To the Graduate Council:

I am submitting herein a dissertation written by Bruce Alexander Munro entitled "A Methodology for the Computer Solution of Radiation-Induced Thermal Stresses in Fusion Reactor Components". I have examined the final copy of this dissertation for form and content and recommend that it be accepted in partial fulfillment of the requirements for the degree of Master of Science, with a major in Nuclear Engineering.

Paul N Stevens

Dr. Paul Stevens, Major Professor

We have read this dissertation
and recommended its acceptance:

Accepted for the Council:

C W Minkel

Associate Vice Chancellor and
Dean of the Graduate School

A METHODOLOGY FOR THE COMPUTER SOLUTION OF RADIATION-INDUCED THERMAL STRESSES IN FUSION REACTOR COMPONENTS

A Thesis

Presented for the

Master of Science

Degree

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ABSTRACT

This report describes the creation of a methodology for determining thermal stresses of complex shapes given a neutron radiation environment. This report also examines the results of its application to a specific case, the TPX (Tokamak Physics eXperiment) lower hybrid waveguide system, which is considered a high-risk case for buckling because of it's structure of multiple thin, flat plates. The thesis describes the computer codes used, namely the Monte Carlo code MORSE for calculating the radiation field and the code ANSYS to calculate the thermal stresses, as well as the linking code created to join them. It discusses the value of a simple, one-dimensional discrete ordinates code such as ANISN in solving the neutronics problem, and the failure of ANISN to provide adequate data for the example problem. Results obtained indicate that the TPX waveguide system survives radiation-induced heating without buckling, and that the linking code can serve a useful purpose in thermal stress analysis. It also looks at the extreme case of treating the charged-particle flux as a surface flux being absorbed on contact, which still yields temperatures within the safe range.

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I. INTRODUCTION AND THEORY

1. General Problem

One of the main problems involved in the design of fusion reactor components exposed to an intense radiation field is the stresses induced as the deposited energy creates spatial distributions in temperature. To determine the magnitude of the stresses created, the neutronics component of the problem must be first solved in order to determine where the energy is deposited in the structure of interest. Charged-particle effects can, as a limiting case, be treated as a surface heat flux, but the high-energy neutrons of a D-T reaction may penetrate deep within reactor structures before depositing all their energy. After the neutronic analysis has determined the heat source distribution, the distribution is used as an input to the heat-flow and temperature distribution component of the problem. Finally, the temperature distribution is used as input to a thermal stress calculation.

Codes are available for solving the individual parts of the total problem. Discrete ordinates codes such as ANISN and TORT-DORT, or Monte Carlo codes such as MORSE can be used to solve a range of neutronics problems. By using both types of codes, and comparing the outputs, accurate results may be obtained.

Discrete ordinates and Monte Carlo methods used in parallel are a necessity in dealing with objects of complex shape. While Monte Carlo codes can easily handle complex geometries, they are limited in describing a detailed radiation profile. Discrete ordinates codes, although providing the spatially detailed radiation information desired for a thermal distribution, are practically limited in their capacity for geometric complexity. It is desirable, when possible, to simplify the geometry to one- or two-dimensional discrete-ordinates codes, using the Monte Carlo results to provide a check on the accuracy of the discrete ordinates results. This, however, is not always feasible.

Both the heat flow and thermal stress problems can be handled by finite-element thermal stress codes such as ANSYS. It should be feasible to link these codes in order to determine the deflections given the radiation environment, without having to perform a lengthy conversion of the neutronics output to the input required by the thermal stress code.

A system of codes, allowing for the accurate determination of the thermal stresses and other effects of interest that result from a known radiation environment, would be of considerable value in the design of fusion reactor components.

To serve as an application for the methodology developed in this study, the TPX (Tokamak Physics eXperiment) lower hybrid waveguide system was selected. This structure provides RF heating to the plasma, and the multiple thin, flat plates of it's antenna array are directly exposed to the radiation flux. Several such structures will be set into the walls around the circumference of the Tokamak, as shown in figure 1. Each TPX structure consists of multiple thin, flat plates composed of a mainly copper alloy, separated by vacuum gaps and forming an antenna array embedded in a larger structure containing channels for cooling water (fig.2). This structure should provide a stringent test of such a methodology, the neutronics being complicated by the multiple vacuum gaps and flat vertical plates with a high height-thickness ratio being at a considerable risk of buckling due to thermal expansion. This buckling could cause arcing between the plates, so it is necessary to determine under what conditions it could occur, and whether curving the plates in the same direction, as shown in fig.3, might solve the problem.

The waveguide system has several difficulties in addition to those problems specific to the general case. The arrangement of thin vertical plates with sizeable vacuum gaps provides many streaming pathways for the neutrons, and the discrete ordinates solution in two or three dimensions may suffer from "ray effect" difficulties. Therefore, a one-dimensional discrete ordinates solution is desirable. A three-dimensional Monte Carlo calculation can be used as a benchmark in adapting one-dimensional discrete ordinates solutions to the three-dimensional problem, and to provide a separate or solution. However, inadequate flux results from the ANISN one-dimensional model make the Monte Carlo method mandatory.

