A Simplified Model of Nuclear Heating in Tokamaks

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Abstract

The determination of the distribution of heating in fusion devices is important for thermo-mechanical design and analysis. The nuclear heating is a component of the total heating and is the concern of this paper. For toroidal geometries, such as a tokamak, this is difficult to approximate using simple models, and typically requires a two-dimensional neutron-gamma transport calculation. The particular problem is to incorporate the poloidal variation of the heating. A simplified model is developed to approximate the nuclear heating as a function of the poloidal angle and depth into the structure of interest. The technique uses ray-tracing and one-dimensional neutron-gamma transport calculations. This method has great advantage over the multidimensional neutron-gamma transport calculations, particularly for a frequently changing design. The 1.75 m Compact Ignition Tokamak (CIT) design is used as an example, and comparisons with two-dimensional results are given.

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1 Introduction

The determination of the distribution of heating in fusion devices is important for thermo-mechanical design and analysis. Both the absolute temperature and the gradients affect the stresses and relaxation in all structure surrounding the plasma. Different heating mechanisms can provide varying contributions in different locations. In addition, these contributions are a function of time. The two primary heating mechanisms are resistive and nuclear, the former being due to I^2R losses from electrical currents, and the latter from neutron and gamma interactions with the material. The primary regions of interest are the structure and the magnets. If the magnets are normal conductors resistive heating will dominate, while superconducting magnets will be predominantly heated by neutrons and gamma rays. The structure will typically be heated solely by nuclear processes, except for the first wall which will receive a substantial surface heating from plasma particles. In present-day and even next-generation tokamaks, such as the Compact Ignition Tokamak (CIT), the heating will be dominated by the resistive component. In the case of CIT the toroidal field coils also act as a shield, and so although the heating is mostly resistive losses, it provides a good example of a blanket for analysis. However, heat generation in fusion reactors, with superconducting coils and thick blankets, is likely to be dominated by nuclear processes. In either case, the nuclear heating distribution must be known to assess the thermal and structural behavior of the device.

For some device geometries, such as a spherical laser fusion chamber or a straight cylindrical mirror reactor, one-dimensional nuclear transport models can be very accurate. However, toroidal geometries, such as a tokamak, are difficult to approximate. The particular problem is to incorporate the poloidal variation of the heating.

There are a number of ways to deal with toroidal geometry for the calculation of nuclear heating. On the most detailed level, one may set up three-dimensional deterministic or Monte Carlo models for the neutron-gamma transport. These are extremely expensive computationally and are not considered feasible for global calculations where the fluxes are required at many points in space. Two-dimensional neutron-gamma transport models can effectively determine the fluxes over large regions of space, by ignoring some three-dimensional effects, such as those due to ports or limiters. These are still expensive computationally. In addition, the man-hours spent to con-

struct the geometric model are considerable, particularly for an evolving machine design. One-dimensional models are good for survey calculations, but they cannot account for poloidal variations, which are important in tokamaks. However, these calculations are very rapid, a great number of which can be done in the time it takes for a two-dimensional calculation.

A semi-empirical formula to approximate the heating distribution without having to run a multidimensional transport calculation, would be quite useful and could be easily updated for a frequently changing tokamak design. The purpose of this paper is to arrive at a reasonably accurate expression for the last method. The 1.75 m CIT design is used as the example, and the toroidal field coil heating will be of main concern.

2 Nuclear Heating Analysis of the Compact Ignition Tokamak

The Compact Ignition Tokamak (CIT) is planned to be a compact, high toroidal field, and high plasma current device that will achieve ignited D-T plasmas. It will provide the stepping stone to the following generation of engineering test reactors by providing a study of the physics of ignited plasmas. Some design parameters of this device are listed in Table 1. It should be emphasized that these do not represent the latest design, which presently has a major radius of 2.1 m.

