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AUTHOR(S): DONALD J. DUDZIAK

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FUSION REACTOR NUCLEAR ANALYSIS METHODS AND APPLICATIONS

by

Donald J. Dudziak
University of California, Los Alamos Scientific Laboratory
Los Alamos, New Mexico 87544, U.S.A.

ABSTRACT

Extensive neutronic and photonic analyses have been performed for conceptual theta-pinch fusion reactors. Characteristics investigated include tritium breeding, neutron/gamma-ray energy deposition, transmutation, primary radiation damage effects, biological shielding, radioactivity, and afterheat. In the process of performing these analyses a nucleonics method has evolved into what is known as the Los Alamos CTR Nuclear Analysis System. Although such a system is being continually improved as the nuclear analysis technology advances, its present capabilities are described in detail. Included in the system are data preparation, neutron and gamma-ray transport (discrete-ordinates and Monte Carlo), and postprocessing codes to determine reaction rates of interest for fusion reactors. A self-consistent set of codes to process multigroup neutron and gamma-ray interaction cross sections, gamma-ray production matrices, and kerma factors from pointwise ENDF data is described. Discrete-ordinates calculations use the one-dimensional DTF-IV code, and a postprocessor code (TR3) has been developed specifically for fusion reactor analysis. Nucleonics studies of laser-fusion reactors have used the same nuclear analysis system, but they may require additional time-dependent capabilities for analysis of shock waves during and after nuclear energy deposition. Specific examples of the system's application to a toroidal Reference Theta-Pinch Reactor design and a Linear Theta-Pinch Reactor hybrid are discussed. Lastly, some future development needs in CTR nuclear analysis are summarized.

INTRODUCTION

Conceptual fusion reactor studies (1-3) at Los Alamos have evolved over the last decade into an engineering design study(4) of a toroidal theta-pinch reactor. Recent work has also included feasibility studies of linear theta-pinch fusion/fission (hybrid) reactors,[5] as well as laser-fusion reactors.[6] All blanket neutron transport calculations have been performed with multigroup discrete-ordinates codes, beginning with the work of Bell,[7,8] who explored various blanket concepts. His work introduced the concept of using beryllium multipliers, which have found renewed interest recently for enhancement of not only tritium breeding but also energy production. In addition, he chose refractory structural materials, niobium and molybdenum, and estimated resonance self-shielding effects[2,8] for niobium.

More recent nucleonic analyses[9-13] of a Reference Theta-Pinch Reactor (RTPR) have included detailed assessments of tritium breeding potential, nuclear heating per D-T neutron and its spatial distribution, primary radiation damage effects, biological shielding, induced structural radioactivity, and nuclear afterheat. The intent of this paper is to discuss the methods used, data sources, and areas of current and future development. A characterization of specific blanket designs has been presented in the references cited, so particular results are included in this paper only for illustration of important neutronic parameters and their uncertainties.

Examples of blanket designs which have been subjected to analysis using the methods described here are given in Figs. 1 and 2. Figure 1 shows a cutaway perspective view of a blanket segment and associated magnet coils for the RTPR. Each segment is 2 metres long, and 100 segments placed radially about the plasma chamber form a 2-m-long module in the toroid. The RTPR, in turn, is composed of 176 such modules, forming a torus of major diameter 112 metres and aspect ratio 112. Inner and outer radii of the blanket are 0.5 m and 0.89 m, respectively; the implosion-heating and compression coils together with their insulation are 0.46 m thick. A biological shield of steel and concrete, not shown in Fig. 1, is placed around the toroidal tunnel containing the blanket, coils, and associated leads, pipes, ducts, etc. Figure 2 is a schematic cross-sectional view of a Linear Theta-Pinch Reactor (LTPR) hybrid blanket. Much less engineering design and optimization work has been performed for the LTPR hybrid, so neutronic studies[5] have been confined to tritium breeding, direct energy production, ^{238}U-to-^{239}Pu conversion, and magnet coil radiation damage. The fertile seed shown in Fig. 2 is essentially an LMFBR blanket in material composition (i.e.,

UO_2/PuO_2 in stainless steel structure), except sodium coolant has been replaced with lithium.

Other CTR-related studies have employed the Los Alamos CTR Nuclear Analysis System. These include presently active studies of a ^{232}Th-^{233}U subcritical, graphite-lattice (thermal) hybrid blanket for a LTPR. Also, evaluations of irradiation test facilities for CTR materials have included calculations of anticipated neutron flux spectra and intensities, transmutation rates in candidate structural materials, and neutron dosimetry. One such study was for a 14-MeV Intense Neutron Source,[14] while others have been for the LAMPF beam stop CTR Radiation Effects Facility.[15,16] Nucleonic analyses by the laser-fusion group at Los Alamos[17] have employed the same CTR Nuclear Analysis System, both the nuclear data and the codes. This same system will soon be applied to shielding and activation studies of a D-T burning theta-pinch test reactor.

