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THE STATUS OF REACTOR SHIELDING RESEARCH IN THE UNITED STATES*

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ABSTRACT

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 appproach is being used to develop a sophisticated fusion reactor shielding technology and to study shields for possible future space power reactor systems.

INTRODUCTION

Following development strategies described at the Fifth International Conference on Radiation Shielding, reactor shielding research in the United States continues in a number of diverse but inter-related programs using basically the same calculational and experimental approaches. Refined versions of the large radiation transport computer codes developed for particular shielding programs in the 1960s and 1970s are now being applied to many reactors, and the ENDF/B nuclear data used as input in the codes are consistently being improved. With this calculational capability available, and with the large advances in the computers themselves, an iterative procedure is possible which includes a preanalysis of a shield concept that in turn aids in the design of an experimental test of the concept, which is then followed by an analysis of the as-tested shield configuration (repeated as necessary). This procedure helps both to ensure that the shield design is adequate and that the design is calculable. The same approach is used in designing experiments to test calculational methods or nuclear data.

The validity of this approach has already been demonstrated for the shield of the now-operating Fast Flux Test Facility (FFTF), and it has been applied to the design of shields for the Clinch River Breeder Reactor (CRBR) and the Gas Cooled Fast Breeder Reactor (GCFR). It is now also being applied to several thermal converter reactor systems, for example, to determine possible additional shielding needs at operating light water reactors (LWRs), particularly for post-accident conditions; to estimate radiation effects in LWR pressure vessels; and to investigate the effects of neutron streaming through coolant holes in the High Temperature Gas Cooled Reactor (HTGR).

The approach is also being adapted to the back end of the fission fuel cycle, specifically to examine spent fuel storage and transportation facilities and fuel reprocessing facilities. Looking forward to the deployment of fusion power reactor systems, current calculational and experimental techniques are also being used to develop a sophisticated fusion shielding technology. Finally, it appears that renewed U.S. interest in space power reactor systems may result in the technology again being applied to small reactors for which light shielding materials and optimized shapes are especially important.

A brief review of each of these areas is given below.

CALCULATIONAL CAPABILITIES

Discrete Ordinates Methods

The venerable discrete ordinates radiation transport method remains the standard method for calculating detailed spatial and energy distributions of particle fluxes within a radiation shield. Since the announcement of its availability at the Fifth International Conference on Reactor Shielding,[1] the DOT 4 code has been used widely in various types of deep-penetration shield calculations. A more recent version of the code, DOT 4.3 (ref. 2), features faster coding and an adaptation of the advanced diffusion synthetic acceleration (DSA) method developed at Los Alamos.[3] The discrete ordinates method has proven very adaptable to vector computers, particularly to the Cray-1, on which the flux calculation section of DOT can be run at a speed 7.4 times faster than it can be run on the IBM 3033.

While the weighted difference formulation is the mainstay of DOT calculations, on-going research into nodal methods promises to reduce the number of space cells required in future calculations. Pevey[4] has used an expansion in exponential basis functions to achieve good results with very large meshes, and Walters and O'Dell[5] have used linear expansions of both internal node fluxes and boundary currents to arrive at formulations which provide good accuracy without undue computational complexity. Theoretical work by Larsen[6] has brought understanding of the convergence difficulties faced in diffusion synthetic acceleration (DSA), and Alcouffe[7] reports rapid convergence of DSA iterations using a multigrid method which he has successfully applied in very difficult problems.

Improvements in computational approaches have made 3-D deep-penetration discrete ordinates calculations realizable. The THREETRAN code[8] introduced 3-D discrete ordinates calculations in the U.S., but the development of THREETRAN has been directed toward reactor core calculations rather than deep-penetration shield calculations. The Oak Ridge TORT code is designed for the latter, but it is not yet in general use. TORT may also include a special treatment of internal voids, work by Clark[9] having indicated the success of such treatments. Similar interest in the development of 3-D discrete ordinates methodology has resulted in the Japanese code ENSEMBLE, which was reported on at the Sun Valley meeting[10] and in an earlier paper presented at this meeting.[11]

