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AN ASSESSMENT OF NUCLEONIC METHODS AND DATA FOR FUSION REACTORS*

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An assessment is provided of nucleonic methods, codes, and data necessary for a sound experimental fusion power reactor (EPR) technology base. Gaps in the base are identified and specific development recommendations are made in three areas: computational tools, nuclear data, and integral experiments. The current status of the first two areas is found to be sufficiently inadequate that viable engineering design of an EPR is precluded at this time. However, a program to provide the necessary data and computational capability is judged to be a low-risk effort.

INTRODUCTION AND SUMMARY

The goal of this assessment is to pinpoint the areas of existing nucleonic methods and data that, in our judgment, require further research and development for application to the design of fusion reactors. Much of the present state-of-the-art as applied to general feasibility studies has been discussed previously,[1,2,3] so the emphasis here is on developments specifically required for the detailed design of an Experimental Power Reactor (EPR),** and for credible system studies of Commercial Power Reactors (CPR). Some consideration is also given to nuclear data needs for radiation effects experiments, particularly if stripping and spallation neutron sources[4,5] are used.

The assessment is divided into three general areas: nuclear data, computational tools, and integral experiments. Within each area, gaps are identified and suggestions made as to the tasks necessary to close these gaps. The development program outlined is based largely on the recommendations of the Neutronics Working Group, at the DMFE Blanket/Shield Workshop.[6] It should be noted that most R & D tasks discussed are equally applicable to magnetically and inertially confined reactors. Fusion/fission hybrids, on the other hand, are entirely excluded from consideration since they open up the entire area of fission reactor analysis.

Scope of the Assessment

Topics considered here under the generic title of nucleonics include:

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**The term EPR is used here in a broader sense than in the current literature on EPR's; it includes all anticipated next-generation fusion reactors whether they be called an EPR, ITR, TETR, or other acronym.

- neutron and photon transport
- responses due to neutron and photon interactions with matter (including, therefore, transport of secondary charged particles)
- sensitivity and optimization
- integral experiments
- nuclear data required for all the above analyses.

Thus, we include all the usual nuclear engineering aspects of reactor design and analysis associated with

- tritium breeding
- nuclear heating
- radiation damage effects by atomic displacement and transmutation
- radiation shielding
- activation and afterheat
- radioactive corrosion product (CRUD) transport and deposition
- perturbation of all the above design parameters.

In assessing nuclear data requirements for fusion reactor nucleonics, we consider the impact of nuclear data not only on transport processes, but also on response functions (e.g., kerma factors and radiation damage) and perturbation calculations (e.g., covariances for uncertainty studies). Under computational methods we evaluate transport methods and codes, as well as the so-called response codes; i.e., codes which convolute neutral particle fluxes from transport codes with response functions. Within this latter category fall the codes for determining integral nuclear heating, radioactivity and afterheat, and radiation damage. Also assessed is the status of perturbation theory codes used for cross-section sensitivity and design sensitivity analyses. Finally, a few comments are made on the role of integral experiments (or more precisely, prototypic blanket/shield mockup experiments) in verifying design calculations.

So far little mention has been made of shielding data and methods per se. However, it is apparent from the preceding paragraphs that fusion reactor analysis is more akin to the traditional shielding discipline (reactor, accelerator, and weapons related) than to conventional fission reactor physics. In particular, fusion reactor calculations generally involve solution of the inhomogeneous Boltzmann equation for moderate to large numbers of mean-free paths of neutral particle transport, as opposed to the eigenvalue problems of fission reactor calculations. Thus, a fusion reactor research and development program necessarily shares with shielding research many of the same methods and data. Shielding problems unique to fusion reactors have been discussed recently,[3] and will not be thoroughly assessed in this review.

