerland, "Physics Experiments on LWHCR Lattices at EIR"

R. Karam, Georgia Institute of Technology, "LMFBRs Operated on Extended Burnup Cycle"

P. A. Krois, Engineering Physics Division, "Correlates of Job Design with Absenteeism and Job Satisfaction"

R. L. Kustom, Physics Division, Argonne National Laboratory, "Accelerator Design for a New Argonne Superintense Pulse-Neutron Source," Joint Seminar with Solid State and Physics Divisions

R. P. Leinius, Computer Sciences, "Status of Central Computing Facilities"

F. C. Maienschein, Engineering Physics Division, "Cost Effectiveness of Protective Measures or What Differences Do They Make?"

F. R. Mynatt and R. K. Adams, Instrumentation and Controls Division, G. W. Morrison, Computer Sciences Division, and W. W. Engle, Jr., Engineering Physics Division, "Informal Discussion of Personal Computers (PCs) and How Their Use May Grow at ORNL"

P. Otaduy, Instrumentation & Controls Division, "Tools for Artificial Intelligence"

F. G. Perey, Engineering Physics Division, Series of three lectures on "Logic of Physics"

V. Protopopescu, Boston University, "Spectral Methods in Transport Theory"

H. Rabitz, Princeton University, "Sensitivity Analysis with Applications to Problems in Chemical Physics"

H. Rief, Sector Head, Code Assessment and Evaluation Joint Research Center, Ispra, Italy, "Shielding Benchmark Experiments and Their Analysis (Including Monte Carlo Sensitivity)"

Y. Ronen, Ben Gurion University, Beer Sheva, Israel, "Developments in High Converter Water Reactors"

R. Schuttler, Institut National des Sciences Appliquees, Toolouse, France, "A General Least Square User-Oriented Program"

J. Schwartz, Courant Institute, New York University, "Recent Work on the NYU Ultracomputer, a Large-Scale Parallel Machine"

L. Shapiro, Virginia Polytechnic Institute, "The Use of Num ric Relational Distance and Symbolic Differences for Organizing Models"

H. J. Sheppard, Seville Research Corporation, "Development of a Model for the Prediction of Transfer of Training"

M. Skorska, University of Tennessee, "Analysis of Fluctuations in a Combustion-Driven Open-Cycle MHD Generator"

M. V. Tulloch, Institute for Nuclear Power Operations, "Human Factors in the Nuclear Utility Industry"

D. Wagner, JBF Associates, Inc., "Flood Risk Analysis Methodology"

S. Wender, Los Alamos National Laboratory, "Bismuth-Germanate as a High Energy Gamma-Ray Detector"

B. Zeitnitz, Institute fur Kern und Teilchenphysik, Karlsruhe, Germany, "Neutrino Physics at the Pulsed Spallation Neutron Source SNS"

PUBLICATIONS*

ALSMILLER, R. G., JR., J. BARHEN, J. E. HORWEDEL,** J. L. LUCIUS,** AND J. D. DRISCHLER

"The Application of Adjoint Sensitivity Theory to a Liquid Fuels Supply Model," ORNL/TM-8850 (September 1983). **(6.3)**

ALSMILLER, R. G., JR., J. BARISH,** J. D. DRISCHLER, J. E. HORWEDEL,** J. L. LUCIUS,** and J. W. McADOO**

"Sensitivity Theory and Its Application to a Large Energy-Economics Model," *Operations Research* 31(5), 915 (1983). **(1982)**

ALSMILLER, R. G., JR., R. T. SANTORO, J. BARISH,** AND J. M. BARNES**

"Dose Rates from Induced Activity in the Elmo Bumpy Torus Proof-of-Principle Device," ORNL/TM-8505 (October 1982); *Nucl. Tech.* **4**, 491 (1983). **(3.5)**

ALSMILLER, R. G., JR., R. T. SANTORO, R. L. CHILDS,** J. M. BARNES,** AND J. L. LUCIUS**

"Sensitivity Analysis for a Specific Fusion Reactor Shielding Experiment Containing Tungsten," ORNL/TM-8105 (June 1982); *Nucl. Sci. Eng.* **83**, 389 (1983). **(1982)**

AUCHAMPAUGH, G. F.,** M. S. MOORE,** J. D. MOSES,** R. O. NELSON,** C. E. OLSEN,** R. C. EXTERMANN,** N. W. HILL,** AND J. A. HARVEY