NUCLEAR DATA SOURCES AND PROCESSING

With minor exceptions, nuclear cross sections and response functions are used in the nuclear analysis computer codes as multigroup data. No standard energy-group structure suitable for all transport calculations has yet been devised. Five energy structures are currently in use:

i. A 100-group neutron data set as discussed in Ref. 12. These fine-groups are more than adequate for transport of high-energy neutrons, for treatment of the resonance region, and for accurate representation of threshold reactions at high energies [e.g., Be(n,2n), ^{7}Li(n,n')αt, and higher energy (n,2n), (n,p), (n,α), (n,n'p), and (n,n'α) reactions].

ii. A 21-group gamma-ray interaction and kerma data set[18] which is also more than adequate in detail. By the nature of the regularity in gamma-ray interaction coefficients, as well as the large uncertainties in gamma-ray emission spectra, finer detail is not warranted.

iii. A 25-group neutron transport and reaction-rate data set[5] for use in analyzing unmoderated hybrid blankets. Like the 100-group set, it contains only one thermal group and has no upscatter matrix component. Yet the 14-MeV region, fission spectrum, and resonance region are adequately treated.

Above 184 keV are 11 groups with lethargy widths of 0.5 or less (0.11, 0.12, and 0.18 in the top three groups); below they are 1.0, except for group 24 which has $\Delta u = 2.0$.

iv. A 19-group neutron transport and reaction-rate data set for analysis of thermal-lattice hybrid designs. These data have the same 11 groups as the 25-group set from 184 keV to 14.9 MeV, four epithermal groups below 184 keV, and four thermal groups (allowing upscatter) below 2.38 eV. Thus, thermal spectra, resonance self-shielding, and disadvantage factors in ^{232}Th-^{233}U-graphite lattices can be well represented.

v. A special 41-group neutron transport data set for analyzing high-energy neutron spectra at accelerator-target neutron sources (i.e., irradiation facilities). These data consist of eleven groups between 17 and 800 MeV, and thirty groups from 17 MeV down through thermal energies. They have been used for discrete-ordinates and multigroup Monte Carlo analysis[19] at LAMPF.

Transport cross sections for all the above multigroup data sets include scattering matrices up to P_3. The master CTR Multigroup Data File at Los Alamos includes response functions such as kerma factors, activation cross sections, atom displacement functions, and gamma-ray production matrices in the 100-group structure. Special group collapsing codes[20] then reduce these data to the other substructures of 41, 25, or 19 groups.

Basic data sources

Figure 3 shows the CTR Multigroup Data Preparation System as it is presently structured. Basic data sources are shown in the left column of circles, the multigroup processing codes in the next column of rectangles, and the multigroup output in the last column. Most source data are being accumulated in the ENDF system, so all the named codes in the second column accept ENDF-formatted input as an option, if not exclusively. The Los Alamos Master Data File contains an extensive library of neutron data from other laboratory programs, much of which is not yet available in the ENDF system. Often, for *ad hoc* studies (e.g., induced radioactivity in structural materials) multigroup cross sections are processed[20] directly from experimental data reported in the literature or from the results of nuclear model calculations. Also, data have been received in the 100-group structure, principally from the fusion reactor design group[21] at the University of Wisconsin.

Multigroup processing codes

Multigroup processing of ENDF neutron cross-section data is now performed with the MINX code,[22] although much of the existing library was generated in the past with the ETOG code.[23] Among many advantages of the MINX code is the capability to compute Bondarenko f-factors for resonance self-shielding. By simply interpolating tabulated (energy group) values as a function of temperature, T, and the "dilution factor" or "background cross section," σ_0, resonance self-shielded cross sections can be prepared for particular design analyses.

Kerma factors are computed by the MACK code,[24] which uses individual reaction cross sections in ENDF to compute kinetic energy release. Similarly, the DON code[25] has options to process ENDF cross-section data into recoil atom spectra, and to partition energy according to Lindhard theory. Atom-displacement rates can then be estimated (cf. Ref. 12, for example) by applying selected secondary displacement models to the total elastic energy* available, T_d.

Multigroup processing of gamma-ray production cross sections is accomplished with the LAPHANO and LAPHAN codes,[26] which create NxG multigroup matrices directly from ENDF pointwise data (N neutron groups, G gamma-ray groups). Although LAPHAN has the capability of computing Legendre moments of the matrix up to P_4, experience has shown[27] that P_0 is sufficiently accurate for energy-deposition calculations.** LAPHAN is often used in its P_0 mode, rather than using the P_0 code LAPHANO, because of programming efficiencies unique to LAPHAN. Photon interaction cross sections for multigroup gamma-ray transport calculations are prepared by the GAMLEG-X code, a version of the GAMLEG code[28] that treats pair production as a $(\gamma,2\gamma)$ reaction with isotropic emission.

