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A COMPARISON OF 1-, 2-, AND 3-DIMENSIONAL MODELING
OF THE TFTR FOR NUCLEAR RADIATION TRANSPORT ANALYSIS

By

L.P. Ku, J.G. Kolibal, S.L. Liew

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L. P. Ku, J. G. Kolibal, S. L. Liew
Plasma Physics Laboratory, Princeton University,
Princeton, New Jersey 08544

ABSTRACT

The computational models of the TFTR constructed for the radiation transport analysis for the Q~1 demonstration are summarized and reviewed. These models can be characterized by the dimensionality required to describe the geometry, and by the numerical methods of solving the transport equation. Results obtained with these models in the test cell are compared and discussed.

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I. INTRODUCTION

During the course of analyzing the radiation characteristics for the TFTR, computational models of varying complexity have been used in the neutron transport calculations. These models were derived by retaining the essential portions of the machine and imposing certain degrees of symmetry to achieve savings in computational time. The choice of a model depended upon the accuracy desired and the quantities analyzed; inevitably this choice was based on the availability of computational tools and resources. This report reviews and summarizes the major models constructed for the TFTR tokamak system that have been the bases for various neutron transport analyses for $Q=1$ operation. Results in the test cell obtained with these models are compared and discussed.

Although many attempts have been made in the past to compare results from different modelings for fusion reactors,¹ there are notable differences from the effort here. The analysis done for TFTR covered a wide range of geometrical and energy spaces; on many occasions the differential quantities are required, and are of major interest. Most comparison studies for fusion reactors are concerned with integral quantities near the tokamak. Furthermore, the TFTR has a high leakage configuration for the neutrons, as opposed to the other comparison studies dealing with conceptual designs, which normally have a high percentage of blanket and device shield coverage. This difference makes the ability to model TFTR in one-dimensional geometry particularly interesting.

II. COMPUTATIONAL MODELS AND METHODS

A. General Considerations

Figures 1 and 2 show two cuts of the TFTR system in engineering detail, as a reference for comparison. Clearly, the myriad pipes, wires, and instrument arrangements make a complete modeling impossible. In fact, only the major components can be retained in even the most definitive studies. The distributed materials can be treated at best as perturbations. The degree of simplification depends on the analysis requirement to describe the geometry. Since the analysis depends on the numerical procedures to solve the neutron transport, the models have to be compatible with the methods and the codes used for solving the problems. Typically, the geometry can be characterized by the dimensionality involved in the description.

On the most sophisticated level, analysis involving a three-dimensional representation of the tokamak is needed. The only numerical method generally available for this purpose is the Monte Carlo simulation. The Monte Carlo solution to the radiation transport equation has the advantage of geometrical flexibility, and also provides a direct picture of the particle histories. However, since Monte Carlo calculations can only give reliable results in limited regions of phase space, they have been used primarily for establishing the baseline reference values, analyzing problems which cannot be reduced by symmetry considerations to lower dimensions, or where a high level of accuracy is desired locally.

Modeling the machine in two-dimensional geometry results in intermediate complexity. The two-dimensional description lends itself readily to the available deterministic method of solving the transport equation, such as the discrete ordinates technique. The discrete ordinates solution to the neutron transport equation is an efficient method of obtaining fluxes and reaction rates at a large number of spatial mesh points. Thus, it is ideal for characterizing the global radiation environment.

Obviously, the simplest model is to represent the machine in one-dimensional geometry - for example, as concentric spheres. Such simple models are extremely useful for many problems, such as scoping studies for the material attenuation properties, or for the calculations of radiation responses far away from the tokamak. One-dimensional calculations usually give reasonable results, although the geometry seems unrealistically simple.

B. Description of the TFTR Models

1. 3-D Modeling

The three-dimensional description of the TFTR was a Monte Carlo model compatible with MCNP² code. The MCNP code offers the option of using up-to-date nuclear cross-section libraries in pointwise energy grids, thus eliminating the errors associated with the multigroup procedure used by the deterministic method or by some other Monte Carlo codes, such as MORSE.³

Figure 3 shows the horizontal and vertical cuts of the basic model used here for the benchmark comparison. Using specular reflection, only half

of the machine was modeled. The torus vessel, the toroidal field (TF) coils, the shear compression panels, and the building walls and floors were included. The test cell was modeled as a right cylinder with a major radius of 17 m. Boration was applied on the floors, but not on the ceiling and the walls. This arrangement is consistent with the projected operations until 1988. The poloidal field (PF) coils, the central post, the support columns, the bellows segments, and the internal hardware, such as the limiters and cover plates, have not been included. Since the modeling is primarily for the study of the radiation in the test cell, particularly at the midplane, omission of those components should not significantly affect the results.

The compositions of the components used in the 3-D calculation are listed, along with those used in the 1- and 2-D analyses, in Table I. The MCNP calculations have been carried out using pointwise cross sections.

The results from this model were used to benchmark the 1- and 2-D calculations and to investigate the errors associated with the multigroup cross-section data. The model has also been used to derive angular distributions of the neutron current for the study of the noise levels induced by the nuclear radiation in the diagnostic instruments,⁴ and for the calculation of the radiation streaming through test cell floor penetrations.⁵ Modified versions of this model have been applied to the design studies of the diagnostics shielding,⁶ and to compare results obtained by measurements in the basement.⁷

2. 2-D Models

The two-dimensional descriptions of the TFTR were discrete ordinates models compatible with the DOT code.⁸ Since DOT can only solve problems in two dimensions, a preferential selection of the axis of symmetry is required in order to represent a three-dimensional machine. There were two basic 2-D models created. One, which has been called the Entire Toroidal (ET) model, has a rotational symmetry line on the toroidal axis. The other has the rotational symmetry line along the minor axis, and was referred to as the Segmented Poloidal (SP) model. The ET model extended the spatial domain to cover the entire basement and the outer surfaces of the walls. Details of the models and the results of the analyses have been summarized and compiled in Ref. 9. Only a brief summary of the models will be given here.