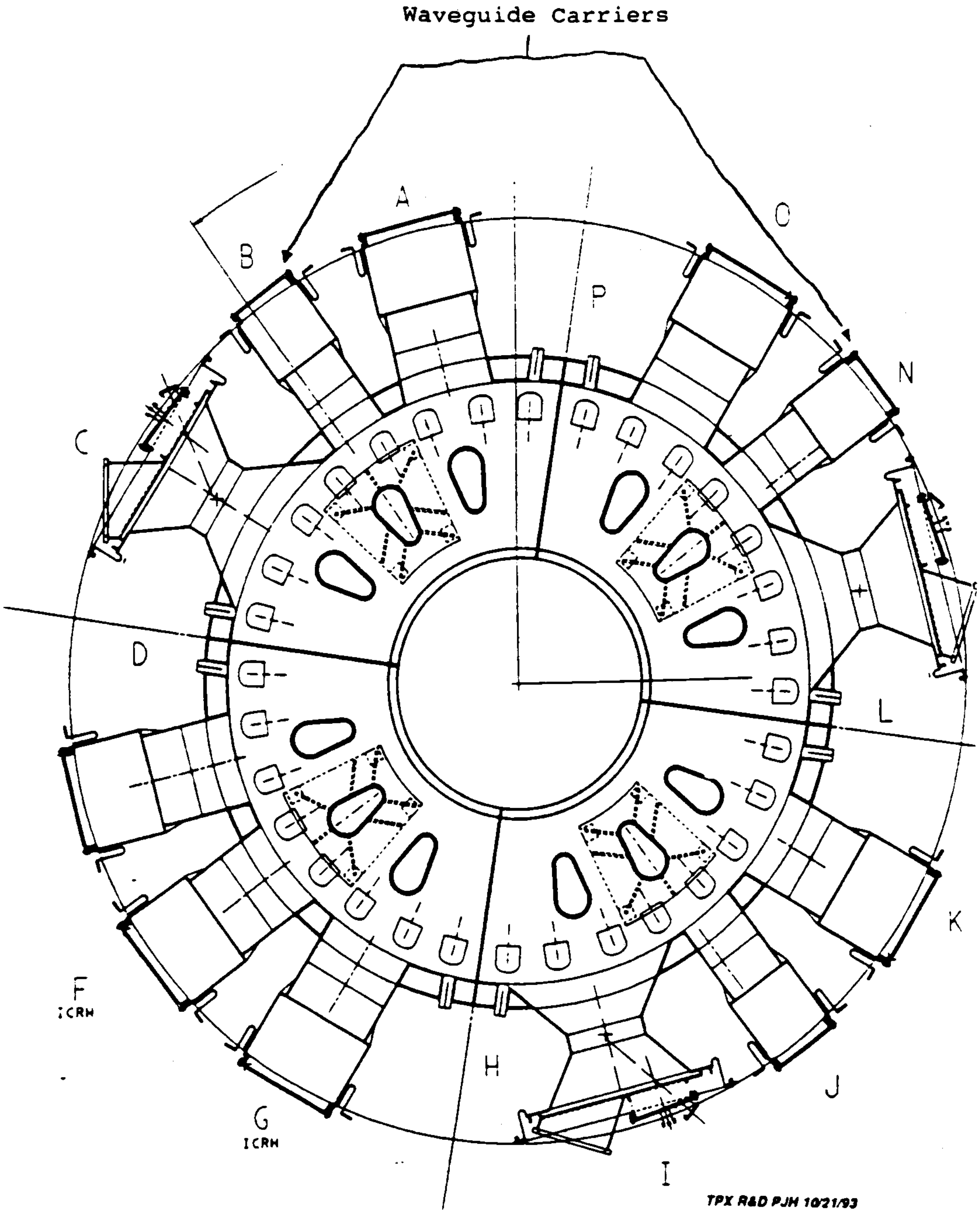

FIGURE 1: The TPX System in the Tokamak

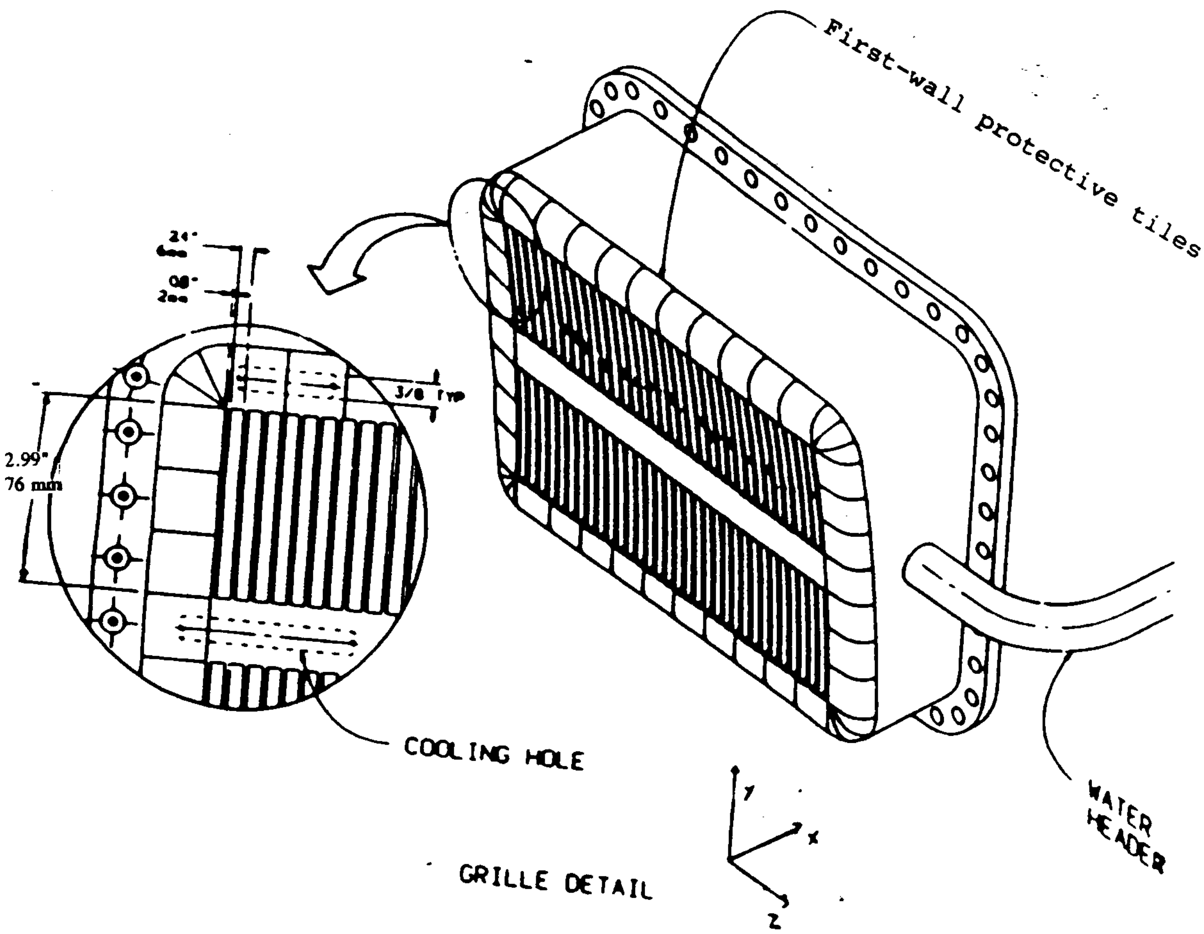

FIGURE 2: The TPX Lower Hybrid Waveguide System

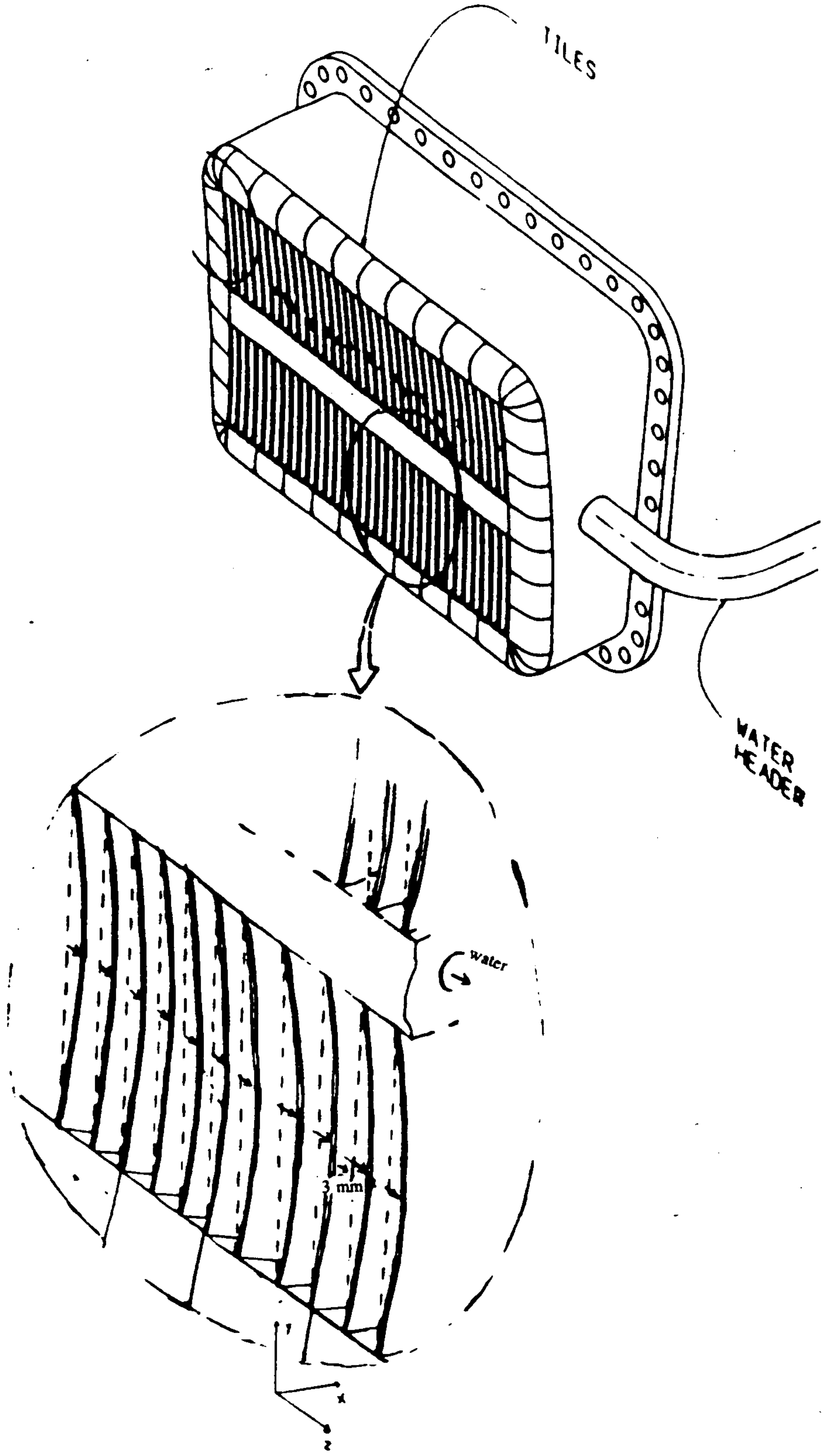

FIGURE 3: Curved-septum TPX Waveguide System.

2. The Discrete Ordinates Method

The theory of discrete ordinate as applied to neutronics consists of effecting a numerical solution to the Boltzmann transport equation. The one-dimensional code ANISN solves the steady-state energy-dependant linear version of the Boltzmann equation. The iterative solution is derived from the analytic transport equation, which expresses the rate of change of the number of particles in a system as the difference between the production rate and the loss rate for a point in space at a particular energy and angular direction. Information on the specifics of ANISN operation can be found in the handbook for ANISN-ORNL, code CCC-254, available from the RSIC computer code collection at Oak Ridge National Laboratory. (5)

The discrete ordinates codes give detailed information, such as group fluxes, and energy released by activation for each spatial mesh segment. The fineness of detail is limited only by the degree of spatial and energy discretization, and the associated computational costs.

Discrete ordinates has serious limitations. It is limited in it's ability to handle three-dimensional structures. First, there is the problem of grid fineness. To model a three-dimensional, complex object with a fair degree of accuracy requires a large number of meshes, leading to very long run times and expensive programs. More serious for someone working in multiple dimensions is the ray effect, so named because particles in a discrete ordinates calculation travel along discrete directions or rays. Fluxes at points not intersected by allowed directions are effectively zero, but are nonzero along allowed directions nearby. Such rapid but nonphysical changes in flux can severely distort the flux behavior, and may be a serious problem in a thermal stress calculation, in which a detailed picture of the flux's spatial variation is desired. The ray effect, of course, vanishes for a one-dimensional model.

The problem at hand is three dimensional, and the TPX system is large enough for attenuation to be considerable. The ray effect, and the time consuming nature of any three-dimensional code make the use of multidimensional codes such as TORT-DORT discouraging. The loss of particles streaming down the vacuum gaps in directions not included in the discretization promises to severely distort the flux picture, and the need to include channels for water further complicates the modelling. If the model could be simplified to a one-dimensional form, these problems would be greatly reduced.

One way to do the simplification would be to replace the open framework with a distributed mass, replacing a structure consisting of alternating areas of high molecular density and vacuum with a homogenous, low-density material. This structure will be exposed to a monoenergetic flux of 14-mev neutrons, with the order of quadrature (S value) 4, maximum order of scattering (P value) 3. Determining the accuracy of such an approximation is one of the major issues of this research, and depends on the use of Monte Carlo results as a standard of accuracy.

3. The Monte Carlo Method

Monte Carlo codes operate rather differently from discrete ordinates codes. The accuracy of Monte Carlo codes for most problems is high, given long enough running times, so they can serve as checks on the more problem-dependant discrete ordinates codes. However, they suffer from limitations in attempting a solution for this kind of problem.

It is to be noted that Monte Carlo solutions are never exact, being statistical in nature. Since there are no approximations involved in the solution method, an accurate solution will be approached if a large enough number of particles are employed. This may not be always practical, especially for systems that require detailed results.

Monte Carlo codes mathematically simulate individual particles through their lives, from the point where they enter the system until they leave it by leakage or absorption. All possible events are modelled as appropriate probability distributions. The "random walk" is repeated for as many particles as precision requires or computational cost allows, and if certain statistical requirements are met, the average behavior of the particles will predict actual flux behavior fairly accurately.

Once a source particle is generated, all subsequent events are determined by a statistical process, in which random numbers are generated to determine the events that will follow.