Shown in Fig. 1 is a cross sectional view of the 1.75 m CIT design. The first wall is made of carbon tiles mounted on an Inconel 625 vacuum vessel. Outside of this on the upper, lower, and outboard sides is a toroidal field coil support made of stainless steel. Outside of this is the set of TF coils, with a slightly different material composition inboard than outboard. Surrounding this are the poloidal field coils and their support structure, which is surrounded by more structure and a cryostat. A concrete "igloo" then surrounds the entire tokamak to provide a radiation shield.

A two-dimensional model of this design was constructed using the 2-D discrete ordinates neutron-gamma transport code, DOT 5.1[1]. Since the tokamak is nearly up-down symmetric, it is only necessary to model half of the device. This model is shown in Fig. 2, which has 21 geometrical/material zones. Materials are homogenized using average volume fractions. The grid

is made up of 166 radial intervals and 129 vertical intervals. A combination of fully symmetric S_8 and S_4 angular quadrature sets is used in the angular discretization, and P_3 Legendre expansions are used for treating the anisotropy in nuclear cross sections and angular flux. The neutron and gamma cross sections are extracted from the VITAMIN-E data set[2]. These are collapsed into the standard PPPL 25 neutron group and 12 gamma group structure[3] and [4].

An isotropic volumetric neutron source for the first neutron energy group (13.5 to 14.9 MeV) is used with the spatial distribution given by

$$S(R, Z) = S_o \left[1 - \left(\frac{R - R_1}{R_2} \right)^2 \right]^3 \left[1 - \left(\frac{Z - Z_1}{Z_2} \right)^2 \right]^3. \quad (1)$$

where S_o is the source normalization. In this case $R_1=1.75$ m, $R_2=0.55$ m, and Z_1 and Z_2 are chosen at each R so that the source is nearly zero at the plasma boundary, which has an elongation of 2.0 and a triangularity of 0.3.

This 2-D neutron-gamma transport model for the CIT design will be used as the baseline for examining the heating behavior and for comparison with the simplified model. The 2-D analysis is itself an approximation, depending on the discretization of space, angle, and energy, and does not necessarily represent the actual situation with 100% accuracy. In addition, it would require a series of 2-D runs with varying discretization to accurately quantify the error; however, this is not considered useful. Thus, it should be kept in mind that the comparisons are only relative.

The neutron and gamma fluxes obtained from the 2-D transport calculation are folded with the kerma factors based on MACKLIB[5] to give the heating at the inner side of the TF coil. This is shown in Fig. 3, with and without the first wall and vacuum vessel. The poloidal angle will be defined as 0° at the outboard and 180° at the inboard. Clearly, in the presence of the first wall and vessel the heating is lower; however, the poloidal variation also reflects the discontinuities in the vessel geometry. For example, the severe drop in the heating around 40° to 50° is due primarily to the "dog-leg" in the vacuum vessel where the neutrons must travel through a thicker section. At 90° to 100° the vessel becomes thicker and another drop in the heating occurs, although this is also due to the distance between the TF coil inner surface and the plasma, since a strong drop also occurs in the no vessel case. At the inboard the heating increases because the plasma is closer to the TF

coil there. The toroidal geometry introduces much of the poloidal variation, while the plasma and structure shape provide more detailed variation. Shown in Fig. 4 is a plot of the CIT cross section with chords drawn from the plasma center at various angles. By following the chords through the first wall and vacuum vessel, one can understand the poloidal variations in the heating, considering the distance from the plasma to the TF coil inner surface, the thickness of the first wall and vessel, and the angle the chord penetrates the first wall and vessel. Although the actual neutron source does not emanate from the plasma center only, and the heating is not only due to the incident neutron flux, this is still useful for interpreting the gross features of the poloidal variations in the heating. Shown in Fig. 5 is a contour plot of the heating throughout the cross section of the CIT device. The materials are all taken as copper since the TF coil is of interest.