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*Energy imparted to electronic excitation may also be of interest for investigations of electrical insulator response to radiation in theta-pinch reactors.

**Also, at this time few ENDF evaluations contain angular distributions for gamma-ray emission.

In addition to the schematic system shown in Fig. 3, there are the usual service and file management routines. While all data are stored on mass storage devices (e.g., tapes, disk, and photostore), input to some transport and reaction-rate codes is most efficiently (in file-searching time) done from special files or cards. Figure 4 shows the principal inputs as they are structured for the Los Alamos CTR Nuclear Analysis System.

Present and future developments

Significant deficiencies occur in the nuclear data available for nucleonic analyses, and many of these deficiencies are the object of national or international research and development programs. For example, cross-section error estimates are projected to be included in future ENDF evaluations. Cross-section sensitivity studies, principally by first-order perturbation theory, are an essential part of any general assessment of cross-section measurement requirements. Such studies depend upon rational estimates of probable errors in particular reaction cross sections in ENDF. At present, the error estimates are usually made by the investigator interested in the sensitivity of blanket parameters such as total recoverable energy release, tritium breeding, or primary radiation damage effects, with no formal error information from the cross-section evaluator. Extensive fusion-reactor blanket sensitivity studies are just now getting under way, and for the next few years they will still depend heavily on informal error estimates.

Particular problem areas, due either to complete lack of data or to known inadequacies, have already been identified. For example, molybdenum is considered among the prime candidates for fusion reactor structures, yet charged-particle reaction data are not yet available in ENDF. As a result, meaningful kerma factors and primary radiation effects response functions cannot be computed by the MACK and DON codes, respectively. Additionally, secondary energy-angle distribution from Be(n,2n) reactions have an unacceptable uncertainty. Sensitivity studies of blanket parameters as a result of variations in these distributions[29] have shown possible large (5 to 10%) effects on tritium breeding, energy deposition, and radiation damage. Because of the renewed interest worldwide in beryllium multipliers (cf. Ref. 30, for example), measurements of secondary neutron spectra at incident neutron energies of 14 MeV and below are planned for the near future.

Another area of development anticipated for the near future is that of creating a minimal-sized neutron energy group structure that will preserve selected integral blanket parameters. In order to accomplish this task, the range of blanket studies must be delimited (e.g., should thermal-spectra hybrids be included?), and criteria for acceptable errors in integral parameters such as energy release must be formulated. Care must be exercised, expecially with respect to spectrum averaging over resonance and high-energy threshold reactions.

NEUTRON AND GAMMA-RAY TRANSPORT AND REACTION RATES

An overview of the CTR Nuclear Analysis System is given in Fig. 4 for the discrete-ordinates calculational path. Not included in the figure are the Monte Carlo and perturbation-theory paths. In general, one-dimensional cylindrical geometry calculations have been sufficient for the blanket studies undertaken so far. Multidimensional S_n and Monte Carlo codes will be necessary, however, when detailed engineering designs define the geometries of coolant and refueling ducts, compression-coil gaps, etc. Extensive experience with these codes has been accumulated over the last decade in fission reactor analysis,[31] so the capability exists when needed. Several first-order perturbation codes (e.g., cf. Ref. 32) for blanket sensitivity studies (i.e., to material, dimensional or cross-section perturbations) likewise exist. In particular, codes[33] linked to output of the DTF-IV discrete-ordinates code[34] are used at Los Alamos.

Monte Carlo codes are routinely applied in analyses being performed for the LAMPF Radiation Effects Facility. Neutron production and transport calculations for neutrons with energies above 20 MeV employ a local version of the NMTC code.[35] Successive transport of $\leq$20-MeV neutrons is computed by the MCN continuous-energy Monte Carlo code.[36] The latter code will also be applied to neutron streaming problems in laser-fusion reactors.[17]

Discrete-ordinates calculations

One-dimensional discrete-ordinates codes have been universally applied to fusion-reactor blanket neutronics and photonics. The present production code, DTF-IV, is a FORTRAN-IV direct descendant of the DSN code used by Bell in his original (1965) blanket studies.[7] Extensive discussion of discrete-ordinates theory, advantages, disadvantages, geneology, and state-of-the-art can be found in a review paper[37] by Lathrop. Suffice it to say here that the current generation of such codes are highly developed and efficient in terms of computer running time. As a result, most parameter studies on blanket neutronics have used repeated forward solutions as opposed to adjoint solutions and perturbation theory. Typical CDC 7600 running times for RTPR design calculations were ≈5 minutes.