"High-Resolution Measurements and R-Matrix Analysis of the Total and Fission Cross Sections of ^{237}Np + n from 1 to 600 eV," Los Alamos National Laboratory Report, LA-9756-MS (May 1983). **(1.23)**

BARHEN, J., R. G. ALSMILLER, JR., C. R. WEISBIN, J. E. HORWEDEL,** J. L. LUCIUS,** B. C. TONEY,** J. D. DRISCHLER, F. MORRA,** V. A. KUUSKRAA,** D. M. NESBITT,** AND R. L. PHILLIPS**

"Design of a Liquid Fuel Data Supply Model for U.S. Policy Analysis," ORNL/TM-8407 (August 1982); *Energy* **8**, 169 (1983). **(1982)**

BARKS, D. B.,** F. E. GOMER,** E. J. KOZINSKY,** AND G. F. MOODY**

"Nuclear Power Plant Control Room Task Analysis: Pilot Study for Boiling Water Reactors," ORNL/SUB/79-40432/1 (NUREG/CR-3415) (September 1983). **(2.39)**

*Numbers shown in parentheses following the publication description correspond to an abstract in this report. In some 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.

BARTINE, D. E., J. C. CLEVELAND,** J. L. ANDERSON,** D. L. MOSES, H. I. BOWERS,** M. L. MYERS,** E. C. FOX,** D. J. NAUS,** R. S. HOLCOMB,** P. L. RITTENHOUSE,** F. J. HOMAN,** G. M. SLAUGHTER,** O. H. KLEPPER, AND B. A. WORLEY

"Characterization of an HTGR for Deep Base Application," ORNL/GCR-83/2 (July 1983).

BARTTER, W. D.,** A. I. SIEGEL,** P. J. FEDERMAN,**

"Job Analysis of the Maintenance Supervisor and Instrument and Control Supervisor Positions for the Nuclear Power Plant Maintenance Personnel Reliability Model," ORNL/TM-8299 (NUREG/CR-2668) (November 1982). **(2.43)**

BEARE, A. N.,** D. S. CROWE,** E. J. KOZINSKY,** D. B. BARKS,** AND P. M. HAAS

"Criteria for Safety-Related Nuclear Power Plant Operator Actions: Initial Boiling Water Reactor (BWR) Simulator Exercises," ORNL/TM-8195 (NUREG/CR-2534) (November 1982). **(2.33)**

BEARE, A. N.,** R. E. DORRIS,** E. J. KOZINSKY,** J. J. MANNING, AND P. M. HAAS

"Criteria for Safety-Related Nuclear Power Plant Operator Actions: Initial Simulator to Field Data Calibration," ORNL/TM-8599 (NUREG/CR-3092) (February 1983). **(2.34)**

BEER, H.** AND R. L. MACKLIN

"^{178,179,180}Hf and ^{180}Ta(n,γ) Cross Sections and Their Contribution to Stellar Nucleosynthesis," *Phys. Rev. C* **26(4)**, 1404 (1982). **(1.9)**

BELBLIDIA, L. A.,** J. M. KALLFELZ,** AND D. G. CACUCI

"Generalized Perturbation Theory with Derivative Operators for Power Density Investigations in Nuclear Reactors," *Nucl. Sci. Eng.* **84**, 206 (1983). **(2.12)**

BELL, Z. W.**

"Cross Sections for the Mo(n,xn) Reactions Between 3 and 21 MeV," ORNL/TM-8863 (October 1983). **(1.32)**

BELL, Z. W.,** J. K. DICKENS, D. C. LARSON, AND J. H. TODD**

"Neutron-Induced Gamma-Ray Production in ^{57}Fe for Incident-Neutron Energies Between 0.16 and 21 MeV," *Nucl. Sci. Eng.* **84**, 12 (1983). **(1.29)**

BELL, Z. W.,** J. K. DICKENS, J. H. TODD,** AND D. C. LARSON

"Neutron Flux Measurements at the 22-Meter Station of the Oak Ridge Linear Accelerator Flight Path No. 8," ORNL/TM-8514 (May 1983). **(1.66)**

BELL, Z. W.,** J. W. McCONNELL,** AND E. D. CARROLL**

"A Digital Pulse-Pair Detecting Circuit," *Nucl. Instrum. Methods* **211**, 551 (1983). **(1.65)**

BENJAMIN, R. W.,** J. A. HARVEY, N. W. HILL,** M. S. PANDEY,** AND R. F. CARLTON**

"The Total Neutron Cross Sections of ^{249}Bk and ^{249}Cf Below 100 eV," *Nucl. Sci. Eng.* **85**, 261 (1983). **(1.21)**

CACUCI, D. G.