MORSE, ORNL version 6174, (6), is the Monte Carlo code with which this study is concerned. It is not entirely user-friendly, requiring user-written subroutines to describe the source and estimate the desired result. However, this can be seen as advantageous in cases for which the complexity of the situation makes the use of standard source distributions inadequate.

The TPX waveguide system should prove amenable to a MORSE solution. However, since MORSE tracks particles from source to some defined detector region, a detailed solution of the radiation profile would be hard to determine. To determine the flux throughout the system, it would be necessary to break the system up into many small detector regions. For each detector, only a small percentage of the particles would contribute to the estimation. Therefore, to obtain good statistics for all detectors, a large number of particles is required, leading to lengthy runs.

The MORSE code may be used to obtain fluxes at certain depths into the system, using large detector areas, which may be considered fairly accurate. These can serve as a check on the ANISN results. If group fluxes fall off at a comparable rate, it will greatly strengthen the case for the ANISN model.

4. Thermal Stress Theory

Once the radiation profile has been determined by discrete ordinates or the Monte Carlo method, the resulting thermal stresses must be determined. A code is required that can handle two separate problems. First, the code must determine the temperature distribution in the solids. Secondly, it must calculate the actual stresses resulting from the temperature gradients. The code chosen for this purpose will be the finite element code ANSYS, which can perform either joint thermal-stress calculations or separate thermal and afterwards stress calculations, the latter allowing examination of the temperature profiles obtained.

MORSE calculates the activation caused by neutron and gamma radiations. This can be converted by a linking code into a distribution of energy deposition rates, joules/sec/cm^3, that can be used as an input to a finite-element numerical calculation of temperatures. This will require the solution of the classical steady-state heat conduction equation.

There are a number of boundary conditions to consider. The water channels passing through the structure can be considered for a first estimate as having high flow rates, and acting thereby as fixed-temperature heat sinks, in this analysis being fixed at 125 Celsius. Then there are the surfaces exposed to charged particle and gamma radiations, which are treated as a surface heat flux, falling off along the septa according to the relevant view factor. Given the fact that heat flow out through the water channels will be rather more important than flow in the reactor wall, and that radiation losses from each septum will be counterbalanced by gains from the neighboring septa, the model will look at a single vertical subsection of the structure, (fig.4), rather than the whole, as shown in fig.2. This subsection consists of a single septum with associated water channels. The top and bottom, being part of a solid structure, will be restrained to prevent unrealistic deformation.

The model used will assume no important heat losses by radiation, and also assumes a fixed temperature at a depth of 60 cm, corresponding to the antenna's connection to the larger structure. The heat escaping in this manner will be considered comparatively insignificant.

The thermal stress code will use this model to determine the thermal profile of the structure, which can then be used as input to the stress-strain calculations. The stress calculations are mainly of importance in determining deformation in the thin septa.

Stress calculations may also be solved by iterative, finite-element methodologies. The fundamental equations for thermal stress are

$$\epsilon_x = (1/E)(\delta_x - \nu\delta_y - \nu\delta_z) + \alpha T \qquad (1.)$$

$$\epsilon_y = (1/E)(\delta_y - \nu\delta_z - \nu\delta_x) + \alpha T \qquad (2.)$$

$$\epsilon_z = (1/E)(\delta_z - \nu\delta_x - \nu\delta_z) + \alpha T, \qquad (3.)$$

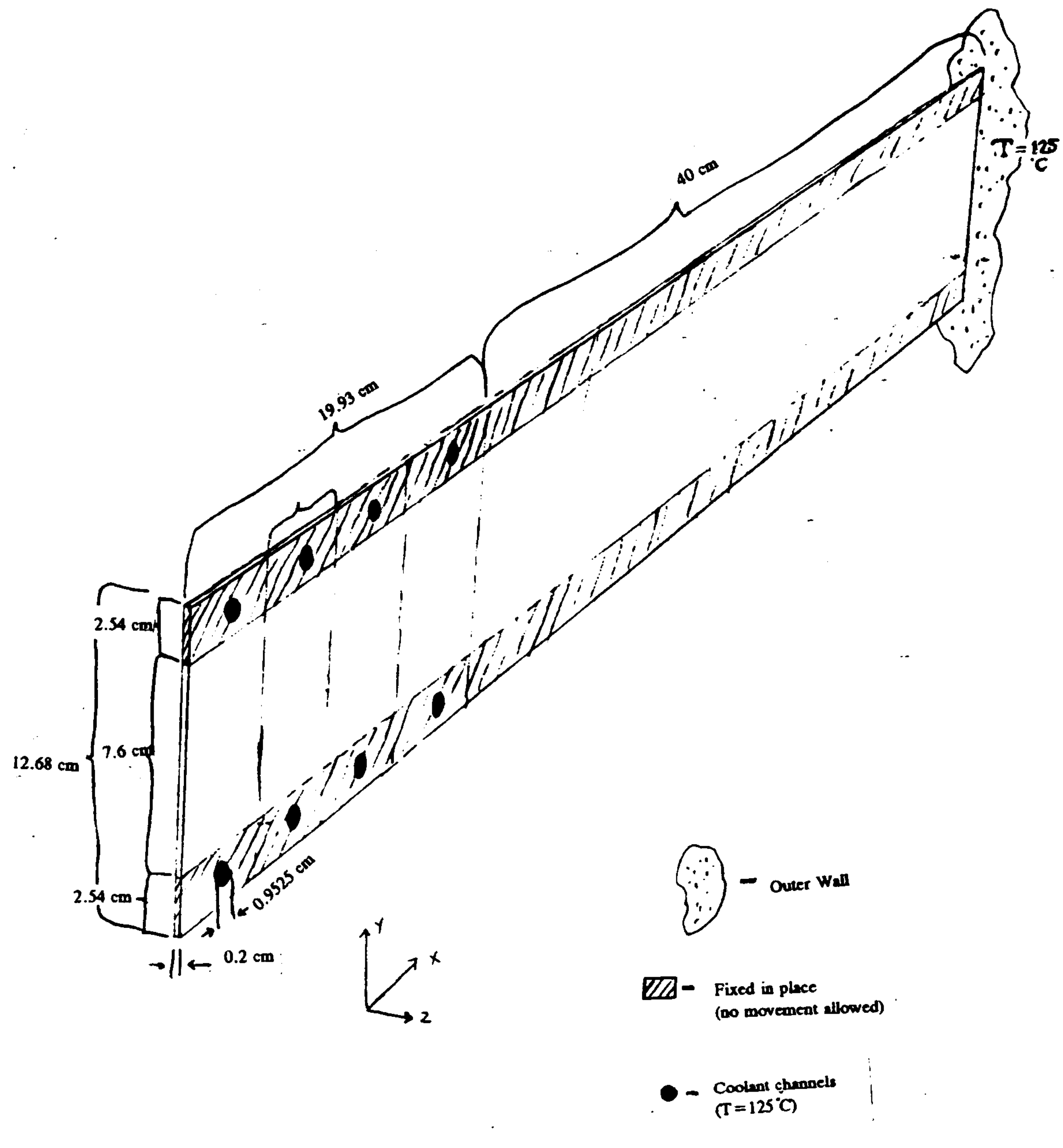

FIGURE 4: Solid model for ANSYS thermal-stress analysis.

where

E = thermal loading

ν = displacement

ϵ = observable strain

δ = stress due to temperature effects

T = temperature change

α = free expansion coefficient

αT = unit expansion of calculational interval in case of unrestricted expansion.

The stress equations are solved to determine the deformation of the septa. To maintain physical continuity (this research does not include such special cases as splitting or tearing of the material) the free expansion of the material (described by αT) must be linked to the equations describing the stress-strain distribution. This is generally achieved through the method of equivalent loads, in which the normal stresses (δ) are replaced with "displacement stresses", modified to include a free expansion factor:

$$\delta_x = \delta_x' - E\alpha T/(1-2\nu) \qquad (4.)$$

The thermal stress code ANSYS is fairly simple to use, although somewhat lacking in on-screen tutorial help. This code has a relatively user-friendly graphic modelling package, which makes solid-model construction somewhat less of a chore than setting up the MORSE model.It consists of four linked packages, a preprocessor, a solution module, a postprocessor and a "special tasks" section. There are some limitations.

Most importantly, the solid model, once constructed, cannot be easily modified.

5. The Linking Code

A major portion of the research is the creation of a translator code which will take the output from the neutronics code and transform it into an appropriate input for the thermal stress code. This will require opening the output file for the neutronics code, locating the activation data, and converting it to appropriate energy density units. Then some form of mapping procedure is used to fit the data points from the neutronics solution to the probably far larger number of elements in the thermal stress model. Two linking codes were created, one handling output from ANISN, the other from MORSE. In both cases the data are transformed to joules/cm^3, and these energy densities are returned with the depth into the homogenized structure for ANISN, or locator detection for MORSE. The MORSE translator will be the main subject of discussion, given a failure to obtain good ANISN results.

The code for reading MORSE output operates by reading the output file line by line until it finds the code string '(detecto'. These are detector responses, activation per source particle, in grays/source particle. This response is multiplied by a correction factor, rho, which provides a solution in terms of energy/cm^3 per source particle. Rho is obtained as follows:

Energy per unit volume = R*D*S*A, (5.)

where

R = MORSE response, j/(kg-particle)

D = density copper (kg/cm^3)

S = equivalent surface source associated with the fusion power, particles/cm^2

A = cross-sectional area of the MORSE model, cm^2.

An answer is obtained in terms of j/cm^3. Since the equivalent surface source, S, is entered by the user, and R is obtained from MORSE output,

rho = D*A =.00896 kg/cm^3*695 cm^2 = 6.232 kg/cm (6.

This will give the volumetric heat generation as an average for the detector region. In the case where the source neutron flux is distributed in energy, this formulation still holds, since what the MORSE code calculates is an average of energies.