In some sense the nucleonics development program involves fewer uncertainties than other engineering problems in fusion reactor research, such as plasma heating and superconducting magnetics. Nucleonics research and development is almost a priori a low-risk venture, because most required *theoretical methods exist in principle*. Moreover, given the wealth of theory, methods development, codes, and data from fission programs, research and development requirements can be predicted with more confidence than in most other fields. It is precisely this predictive capacity which allows us to conclude that the present technological base of *engineering design tools for nucleonics is inadequate* for an assured viable design of the EPR. Even for conceptual system studies or point designs, nucleonic uncertainties exist which could affect conclusions on technical or economic feasibility.

Nuclear Data and Integral Experiments

Specific deficiencies have been identified in nuclear data and computational methods now used for design studies. Examples in the area of nuclear data include: (1) errors in transport and activation cross sections, which can cause errors of 50% and more in prediction of personnel dose rates, even for thin shields;[7] (2) cross-section uncertainties in evaluated data in ENDF[8] for key fusion reactor materials such as ^{7}Li, ^{11}B, C, and Fe, which are known to be inadequate for accurate calculation of tritium breeding, shielding effectiveness, etc.;[9,10] and (3) inconsistencies between neutron kerma factors and gamma-ray production data, resulting in nonconservation of energy and thus erroneous nuclear heating (but usually < 2% error[11]). A measurement and evaluation program[6] for selected reactions will need to be undertaken prior to any final reactor designs. However, there may still need to be a carefully planned program of integral experiments to verify the total attenuation of the primary (magnet) shielding.

Computational Tools

Computational methods for neutron and photon transport have been principally based upon Carlson's S_n method,[12] in the present discrete-ordinates form. They are mostly for calculations in one spatial dimension, using a variety of established codes. Although some difficulties have occurred in comparisons with benchmarks and integral experiments, 1-D methods appear to be well developed. However, 1-D calculations are useful mainly for scoping studies, whereas 2- and 3-D calculations are essential for blanket/shield design and streaming calculations. Difficulties have been experienced in using present 2-D discrete-ordinates codes for EPR blanket design, where a typical reactor[13] will have a poloidal cross section about 7-m in diameter (cf. Fig. 1 for illustration of geometry). Even with symmetry about the horizontal midplane, calculation of detailed flux spatial distributions in 1-m-thick blankets would require tens of thousands of mesh points. Development is thus required for 2-D codes that will efficiently represent circular and irregular geometries (e.g., by triangular meshes[14]), employ advanced acceleration methods,[15] have toroidal geometry capabilities, and of necessity have hierarchical storage. In addition, studies of potential ray effects or other computational anomalies may well be required when extensive 2-D analysis is undertaken. The importance of basic research in numerical transport methods to circumvent some of these difficulties cannot be overemphasized. Deterministic transport methods for treating streaming in voids may also have a large benefit.

Most calculations of streaming in voids, and of neutron transport in complex blanket geometries, are now performed with Monte Carlo methods. Numerous codes exist for such calculations, including ones with explicit toroidal geometry capabilities.[16] However, the difficulties with such codes are due not so much to deficiencies of the calculational techniques nor to the inability to model complex geometries, but rather to practical considerations of problem set-up time, input errors due to complexity, and computer time requirements. Minimization of computer time for a given variance in edit parameters presently depends most strongly on the cleverness of the user in biasing the transport process. Development is urgently needed to simplify input specifications for such Monte Carlo codes, and to put biasing schemes on a more systematic