The RTPR neutronic model discussed in Ref. 12 consisted of 100 energy groups, P_3 cross-section expansion, 133 spatial mesh points, 36 regions, and 10 materials. A one-dimensional radial traverse through the blanket segment shown in Fig. 1 was used for these calculations. Some resulting neutron flux distributions through the blanket and coils are shown in Fig. 5, which illustrates the penetration of neutrons >10 MeV (mostly ≈14 MeV) through the blanket, as well as the buildup of lower-energy neutrons in the graphite and coil regions. A reference neutron wall loading, I_w, of 2.0 MW/m^2 has been chosen for normalization of all fluxes and reaction rates in the RTPR.

Table I illustrates a typical parameter study performed for the RTPR design analysis. Because of the large energy storage and transfer requirements of the RTPR (≈177 MJ/m in the magnetic field), as well as joule losses, a strong incentive exists for minimizing blanket thickness. One way to accomplish this objective is to examine the tradeoff between decreased energy losses in the magnetic energy storage and transfer system versus decreased recoverable thermal energy per D-T neutron, as the blanket is thinned. Coil heating and radiation damage concurrently increase also. To a first approximation, the increased waste (≈300 K) heat in the coils is equal to the decreased recoverable (≈1100 K) heat in the blanket. Tritium breeding has been shown[10] to be easily maintained at a value >1.0 by adjusting the beryllium-region thickness, so values of T < 1.0 in Table I are not of concern. Similarly, the recoverable and waste energies will both increase with increased beryllium thickness;[10] e.g., a 10-mm increase in beryllium thickness provides a 1.1-MeV increase in recoverable energy per D-T neutron and a 0.07 increase in tritium breeding ratio. Coil heating and transmutation rates, given as percent changes in Table I because they are independent of wall loading, vary linearly with small (20 mm) changes in graphite thickness as shown for a niobium structure. Thus, first-order perturbation theory involving only two transport calculations, forward and adjoint, could provide this same information. However, for investigations of molybdenum structures[12] large changes of the graphite region were considered, and the effects were nonlinear as can be seen from Table I. In general, application of perturbation theory to moderate (10 to 20 mm) changes in adjacent beryllium and lithium regions of the RTPR, and to substitution of structural materials, has proven successful.[32] Unlike Δk_{eff} calculations in fission reactors, for example, perturbation theory has not been found necessary in order to circumvent convergence problems that mask small changes in an integral blanket parameter.

The system shown in Fig. 4 implies separate transport calculations for neutrons and gamma rays, although such calculations can be performed simultaneously by using coupled multigroup cross-section sets. Which path to take is a matter of choice, but with complex blankets and large numbers of energy groups, computer storage capability rapidly becomes limiting. This fact, coupled with the need for many neutron transport survey calculations prior to the need for gamma-ray transport, has led to the adoption of separate calculations in most cases. In fact, even for 100-group neutron transport, a premix of region materials (PREMIX code in Fig. 4) is performed prior to operation of the DTF-IV code; this is simply a device to allocate the maximum possible storage to finer meshes (energy, angle, and space).

Similarly, postprocessing of the output flux files is most conveniently performed in a separate operation. For this purpose, a series of code modules, generically denoted TR3, were developed as the need arose. Typically, a reference design calculation of the neutron flux distribution will be used in repeated reaction rate calculations for varying wall loadings, operating and shutdown time in the case of radioactivity, etc.

Most fusion reactor transport calculations are simply inhomogeneous fixed-source calculations, which are performed to a pointwise flux convergence criterion, in most cases $\approx 10^{-4}$. Hybrid studies,[5] however, have also involved eigenvalue calculations to determine degree of blanket subcriticality, including postulated accident cases.

Numerous studies[10,38-40] of order of quadrature in S_n, order of Legendre expansion of cross sections, and source spatial and energy distribution have been carried out for nonfissile blankets. With minor exceptions such as reaction rates at deep penetrations, an S_4-P_3 transport calculation predicts integral parameters of interest (e.g., tritium breeding) to within ≈2% of a reference high-order calculation (S_{16},P_5). Similarly, variation of source spatial[38] and energy[40] distributions within physically reasonable bounds has a negligible effect on integral parameters. Studies of fusion reactor neutronics at Los Alamos have thus consistently included the assumption that the D-T neutron source is spatially uniform in the plasma and is Doppler-broadened[40] based upon an ion temperature of 19 keV.

Postprocessor development

The TR3-series of code modules have been specifically tailored to analysis of cylindrical fusion-reactor blankets.* Output includes quantities such as (i) fluxes within specified energy bands; (ii) transmutation, primary radiation damage and other reaction rates by space point and energy group, as well as summed over energy groups; (iii) kerma by space point and material, and integrated over specified regions (i.e., recoverable and waste thermal energy can be separated); and (iv) gamma-ray sources by space point and (gamma) group.