To examine the contributions from the gammas, neutrons, and first group neutrons, to the total heating at the TF coil inner surface, a plot of these is given in Fig. 6, as a function of poloidal angle. In both cases, with and without the first wall and vacuum vessel, the gamma contribution dominates. The first group and total neutron contributions show the same shape in both cases. For neutron heating, neutrons in the first group dominate, contributing up to 80% for no first wall or vessel. The rest of the neutrons contribute an almost constant background. Thus, the poloidal variation in the neutron heating is due almost entirely to the variation in the first group neutron flux. The reduction in the neutron heating with the first wall and vessel is a result of the attenuation in the first group flux by these structures. The fractions that the gammas and neutrons contribute are different for the two cases. The first wall and vessel basically remove neutrons or scatter them into lower energy groups where their heating potential is lower. Fewer neutrons reach the TF coil inner surface to produce gammas. This is clear from the shape of the gamma heating curve with the first wall and vessel present. A dip around 40° to 50° occurs precisely where the vessel becomes thicker and should more strongly attenuate incident neutrons. This indicates that the gamma heating is responding directly to the incident neutrons. It is observed that the gammas produced in the vessel do not appear to contribute much since they would cause the dip to be less pronounced. The shape of the gamma heating curve without the first wall and vessel is very similar to the first group and total neutron heating, again indicating that it is following the incident neutron behavior.

In addition to the poloidal variation in the heating on the TF coil inner surface, there is the attenuation of the heating into the TF coil. Shown in Fig. 7 are two attenuation curves of the heating into the TF coil for poloidal angles of 0° and 120° . These represent the heating along a chord from the plasma center. Included are the gamma and neutron contributions to the total heating. The gamma contribution clearly dominates, although near the TF coil inner surface the neutron contribution is not negligible. However, deeper into the coil the gamma heating completely determines the total heating. The slopes of the lines are different for different poloidal angles, however, they are substantially different only around 75° and 120° to 150° where the chord along which the attenuation is found does not lie normal to the heating contours. This can be seen in Fig. 5, where heating contours are included with chords emanating from the plasma center.

In addition, the neutron and gamma energy spectra can be used to further understand the contributions and their poloidal distribution. The neutron energy spectrum is shown in Fig. 8a with and without the first wall and vessel at 0° . The only major spectrum change occurs in the first group, although an increase in the fifth group flux (8.2 to 10.0 MeV) due to inelastic scattering in the graphite first wall can be seen, the remaining spectrum is fairly well preserved. This has been observed to be true of all angles, and indicates the drop in the neutron heating is mainly due to the drop in the first group neutron flux. Figure 8b shows the gamma spectrum at 0° and shows that the spectrum shape is preserved, but there is an overall drop in the gamma flux, reflective of the drop in the first group neutron flux.

As a comparison to the 2-D results and to illustrate a "typical" one-dimensional analysis of the problem, ANISN[6] was run with vertical cylinder geometry, where the axis of symmetry was coincident with the machine centerline. The material/zone and source information from the first vertical interval of the 2-D problem was used. A volumetric source was used with normalization corresponding to the equivalent of 300 MW of total fusion power and a vertical plasma height of 2.2 m. Shown in Fig. 9 is the heating along the major radius coordinate. All materials are treated as copper. The heating of the TF coil inner surface for the outboard is 30.3 W/cc and for the inboard is 40.1 W/cc. These compare quite well with the 0° and 180° 2-D results of 27.8 W/cc and 41.1 W/cc. However, to use an average heating value of 35.2 W/cc would severely overestimate the entire outboard and especially around 90° to 120° . Another possible averaging method would be

to use 30.3 W/cc from 0° to 90° and 40.1 W/cc from 91° to 180° . These are shown superimposed in Fig. 3. Neither would represent the poloidal variation of the heating even in an average sense. Both methods have a root mean square deviation and percent deviation of 10 W/cc and 54%, respectively.

The 1-D transport calculation treats the plasma neutron source as infinite in the Z direction, the dividing of the plasma source by the plasma height is to compensate for this. However, the region of the tokamak above the plasma is neglected in these calculations and cannot be re-introduced. Only the $1/R$ effect of the toroidal geometry is represented, but there is actually additional divergence. Introducing poloidal variation is not possible and this is the main drawback of using 1-D transport calculations only. One-dimensional models would be useful if only an average heating was required say for bulk power balance; however, we are interested in the poloidal variation of the heating, which requires more information than can be provided by 1-D analysis.