Improvements in the calculational techniques employed in discrete ordinates calculations have also led to improved techniques for coupling separate calculations along a common boundary. One technique, called the bootstrapping technique,[12] uses the output angle-dependent flux of one calculation as the input flux in a succeeding calculation. Initially developed so that large-scale air-transport problems could be performed, it was subsequently used for the analysis of the FFTF and has since been adapted to other systems. A second coupling technique, the forward-adjoint folding technique, developed during the analysis of the FFTF, has recently been extended to off-axis cylinders,[13] such as in the geometry shown in Fig. 1. This approach has been applied to the calculation of the activation of components in large LMFBRs. The large forward calculation covered the reactor and other components inside the containment vessel, while the adjoint calculation covered the particular component being investigated, such as a heat exchanger. A newer coupling technique, identified as the forward-forward coupling technique,[14] provides for the coupling of cylinders intersecting at a 90-deg angle.

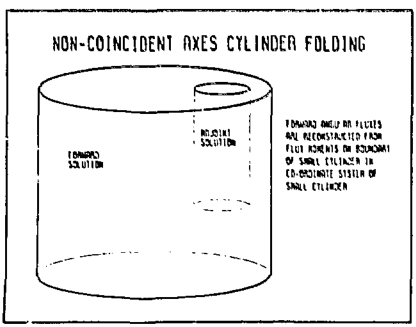

Fig. 1. An Illustrative Geometry for Off-Axis Cylindrical Coupling of Forward and Adjoint Discrete Ordinates Calculations.

In the area of sensitivity analysis, a new semi-empirical approach has been developed in which sensitivity coefficients are used to deduce the change in the response calculated for a deep-penetration shielding problem that would be caused by changes in the properties of the shield. The application of this approach to several problems has given encouraging results.[15]

Monte Carlo Methods

Several Monte Carlo radiation transport code systems are available in the U.S., including MORSE[16] from ORNL, MCNP[17] from Los Alamos National Laboratory, and TRIPOLI,[18] which was developed in France. Development continues on each of these systems.

MORSE uses coupled neutron-gamma multigroup cross sections in Legendre expansion and combinatorial geometry. It can operate in the adjoint mode and can be coupled to discrete ordinates codes through the DOMINO code[19] or similar codes. MORSE is operational on any of the well-known main-frame computers. Current development of the code includes implementing an adjoint in-group energy biasing scheme[20] and preparing an automated biasing procedure with biasing parameters generated from a one-dimensional S_n adjoint calculation. These improvements are needed for the analyses of shields on spent fuel shipping casks.[21]

MCNP is a continuous-energy generalized-geometry code that also uses coupled neutron-photon cross sections. It is operational on the CDC-7600, the Cray-1, and the Cyber-176 computers and is currently being implemented on an IBM-3033 computer at Oak Ridge. A code for plotting the geometry is also available and can be run interactively on the Cray. The current version is in FORTRAN 77, which makes it more adaptable to different computers.

TRIPOLI is also a continuous-energy code. It is being modified to perform coupled neutron-gamma calculations and to interface with discrete ordinates codes. Originally limited to CDC computers, it is now being adapted to other types of computer systems.

Cross-Section Libraries

Version V of the ENDF/B file data is now available and includes considerable quantitative and correlated uncertainty information. The Vitamin-E coupled fine-group library (174 neutron groups, 38 gamma-ray groups) based on ENDF/B-V is also largely completed.[22]

Changes in the Version V data relative to Version IV are currently being evaluated. Comparisons of calculations using both sets of data with experimental benchmarks show only moderate changes in the data for sodium and iron.[23] By contrast, comparisons made as part of the ORNL-GCFR shielding program[24] show that the cross sections for thorium are improved, especially in the important 3-MeV to 30-keV region (see Fig. 2).

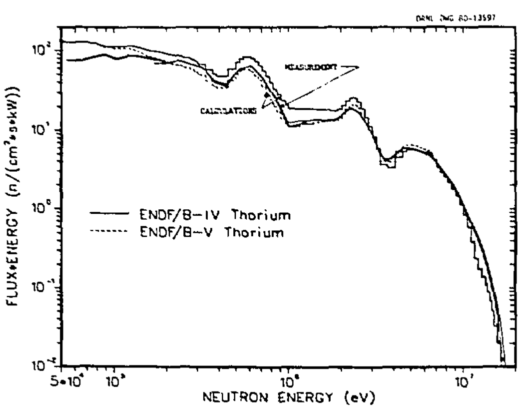

Fig. 2. Comparison of Calculated and Measured Neutron Spectra Behind a 45-cm-thick GCFR ThO_2 Radial Blanket Mockup. (Note: All non-Th data in calculations were ENDF/B-IV.)