A particularly specialized module is TR3A, which computes total and specific (per watt of thermal output, Wt) radioactivity inventories, specific biological hazard potentials (km^3/Wt), and afterheat power fraction of operating power. Figure 4 depicts schematically the calculational flow, as well as the principal input data required for each module. An additional input is the plasma neutron emission rate ($m^{-1}s^{-1}$), which is used to normalize all fluxes and reaction rates to the reactor power level. For example, Fig. 5 was traced from TR3 output plots for a neutron wall loading of 2.0 MW/m^2, or a plasma neutron emission rate of 2.804 x 10^{18} $m^{-1}s^{-1}$. Parametric studies of RTPR radioactivity as a function of wall loading, operating time, and structural material (Nb-1%Zr and V-20%Ti) have been performed[41] with TR3A.

*Versions modified by the laser-fusion group handle spherical geometries also.

MAJOR UNCERTAINTIES AND REQUIRED DEVELOPMENT

Sensitivity studies

Cross-section sensitivity studies will soon be placed on a systematic basis for the major fusion reactor concepts, in order to formulate a long-range plan of cross-section measurements and evaluations. A useful output format[42] from first-order (linear) perturbation codes is the fractional change of a reaction rate, δR, for a given fractional change in a selected cross section, $\delta\Sigma$, as a function of neutron energy. This gives a "sensitivity profile"[42] useful for determining energy ranges over which evaluation and measurement efforts should be concentrated. While recent work has shown that tritium breeding is not usually a critical blanket parameter, the application of this technique to energy release per D-T neutron has significant import in that net electrical output is directly proportional to energy release.

Another important class of sensitivity analysis is that associated with system studies of particular blanket designs. Variations in structural materials, moderators, lithium compounds and region dimensions all need to be explored in blanket optimization studies. For given blanket integral parameters as discussed above, trajectories in parameter space starting at a reference design point can best be determined by perturbation theory. While the perturbation may appear in the Boltzmann equation as changes in total or differential scattering cross sections, they represent actual physical changes in design as opposed to uncertainty estimates.

Specifically of interest in theta-pinch fusion reactor system studies are the minimum blanket thicknesses consistent with tritium breeding, recoverable thermal energy, and coil radiation damage. A beginning of this investigation was presented in Table I. Also, resonance self-shielding[43,44] and temperature effects[44] in niobium are being evaluated to determine their influence in the RTPR, primarily on tritium breeding and recoverable energy per D-T neutron. The self-shielding and temperature broadening calculations[44] utilize Bondarenko $f(T,\sigma_0)$-factors from the MINX code.

Additional investigations are under way at LASL to examine the uncertainties in ^{94}Nb (2.0×10^4 y) long-term radioactivity and ^{95}Nb(35.1 d) short-term radioactivity and afterheat. Available resonance data[45] for the ^{94}Nb(n,γ) cross section have been applied to possible models for the distribution of resonances above ≈22 eV (known resonance parameters are at 11.6 eV and 22.6 eV, and the resonance integral has been measured). Three reasonable distribution patterns have been chosen, all consistent with the resonance integral, to bracket the expected behavior of the cross section in the resonance region. These three cross-section sets are now being applied to calculations of RTPR radioactivity and afterheat.[46]

Hybrids

Conceptual design and analysis of a linear theta-pinch fusion/fission hybrid reactor based upon a ^{232}Th-^{233}U cycle is now under way. A thermal neutron spectrum is most advantageous for this cycle for several reasons, including: (i) low production of radiologically undesirable isotopes (e.g., ^{231}Pa, ^{230}Th, ^{232}U) relative to a fast neutron spectrum; (ii) a relatively high value of η (the average number of neutrons emitted in fission per neutron absorbed) for ^{233}U at thermal energies; and (iii) smaller fissile inventories required per unit electrical power output, with attendant radiological and economic advantages. Critical fission reactors fueled by ^{233}U produced in hybrids have some of these same attractions *vis-à-vis* fast reactors fueled by ^{239}Pu. On the negative side, ^{232}Th does not have the advantage of a high fission cross section for ≈2 to 14 MeV, as does ^{238}U; also, the (n,2n) and (n,3n) reactions lead to radiologically objectionable isotopes. Thus, all considerations lead one to a thermal spectrum in the blanket fertile seed. As a result, the reactor physics of these blankets must encompass a correct treatment of the entire neutron energy range from thermal to 14 MeV. Thermal spectra must be calculated in fine detail, using appropriate scattering kernels, and disadvantage factors in heterogeneous fuel beads and/or rods must be estimated. Resonance self-shielding in the fertile/fissile materials is likewise important. Because advantage cannot be taken of the 14-MeV neutron energy for fast fission of ^{232}Th, current design concepts under study at Los Alamos use those neutrons as effectively as possible for tritium breeding via ^{7}Li(n,n'α)t reactions and Be(n,2n) neutron multiplication.