3 Simplified Model and Computational Procedure

The heating is determined by folding the neutron and gamma group fluxes with energy-dependent kerma factors. The neutron-gamma transport and scattering, as well as the energy structure of the kerma factors makes an analysis of the contributions of heating terms very complicated. Even to examine heating from gammas and neutrons separately is involved because the gammas dominate the heating, yet they are produced by scattering and absorption of neutrons of all energies. This does not allow them to be easily correlated to the source of neutrons. Observations in the previous section led us to believe that the poloidal variation in the heating could be accounted for by treating it as a sum of two contributions; a local or primary, and a distant or secondary. Each of these is composed of both neutron and gamma parts that will not be resolved. The primary term accounts for the direct heating effects of neutrons incident locally and the secondary term represents the effects of neutrons incident on all other regions. Attached to this is an exponential decay term to account for the drop of the heating into the TF coil.

Hence, the heating at a given poloidal angle θ is represented by

$$H(x, \theta) = (H_1(\theta) + H_2(\theta)) e^{-\frac{x}{\lambda}}, \quad (2)$$

where H_1 is the primary and H_2 is the secondary heating terms, λ is the attenuation length for the heating, and x is the coordinate along a chord from the plasma center at θ into the coil ($x = 0$ is at the coil inner surface). As a first approximation we have assumed that λ is independent of θ . One way of expressing H_1 and H_2 explicitly is,

$$H(x, \theta) = \left[C_1(\theta)J(\theta) + \int_{\theta' \neq \theta} C_2(\theta, \theta')J(\theta')d\theta' \right] e^{-\frac{x}{\lambda}}, \quad (3)$$

where $C_1(\theta)$ is the heating rate at the TF coil inner surface per unit incident 14 MeV neutron current at θ , and $C_2(\theta, \theta')$ is the heating rate at θ per unit incident 14 MeV current at θ' , per unit angle. $J(\theta)$ is the incident 14 MeV neutron current at the TF coil surface, and the integral is over all poloidal angles not equal to θ . The neutron current can be expressed as $J(\theta) \equiv \bar{J}f(\theta)$, where $\bar{J}$ is the average current on the TF coil surface, and $f(\theta)$ is the angular neutron current distribution normalized to the average. Approximating the integral by a sum gives.

$$H(x, \theta) = \left[C_1(\theta)f(\theta) + \sum_{i=1, \theta'_i \neq \theta}^{N_\theta} C_2(\theta, \theta'_i)f(\theta'_i) \right] \bar{J}e^{-\frac{x}{\lambda}}. \quad (4)$$

$C_2(\theta, \theta')$ is defined here to be the heating rate at θ per unit incident 14 MeV current at θ' , the angular increment being absorbed. At this point there are a number of possible further reductions. The present form with the sum over a large number of angles is still very difficult to evaluate. In order that C_1 and C_2 can be evaluated with simple 1-D transport models, the expression is simplified into two terms, primarily because the 1-D transport analysis can provide only two heating values, one at 0° and one at 180° . This removes the θ dependence from the heating terms C_1 and C_2 and makes them strictly outboard and inboard terms. The $f(\theta)$ then provides the only poloidal variation. 90° is arbitrarily chosen as the dividing line between the inboard and outboard. For angles between 0° and 90° the primary heating term is $C_1(\theta) = C_1(0^\circ)$ and the secondary term is $C_2(\theta, \theta') = C_2(0^\circ, 180^\circ)$. For angles between 91° and 180° the primary heating term is $C_1(\theta) = C_1(180^\circ)$

and the secondary term would be $C_2(\theta, \theta') = C_2(180^\circ, 0^\circ)$. With respect to the 1-D transport geometry, the primary heating for the outboard angles is considered due to a source emanating from the outboard side of the plasma, all neutrons launched to the right. The secondary heating for the outboard angles is due to a source emanating from the inboard side of the plasma and being scattered back to the outboard side, all neutrons launched to the left. The converse is true for the inboard heating. This is illustrated in Fig. 10. The heating breaks up into two expressions, one for outboard and one for inboard.