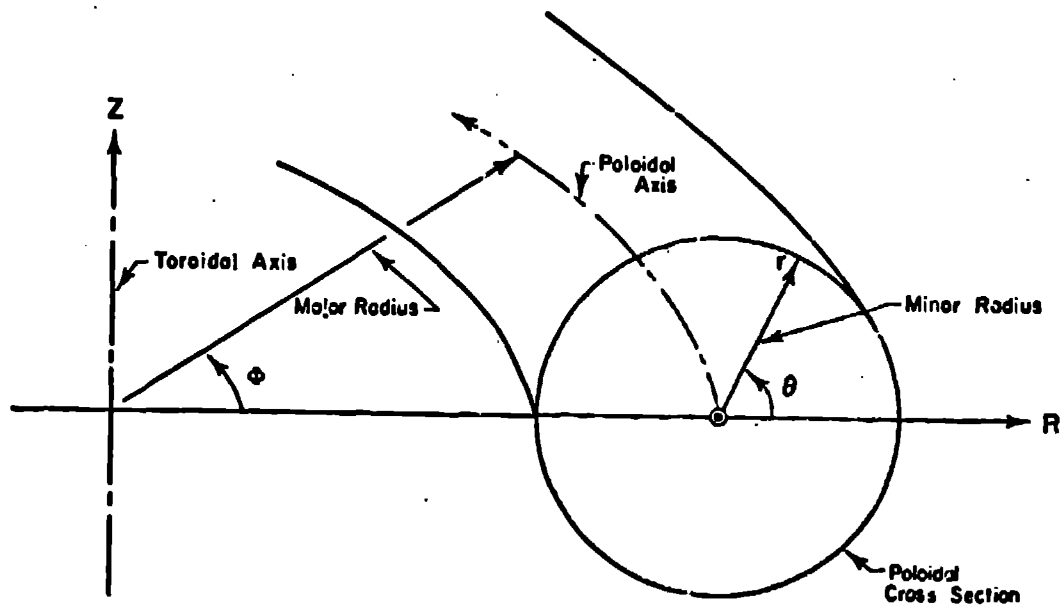

FIGURE 1. Coordinate Geometry Convention for Toroidal and Poloidal Sections

and user-oriented basis. Moreover, since many design problems involve both simple and complex geometries, but in different regions, procedures for efficiently coupling existing S_n and Monte Carlo codes need to be perfected.

An area in which much productive work has occurred is the application of perturbation theory to cross section and design sensitivity,[17,18] as well as to blanket/shield optimization.[19] Further development of such codes, especially in 2-D and with respect to secondary energy distribution sensitivity, will almost certainly provide a large benefit for both EPR and CPR design studies. Advanced methods that transcend the limits of first-order perturbation theory (e.g., mathematical programming techniques with constraints, or the inclusion of higher order terms) may prove valuable for CPR analysis, and a research program in this area appears to be warranted.

Many areas exist in which straightforward code development, data format and interface file standards specification, and data distribution/coordination need to be undertaken. These should evolve from program planning in terms of both CPR and longer range needs. In this process, the nucleonics development needs will probably shift, especially in nuclear data. But certainly the major methods and code advances will continue to prove valuable.

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TRANSPORT METHODS AND CODES

Generally discrete ordinates and Monte Carlo have been the methods of choice for fusion reactor transport calculations. Some analytical approximations and neutron albedos have been used for streaming calculations, but mostly in irradiation facility conceptual shield design. Also, interesting ray-tracing (simplified integral transport) calculations have been performed[20] to determine poloidal flux distributions, spatial and angular, on fusion reactor first walls. Production calculations have, however, been predominantly by discrete ordinates.

One-Space Dimension

Conceptual design and scoping studies have usually employed a stable of 1-D discrete-ordinates codes (ANISN, DTF-IV, ONETRAN). These are standard production codes in the nuclear industry, providing accurate and inexpensive flux distribution results when intelligently used. Although they are already economical of computing time, improvements in computational speed by a factor of a few are on the horizon.[15] These acceleration techniques will be most valuable if, as it appears now, they can be extended to 2-D.

One notable exception to the ascendancy of discrete-ordinates codes for 1-D analysis is the extensive use of the TART Monte Carlo code by Lee,[21] Maniscalco,[22] and colleagues at the Lawrence Livermore Laboratory. They find such 1-D calculations competitive with discrete ordinates in computer running time, with standard deviations of ≤ 2 % in blanket nuclear parameters.