The procedure is carried out for all the detector regions included in the MORSE model. This gives what is basically a step function varying with one coordinate. The ANSYS computer code will require input in the form of nodal or element inputs, so the translator takes the form of a macro written in the ANSYS command language which will take these data and calculates a corresponding heat generation rate for each of the model's nodes or elements. Since the mesh of elements will be much finer than the rather large detector regions, the responses are taken as being midpoint values for each of the detector regions, and linear extrapolation can be used to obtain relatively accurate values for intermediate points. A more complex curve fitting would be required in the case of a rapidly or erratically changing activation rates, but the actual case is smooth and gradual enough to make this unnecessary. The results are estimated as falling off as a function of $x^{-0.6}$ in the region beyond 20 cm into the model, for which data are lacking. This probably overestimates the heat deposited in this region, but the energy deposited is minor compared to the heat loading at the front end. The linking code is presented in appendix A. Due to difficulties in obtaining high-quality ANISN results, this is the code upon which the final package is based.

The ANISN code is similar in structure, the ANISN data being in the form of grays (joules/kg) per unit flux. Although difficulties in bringing the flux in line with MORSE estimates has made it unreasonable to use ANISN as the neutronics code, the difference in magnitude between ANISN and MORSE estimates of activation is within the same order of magnitude, which may indicate that the homogeneous one-dimensional model, although presently inadequate, might be amenable to improvement, as described in the analysis.

The present code can fairly easily be called by the MORSE code through the introduction of a CALL statement before the END statement in the main MORSE program, thereby treating it as a subroutine. However, this will complicate the link with the thermal stress program, since it will require the program to be called from a subroutine. Furthermore, as described in the analysis, the user would still have to enter ANSYS separately to construct the solid model. The system as it runs now requires the user to call on the code ANSYS separately after exiting MORSE. No modification of the MORSE output is required, and the data will be read into ANSYS in the appropriate form by using the command *USE,MACRO2,fluxvalue (flux per cm^2) while in the preprocessor. The biggest hurdle to the user will be running MORSE in the first place, it not being a user-friendly program: making it so would be a project of quite significant proportions.

II. RESULTS OF THE COMPUTER ANALYSIS

1. ANISN vs MORSE

The ANISN computer code suffers from some serious difficulties when an attempt is made to compare it's flux calculations with those of the MORSE model. The fluxes in the program output are given by group and by interval, and the translator program sums the group fluxes to get total flux per interval. The flux should have undergone a normalization, presumably by the division of local flux by the total source flux, as described by the manual. This should yield flux values ranging from zero to, in cases near the front face where backscatter is considerable, somewhat over one. However, after a run with maximum inner and outer iterations set to 35 and 50 respectively, the flux magnitudes that were obtained varied between zero and 7.22 (100% solid density), an unlikely result. Varying the source flux and the density failed to bring the maximum flux below 3.0, although the peak flux did decrease for lower-density cases. Some graphs of ANISN flux magnitudes are shown in fig.5 for distributed-mass calculations, with the structure being modelled as a 20.01-cm. thick copper block, with density varying between 0.3 and 0.6 times actual copper density.

The improbable flux magnitudes obtained from ANISN make any use of ANISN data a gamble, making it necessary in this work to rely principally on MORSE calculations. Some possible

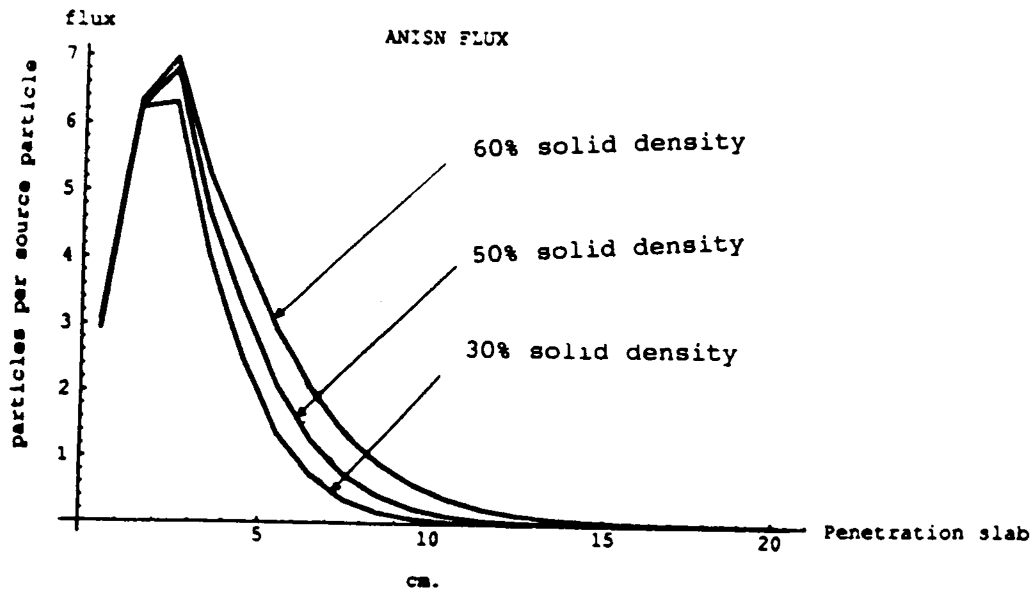

FIGURE 5: Variations of ANISN per unit source flux with changing model density.

explanations for the peculiar ANISN results will be discussed later.

The Monte Carlo code is a statistical process involving none of the approximations associated with discrete ordinates codes, so if the model is realistic the solution obtained should approach reality as the number of particle histories increases. Of course, the fact that the Monte Carlo estimate only provides data for detector regions necessarily coarsens the data, since the flux or activity obtained will be an average over a detector region or volume rather than the point data available from a discrete ordinates calculation.

The MORSE model consists of a copper block with appropriate vacuum channels and water-filled channels cut through it as shown in fig.6.

Calculations are carried out for septa thickness of 2 mm, the results indicating that thicker septa will probably be unnecessary. Activities show a considerable drop over the 30-cm. depth of the model. The calculation uses a collision-density estimator, and divides up the structure into 10 detector volumes.

2. Monte Carlo Details

For the Monte Carlo calculation, the kerma factors and cross section data were obtained from DABL69 (4), a broad-group neutron and photon cross section library, an ORNL library originally designed for nuclear defense applications.

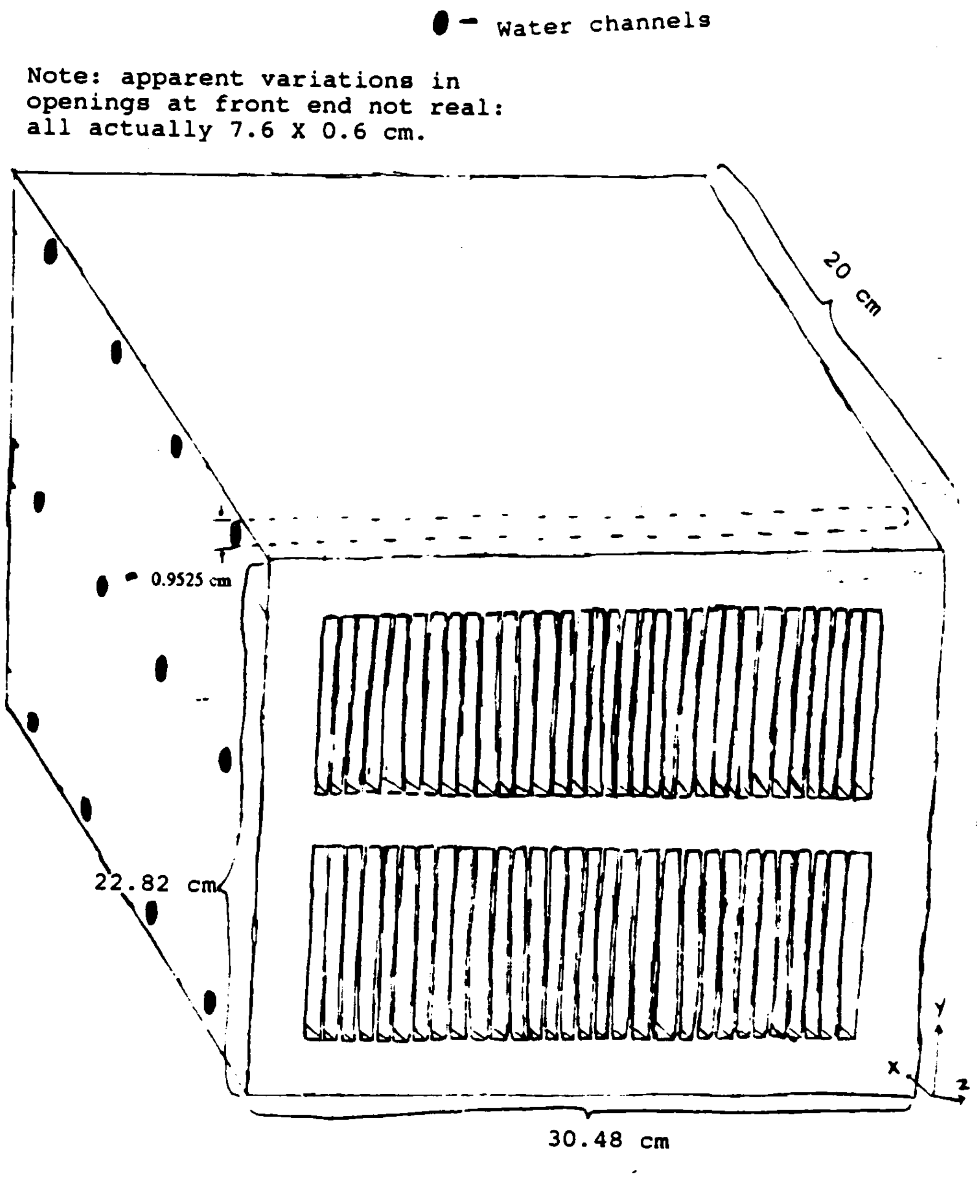

FIGURE 6: Monte Carlo analysis model of the TPX Waveguide System.