$$H(x, 0^\circ \leq \theta \leq 90^\circ) = [C_1(0^\circ)f(\theta) + C_2(0^\circ, 180^\circ)f(180^\circ)] \bar{J}e^{-\frac{x}{\lambda}} \quad (5)$$

$$H(x, 91^\circ \leq \theta \leq 180^\circ) = [C_1(180^\circ)f(\theta) + C_2(180^\circ, 0^\circ)f(0^\circ)] \bar{J}e^{-\frac{x}{\lambda}} \quad (6)$$

However, the secondary heating terms are due to scattered neutrons and gammas from one side of the tokamak to the other and have essentially lost memory of their source, while $f(\theta)$ represents the poloidal variation of the 14 MeV incident neutron current. Thus, the secondary term should not be related to the incident neutrons, but rather treated as a constant background. The above expressions then reduce to,

$$H(x, 0^\circ \leq \theta \leq 90^\circ) = [C_1(0^\circ)f(\theta) + C_2(0^\circ, 180^\circ)] \bar{J}e^{-\frac{x}{\lambda}} \quad (7)$$

$$H(x, 91^\circ \leq \theta \leq 180^\circ) = [C_1(180^\circ)f(\theta) + C_2(180^\circ, 0^\circ)] \bar{J}e^{-\frac{x}{\lambda}} \quad (8)$$

As shown in the following sections, $f(\theta)$ and $\bar{J}$ will be obtained from a ray-tracing procedure, and C_1 , C_2 , and λ will be evaluated using 1-D transport models.

3.1 Toroidal Geometry Ray-Tracing

The poloidal variation of neutron current $f(\theta)$ can be obtained from simple ray-tracing calculations[7] and [8]. The ray-tracing calculation only treats the neutron streaming problem for the 14 MeV neutrons. Given a specified neutron source distribution within the plasma, the algorithm determines the angular fluxes, $F(\chi, \phi)$ (χ and ϕ are the angles in a spherical coordinate system at a point) and $F(\chi)$, and the scalar flux F and current J , at various poloidal angles θ on a specified surface. The geometry is illustrated in Fig. 11. These quantities are only for the 14 MeV uncollided neutrons, so the

first wall and vacuum vessel are neglected. In addition, an average neutron wall loading $\bar{J}$ is calculated as the ratio of the total neutron power to the surface area. Since the ray-tracing is three-dimensional the actual toroidal geometry can be accounted for. The ray-tracing calculation is done for the plasma surface and again for the TF coil inner surface, with the neutron source distribution within the plasma given by

$$S(R, Z) = S_0 \left[1 - \left[\frac{(R - R_0)^2}{a^2} + \frac{Z^2}{\kappa^2 a^2} \right] \right]^3, \quad (9)$$

where R_0 , a , and κ are the plasma major radius, minor radius, and elongation, respectively. (R, Z) is the position inside the plasma. The source specification used in the 2-D transport model is slightly different[1], however the difference is in the fourth order and is found to be negligible. The scalar neutron current at the TF coil inner surface is represented as $J(\theta) \equiv \bar{J}f(\theta)$, and the $f(\theta)$ term is used to weight the primary heating term. This is based on two premises; that the poloidal variation of the heating will have similar shape to the 14 MeV uncollided neutron current, and that the poloidal variation of the neutron current will not change significantly with first wall and vessel present. However, the variation is not as sharp for the heating as it is for the neutron current, so the secondary term is required to reduce the degree of variation. The plasma ray-tracing results are used in the 1-D analysis discussed in the next section.