Time-dependent transport calculations in one space dimension are now performed with the TIMEX code,[23] and although somewhat lengthy in computer time, such calculations are relatively infrequently required. Their principal use is in shock and stress analysis of inertial fusion devices.[24] Explicit time-dependent methods using discrete ordinates appear adequate, and time-dependent multigroup Monte Carlo calculations are often a competetive alternative method[24] for such analysis.

Two-Space Dimensions

Once a reactor design is beyond the scoping parametric study stage, 1-D transport calculations prove inadequate for the complete engineering design. Even though they may be useful for energy deposition distributions or other design details in specific modules, such calculations must at least be normalized to poloidally* varying first-wall fluxes or currents.[20] Even accounting for poloidal variations cannot determine the effects of blanket inhomogenieties in directions transverse to the 1-D model. Thus, engineering designs like those evolving for EPR's must employ two- or three-dimensional analysis tools for such purposes. Monte Carlo codes provide a relatively straightforward avenue to multidimensional analysis,[25] but suffer from an inherent limitation on the information content available in practical edit procedures. To circumvent these limitations, existing 2-D discrete-ordinates codes (DOT and TWOTRAN) have been applied to large toroidal reactors, revealing marked differences in local heating and breeding. It has also been shown[26] that calculations in X-Y geometry are inadequate due to neglecting toroidal effects. For systems symmetrical in the toroidal direction, ϕ, existing cylindrical geometry codes can be used to solve the toroidal geometry problem by using R-Z coordinates. Here R is along the major radius and Z is along the major (toroidal) axis. Observed toroidal effects can be divided into two categories; those caused by different volume elements on the inner and outer portions of the blanket, and those caused by different streaming operators (with respect to ϕ, as opposed to Z in an X-Y calculation or Z in a 1-D cylindrical calculation). The latter curvature effect for tori of aspect ratios three and five has been shown[25] to be small, at least for uniform source distributions; i.e., correcting for volume element differences gives good agreement between cylindrical (r-θ, a poloidal section) and toroidal calculations of reacton rates.

What then, are the principal drawbacks of current 2-D discrete-ordinates codes such as DOT or TWOTRAN? First, there are difficulties in modeling the geometric boundaries in a poloidal section with orthogonal coordinate systems, for either circular or irregular cross sections.[26] Second, the tokamak reactors are all very large relative to neutron mean-free paths, thus requiring large numbers of mesh points. The result is a taxing of computer computation

*See Fig. 1 for definition of poloidal and toroidal coordinates.

time and storage. In order to account for toroidal geometry, the only symmetry which can be represented is that about a horizontal midplane, thereby reducing the options for decreasing problem size.

Given the above difficulties, what developments appear possible to alleviate them? No immediate solution to the model size problem occurs to this author*, other than to treat the symptoms directly. For example, any developments in hierarchical storage and data transfer efficiency will prove very valuable in fitting large problems on existing computers, as well as speeding execution times. Also, advantage should be taken of class 6 computers with vector processing as soon as possible, because the discrete-ordinates equations along different characteristics are ideally suited for vector operations. Of equal importance is to test promising convergence acceleration methods, such as the synthetic one,[15] which offer possibilities of reducing computation times by a factor of a few. In many cases the combination of one or more of these development efforts can mean the difference between being able to undertake a particular important 2-D analysis, or not.

The above symptomatic remedies, although valuable, still do not address the modeling problem. For a reactor with a circular cross section, for example, a simple mathematical exercise demonstrates that no polygon, no matter how fine, can represent both the first wall radius and surface areas correctly in the limit (e.g., if radius is preserved with equal R and Z mesh spacing, the surface area is high by a factor $\sqrt{2}$, an interesting academic curiosity at least). From a practical point of view, almost all present reactor designs can be modeled more easily in the R-Z plane using triangular meshes rather than orthogonal meshes. Also, fewer mesh points will generally be required for the same accuracy if mesh spacing is determined not by optical thickness considerations, but rather by fidelity in representing region boundaries.