The international LMFBR calculational shielding benchmark effort[25] of the Nuclear Energy Agency Committee on Reactor Physics is emphasizing the changes due to different cross-section data bases and processing codes, and also the effects of collapsing the data from fine-group to broad-group energy structures using zone-averaged flux weighting. In another project, the International Atomic Energy Agency is also comparing the effects of different processing codes and methodologies.[26]

ENDF/B-VI will slowly evolve from these continued evaluations, together with additional measurements of nuclear data. The need for improved evaluations in ENDF/B-VI is known for some elements (for example, for niobium and tungsten), and more complete gamma-ray-production data are needed for other elements (for example, cadmium and thorium). The extension of a number of data sets to higher energies is required for fusion energy applications, and for some elements detailed gas-production and activation files are essential. Substantially improved covariance files are also required.[23]

Details of the various types of multigroup data sets and their availability are being presented in another paper at this conference.[27]

FISSION REACTOR SHIELDING RESEARCH

Fast Breeder Reactors

Liquid-Metal Fast Breeder Reactors (LMFBRs). A milestone in LMFBR shielding analysis was achieved when ORNL predictions of radiation levels at the FFTF* compared favorably with the radiation levels measured at the facility in 1981 and 1982 (refs. 28,29). This success indicated that the methods and data employed in the FFTF design and supporting analysis were indeed adequate and that the same techniques could be applied to the shield design for the CRBR with confidence. During the design of the CRBR shield, new data on material irradiation damage became available which reduced the shielding requirements and allowed for an optimized in-vessel shield design that reduced plant costs.[30] Overall, the U.S. experience with LMFBR shield designs indicates the importance of early examination of the shield design and early and frequent examination of interfaces between shielding and other systems and components before major mechanical features are fixed. A recent evaluation of FFTF shield problems has indicated that most are associated with interfaces -- interfaces between fixed and movable shields, between engineered components, and between contractor designs and/or equipment.[31]

Many challenges remain in designing large commercial LMFBRs to be efficient and competitive power plants. In the area of shielding, interest is focused on the tradeoff between shields around the core and on individual shields for radiation-sensitive components, such as the decay heat removal heat exchanger. Alternate shielding materials appear to be attractive for reducing shielding costs, size, and weight. Monitoring systems for initial startup and startups after refueling will require sophisticated radiation transport analysis using validated methods.

Gas-Cooled Fast Breeder Reactors (GCFRs). Although the U.S. GCFR program was cancelled in 1980, some of the information gained in a series of GCFR shielding experiments performed at the ORNL Tower Shielding Facility between 1975 and 1980 (ref. 24) is generic and therefore applicable to other systems. The experiments were performed to evaluate: (1) neutron streaming in the helium coolant passageways in the GCFR core, (2) the effectiveness of the shield designed to protect the reactor grid plate from radiation damage,

*The FFTF, a liquid-metal-cooled fast reactor that is a forerunner of the CRBR, does not have a breeding blanket.

(3) the adequacy of the radial shield in protecting the prestressed concrete reactor vessel (PCRV) from radiation damage, (4) neutron streaming between abutting sections of the radial shield, and (5) the effectiveness of the exit shield in reducing the neutron fluxes in the upper plenum region of the reactor. In general, good agreement between the calculated and measured fluxes was obtained for all experiments.