The ET model included most of the machine major components which are toroidally contiguous. Figure 4 illustrates the entire model and Figure 5 shows the details near the tokamak. The poloidal field/equilibrium field coils were faithfully reproduced. The TF coils, by virtue of the discrete arrangement, cannot be modeled anywhere except at the inboard portion where the coils converge. The discrete shear compression panels were modeled as contiguous rings, thus partially taking neutron attenuation by the coils into account. The model, including the full basement, the ceiling, and the walls, had about 50,000 mesh points. The results from this model have been used to generate flux and reaction rate contours in the test cell and in the "perfectly shielded" basement. The data were also used extensively to examine the radiation environment around the entire TFTR site. In addition, the scalar flux spectra were coupled to the penetration calculations,¹⁰ and to the initial diagnostic shield "box" analysis.⁴

The SP model, as shown in Fig. 6, was a cylindrical representation of half of a typical bay. The torus was stretched into an infinitely long cylinder with a symmetric arrangement of the TF coils and the bay openings. Because of the distortion of the curvature of the major circle of the tokamak, the TF coil orientation was also distorted, i.e., the coils were parallel to each other instead of converging towards the center. The advantage of this model is that the internal hardware, which is usually arranged in the poloidal direction, could be included. Including the internal hardware is important for the global characterization of the induced radioactivity. The disadvantage of this model is that the toroidal curvature was not properly represented, and toroidally contiguous objects could not be included. Furthermore, the floor and the wall could not be separated in this model, resulting in the difficulty of compromising both the material and the location for the wall and the floor. This model has been used primarily for deriving quantities near the machine. In particular, it was used extensively to map the spatial variations of the induced radioactivity near the TFTR.¹¹

The major differences between the ET and 3-D models are: (1) the representation of the TF coils, viz., the ET model did not have the outboard legs; (2) the representation of the vessel and the associated internal hardware, viz., the ET model includes a carbon limiter, and the vessel was modeled as a square; and (3) the surface boration, viz., the entire inner surface of the test cell in the ET model was covered by some forms of boration (proposed for the original TFTR program). The major differences between the SP and 3-D models are: (1) the toroidal curvature for the vessel and the TF coils was not properly accounted for in the SP modeling, (2) the floor and

ceiling were combined with the test cell wall in the SP modeling, and (3) boration was assumed for the wall in the SP model.

All the discrete ordinates calculations were performed using a (25n, 12γ) group structure and the PPPL working data library, PRAX.¹² This group structure is given in Table II.

3. 1-D Models

The one-dimensional description of the TFTR was a spherical representation of the entire facility, compatible with either the discrete ordinate code ANISN¹³ or the Monte Carlo code MCNP.

Figure 7 illustrates schematically the one-dimensional arrangements. The major radius of the vessel was reduced such that the neutron current density across the vessel wall was the same as that for the 3-D geometry. The center of the sphere was filled with a homogenized coil/case material to simulate the inboard cluster of the structures. Concentric shells were modeled to simulate the vessel, the air, the wall, and the floor. A shell of homogenized coil/case material, with a thickness of 0.5 meters and a density factor of 10%, was included in the modeling to simulate the distributed material and the TF/PF coils presented in the outboard section. Because of the spherical symmetry, the walls and the floor had to be uniform in composition and thickness. Corrections due to incomplete coverage of the surface boration can be obtained from other studies.¹⁴

The one-dimensional model has been used extensively for material studies, and for the source term calculations in the 3-D labyrinth design analysis.¹⁵

III. RESULTS AND COMPARISONS

A. Comparisons of Differential Quantities

Graphic comparisons of the neutron flux spectra at the vessel are given in Fig. 8 and at the pedestal edge in Fig. 9, both for a DT neutron source. Tabulated comparisons for the same locations, both for DD and DT sources, are given in Tables III to VI. All the comparisons are made using the PPPL 25 group structure, although the Monte Carlo calculations were performed using the continuous cross section option. For the Monte Carlo calculation, the vessel results were the volume averaged fluences (track length estimates), whereas the fluences at the pedestal edge were averaged over a cylindrical surface (surface crossing estimates) which is 1 m high and symmetric with respect to the machine midplane and major axis. For the 2-D ET model, the vessel results are given at the midplane for both the inboard and outboard segments, instead of the vessel average. For the 2-D SP model, the results shown are those averaged over the two sections, the plate section and the bellows, which are in the shadow of the TF coils. For the one-dimensional case, the results are also the volume averaged fluences.

1. Neutron Spectra at the Vacuum Vessel

Examining the spectra at the vacuum vessel shows that, in general, the results agree well for fast neutrons, both at the source energy and in the region between 100 keV and 5 MeV. Since the dominant collision mechanism which leads neutrons into the ~ 1 MeV region is the rapid loss of energy due to inelastic scattering, it is the amount of metal around the vessel that is significant in the determination of the spectrum for the fast neutrons.

Examining the fluences at the vessel also reveals that there are two regions where differences exist, one in the vicinity of 10 MeV, the other below 100 keV. Neutron transport in both regions involves many elastic collisions.