Development of a 2-D triangular-mesh, finite-element transport code specifically designed for fusion reactor analysis in toroidal geometry is currently underway This development is based upon an existing such code,[14] TRIDENT, designed for analysis of fission reactors with hexagonal modules. A significant effort is being expended to relax the banded triangular structure in TRIDENT, and/or to incorporate an automatic mesh generator.

If history is any indicator, new developments will be accompanied by pathological side effects, as in the case of the ray effects[27] discovered in early 2-D discrete-ordinates calculations. Although ray effects can be avoided in TRIDENT by use of the fictitious source option (making the equations similar to spherical harmonic equations), the designer's judgment must enter into the choice of the degree of such alteration. One also has a vague expectation of new computational anomalies!

Some interest naturally arises in toroidal geometry codes using the r-Θ plane of a poloidal section as the explicit coordinates, with the toroidal angle ϕ being implicit. Such calculations may be of limited interest in a reactor of circular cross-section, but are not appropriate for most tokamak calculations[26] because of geometric modeling difficulties.

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*But cf. below section on Other Deterministic Methods.

Inertially confined fusion reactor designs will, as they approach the "EPR" stage, require multi-space-dimensional, time-dependent transport calculations. Some effort at LASL has been devoted to such a code, TWOTIME, in two space dimensions.

Other Deterministic Multidimensional Transport Methods

There may be a high payoff to research in application of integral transport theory to streaming problems. Neutron streaming in relatively narrow voids is difficult to treat in 2-D discrete-ordinates codes, even in a few instances where such analysis is appropriate (e.g., for divertors where toroidal symmetry holds). For streaming in vacuum or injector ports, Monte Carlo has been the method of choice, with one notable exception; Calculations have been performed for streaming in injector ports,[28] using TWOTRAN in R-Z cylindrical geometry and with the port along the Z-axis. Hence, the plasma is a disk and the geometry of regions around the port can be well represented, giving a realistic model.

Also, the very large size of tokamak poloidal cross sections suggests the possible development of nodal methods. These would be particularly useful where fine detail within blanket modules is not of interest, but rather the power factors from module to module, for example. Nodal diffusion methods as used for large fission reactor core three-dimensional (two space, one time) analysis are suggestive of similar applications to fusion reactors.

It is probably worth stating that analytical and semi-empirical formulae will continue to be useful for calculations of streaming in regular geometries, and some new neutron albedo information may be valuable. This would be a simple extention of previous Monte Carlo calculations of albedos[29] to new materials and energies.

Monte Carlo

Several Monte Carlo codes, both continuous energy and multigroup, have been applied to fusion reactor nucleonics. A partial list of U. S. codes includes MCNG, MCMG, MORSE, SORS, and TART. All have proven satisfactory in assorted applications, from 1-D blanket design[21,22] to studies of 3-D toroidal effects[25] and time-dependent energy deposition.[24] The main difficulties users have experienced[6] seems to be associated with problem setup of complex geometries, as well as selection of biasing schemes for variance reduction. One acquaintance of the author likened the use of Monte Carlo codes to flying an experimental aircraft with unknown flight characteristics. Systematic schemes for biasing neutron flight characteristics are sorely needed, and should be placed in a user-oriented input format. The difficulty (even impossibility) of providing universally applicable biasing schemes is well recognized by users knowledgeable in Monte Carlo methods. However, a concerted effort needs to be made in this direction, and a movement made toward structuring the codes for use by a competent designer who is not a Monte Carlo expert.

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RESPONSE AND UTILITY CODES

Once the distribution of neutrons or gamma rays is determined in the appropriate phase space for a problem, the particle flux density is available. Many auxiliary codes exist to process such output fluxes, in conjunction with response functions and other data, into useful engineering data such as energy deposition, radioactivity, and afterheat. These postprocessing codes, although somewhat mundane, serve an important enough function to deserve consideration for future development. Similarly, a class of utility codes serve miscellaneous functions such as linking transport calculations, converting formats, etc.