3.2 One-Dimensional Transport Models

Three one-dimensional geometries are available; slab, sphere, and cylinder. The neutron source can be specified as a volume or shell source, although only the shell source can accommodate an anisotropic angular distribution. The slab and vertical cylinder models have infinite extent vertically. These calculations determine the neutron and gamma fluxes along one coordinate, with which the kerma factors can be folded to calculate heating or any other response. Thus, the spatial dependence of the fluxes, currents, and heating are available; however, only the 0° and 180° from the 2-D model can be simulated.

Since the ray-tracing provides the angular neutron flux $F(\chi)$ as a function of angle χ (or $\mu = \cos\chi$), for various poloidal angles, this can be used in a

shell source for 1-D transport. The angular neutron source distribution at the plasma surface from the ray-tracing at 0° is then used in a shell source at the outboard plasma edge of a 1-D transport calculation. The resulting heating on the outboard is the outboard primary heating [$C_1(0^\circ)$] and the heating on the inboard is the inboard secondary heating [$C_2(180^\circ, 0^\circ)$]. This is repeated with the 180° angular neutron source distribution used in a shell source at the inboard plasma edge. The inboard heating is the inboard primary [$C_1(180^\circ)$] and the outboard heating is the outboard secondary [$C_2(0^\circ, 180^\circ)$]. This is schematically illustrated in Fig. 10. This only provides two different total heating values. The 1-D transport calculations are done with 1 MW/m^2 current shell sources at the plasma, so the heating at the TF coil is per unit current at the plasma surface. In addition, the ray-tracing gives the average neutron wall loading at the plasma surface. The correlation requires these quantities to be per unit current at the TF coil, and the average wall loading on the TF coil, respectively, so they must be renormalized. The renormalizing factor is taken from the drop in the first group current from the plasma surface to the TF coil calculated from a slab 1-D transport model.

The 1-D analysis also gives the heating spatially into the TF coil at 0° and 180° . The slope of the decay in the heating is the attenuation length λ , used in the correlations. An average is taken between the inboard and outboard values, although they could be used separately since the correlations are separate.

4 Results

The model views the heating as due to a local and a distant source. Due to the limitations of 1-D transport, only the 0° and 180° from the 2-D analysis can be simulated, so that the local and distant heating terms are derived for these poloidal angles only. In effect, the primary heating term is correlated to the incident 14 MeV neutron current, and so the poloidal variation factor [$f(\theta)$] from the ray-tracing, is attached to this term. The secondary heating has lost memory of its source and so it is represented as a constant background added on. This justification is not without merit. However, the complicated energy structure of the fluxes and kerma factors coupled with the geometry of the tokamak make its proof from first principles impossible.

The heating is dominated by gammas, not the 14 MeV neutrons, and the gammas are not produced only by the 14 MeV neutrons. Thus, at first glance there appears to be no reason why the poloidal heating variation should follow the distribution of the 14 MeV neutrons. However, they do show similar behavior, illustrated in Fig. 3, where without the first wall and vessel the shapes of the gamma heating and 14 MeV neutron heating are quite similar. When the first wall and vessel are included, the gamma heating reflects the discontinuities in these structures more than the 14 MeV neutrons. From the neutron spectrum results, the first wall and vessel mainly remove 14 MeV neutrons while leaving the remaining spectrum unchanged. This would infer that the drop in the total and gamma heating is, in fact, due mainly to the drop in the 14 MeV neutron flux reaching the TF coil inner surface. This would be a critical function of the first wall and vessel thickness, since as it got thicker it would begin to influence other groups more strongly.

The simplified model is based on trends from the 2-D transport analysis, as well as the available calculations that can be made without actually doing a 2-D calculation.

Shown in Fig. 12 are the 14 MeV scalar flux and current from the ray-tracing calculation as a function of poloidal angle for 300 MW total fusion power for the CIT geometry. These are for both the plasma surface and at the TF coil inner surface. The flux and current at the TF coil is lower than that at the plasma surface due to geometrical divergence, and the outboard values drop much faster than the inboard since the distance to the TF coil is larger on the outboard. The curves stress the fact that the flux and current distributions are strongly dependent on the surface chosen.