First, consider the region in the neighborhood of 10 MeV. Since the multigroup discrete ordinates method involves approximation of the scattering angular distribution, the one-dimensional spherical model was run using MCNP, which uses exact distribution functions in order to study the effect. Comparisons of the results obtained from MCNP and ANISN analyses show that about 75% of the difference in the flux between 7-11 MeV could be due to the inadequacies of representing the forward elastic scattering and the redistribution of neutrons using finite discrete angles and Legendre polynomial expansions in the discrete ordinates calculations. By putting additional layers of material inside the vessel in the one-dimensional model, on the other hand, it could be shown that including internal hardware, such as the limiter or the cover plate in ET and SP models, contributes no more than 30% difference in this energy range.

The difference in the fluences below 100 keV are more significant. Obviously, the difference below ~ 1 eV is primarily due to the difference in boration. However, in the resonance region, since the multigroup data set has a very coarse structure, accurate calculations of group fluxes require that correct weighting functions be used when the group constants are derived from the pointwise cross sections. Although the weighting functions used to derive the PRAX cross sections were based on the representative spectra in a one-dimensional geometry,¹² the extent that these spectra are representative in a complex, heterogeneous material cannot be known a priori. The comparisons show clearly, however, that accurate flux spectrum information can only be obtained if the detailed material distribution is properly accounted for.

2. Neutron Spectra at the Pedestal Edge

Comparison of the neutron spectrum at the edge of the pedestal shows that the general agreement is reasonably good for the results obtained from different models. Again, the difference towards the low energy end is primarily due to the modeling of the boration. The modeling of the outboard TF coils in the 3-D clearly has the effect of increasing the neutron down scattering probability, thus shifting the spectrum toward lower energies relative to the 2-D results. As a consequence, even though the total fluences obtained from different calculations are not much different, the energy dependent responses, such as the dose, would be different. Typically, this is less than 50%. It can also be observed that while using a 10% density factor for the outboard coil material distribution gives reasonable results for DT neutrons in the one-dimensional modeling, it gives too much attenuation for

the DD neutrons near the source energy because of the difference in cross sections at two source energies.

Since the material densities are much lower beyond the pedestal, the flux attenuation is primarily due to the geometrical divergence. It is expected that the above observations will be true for other locations in the test cell.

3. Angular Distributions of the Neutron Currents

Although the discrete ordinates method provides a means of calculating the flux spectra in the entire problem space, it does not provide sufficiently fine angular information associated with the neutron flow. The detailed knowledge of the neutron angular distribution in many instances is very important, for example, in the design of the subsystems involving penetrations. For a tokamak in a test cell, the material density is higher near the tokamak and much lower in the test cell, so that large numbers of scatterings for fast neutrons will occur only near the center of the test cell. The neutrons away from the tokamak will flow outward radially, resulting in an outward current that is highly forward peaked. This kind of information can best be obtained by the Monte Carlo method.

Examples are given in Figs. 10 and 11 showing the angular distributions of the neutron currents, both forward and backward, across cylindrical surfaces which are 1 m in height, symmetric with respect to the machine midplane, and with major radii 5.82 and 10 m, due to a DT source. The results at a major radius of 10 m due to a DD source are shown in Fig. 12,

with finer angular bins for the purpose of illustrating the degree of peaking which occurs at the midplane.

Clearly, the outgoing current is highly forward peaked for neutrons with energy > 100 keV. For the case of DD, the outgoing current for $E > 1$ MeV can be approximately represented by n^{38} , where n is the cosine angle with respect to the surface normal. The angular distributions for the backward current, on the other hand, are almost cosine. Generally speaking, the angular fluxes are almost isotropic only for energies below ~ 1 keV, where the ratio of the outgoing to incoming current ($J+/J-$) approaches ~ 1 . The ratio of $J+/J-$ increases as the energy increases and decreases as the major radii increases. Typically, $J+$ decreases like $1/R^2$, whereas $J-$ remains almost constant. For energies near the source, the flux and current are nearly identical. Even for energies in the 1 MeV region, the forward current is still more than an order of magnitude higher than the backward current. This feature has important implications in the local shield box design in the determination of the thickness required for the side and the back of the box.

B. Comparisons of Integral Quantities

1. Activation Reactions at Vacuum Vessel

The production rates of the major radioactive isotopes, which are important in determining the delayed dose rate near TFTR, are shown in Tables VII and VIII for the DT and DD case, respectively. Since most of the global activation calculations were performed and reported using the SP model,¹¹ only the results based upon this model are compared with the Monte Carlo results.

The Monte Carlo results were based on continuous cross section treatment, both for the transport and for the reactivity calculation. The reaction cross sections were mainly based on the most recent evaluation of the ACTL library.¹⁶ The SP calculations were based on the 25 group version of the MONTAGE data set.¹⁷

Table VII shows that, for the DT neutron case, the results for most of the threshold reactions tend to agree within 30%. The only noticeable difference is the Cr-50(n,2n) reaction, which has a threshold of about 14 MeV. Obviously, the weighting function used in deriving the source neutrons would have much stronger influence on the reaction rate calculation for this type of reaction.¹⁸ For the case of DD neutrons, the effects due to the cross-section averaging and the inaccuracy of the data set are striking for the Ni-60(n,p)Co-60 reaction (~ 4 orders of magnitude difference). An equally interesting reaction is the Cu-63(n, α)Co-60 for a DD source. Although this reaction is exothermic ($Q=1.1$ MeV), the cross section does not become appreciable until the neutron energy is greater than about 4-5 MeV, resulting in significant difference in the prediction of the radioactivities. Since both of these reactions produce the long-lived isotope Co-60, the prediction of long term activity, which is induced by the DD fusion neutrons, may be different by a large factor when the 25 group version of the MONTAGE data set is used instead of the pointwise ACTL library.