Discrete-Ordinates and Monte Carlo Coupling

This is a practical concept for linking transport calculations performed in adjacent regions, where one region has simple geometry (discrete ordinates) and another complex geometry (Monte Carlo). Advantage is thus taken of each method in its domain of intrinsic applicability. Such linking has been used to a limited extent in fission reactor shielding,[30] at least in the forward transport modes. A relatively minor development should be expended to provide linking of current generation codes used for fusion reactors, and to implement forward and adjoint linkage. Adjoint Monte Carlo can provide invaluable information for design of shielding around vacuum and injector ports, obviating the need for extensive iterative design calculations in these areas. Coupled adjoint calculations similary can provide insight into the importance of spatial and energy regions to transport in complex systems. Development of this basically straightforward method using linked calculations should be taken to the stage of providing a relatively routine design tool.

Radioactivity and Afterheat

Numerous codes have been devised to compute radioactivity and afterheat. The major effort now involves devising decay-data libraries (See Nuclear Data Section). Once these data become available in ENDF, an intermediate processor will be required to produce libraries for radioactivity codes. These codes generally compute spatial distributions and integrals of radioactivity, afterheat, biological hazard potential, and other weightings of the basic radioactivity values.

Shield Optimization

A strong incentive exists to optimize shield arrangements and materials to achieve a given effective attenuation in minimum thickness. For example, such optimization may have a high payoff on the inner shields of tokamaks,[13] and around streaming ports. Development effort aimed at 2-D perturbation-theory optimization codes should prove valuable for these classes of problems, including guiding 3-D streaming calculations. Advanced methods such as mathematical programming models with constraints may prove valuable for CPR design, and hence appear to be worth long-range development efforts.

Code and Data File Standardization

It seems apparent that any well-conceived plan for standardizing coding practices and data file interfaces should be desirable. The objective of any effort in this direction is to avoid duplication of effort by making codes easily interchangeable among computers and installations (exportability), and making data files similarly machine independent. Data file standardization allows linking of codes from different installations, but just as important, it provides a communication medium for distributing data such as the CTR Cross-Section Library. A liaison has been established between the fission and fusion reactor communities, in the form of the Committee on Computer Code Coordination (CCCC). Future cooperative efforts should make much of the code and data development work in the fission area immediately applicable for fusion, and vice versa.

NUCLEAR DATA

Experience in the nuclear data field for fission work has provided an understanding of the need to focus early on a limited and achievable set of goals. This implies strict priorities, and nonproliferation of "laundry lists." As a corollary, priorities must follow from needs which are well defined and substantiated. Requirements for measurements and evaluations should result from assessments which are as quantitative as possible. One mechanism for quantifying such needs is sensitivity and uncertainty analysis. These methods are extremely valuable when there is some definition of the design under consideration; and in the case of uncertainty analysis, where first-order perturbation theory applies. Projecting longer-range needs, however, still relies heavily on good engineering judgment in identifying materials and nucleonic effects of importance, as well as in estimating the quality of pertinent evaluated data files.

One difficulty with identifying nuclear data needs has been the long lead times required for the measurement and evaluation process. For example. if any data deficiencies had been identified for the TFTR activation calculations[7] (none were), it would be difficult to rectify them prior to the engineering design, except perhaps on an ad hoc basis.

The foregoing discussion is directed at nuclear data measurements for fusion reactor programmatic needs. It is not in the scope of this review to comment on the priorities for measurements of interest in low-energy physics, but which have only incidental engineering application in the fusion program. Below are summarized measurement and evaluation recommendations from Ref. 6, where only direct programmatic judgments are involved. Top priority was assigned almost exclusively to EPR design requirements, with ^{7}Li being the notable exception. It is worth commenting that continuing developments in nuclear-model calculations may ease some of the measurement program by filling in data gaps satisfactorily.