Radiation damage

Primary radiation damage effects can be predicted for most materials of interest in existing fusion reactor studies,[12] with the exception of molybdenum. These effects include damage energies (and consequent atom displacements, depending upon what model is chosen), as well as transmutations and gas production (H and He). However, atom-displacement response functions still need to be calculated with the DON code for Al_2O_3, Be, Cu, and some of the newer materials being considered (e.g., Li_2O).

Atom sputtering or chunk ejection by 14-MeV neutrons is potentially the most limiting form of radiation damage, at least in niobium,[12] yet sputtering coefficients are still not well established.[47]

Miscellaneous nucleonics

Bulk biological shielding is not a critical area of either pure fusion or hybrid reactor design, and has consequently received little attention.[11] However, detailed shield design calculations will be required for the first experimental, D-T burning, theta-pinch test facility. Even more important, activation of the reactor structure will affect accessibility for experimentation and maintenance, and thus must be accurately determined. Theta-pinch reactors, unlike Tokamaks, have no superconducting magnet coils requiring shielding, but energy deposition and primary radiation effects in the room-temperature coils must be determined accurately.

Traditional nucleonic interests such as coolant activation and equilibrium activity, deposition of long-lived radioactivity on primary system components (with attendant shielding, maintenance, and disposal problems), and dispersion of radioactivity into the environment have only begun to be addressed.

CONCLUSIONS

A relatively large effort has been invested in a CTR nuclear data and computer code system for fusion reactor nuclear analysis. Successful application of this system has been made to analysis of a toroidal Reference Theta-Pinch Reactor (RTPR), a Linear Theta-Pinch Reactor (LTPR) hybrid, and radiation test facilities. The system further serves as a skeleton for further developments necessary in anticipated future system studies. Most efforts at the present time are being devoted to sensitivity studies as part of a postdesign assessment of the RTPR. Analyses are being focused on identified uncertainties caused by ^{93}Nb resonance self-shielding and temperature effects, $^{94}Nb(n,\gamma)$ cross-section assumptions, and primary radiation effects in niobium, alumina, and copper. Sensitivity analyses are being applied in an effort to minimize the RTPR blanket thickness. Hybrids based upon the LTPR concept are being evaluated also.

Future developments in the CTR nuclear data and analysis areas depend upon the trends of reactor systems studies, but must also anticipate needs of such studies. Specifically, first-order perturbation methods are being instituted for design sensitivity investigations, as well as for more general cross-section sensitivity studies in assessment of data needs. Also, a capability to calculate thermal lattice parameters accurately is being developed in conjunction with ^{232}Th-^{233}U hybrid system studies.

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TABLE I

EFFECT OF GRAPHITE THICKNESS ON TRITIUM BREEDING, COIL HEATING, AND COIL TRANSMUTATION IN THE RTPR

STRUCTURE	OUTER GRAPHITE (m)	T[a]	COIL HEATING % CHANGE[c]	Cu(n,2n) % CHANGE[c]	Cu(n,γ) % CHANGE[c]
Nb[b]	0.1474	1.108	—	—	—
	0.1374	1.106	9.0	10.	9.0
	0.1274	1.105	18.	20.	18.
Mo	0.1474	0.924	—	—	—
	0.1274	0.943	20.	22.	19.
	0.1074	0.961	42.	50.	40.
	0.0874	0.979	69.	83.	66.

(a) Including an estimated[10,32] correction of 0.07 for resonance self-shielding.

(b) RTPR reference design case, for which waste heat (>99% in the coils and insulation) is 0.80 MW/m and recoverable heat is 9.6 MW/m.

(c) Relative to 0.1474-m thickness.