For capture reactions, the ratio of the results from the two calculations generally fall within the range of one to four. The obvious reason is the overall boration used in the SP model (proposed for the original TFTR operations). However, since the spectrum near the vessel has not reached

equilibrium due to the high leakage configuration, the contribution to the reaction from the epithermal neutrons tend to decrease the difference that would otherwise be observed if the tokamak system were to be entirely closed.

2. Fluence and Dose in Test Cell

Comparisons of the spatial distributions of the neutron fluence, gamma-ray fluence, and the neutron dose are given in Figs. 13, 14, and 15 for the results obtained by the Monte Carlo simulation and by the 2-D discrete ordinates ET model calculation. The results are clearly consistent and in agreement. Thus, the omission of the outboard TF coils in the 2-D calculation does not give rise to any significant error for the integral quantities. Indeed, examining the 2-D contour lines suggests that although the existence of the PF coils or the lintel in the 2-D model results in local perturbation, the global variation of the radiation is not affected.

The same observation can be made for the 1-D spherical approximation. Using a homogeneous representation of the outboard coils, though slightly underestimating the source neutrons leaking into the test cell, the spatial behavior on the average is much like that obtained by the 3-D calculation. Obviously, because of the spherical symmetry, the one-dimensional calculation will not be able to identify the difference in vertical or horizontal locations. The results when compared with 3-D, however, show that at the plasma midplane, which is about 3 m above the floor but about 13 m below the roof, the difference in the integral quantities, such as the dose or fluence, is less than 50%.

IV. CONCLUSIONS

The major computational models used in the radiation transport analyses for the TFTR system are discussed in this report. Table IX summarizes the models constructed, and the advantage and disadvantage of using these models. Comparison of the results suggests that, for obtaining global scalar quantities for general use, simplified models of the tokamak usually suffice. The study also shows that a simple model can yield results which can be used to check the correctness of more complicated models, and the calculations derived by using these models. Furthermore, improvement of the results based on the simplified modeling may be obtained by performing 3-D transport calculations for the high energy neutrons and renormalizing the 1- or 2-D results to the 3-D results. Considerable saving in computer time may be realized in this synthetic procedure. For example, local fluxes, such as those on the plasma midplane in the test cell, calculated by the 2-D discrete ordinates method may be distorted by the so-called ray effect. Using the 3-D results for the high energy neutrons to renormalize the 2-D fluxes, much better local flux variation can be obtained.

The results also seem to indicate that the modeling and the multigroup representation of the cross section do not significantly affect the fast neutron spectrum. The spectrum below 100 keV, however, depends strongly on the material arrangement as well as the cross-section representation. Thus, the accurate spectrum information can only be obtained if the detailed material distribution is properly accounted for.

The Monte Carlo calculations show that in the TFTR test cell, the neutron angular flux is peaked in the outward direction. The degree of the flux peaking obviously depends on the neutron energy and the spatial location. However, not until the neutron energy falls below 1 keV, where many of the neutrons are scattered by the concrete building wall and floor, does the angular distribution become isotropic. This feature has important ramifications on the analysis of neutron streamings through the test cell floor, and on the analysis of the noise levels for the diagnostic instruments. All these topics are under investigation.

Throughout the discussion, the 3-D calculation has been used as a reference for comparison because it has the least errors in the treatment of cross section, geometry, and the transport process. However, a true benchmark can only come from measurements that are designed to test the validity of each model in different regions of phase space. Sensitivity and perturbation studies are also required in order to realize the effect of omission of various components from even the 3-D model.

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TABLE I
COMPOSITION OF MATERIALS
(atom/b-cm)

Element	Vessel & Column	Bellows	Pseudo- Cover Plate	Coil Casing	Coil TF	Column Filler	Wall and Floor
H	---	---	---	---	4.397-3 ^a	8.246-3	7.830-3
B	---	---	---	---	3.432-4	1.097-3	1.058-3
C	---	2.116-4	---	---	1.840-3	1.052-2	0.0
O	---	---	2.494-3	---	4.798-3	4.322-2	4.396-2
Na	---	---	---	---	---	1.310-5	9.955-4
Mg	---	---	---	---	---	1.882-4	1.486-4
Al	---	3.768-4	2.349-3	---	2.340-4	2.634-4	2.366-3
Si	---	5.430-4	---	1.662-3	1.178-3	1.091-3	1.507-2
S	---	---	---	---	---	8.077-5	5.635-5
K	---	---	---	---	---	1.386-5	6.778-4
Ca	---	---	---	---	4.212-4	1.308-2	2.975-3
T	---	2.123-4	1.180-8	---	---	---	---
Cr	1.738-2	2.151-2	1.422-2	1.616-2	---	---	---
Mn	1.188-3	2.313-4	6.594-4	1.105-2	---	---	---
Fe	6.132-2	2.276-3	2.988-2	5.434-2	---	1.499-4	2.912-4
Co	---	4.313-5	---	---	---	---	---
Ni	6.793-3	5.346-2	3.112-2	2.386-3	---	---	---
Cu	---	---	2.583-3	---	7.215-2	---	---
Zr	---	---	2.082-4	---	---	---	---
Nb	---	2.025-3	2.614-4	---	---	---	---
Mo	---	4.769-3	---	---	---	---	---
Ba	---	---	---	---	---	1.245-4	1.184-4

^aRead as 4.397×10^{-3} .