It contains 46 neutron groups from 10^{-5} ev to 19.64 mev and 23 gamma groups, from 10 kev to 20 mev. It includes gamma and neutron kerma factors in grays per cm^2 per neutron or gamma ray (unit flux) for copper, hydrogen, oxygen, and a wide variety of fusion-related materials. More details may be found in the ORNL handbook, id number DLC-130, ORNL/TM-10568 (4).

The activities calculated for a flux of 17.6 Mev D-T fusion neutrons, for the varying septum thicknesses, were in the range of $4*10^{-15}$ to $8*10^{-15}$ grays per neutron/cm^2 at the front end, falling off by a factor of 10-20 over the structure length. The total activation must be obtained by multiplying by the area of the structure, as well as correcting for mass. ANISN activities are not equivalent, being given in terms of reactions per cm^3 per second. To obtain numbers for comparison, it is necessary to multiply the group fluxes by the appropriate kerma factors.

A value for the equivalent surface source is not well established, since estimates for the power output of experimental reactors vary widely. Two fairly typical cases (8), with power output/unit area wall of respectively 5 and 0.6 MW/m^2 were chosen. These can be considered as high and low estimates for power production in an experimental reactor. Given that a neutron and an alpha particle are produced for each D-T reaction, with a total energy of 17.6 Mev, and ignoring rarer reactions and charged particles, the equivalent surface source per cm^2, S, can be calculated as

$$S = \text{Power}(j/m^2)/((17.6\text{Mev/reaction})*(1.602*10\text{-}13j/\text{Mev}))$$
$$* \ 2 \text{ particles/reaction}*(1m^2/10{,}000cm^2). \quad (7.)$$

This works out to equivalent surface sources of $4.256*10^{13}$ and $3.5467*10^{14}$ particles per cm^2 per second impinging upon the TPX lattice.

Using output from a 400-minute MORSE run on the vax mainframe at the FEDC at ORNL, with 2-mm septa, the resulting heat generations varied from 0.088 to 1.7024 j/cm^3 in the low-energy case and 0.732 to 14.187 j/cm^3 in the high-energy case.

3. Thermal-Stress Profiles

The thermal stress results were promising. For a 0.6 MW/M^2 flux, there was hardly any temperature rise worth speaking of, as shown in fig. 7 . For the 5 Mw/M^2 flux, a temperature rise of about 20 degrees took place, causing a fair degree of swelling, as shown in figures 8 and 9. Although the swelling is not inconsiderable, there was no buckling, the configuration being symmetrical. Therefore, a case was run in which a 9 w/cm^2 surface flux was applied to one surface but not the other. This led to substantial increases in temperatures, as shown in figures 10 and 11. The temperatures on each side were substantially identical, making stress calculations unnecessary. Pairs of nodes on opposite sides of the septum showed temperature differences of less than 1

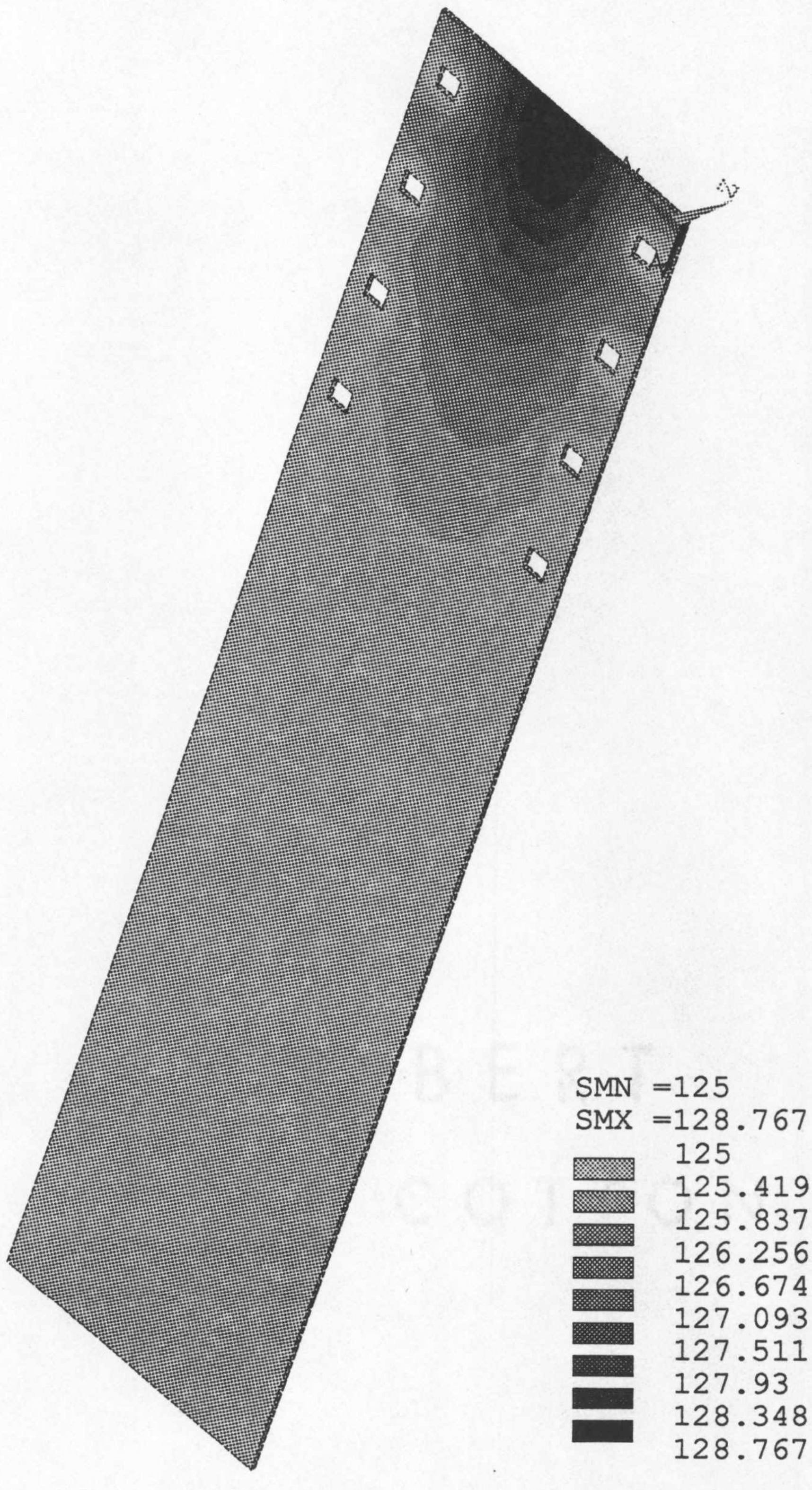

FIGURE 7: Thermal profile of the septum for 0.6 Mw/m^2 -wall.

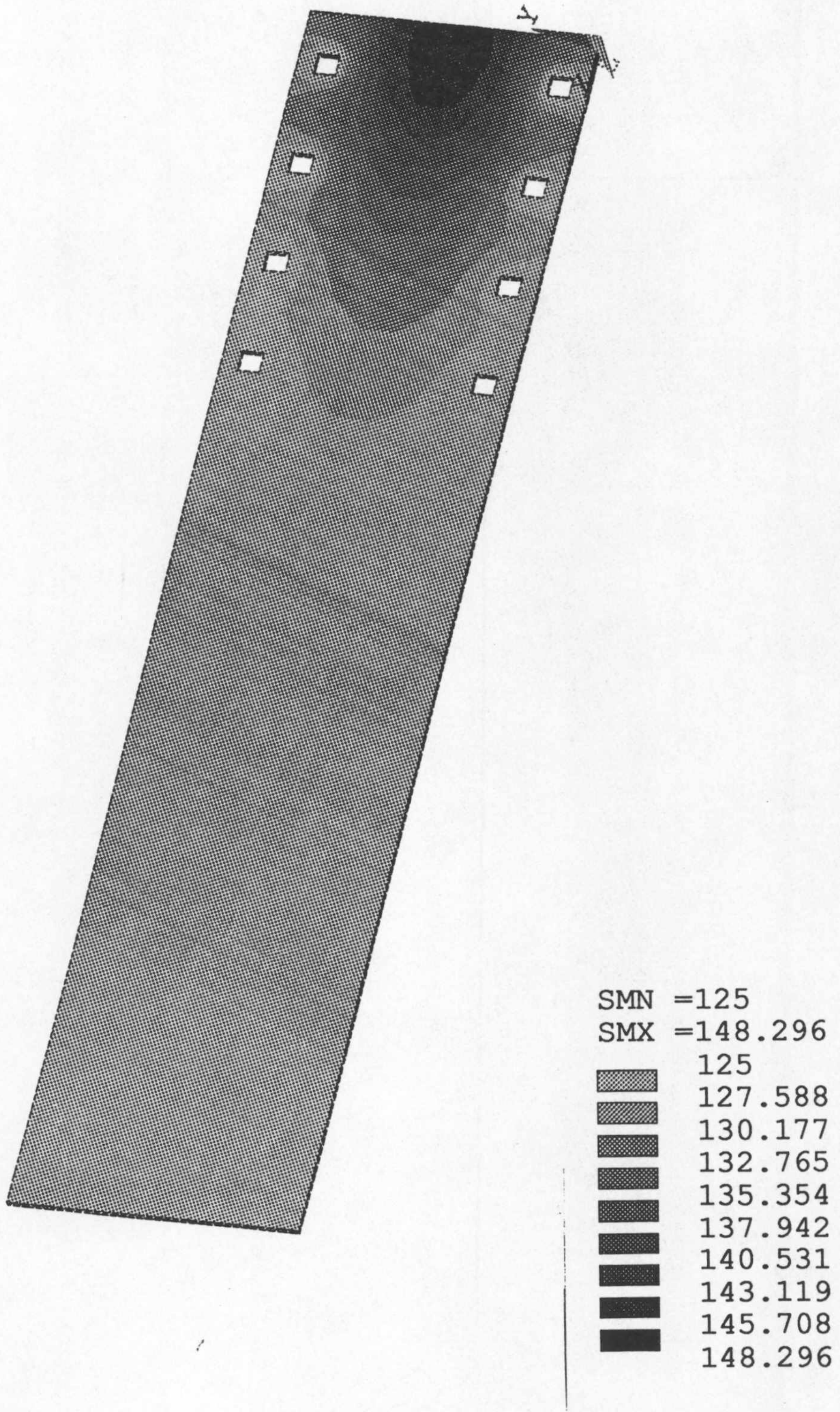

FIGURE 8: Thermal profile of the septum for 5 Mw/m^2 -wall.