In addition to the scalar flux and current, the angular fluxes are available. The particular angular fluxes, $F(\chi)$ [or $F(\mu)$, $\mu = \cos\chi$], at the plasma surface for 0° and 180° are used in the shell source 1-D transport to determine the heating and attenuation. Shown in Fig. 13 are these distributions for the inboard and outboard, at the plasma surface and the TF coil inner surface. These show that the neutron angular source distributions are quite different from inboard to outboard, being heavily peaked toward normal on the outboard. Also, neither distribution would be well approximated by a flat distribution. It should be emphasized that these are neutron streaming calculations and cannot treat material between the neutron source and the surface of interest.

The one-dimensional transport was used with a slab, sphere, and vertical cylinder, using the first vertical interval from the 2-D model for the material and zone information. This material description is shown in Fig. 14. Shown in Fig. 15a and b are the heating profiles across the tokamak from the 1-D analysis, for outboard and inboard shell sources, respectively. These are for cylindrical geometry, and assume the materials are all copper, since the TF coil is of primary interest. The values for the primary and secondary heating per unit current at the plasma surface for all three geometries are given in Table 2, with the renormalized values in parentheses.

Graphs of the nuclear heating rates at the TF coil inner surface from the correlations versus the 2-D results are shown in Fig. 16a and b, the former with the first wall and vessel and the latter without. All three 1-D geometries are shown. The agreement is reasonable considering the simple form of the expression and the complicated quantity being estimated.

The curves for the case with first wall and vessel overestimate near 40°, 90°, and between 140° and 180°. At 40° and 90° the vacuum vessel is thicker than at 0°, which the correlation is based on, leading to the drop in the heating there. The overestimate at the inboard is more complicated and is likely due to the limitations of the 1-D modelling, particularly too large a value for the inboard secondary heating. The "wobble" in the two correlation curves around 90° is due to the switchover from the outboard correlation to the inboard one. The average deviations for no first wall or vessel are 7.4% for spherical, 7.1% for cylindrical, and 9.3% for slab. With the first wall and vessel included the average deviations are 12.6% for spherical, 18.8% for cylindrical, and 15.6% for slab. The average deviation is defined as the sum of the absolute values of the differences between the heating value calculated with a 2-D transport model and the predicted value from the correlation, normalized to the 2-D value. The entire sum is then normalized by the number of points used for comparison. It should be emphasized that the 2-D model is also approximate, so the deviations are relative.

$$\text{average deviation} = \frac{1}{N} \sum_{i=1}^N \left(\frac{|H_{2-D} - H(x, \theta)|}{H_{2-D}} \right) \quad (10)$$

These are reasonable error spreads considering the simplicity of the model.

Shown in Fig. 17 is heating as a function of depth into the TF coil for a few poloidal angles. The differences in the slopes at various angles are due to

the geometry of the TF coil compared to the chord drawn from the plasma center through the coil at a given angle. This is illustrated in Fig. 7 where heating contours are shown with the same chords included. At 120° the chord does not cross the heating contour at a right angle and so a different decay rate than at 0° where the chord is normal is present. If the TF coil were circular, the chords would all cross the contours at 90° angles. The 1-D transport can only give two values for the attenuation length, at 0° and 180° . Since the coil geometry can be complicated, an average between the two values is used as a constant λ for all poloidal angles with the present model. Greater effort can be made to make λ a function of θ by examining the structure geometry. There can be considerable disagreement, particularly near the top of the TF coil.

5 Conclusions

The actual neutron-gamma transport problem in two or three dimensions is quite complicated and it is impossible to reproduce the same environment with 1-D models for toroidal geometry. The multi-energy group scattering makes it possible only to approximate heating values to within some degree of error. In fact, the only reason one may get close values with a simplified model is because the response is a sum over all energy groups, causing fine scale differences between the true system and the simplified model to be smeared out.