TABLE I con't.

Element	Boron Paint	Graphite	Center Column Filler	Grout for Lintel	Type 4 Concrete	PPL 308 Concrete
H			1.089-2 ^a	3.483-2	7.830-3	8.246-3
B	1.626-2	---	---	1.661-3	---	1.097-3
C	---	9.226-2	---	---	---	1.052-2
O	2.439-2	---	4.743-2	6.948-2	4.385-2	4.322-2
Na	---	---	---	---	1.048-3	1.310-5
Mg	---	---	3.712-4	1.342-3	1.486-4	1.882-4
Al	---	---	1.776-4	7.745-4	2.389-3	2.634-4
Si	---	---	1.909-2	1.764-2	1.580-2	1.091-3
S	---	---	8.009-5	2.891-4	5.635-5	8.077-5
K	---	---	---	---	6.932-4	1.385-5
Ca	---	---	2.773-3	1.012-2	2.915-3	1.308-2
Ti	---	---	---	9.461-6	---	---
V	---	---	---	---	---	---
Cr	---	---	---	---	---	---
Mn	---	---	---	---	---	---
Fe	---	---	1.071-4	3.970-4	3.127-4	1.499-4
Co	---	---	---	---	---	---
Ni	---	---	---	---	---	---
Cu	---	---	---	---	---	---
Zr	---	---	---	---	---	---
Nb	---	---	---	---	---	---
Mo	---	---	---	---	---	---
Ba	---	---	---	1.889-4	---	1.245-4

^aRead as 1.089×10^{-2} .

TABLE I cont'd.

Element	Homogenized PF Coils	Homogenized TF Coils & Case	TF Coils	Ti-V-Al	304 SS
H	1.868-2	3.700-3	1.436-2 ^a	---	---
B	8.594-4	1.203-4	2.727-4	---	---
C	1.151-2	1.612-3	3.654-3	---	2.417-4
O	1.452-2	2.576-3	8.826-3	---	---
Na	1.918-5	2.686-6	6.088-6	---	---
Mg	---	2.598-5	5.889-5	---	---
Al	8.215-4	1.150-4	2.607-4	5.893-3	---
Si	2.639-3	3.695-4	8.375-4	---	8.622-4
S	---	---	---	---	3.773-5
K	1.588-5	2.222-6	5.040-6	---	---
Ca	1.067-3	1.493-4	3.386-4	---	---
Ti	---	---	---	4.979-2	---
V	---	---	---	2.081-3	---
Cr	---	5.843-3	---	---	1.740-2
Mn	---	3.993-3	---	---	1.189-3
Fe	---	1.995-2	---	---	6.137-2
Co	---	---	---	---	---
Ni	---	---	---	---	6.778-3
Cu	3.843-2	4.632-2	5.945-2	---	---
Zr	---	---	---	---	---
Nb	---	---	---	---	---
Mo	---	---	---	---	---
Ba	---	---	---	1.889-4	---

^aRead as 1.436×10^{-2} .

TABLE II

Neutron 25-group Structure

<u>Group</u>	<u>Upper E (MeV)</u>
1	1.4918+1
2	1.3499+1
3	1.2214+1
4	1.1052+1
5	1.0000+1
6	8.1873+0
7	6.7032+0
8	4.9659+0
9	3.6788+0
10	2.7253+0
11	2.2313+0
12	1.6530+0
13	1.2246+0
14	1.0026+0
15	8.2085-1
16	6.7206-1
17	5.5023-1
18	3.0197-1
19	1.6573-1
20	3.1828-2
21	3.3546-3
22	3.5358-4
23	3.7267-5
24	3.9279-6
25	4.1399-7
	1.0000-11

Gamma-Ray 12-group Structure

1	1.40+1
2	1.00+1
3	8.00+1
4	7.00+1
5	6.00+1
6	5.00+1
7	4.00+1
8	3.00+1
9	2.00+1
10	1.00+1
11	4.50-1
12	1.00-1
	1.00-2