"On the Finiteness of the Number of Discrete Eigenvalues in Neutron Transport Theory," *J. Math. Phys.* **23**, 2205 (1982). **(1982)**

CACUCI, D. G., P. J. MAUDLIN, AND C. V. PARKS

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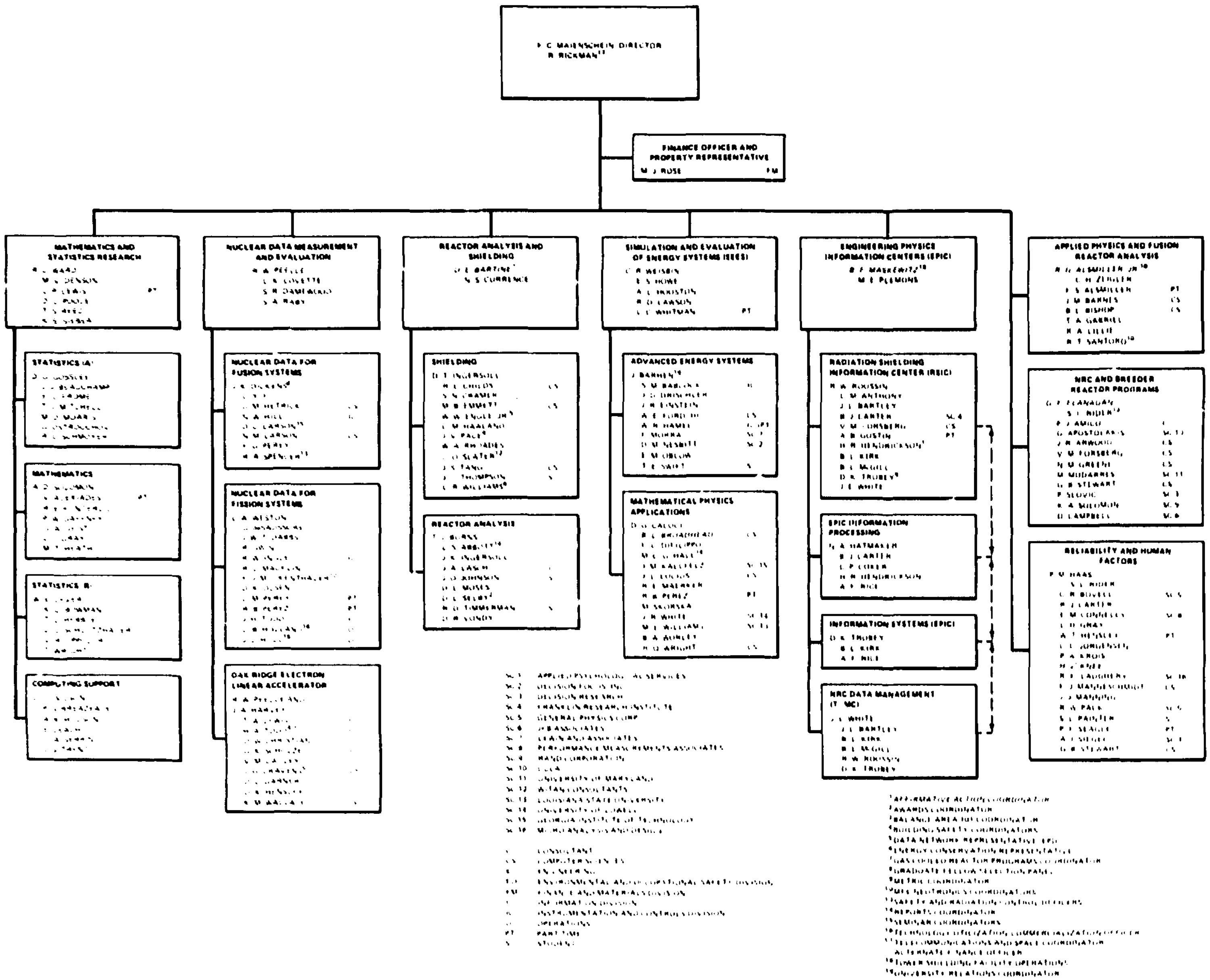

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