Thermal Converter Reactors

Light Water Reactors (LWRs). LWR activities in the U.S., including shielding activities, have been dominated by reaction to the accident at Three Mile Island Unit 2 (TMI-2). Before the accident it had been assumed that a reactor could always be brought to cold shutdown from the control room. However, TMI-2 showed that access to the balance of the plant could be necessary during an accident, which would imply that personnel might have to work in high exposure fields. Much effort has been directed toward minimizing post-accident radiation exposures,[32] which, in turn, has led to reduced exposures during normal operations. Specifically, the design criteria have been based on (1) limiting radioactive fluids to containment in specially designed systems, (2) planning for control of personnel exposures, and (3) adding shielding and revising designs to allow operation of balance of plant equipment without greatly increased exposures. New equipment specifications and qualifications tests have also been prepared to assure equipment performance and lifetimes under postulated temperature, pressure, humidity, and radiation level conditions. In addition, a technique has been invented for using a modified version of a standard gamma thermometer to measure the adequacy of the core cooling mechanism of a pressurized water reactor without compromising the original function of the instrument.[33]

The Electric Power Research Institute (EPRI) has established a significant program to estimate radiation effects in LWR pressure vessels.[34] The program includes measurements, calculations, and a newly conceived adjustment procedure to provide updated estimates and uncertainties from information derived from surveillance dosimetry.[35]

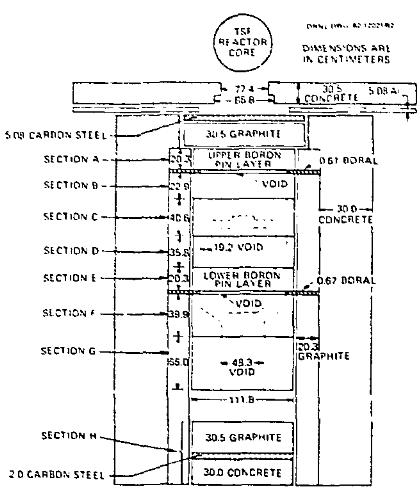

Fig. 3. Configuration for HTGR Core Support Block Area Experiment.

High-Temperature Gas-Cooled Reactor (HTGR). An experimental program has been initiated at the ORNL Tower Shielding Facility to investigate neutron streaming in the core support block region of the HTGR. Design analysis revealed that the streaming of neutrons through large coolant holes in the lower radial reflector, the support block, and the core support post region leads to a high thermal-neutron fluence in the sidewall liner. The experimental configuration shown in Fig. 3 is being used to investigate the effectiveness of boron pin placement and pitch, to measure neutron-streaming effects in coolant holes, and to determine design bias factors.

SHIELDING FOR BACK END OF FISSION FUEL CYCLE

Pool Storage Facilities

Plans for increasing the capacity of storage pools for spent LWR fuel assemblies include both reracking and fuel consolidation. Fuel consolidation is accomplished by pulling the fuel rods from the spent fuel elements and reinserting them in grids that can accommodate twice as many rods as a fuel element grid. Thus, the radiation source densities within the storage pool are doubled. Calculations have shown, however, that because of an increased inventory of uranium in the rod array, gamma-ray absorption within the array itself limits the increase in the gamma-ray dose reaching the pool surface to only about 30%. Thus, the increased shielding (water) that must be added above the stored rods is less than might have been expected.

Experience in operating spent fuel storage pools has revealed an additional source of radiation from contaminated debris floating in the pool. While this source must be accounted for in personnel radiation exposures, it is more of a removal problem (through filtering or some other process) than a pool shielding problem.

Shipping and Dry Fuel Storage Casks

The dry fuel transportation and interim storage cask concept is of current interest both to the U.S. Department of Energy for demonstration purposes and to the electric utilities as a possible solution to their shortage of pool storage capacity. In this concept, the spent fuel elements are transported and stored in the same cask. Most dry fuel cask designs incorporate carbon steel gamma-ray shields and boraflex neutron shields. Calculations have shown that the shields are adequate and that the major design limitation is convective heat transfer.

Some experience is being gained by the utilities in using conventional shipping casks fabricated primarily from lead or uranium. Transfer of spent fuel in the casks between LWR sites has resulted in no significant radiological problems involving the public.[36] Advanced design efforts are concentrating on the optimization of the casks.

Fuel Reprocessing Facilities

A recent review of the shielding aspects of advanced fuel reprocessing plant designs[37] has indicated that substantial cost reductions could be obtained by reducing the margin of conservatism with experimentally benchmarked radiation transport methods and data. A variety of methods will be needed to compute radiation levels in the plant, and a first step will be to collect and tailor existing methods for a range of problems. The primary data deficiencies appear to be material damage response functions needed to define design constraints and estimate equipment lifetimes. A program of integral experiments would verify the computational methods and final shield designs when large uncertainties or stringent criteria justify the expense.