TABLE III

COMPARISON OF GROUP FLUX (n/cm^2) AT VACUUM VESSEL FOR 10^{18} DT NEUTRONS

Group	MCNP	DOT-ET		DOT-SP		ANISN-S
		Inboard	Outboard	Plate	Bellows	
1	1.0+12 (0.2)*	7.6+11	7.1+11	1.0+12	8.2+11	9.1+11
2	4.1+10 (2.3)	6.4+10	4.3+10	4.7+10	4.6+10	3.8+10
3	8.5+9 (5.2)	1.8+10	9.3+9	1.1+10	1.3+10	7.9+9
4	1.1+10 (4.5)	1.9+10	1.7+10	1.7+10	1.9+10	1.2+10
5	2.3+10 (3.1)	5.8+10	4.1+10	4.2+10	4.4+10	3.0+10
6	1.9+10 (3.4)	3.8+10	3.1+10	3.2+10	3.2+10	2.4+10
7	3.8+10 (2.4)	4.7+10	4.2+10	4.0+10	3.7+10	3.5+10
8	5.6+10 (2.0)	5.9+10	5.3+10	5.1+10	4.7+10	4.9+10
9	8.1+10 (1.7)	8.6+10	7.7+10	7.6+10	7.4+10	7.2+10
10	7.5+10 (1.8)	8.5+10	7.3+10	7.4+10	7.4+10	6.6+10
11	1.5+11 (1.4)	1.6+11	1.4+11	1.5+11	1.5+11	1.2+11
12	1.9+11 (1.3)	2.1+11	1.8+11	1.9+11	2.1+11	1.6+11
13	1.7+11 (1.3)	1.8+11	1.5+11	1.8+11	2.0+11	1.4+11
14	1.8+11 (1.3)	1.9+11	1.5+11	1.6+11	1.7+11	1.4+11
15	2.0+11 (1.2)	2.2+11	1.7+11	2.0+11	2.2+11	1.7+11
16	2.6+11 (1.1)	2.5+11	1.8+11	2.4+11	2.5+11	1.8+11
17	5.8+11 (0.9)	5.9+11	3.9+11	5.1+11	5.5+11	4.3+11
18	4.1+11 (1.0)	3.7+11	2.1+11	3.0+11	3.3+11	3.0+11
19	6.1+11 (0.9)	4.0+11	2.1+11	3.4+11	3.9+11	4.4+11
20	3.1+11 (1.3)	1.8+11	8.6+10	9.2+10	9.6+10	3.0+11
21	1.2+11 (1.8)	7.3+10	3.5+10	2.1+10	2.3+10	9.9+10
22	7.8+10 (1.9)	3.3+10	1.8+10	1.7+10	1.5+10	9.3+10
23	4.9+10 (2.2)	1.5+10	9.6+9	9.5+9	8.9+9	5.2+10
24	1.2+10 (3.7)	5.1+9	3.6+9	3.4+9	2.8+9	2.0+10
25	2.5+9 (6.6)	6.3+8	3.9+8	4.6+8	1.9+8	2.6+9
n-Total	4.7+12 (0.4)	4.1+12	3.0+12	3.8+12	3.8+12	4.8+12
γ-Total	2.0+12 (0.4)	2.4+12	2.2+12	1.9+12	1.9+12	2.2+12

*fractional standard error in percent.

TABLE IV
COMPARISON OF GROUP FLUX (n/cm^2) AT Vacuum Vessel for 10^{16} DD Neutrons

Group	MCNP	DOT-ET		DOT-SP		ANISN-S
		Inboard	Outboard	Plate	Bellows	
10	1.3+10 (0.2)	8.8+9	9.1+9	1.3+10	1.0+10	1.2+10
11	2.8+9 (0.9)	3.4+9	2.6+9	2.7+9	2.6+9	2.3+9
12	4.2+9 (0.7)	3.9+9	3.6+9	3.4+9	3.0+9	2.9+9
13	2.3+9 (1.1)	2.3+9	1.9+9	2.3+9	2.7+9	1.7+9
14	1.9+9 (1.2)	2.1+9	1.7+9	2.1+9	2.4+9	1.6+9
15	1.9+9 (1.2)	2.2+9	1.6+9	2.2+9	2.6+9	1.7+9
16	2.3+9 (1.1)	2.2+9	1.5+9	2.2+9	2.3+9	1.5+9
17	4.8+9 (0.8)	4.8+9	3.2+9	4.4+9	4.8+9	3.6+9
18	3.3+9 (1.0)	2.8+9	1.6+9	2.6+9	2.9+9	2.5+9
19	4.7+9 (1.0)	3.8+9	2.0+9	3.1+9	3.6+9	3.7+9
20	2.4+9 (1.3)	1.5+9	7.4+8	8.2+8	8.7+8	2.5+9
21	9.9+8 (1.8)	6.4+8	3.2+8	1.6+8	1.8+8	8.2+8
22	6.7+8 (1.9)	3.0+8	1.7+8	1.1+8	9.2+7	7.8+8
23	4.3+8 (2.2)	1.5+8	9.9+7	6.1+7	5.3+7	4.4+8
24	1.1+8 (3.4)	6.5+7	4.4+7	2.4+7	1.8+7	1.7+8
25	2.6+7 (6.2)	7.8+6	4.5+6	3.5+6	1.3+6	2.2+7
n-Total	4.6+10 (0.3)	3.9+10	3.0+10	3.9+10	3.8+10	3.8+10
γ -Total	8.6+9 (0.5)	8.4+9	8.1+9	6.9+9	6.9+9	8.9+9

TABLE V

COMPARISON OF GROUP FLUX (n/cm^2)
 AT R = 5.82 M, Z = 3.15 M* for 10^{18} DT Neutrons

Group	MCNP	DOT-ET	ANISN-S
1	1.1+11 (1.1)	1.15+11	8.0+10
2	2.8+9 (7.7)	5.3+9	4.8+9
3	8.2+8 (21.0)	1.5+9	1.3+9
4	8.5+8 (13.7)	2.2+9	1.4+9
5	1.5+9 (11.0)	4.2+9	2.6+9
6	1.4+9 (11.5)	3.5+9	2.3+9
7	3.2+9 (8.8)	5.1+9	4.0+9
8	4.0+9 (7.3)	6.1+9	5.7+9
9	5.1+9 (6.0)	9.0+9	8.3+9
10	5.2+9 (6.1)	8.9+9	7.9+9
11	9.8+9 (4.6)	1.6+10	1.3+10
12	1.3+10 (4.1)	2.1+10	1.7+10
13	1.2+10 (4.3)	1.9+10	1.5+10
14	1.3+10 (4.0)	2.1+10	1.5+10
15	1.6+10 (3.7)	2.4+10	1.9+10
16	2.1+10 (3.1)	2.6+10	2.0+10
17	4.7+10 (2.9)	6.0+10	4.4+10
18	3.9+10 (3.0)	3.6+10	3.4+10
19	5.9+10 (2.2)	4.1+10	4.8+10
20	4.0+10 (3.2)	2.3+10	2.9+10
21	2.1+10 (4.5)	1.4+10	1.7+10
22	1.6+10 (5.1)	1.0+10	1.7+10
23	1.2+10 (6.4)	7.0+9	1.6+10
24	7.2+9 (7.7)	3.1+9	1.3+10
25	2.0+10 (6.2)	5.4+8	5.8+10
n-Total	5.0+11 (0.8)	4.8+11	4.9+11
γ -Total	1.5+11 (1.6)	2.8+11	2.1+11

* Coordinates are referenced to centerline of tokamak (R=0.0) and test cell floor (Z=0.0).