A simplified model for approximating the nuclear heating in structure or magnets for toroidal geometries has been developed that can circumvent the need for a multidimensional neutron-gamma transport calculation. This is particularly useful for evolving machine designs where the frequent updating of a rigorous multidimensional model would be very time-consuming. The average deviations, in the example of CIT at the inner side of the TF coil, are about 7-10% without the vacuum vessel and 12-18% with the vessel included. The model relies on ray-tracing and 1-D neutron-gamma transport calculations. The computational times for these are considerably less than a 2-D calculation (typically 5%).

This method can allow for different neutron source profiles in the plasma that cannot be adequately represented in 1-D transport calculations alone.

The present model is based on three primary assumptions: the poloidal

variation in the heating can be related to the poloidal variation in the incident 14 MeV neutron current, that the poloidal variation of this current will not be significantly altered upon passing through the first wall and vacuum vessel, and the decay of heating into the coil can be described by a single attenuation length.

The first two assumptions can be violated when the structures between the plasma and the region of interest become very thick or highly non-uniform. These would strongly affect the 14 MeV neutron current and the ray-tracing would be unable to account for this. The error introduced by using a single attenuation length is not very significant in CIT since the heating value already drops by more than an order of magnitude when the error becomes large. However, it is significant for superconducting coils shielded by a blanket/shield structure. The error introduced can be more than a factor of 2.

The choice of 90° as the switchover point for using the outboard to inboard correlation is arbitrary, and this can be used as an adjustable parameter depending on the structure geometry, material composition, or any other change that could create a different heating rate.

The model may be generalized to include other responses with appropriate changes in the parameters. Physical insight into a particular application may improve the correlation or offer a better form than the one presented here for heating.

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Major Radius	1.75 m
Minor Plasma Radius	0.55 m
Plasma Elongation	2.0
Plasma Current	9 MA
Toroidal Field (baseline)	10.0 T
Plasma Burn Time (up to)	5.0 s
Neutron Wall Loading (at 300 MW)	4.2 MW/m ²
Peak Power for TF and PF Coils	1.1 TW
Number of Full-Field Pulses	3000

Table 1. Selected parameters of the 1.75 m baseline CIT design.

Slab	outboard	$C_1(0^\circ)=545 \text{ J/kg}$ $C_2(0^\circ, 180^\circ)=153 \text{ J/kg}$	(2127 J/kg) (615 J/kg)
	inboard	$C_1(180^\circ)=664 \text{ J/kg}$ $C_2(180^\circ, 0^\circ)=220 \text{ J/kg}$	(2668 J/kg) (859 J/kg)
Sphere	outboard	$C_1(0^\circ)=467 \text{ J/kg}$ $C_2(0^\circ, 180^\circ)=53.6 \text{ J/kg}$	(2230 J/kg) (198 J/kg)
	inboard	$C_1(180^\circ)=655 \text{ J/kg}$ $C_2(180^\circ, 0^\circ)=262 \text{ J/kg}$	(2422 J/kg) (1251 J/kg)
Cylinder	outboard	$C_1(0^\circ)=519 \text{ J/kg}$ $C_2(0^\circ, 180^\circ)=100 \text{ J/kg}$	(2269 J/kg) (431 J/kg)
	inboard	$C_1(180^\circ)=637 \text{ J/kg}$ $C_2(180^\circ, 0^\circ)=257 \text{ J/kg}$	(2745 J/kg) (1123 J/kg)

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FIGURE 17. Attenuation of the heating into the TF coil for a few poloidal angles compared with the correlation.

Fig. 1

R-Z GEOMETRY OF THE TWO-DIMENSIONAL MODEL

Fig. 2

Fig. 3(a)

Fig. 5

Fig. 6(a)

Fig. 6(b)

Fig. 7

Fig. 8 (a)

Fig. 8 (b)

Fig. 9

Fig. 10

Fig. 12

Fig. 13

Fig. 14

Fig. 15(a)

Fig. 15 (b)

Fig. 16(a)

Fig. 16(b)

Fig. 17(a)

Fig. 17(b)

Fig. 17(c)

Fig. 17(d)

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