Nuclear Data Measurements

Priority I measurement recommendations are all for neutron emission spectra and gas production cross sections in EPR materials and in ^7Li. Table 1 presents the materials, accuracy requirements and incident neutron energies for these measurements. Emission spectra are important for transport calculations where the secondaries make significant contributions to the flux. Gas production cross sections are essential for predicting radiation damage and for analyzing correlation experiments in irradiation facilities. The singular exception is ^7Li, where tritium production is of interest.

Table 1. Neutron Cross-Section Measurement Needs (mainly for EPR)[6]

NEUTRON EMISSION ($\geq$ 5 values of $\theta_{n'}$, $E_{n'} \geq$ 500 keV)

Material	Accuracy	Incident Energies, E_n
^7Li	10%	11, 14 MeV
^{11}B	10%	11, 14 MeV
C	10%	11, 14 MeV
Fe	10%	11, 14 MeV
GAS PRODUCTION		
$^7Li(n,n't)$	10%	thresh-15 MeV in 1 MeV increments
$^{11}B(n,xp)$, $(n,x\alpha)$	15%	14 MeV
$^{12}C(n,x\alpha)$,	15%	14 MeV
$^{56}Fe(n,xp)$, $(n,x\alpha)$	15%	14 MeV

Nuclear Data Evaluations

No simple list has been compiled for evaluation needs, as in the case for measurements. A general plea to ENDF evaluators has been made,[6] asking them to pay particular attention to fusion reactor needs in preparing Version 5 of ENDF. Materials identified as of special interest for EPR's are B, C, Si, Cu, and stainless steel (Fe, Ni, Cr); reactions of particular concern are (n, n'), (n, 2n), (n, n' particle), and (n, xγ) cross sections and spectra at $E_n \leq$ 15 MeV (2.4 pJ). More specific recommendations for ENDF Version 5 were:[6]

1. A new ^{11}B evaluation is needed.*
2. New experimental data and more accurate representations of secondary neutron spectra should be incorporated for ^7Li and ^9Be.
3. Correlated error files should be included for all partial cross sections and secondary spectra of the EPR materials identified above.
4. Gas production and activation data evaluations for Mo, using existing experimental data supplemented by model calculations, should be included. (Mainly for CPR system studies and radiation damage correlation.)
5. An evaluation of the T(t, 2n) ^4He reaction is needed.

A large impact will be made upon evaluation requirements if a D-Li neutron source facility is built. As a first step, the materials and reactions for which data will be needed up to ~ 15 MeV could possibly be satisfied by nuclear model calculations, thus requiring only a few experimental tie points.

Processor Codes

Existing processing codes are generally adequate for producing multigroup cross sections, covariance matrices, kerma factors, photon-production matrices, radiation damage functions (via recoil atom spectra), etc. Processing individual portions of these data independently often produces inconsistencies between, for example, kerma factors and photon-production matrices,[11] resulting in nonconservation of energy. This is usually not due to any deficiency in the individual codes, but rather to data evaluation inconsistencies or omissions. Currently under development is at least one code,[31] NJOY, which performs all the above processing tasks concurrently, forcing consistency by its internal procedures. Most processing codes should prove accurate enough for foreseeable future

*Subsequent, but still preliminary, sensitivity studies for an EPR show[10] a weak dependence of important nucleonic parameters on the ^{11}B cross sections.

requirements, especially as compared to data uncertainties. Future development will be required mainly to adapt to new ENDF formats. process new classes of data (e.g., radioactive decay chains, energy emissions, etc.), and incorporate contemporary CCCC standard interface data files.

Sensitivity and Uncertainty Codes

Codes such as SENSIT-1D[32] and SWANLAKE[33] have been highly developed to provide sensitivities of integral design parameters to cross-section data and material arrangements. Their principal objective in cross-section sensitivity analysis is to provide guidance to experimentalists and evaluators in their attempts to improve the nuclear data. Uncertainty analysis, wherein cross-section sensitivity values are convoluted with cross-section covariance matrices, is of primary interest to the nuclear designer. From the resulting uncertainties in nucleonic parameters he has an integral view of the combined effect of all cross-section errors, always within the limitations of the covariance data reliability and linear perturbation theory. Similarly, the sensitivity codes can easily be used to determine the effects of design changes on various nuclear responses, insofar as the changes are small variations of region boundaries, compositions, densities, etc. (i.e., within linear perturbation theory's applicability).