TABLE VI

COMPARISON OF GROUP FLUX (n/cm^2)AT R = 5.82 M, Z = 3.15 M for 10^{16} DD Neutrons

Group	MCNP	DOT-ET	ANISN-S
10	1.3+9 (1.0)	1.1+9	7.4+8
11	1.8+8 (3.6)	2.8+8	2.3+8
12	2.6+8 (2.7)	3.7+8	2.6+8
13	1.7+8 (3.4)	2.4+8	1.8+8
14	1.4+8 (3.8)	2.3+8	1.8+8
15	1.5+8 (4.2)	2.4+8	2.0+8
16	1.8+8 (3.6)	2.3+8	1.8+8
17	4.1+8 (2.7)	5.0+8	4.0+8
18	3.3+8 (2.8)	2.9+8	3.0+8
19	5.1+8 (2.5)	4.0+8	4.3+8
20	3.6+8 (4.3)	2.1+8	2.7+8
21	2.1+8 (5.2)	1.4+8	1.7+8
22	1.5+8 (6.8)	1.1+8	1.6+8
23	1.1+8 (6.0)	7.0+7	1.5+8
24	7.0+7 (8.3)	3.2+7	1.3+8
25	2.0+8 (5.0)	5.5+6	5.8+8
n-Total	4.7+9 (0.8)	4.4+9	4.6+9
γ -Total	8.3+8 (2.3)	8.6+8	9.8+8

TABLE VII

COMPARISON OF ACTIVATION REACTION RATE AT VACUUM VESSEL FOR DT
(arbitrary unit)

Reaction	MCNP	DOT - SP	
		Plate	Bel lows
$^{50}\text{Cr}(n,2n)^{49}\text{Cr}$	$2.6+10^{(1)}$	$1.2+10^{(2.2)^{(2)}}$	$9.5+9^{(2.7)}$
$^{52}\text{Cr}(n,2n)^{51}\text{Cr}$	$3.2+11$	$2.7+11^{(1.2)}$	$2.1+11^{(1.5)}$
$^{55}\text{Mn}(n,2n)^{54}\text{Mn}$	$7.9+11$	$8.3+11^{(0.95)}$	$6.7+11^{(1.2)}$
$^{58}\text{Ni}(n,2n)^{57}\text{Ni}$	$2.8+10$	$2.6+10^{(1.1)}$	$2.1+10^{(1.3)}$
$^{65}\text{Cu}(n,2n)^{64}\text{Cu}$	$9.4+11$	$9.7+11^{(0.97)}$	$7.9+11^{(1.2)}$
$^{54}\text{Fe}(n,p)^{54}\text{Mn}$	$4.7+11$	$5.0+11^{(0.94)}$	$4.3+11^{(1.1)}$
$^{56}\text{Fe}(n,p)^{56}\text{Mn}$	$1.2+11$	$1.2+11^{(1.0)}$	$1.0+11^{(1.2)}$
$^{58}\text{Ni}(n,p)^{58}\text{Co}$	$5.6+11$	$5.5+11^{(1.0)}$	$4.7+11^{(1.2)}$
$^{60}\text{Ni}(n,p)^{60}\text{Co}$	$1.4+11$	$1.5+11^{(0.93)}$	$1.3+11^{(1.1)}$
$^{63}\text{Cu}(n,\alpha)^{60}\text{Co}$	$4.8+10$	$4.2+10^{(1.1)}$	$3.5+10^{(1.4)}$
$^{50}\text{Cr}(n,\gamma)^{51}\text{Cr}$	$1.1+11$	$1.1+11^{(1.0)}$	$9.6+10^{(1.1)}$
$^{55}\text{Mn}(n,\gamma)^{56}\text{Mn}$	$3.0+11$	$1.2+11^{(2.5)}$	$1.1+11^{(2.7)}$
$^{58}\text{Fe}(n,\gamma)^{59}\text{Fe}$	$4.3+10$	$2.7+10^{(1.6)}$	$2.5+10^{(1.7)}$
$^{63}\text{Cu}(n,\gamma)^{64}\text{Cu}$	$2.8+11$	$8.6+10^{(3.3)}$	$9.1+10^{(3.1)}$

Note: 1) Standard error in MCNP < 1%.
2) Ratio of MCNP/DOT.