Operating experience at the only U.S. reprocessing facility currently in use indicates that a major radiological problem is the control of inert gaseous fission products whose release could subject personnel to beta exposure.

Permanent Waste Storage Facilities

The preferred method of geologic disposal of high-level waste is the permanent storage of waste cannisters in holes drilled in the floor of a mined cavity. This disposal concept is favored as the result of an extensive study[38] which took into account radiological effects as well as other effects. Major advantages are that many already tested mining techniques can be adapted for use at the facility and the opportunity will exist for retrieving the waste material if that becomes an economic option. In selecting the facility site and determining its design, long-term effects of radiation on the geologic material will be considered. Investigations of the shielding required to reduce exposures to operating personnel will also be carried out.

FUSION REACTOR SHIELDING

With the continued emphasis on optimizing the U.S. technology for magnetically confined fusion plasmas, it appears that the construction of a U.S. fusion power system will be delayed, and thus the Tokamak Fusion Test Reactor (TFTR) at Princeton University probably is the major fusion device in the U.S. for this decade. (Inertial confinement fusion appears to require a few years of additional study before serious consideration can be given to the design of reactors.)

With the U.S. fusion program thus curtailed, development of a fusion reactor shielding technology must depend upon validating the calculational techniques against integral experiments performed at other facilities and later, if feasible, against measurements made at the TFTR. The development of this technology is already well underway with integral experiments using a 14-MeV neutron source being performed at the ORNL.[39] The program planned for the TFTR will include activation measurements, measurements of residual induced activities, and neutron spectral measurements. For the first years of operation, the TFTR will be without tritium, but shots (single-pulse operations of the Tokamak) with a deuterium plasma should produce enough neutrons for these experiments.

It now appears that a recent technological advance for fusion reactors will alleviate some of the shielding problems. It is now planned to effect plasma heating by rf excitation rather than by neutral-beam injection, and the diameters of the ports for the rf injection system should be significantly smaller than the approximately 1-m-diameter ports needed for the neutral-beam injection system; also the rf ducts can be curved (e.g., Z shaped). Thus the requirements for shielding against neutron streaming through the ports should be reduced. The remaining major penetrations - for pumping channels - can be bent or offset to reduce streaming problems, although at additional cost to the pump operation.

Fusion reactor shielding technology will utilize several calculational methods. For the complicated neutron-scattering angle and geometry relationships involved in fusion reactor configurations, point cross section Monte Carlo methods are highly desirable. The Monte Carlo code MCNP[17] operating on Cray computers at Livermore National Laboratory has been successfully used at ORNL for the analysis of complex integral experiments.[39] This represents one of the first clear examples of the use of a computer network to eliminate the need for transfer of methods between laboratories. Two-dimensional discrete ordinates methods obviously will still have an important place in fusion shielding technology, and three-dimensional discrete ordinates methods also will be useful when they become available.

SPACE REACTOR SHIELDING

Space power reactor systems appear to be receiving renewed interest, and the shielding requirements for such systems are quite stringent. Payload and reactor electronic components must be protected, and the shield composition and shape must be optimized to a minimum weight. If this program is pursued, codes to develop space power reactor shield optimization procedures and methodology must be reactivated, and shielding materials must be examined for their efficiency and durability.

SUMMARY STATEMENT

Current reactor shielding research activities in the U.S. are directed toward: (1)improving available code systems and taking advantage of the capability offered by vector processors and the accessibility of codes through computer networks; (2) obtaining and validating needed cross sections and understanding the effects of data manipulation on calculated results; (3) using the established LMFBR shielding technology to help design an efficient and economic large-scale LMFBR; (4) ensuring minimum personnel exposure at LWRs under both operating and post-accident conditions; (5) investigating neutron streaming effects in the HTGR; (6) designing adequate and economic shielding for the back end of the fission fuel cycle; (7) developing a fusion reactor shielding technology; and (8) responding to a renewed interest in space power reactors, which require special shielding considerations.

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