Significant R & D efforts could be usefully devoted to several areas, including (1) improving the interpretation of results by extracting information in forms most meaningful to experimentalists and evaluators; (2) devising formalisms for calculation of sensitivities to secondary energy distributions, including means for describing shifts in the secondary spectra; and (3) developing the theory to overcome limitations of linear perturbation theory, e.g., incorporate higher-order terms or adopt different variational methods. Work is actively being pursued in all the above three areas, with promise of major advances in the near future - - in time to impact the EPR designs. Sensitivity and uncertainty analysis code systems such as FORSS[34] and LASS,[35] including multigroup covariance processors, are now evolving from the ongoing research.

Extention of sensitivity codes to 2-D is fairly straightforward, and is presently being undertaken. An added benefit from 2-D analysis will be in the area of design sensitivity, where realistic blanket module designs can be treated. Also, both cross-section and design sensitivity of regions near streaming paths such as vacuum ports can provide valuable guidance in selecting and placing local shielding materials.

Before optimum use can be made of these new developments in sensitivity analysis, an extensive data evaluation effort will be required. Not only are error files for most partial cross sections required, but considerable and careful effort will be required to include reasonable estimates of correlations among partials and energy ranges. The ENDF files are presently devoid of such covariance data except for five materials - - C, O, N, Aℓ, and Fe, and even some of these are sparse or not generally available.

INTEGRAL EXPERIMENTS

Integral experiments serve a variety of roles; at one end of the spectrum are "clean" experiments designed to elicit a small amount of high-quality information about a very limited number of cross sections.[36] These and slightly "dirtier" experiments used for data adjustment are important but the subject is too complex for an abbreviated discussion. However, the excellent review of Farinelli[37] is recommended as a comprehensive summary of the role of integral experiments in nuclear data evalutation.

A second class of integral emperiments is one in which generic engineering designs (i.e., selected materials and their laminations) are proof tested in as simple a configuration as possible; i.e., the objective here should be to provide a geometry simple enough to calculate confidently, and then observe the discrepancies with the integral measurements. Such experiments provide a general insight into the adequacy of the combination of cross-section sets and codes; they can also furnish specific engineering data to guide design decisions regarding shield configurations, and for selection of safety factors (degree of conservatism). Justification of this class of experiments is now in question, and strongly depends upon timing. If sufficient development of computational tools and data occur before a detailed EPR engineering design, there should be sufficient confidence in the bulk shield design to obviate the need for more than perhaps one prototypic experiment. Even this experiment may be superfluous if the design tools can be honed on previous experimental results.

It cannot be overstated that benefit should be derived from the experience, including mistakes, of fission reactor experimental programs. That is to say, state-of-art methods and data should be applied to analysis of existing experiments, and new experiments should be designed such that they can be analyzed with existing tools. Experiments that leave one in a quandry which is then used to justify another experiment, ad infinitum, are a technical and economic waste.

The last class of experiments consists of engineering mockups, representing as closely as possible the actual geometric complexities of the blanket/shield. These expensive experiments serve a pure proof-test function, with little hope of resolving calculational versus experimental discrepancies. Their main function is to provide final engineering safety factors, and incidentally to uncover design faults and shoddy calculations.

One area in which either simple prototypic or engineering mockup experiments will almost certainly be required is in streaming paths through voids in the blanket/shield. Neutral beam ports especially are complex and critical enough to justify such experiments.

The experimental techniques, instrumentation and neutron sources to carry out any required experimental program are generally extant. No new development needs to be undertaken, other than design of the experiments so as to make them amenable to analysis with existing tools.

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