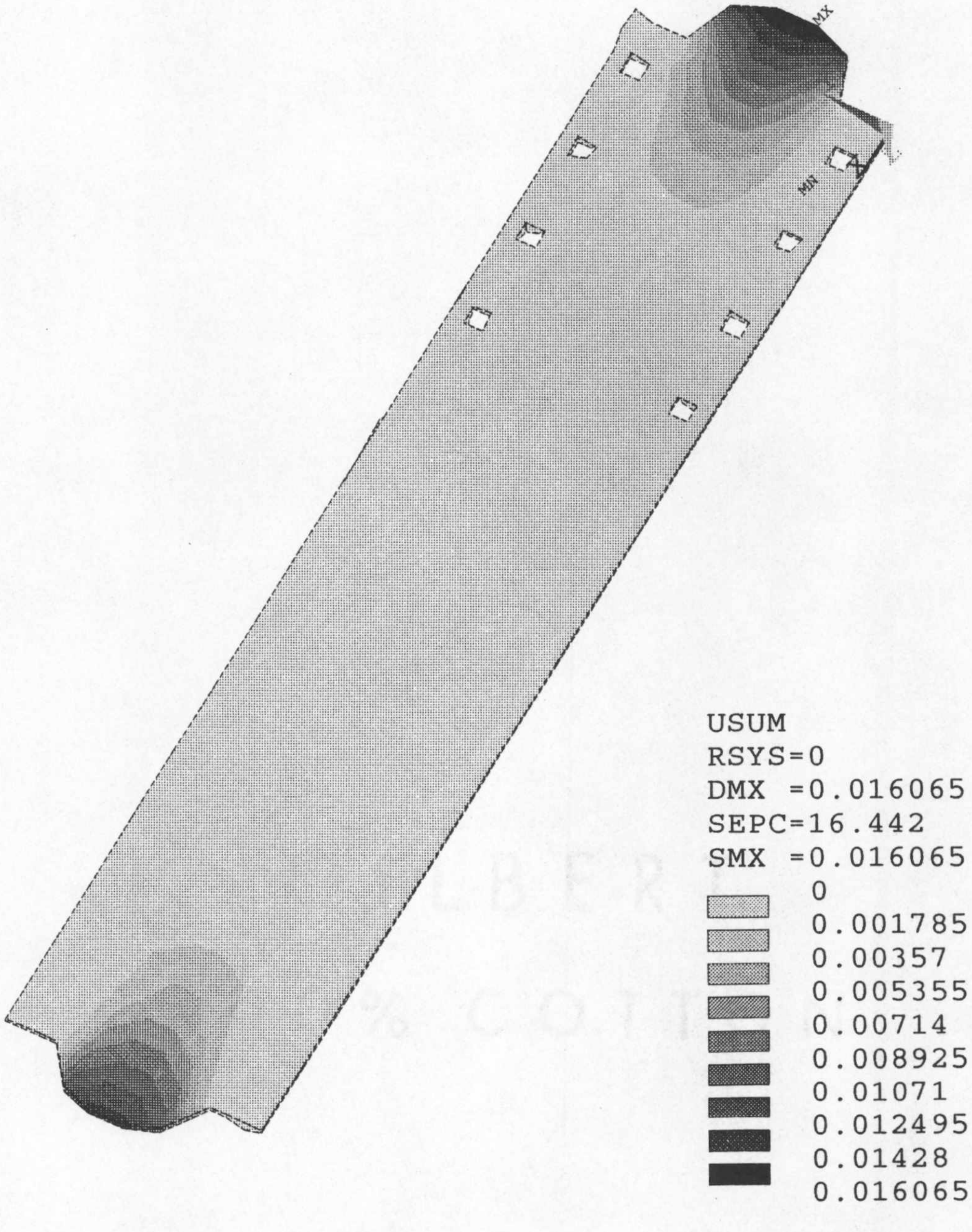

FIGURE 9: Stress profile of the septum for 5 Mw/m^2 -wall (total stress).

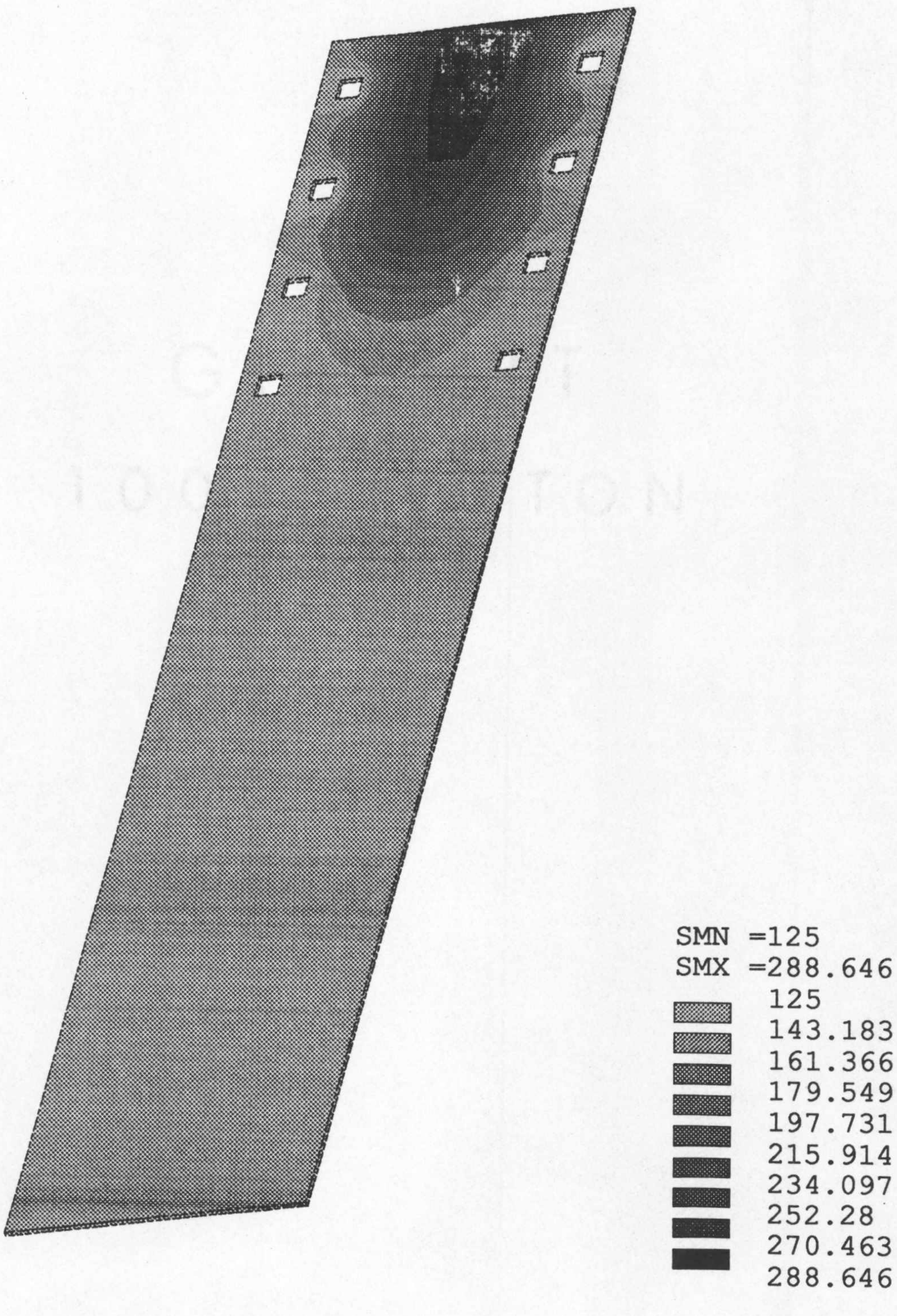

FIGURE 10: Thermal profile of the septum for a 9 w/cm^2 applied flux, heated side.