TABLE VIII
COMPARISON OF ACTIVATION REACTION RATE
AT VACUUM VESSEL FOR DD
(arbitrary unit)

Reaction	MCNP	DOT-SP	
		Plate	Bellows
$^{58}\text{Fe} (n,p) ^{54}\text{Mn}$	$9.3+8^{(1)}$	$8.2+8 (1.1)^{(2)}$	$6.5+8 (1.4)$
$^{58}\text{Ni} (n,p) ^{58}\text{Co}$	$1.5+9$	$1.4+9 (1.1)$	$1.1+9 (1.4)$
$^{60}\text{Ni} (n,p) ^{60}\text{Co}$	$1.1+2$	$6.0+5 (2.0 \cdot 10^{-4})$	$4.7+5 (2.0 \cdot 10^{-4})$
$^{63}\text{Cu} (n, \alpha) ^{60}\text{Co}$	$1.4+4$	$0.0 (\infty)$	$0.0 (\infty)$
$^{50}\text{Cr} (n, \gamma) ^{51}\text{Cr}$	$1.0+9$	$8.6+8 (1.2)$	$7.7+8 (1.3)$
$^{53}\text{Mn} (n, \gamma) ^{56}\text{Mn}$	$2.6+9$	$8.9+8 (2.9)$	$8.2+8 (3.2)$
$^{58}\text{Fe} (n, \gamma) ^{59}\text{Fe}$	$3.5+8$	$1.6+8 (2.2)$	$1.6+8 (2.2)$
$^{63}\text{Cu} (n, \gamma) ^{64}\text{Cu}$	$2.3+9$	$8.1+8 (2.8)$	$8.5+8 (2.7)$

Note: 1) Standard error in MCNP < 1%.
 2) Ratio of MCNP/DOT.

TABLE IX

SUMMARY OF COMPUTATIONAL MODELS FOR TETR RADIATION TRANSPORT ANALYSIS

	MODEL			
	1	2		3
a. Dimensionality				
b. Calculational Method	Multigroup Discrete-ordinates, Monte-Carlo	Multigroup Discrete-ordinates		Monte-Carlo
c. Typical code used	ANISN, MCNP	DOT		MCNP
d. Solution provided for the problem space	Scoping studies with energy and spatial distributions	Specific problems with energy and spatial distributions		Specific problems with energy and angular distributions
e. Source of computational errors	Omission of components, imposed spatial symmetry, cross-section multigroupings, angle discretization	Omission of components, imposed spatial symmetry, cross-section multigroupings, angle discretization (ray-effect), spatial mesh size		Omission of components, statistics
f. Typical computational resources (Cray-1 on MFE) required for the problems discussed in Section III				
1) CPU (hours)	<0.02	2.5	0.2	5
2) Core size (million words)	<0.2	1.2 ET	0.15 SP	1.2
3) File size (million words)	<0.2	100	2.6	1.3

Figure Captions

- Fig. 1 An engineering drawing showing a horizontal view of the TFTR system.
- Fig. 2 An engineering drawing showing a vertical view of the TFTR system.
- Fig. 3 Schematic representations of TFTR in three-dimensional geometry. This basic model has been used for the Monte Carlo simulation for the neutron transport using the MCNP code.
- 1 - plasma, 2 - vacuum vessel, 3 - TF coils, 4 - shear compression panels, 5 - substructure, 6 - floor, 7 - ceiling, 8 - wall.
- Fig. 4 A schematic representation of TFTR in two-dimensional R-Z geometry (the Entire Toroidal Model). The axis of rotational symmetry is on the machine center line. This model has been used for the discrete ordinates calculation using the DOT code.
- Fig. 5 A closeup view of Fig. 4 for the region near the tokamak.
- Fig. 6 A schematic representation of TFTR in two-dimensional R-Z geometry (the Segmented Poloidal model). The rotational symmetry is along the torus minor axis. This model has been used for the discrete ordinates calculation using the DOT code.
- Fig. 7 A schematic representation of TFTR in one-dimensional spherical geometry. This model has been used for both the Monte Carlo simulation and the discrete ordinates calculation.

Fig. 8 Comparison of neutron flux spectrum at the vacuum vessel due to 10^{18} DT neutrons. The MCNP results are volume averaged fluences. The DOT-SP results are volume averaged fluences for the plate section. The DOT-ET results are volume averaged fluences at the inboard and outboard midplane, respectively.

Fig. 9 Comparison of neutron flux spectrum at $R = 5.82$ m due to 10^{18} DT neutrons. The MCNP results are area averaged fluences, with a scoring surface 1 m high and symmetric with respect to the midplane and major axis. The DOT results are mesh averaged fluences at the midplane. The ANISN results are also mesh averaged fluences.

Fig. 10 Comparison of neutron current angular distributions at $R = 5.82$ m for a DT source. The distributions have been averaged over a surface 1 m in height and symmetric with respect to the midplane and the major axis.

Fig. 11 Same as Fig. 10 but at $R = 10$ m.

Fig. 12 Same as Fig. 11 but for a DD source.

Fig. 13 Comparison of the spatial distribution of the neutron fluence in the test cell due to 10^{18} DT neutrons. The contour lines are those calculated by the DOT-ET model. The inserted values are the area averaged fluences calculated by the MCNP Monte Carlo model.

Fig. 14 Comparison of the spatial distribution of the gamma-ray fluence in the test cell due to 10^{18} DT neutrons. The contour lines are those calculated by the DOT-ET model. The inserted values are the area averaged fluences calculated by the MCNP Monte Carlo model.

Fig. 15 Comparison of spatial distribution of neutron dose equivalent in the test cell due to 10^{18} DT neutrons. The contour lines are those calculated by the DOT-ET Model. The inserted values are the area averaged dose equivalent calculated by the MCNP Monte Carlo Model.

Fig. 2

85E0050

Fig. 3

Fig. 4

Fig. 5

Fig. 6

#85E0048

Fig. 7

Fig. 8

#85E0044

Fig. 9

#85E0047

Fig. 10

#85E0046

Fig. 11

#85E0049

Fig. 12

Fig. 13

Fig. 14

TFTR DT Neutron Dose (Sv)

Fig. 15

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