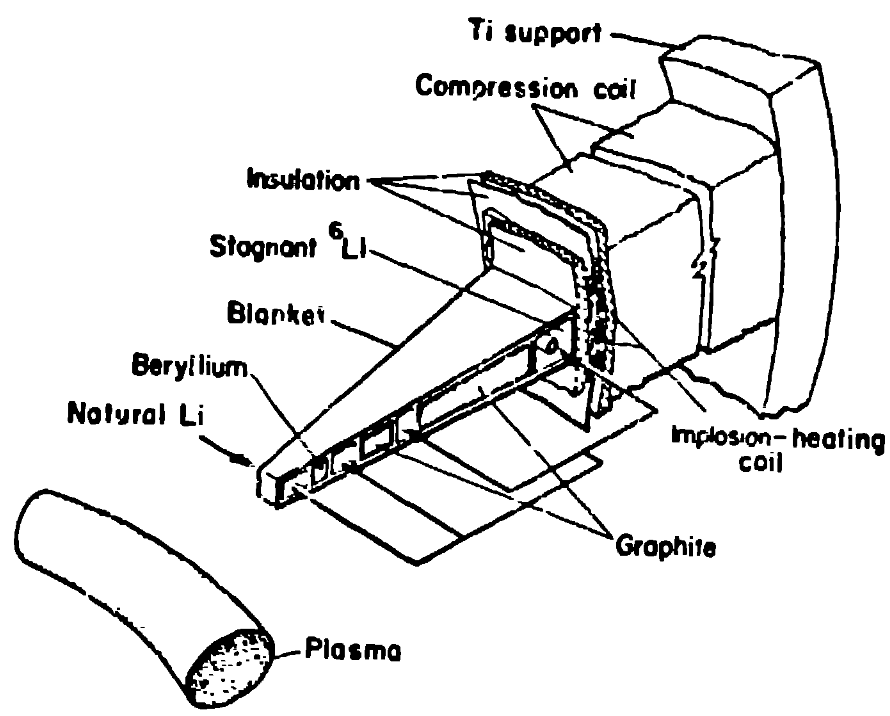

FIG. 1. PERSPECTIVE VIEW OF AN RTPR BLANKET SEGMENT AND MAGNET COILS

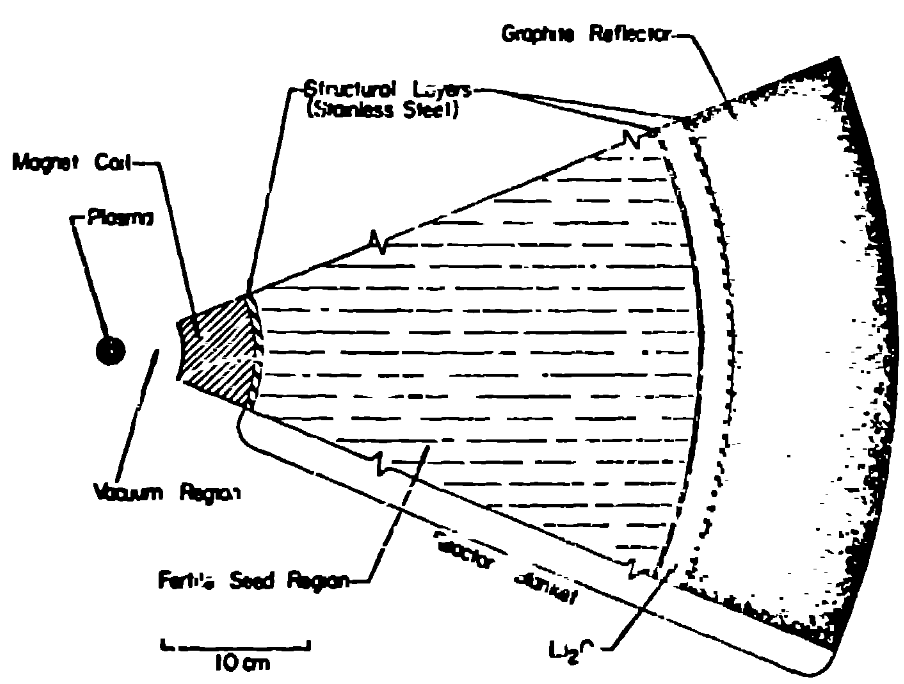

FIG. 2. SCHEMATIC CROSS-SECTION VIEW OF LTPR COIL AND BLANKET

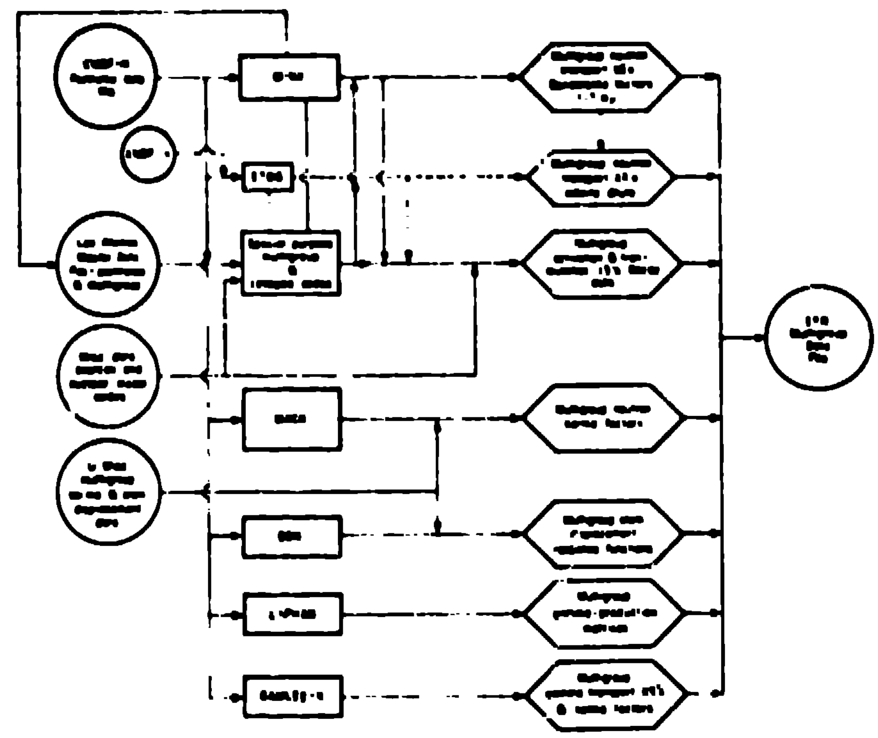