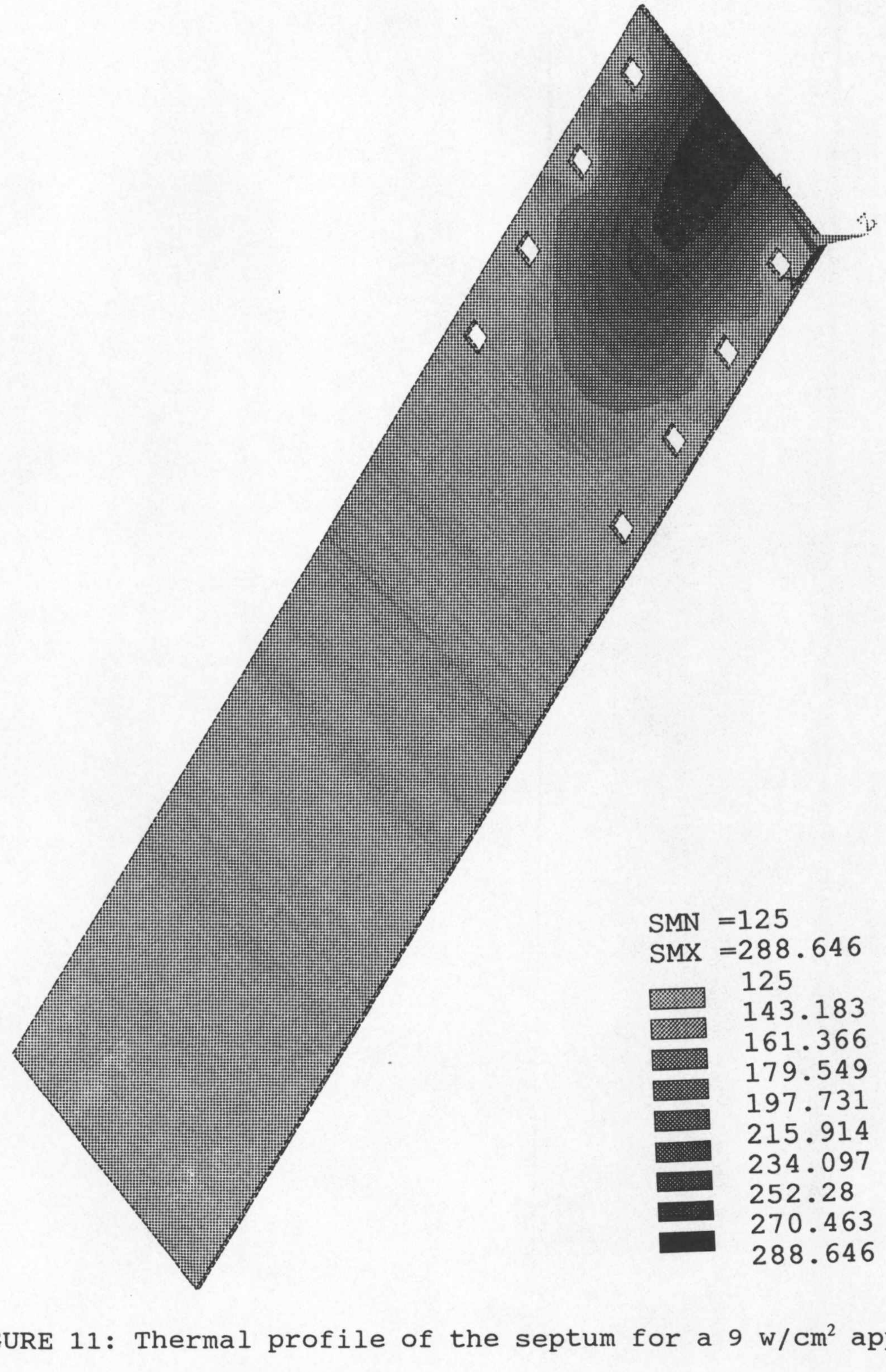

FIGURE 11: Thermal profile of the septum for a 9 w/cm^2 applied flux, unheated side.

°C. The conductivity of copper is high enough, and the septum thin enough, that there is very little in the way of a temperature gradient in the septum. Since in actual operations the asymmetry between the two sides of a septum is likely to be rather less, it can be taken with a fair degree of confidence that uneven heating is unlikely to cause buckling. Sample nodal temperature data are included in appendix C.

One way to "rough estimate" the heating caused by the reactor is to take the charged particle energy per unit area wall as acting like a low-intensity EM flux, reaching the septum walls according to view factors and being absorbed on contact. This is of considerable importance in the present case, since in the case of a D-T reaction, most reactions produce a 3.5 MeV alpha which will be quickly absorbed. As a limiting case, it is assumed that the entire non-neutron fraction of the fusion reaction energy reaches the wall as a heat flux. Using the higher-energy wall loading, and the view factor equation for parallel plates ((9), see appendix B) large increases in temperature were obtained. While the neutron heating had led to a maximum temperature increase of 23 °C, the charged particle flux calculated led to a temperature rise of over 400 °C, as shown in figure 12, indicating a far larger charged particle contribution near the front end.

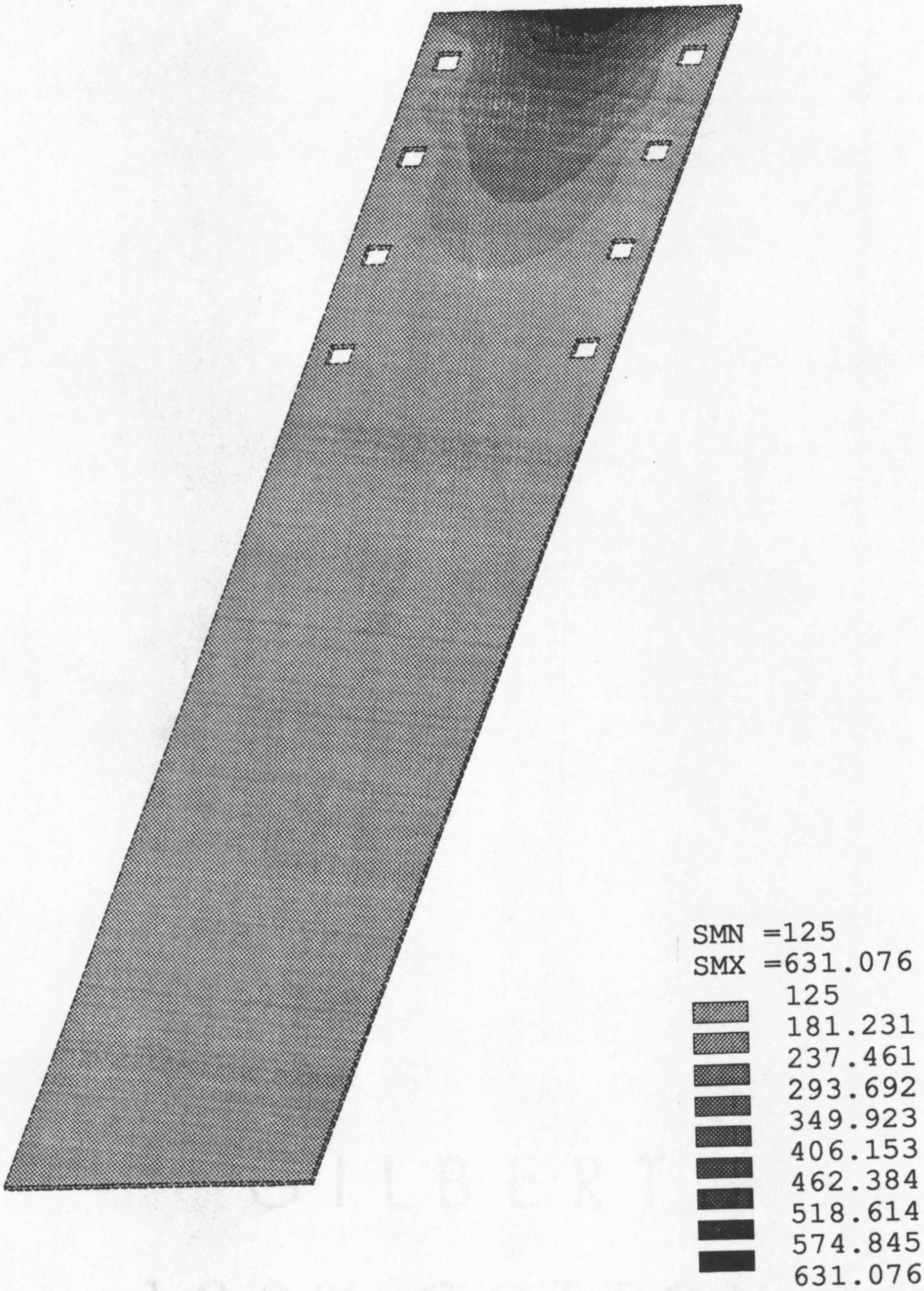

FIGURE 12: Thermal profile of the septum for non-neutron flux.

III. ANALYSIS

1. ANISN PROBLEMS

One of the intended aims of this analysis was the construction of a model that would allow the use of one-dimensional discrete ordinates to solve the neutronics distribution. This was to be accomplished by the substitution of a homogenous mass of reduced density for the mix of solid matter and vacuum gaps in the actual case. Flux results from ANISN, although showing a similar profile to MORSE results, are so unrealistic in their failure to normalize as to discourage the use of ANISN results as input for the thermal stress codes. The activities also differ considerably. For a 5-MW flux, heat generation densities for the partial density closest to the actual mass-volume ratio in the MORSE model (50% solid density) the heat generation rates were substantially higher. For the first two measurement points in the block, equivalent to the middle and endpoint of the first detector region in MORSE, ANISN heat generation rates of 24.6 and 17.83 j/cm^3 were obtained. For the same flux, MORSE gives a first interval average of 14.187 j/cm^3, substantially lower.

Deeper within the structure, the differences become more severe. While MORSE falls off by a factor of about 20 over the first 19 cm, the ANISN has already fallen off by a factor of 20 by 7 cm. into the block, as shown in figures 13 and 14, and has fallen off by a factor of 3500 by 15 cm. into the block.