FIG. 3. CTR MULTIGROUP DATA PREPARATION SYSTEM

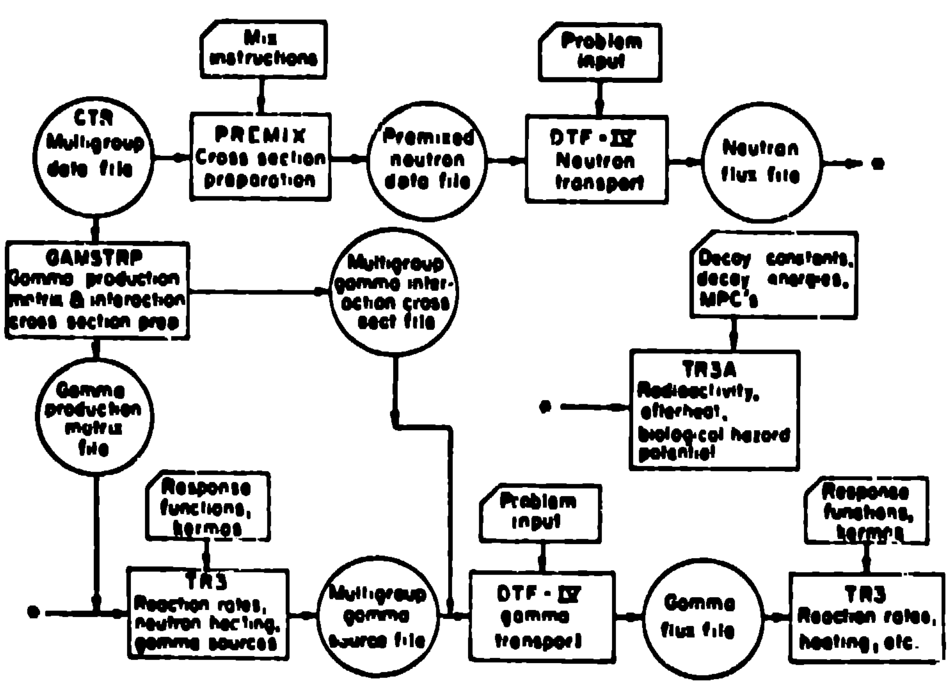

FIG. 4. CTR NUCLEAR ANALYSIS SYSTEM

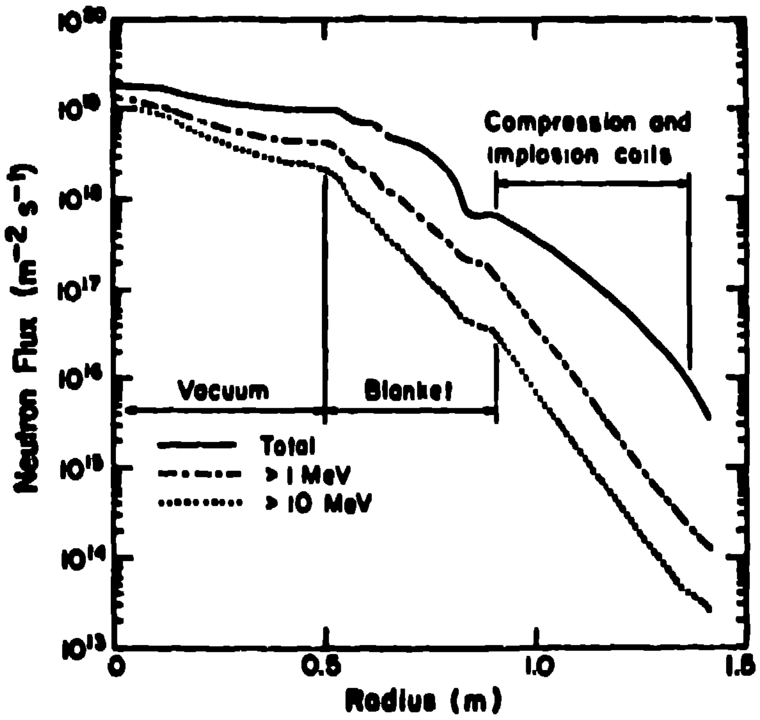

FIG. 5. NEUTRON FLUX DISTRIBUTIONS IN THE RTPR AT I_w = 2.0 MW/m^2