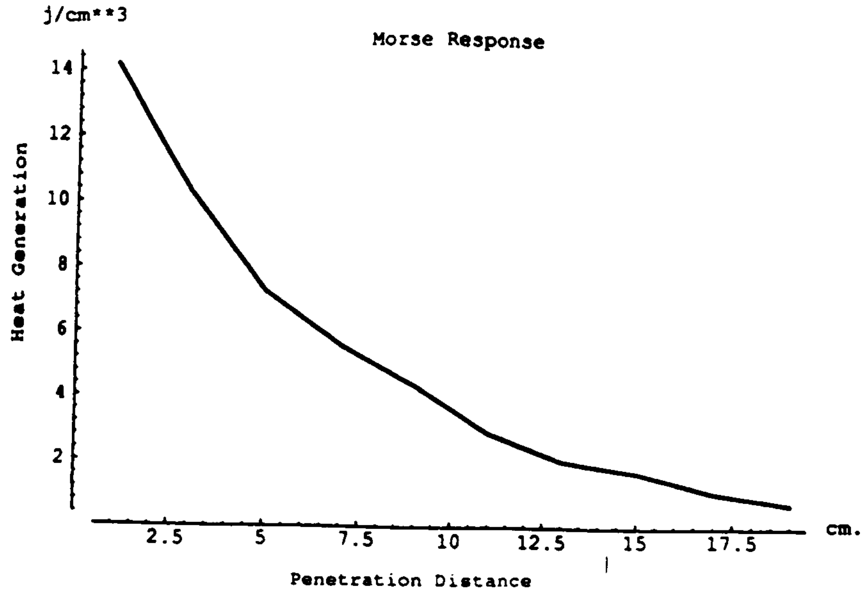

FIGURE 13: MORSE responses in j/cm^3.

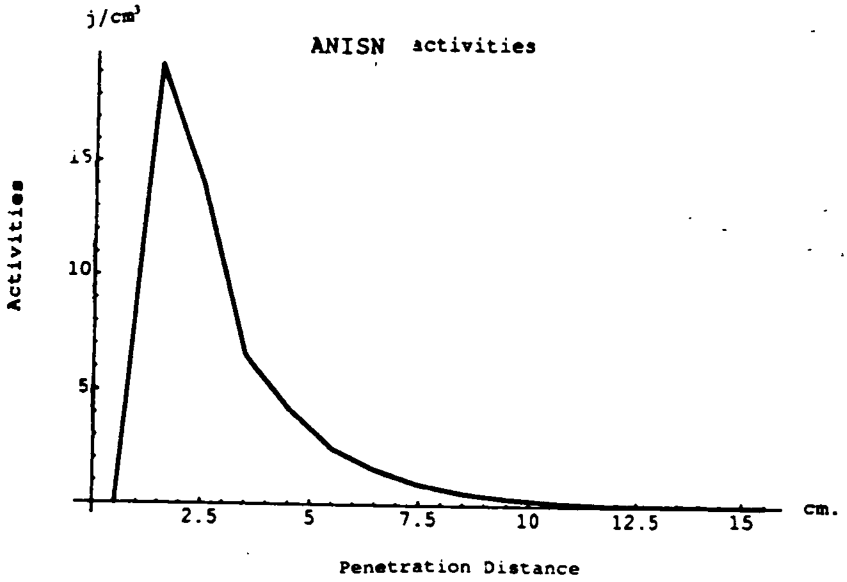

FIGURE 14: ANISN responses in j/cm^3, model at 50% solid density.

The lack of streaming paths in the homogenous model appears to lead to an energy distribution more concentrated at the front end. Of course, the front-end contribution is far more important in both cases, but this rapid drop-off, so inconsistent with a structure so rich in streaming paths, makes it unlikely that the one-dimensional ANISN model is acceptable for thermal-stress calculations.

The ANISN flux distribution, peculiar as it may seem, is probably an artifact of the code setup by the user, rather than a fundamental difficulty with homogenous, one-dimensional systems. ANISN, if running properly, should always normalize the results. The discrepancy in the activities is rather more serious. It appears to indicate that ANISN does not see a decrease in density as being equivalent to an open-latticed structure. If the density is lowered to 10% of that of solid copper, far below the equivalent density, the drop-off in magnitude is closer to the MORSE case (maximum is 18.44 times the minimum), but the maximum heat generation is now only 5.82 j/cm^3. It appears that the discrete ordinates calculations for a homogenous low density block are not equivalent to the results for a Monte Carlo analysis.

The results indicate that the basis for using one-dimensional discrete ordinates could not be established. A homogenous low-density block does not appear to act the same as an open structure with multiple streaming paths, at least not for any great distance. The remaining differences may be correctable, and some ideas to this effect are advanced in the conclusions. However, the results do not warrant the use of ANISN in an induced-stress analysis of the TPX waveguide system.

2. Monte Carlo Results

The Monte Carlo calculation provides more believable results. If the model is realistic, a fair degree of trust into a Monte Carlo analysis is reasonable, since MORSE will eventually converge on the correct solution once enough particle histories have been processed. After a 400-minute run on the FEDC vax, Fractional Standard Deviations have been obtained which vary between 0.02247 and 0.05614 for the first six detectors, which are generally regarded as being statistically acceptable. FSDs increase somewhat for the last four detectors, with a maximum of 0.11387: however, the activation rate has dropped sufficiently by this point that the statistical uncertainty should not lower confidence in the solution obtained to any significant extent. The relevant output is included in appendix C.

The data obtained from shorter runs were poorer in statistical quality and not utilized: for 15-20 minute runs, FSDs of over 30% occurred. This gives rise to the question of how advantageous is the MORSE code in comparison with multidimensional discrete ordinates codes. Given the importance of the ray effect in any discrete-ordinates solution of the TPX structure, MORSE probably retains its primacy. Discrete ordinates codes probably would prove competitive in cases containing more solid matter.

One problem in using Monte Carlo analysis is the lack of fine detail in the radiation profile. In the present analysis data are only provided as averages for large detector volumes, each comprising one-tenth of the entire model volume. However, the model in question is fairly symmetrical, as shown in fig. 5, so there shouldn't be too much variation across the structure's width. There will be a considerable asymmetry in that more neutrons will be absorbed in the solid regions containing the water channels, so the averaged heat generation per volume solid will tend to overestimate thermal loading to the septa. But this, in a case of testing for failure, should lead to a conservative result.

3. The Linking Code

The linking code provides heat generations for the nodes of the thermal-stress solid model. There is not much to say about this since the function it performs is simple enough to not require much elucidation. A commented listing of the code is included in appendix A.

4. Thermal Stress

The thermal stress analysis reveals, most importantly for the problem considered, the lack of necessity for a neutronics analysis. High-energy neutrons, finding the structure relatively transparent, deposit considerably less energy than the charged particles, which will be stopped even by thin septa. The methodology developed in this analysis, although undoubtedly useful in examining other components of a fusion reactor, is actually unnecessary for a thermal analysis of the TPX structure. On the other hand, since a neutronics analysis is necessary in the first place to confirm this, the methodology is not entirely wasted.

ANSYS results have been left out of this analysis. Although the heat generation rates obtained are not too far off from the MORSE case, the greater importance of charged-particle effects leaves little point in examining thermal-stress profiles generated from neutronics data.

Modelling the structure with ANSYS was relatively simple, and the use of this code in future studies is recommended. However, the meshed solid model does take up considerable memory, over 3 megabytes. In doing multiple cases, considerable memory storage capacity will be needed.

In no case examined did the stresses caused by heating lead to buckling. What swelling takes place is a lengthwise expansion at the ends. This will remain the case for all situations up to the point the structure begins to melt, because of the symmetric nature of the problem and the extreme asymmetry that would be required for a serious temperature gradient to take place across the thickness of a septa.

IV. CONCLUSIONS AND RECOMMENDATIONS

It appears from this analysis that a linked neutronics-thermal stress package is indeed feasible and useful in performing analyses of fusion reactor components. The main obstacle to the MORSE novice in performing a rapid analysis lies in the sizeable investment of time and effort involved in setting up a problem for analysis. Also, there is the limited amount of detail available with a reasonable number of particle histories.

Further study of the one-dimensional model is indicated. One approach that might obtain a closer approximation to the Monte Carlo model would be to vary the density of the homogenous model along the length of the slab. Whether this would provide an accurate model in terms of activations remains uncertain.

The linking code performs adequately, given the lack of detail in the MORSE data. It is of course code-specific to ANSYS, although it can be modified fairly easily to take data from ANISN or other neutronics codes. Considering the limited nature of the linking code, the question arises, how much further work is required to create a truly unified code package?

Linking ANSYS and MORSE into a single package is not considered viable, given the necessity of creating a solid model for use in ANSYS. If ANSYS were to be called from MORSE, it would be necessary for MORSE to create a model for ANSYS, presumably by somehow translating the geometry and material data used for the neutronics into ANSYS input data.

The complexity of the problem above is such that this methodology performs the MORSE runs and the construction of the thermal-stress solid model separately, and the linking code is invoked once the ANSYS solution module is entered and is ready to perform a thermal-stress analysis. Therefore, the linking code is a bridge between two stand-alone codes rather than a permanent junction. Given the limitations it would place upon the manipulation of the thermal stress solid model, a truly unified MORSE-ANSYS package may not be practical.

It may be considered as being demonstrated that the TPX waveguide system is not in serious danger of either buckling or melting under the conditions of an experimental fusion reactor. Even if all non-neutron radiations are assumed to be absorbed on contact rather than penetrating, maximum temperatures remain hundreds of °C below melting. Indeed, the contribution made by the charged particles is so much larger near the front end than that of the neutrons that for open-grid structures of this type using D-T reactions, it may be unnecessary to use neutronics calculations for thermal stress calculations. The thinness of the septa prevents a temperature gradient large enough to cause buckling so swelling takes place only at the ends. Indeed, for determining temperature distributions, a two-dimensional model would probably suffice.

Of course, the structure eventually used in actual fusion reactor components may differ considerably from the model used in this analysis. Alloys with lower thermal conductivities may be used, leading to larger temperature gradients. Cooling may be more or less efficient. There are several possible approaches for further research. Some useful subjects of study include the contribution made to swelling by neutron damage, the deposition of charge-particle energy (which *can* be treated as a surface flux), and a more detailed look at cooling mechanisms.

REFERENCES

REFERENCES

1.) ANSYS 5.0 user's manual, Silverado Software,1994

2.) Bahder, Thomas B., Mathematica for Engineers and Scientists. Menlo Park, California: Addison-Wesley Publishing Company, 1995

3.) Bulatt, Bill, ANSYS consultant, SSC. Multiple consultations ANSYS program usage, June-July 1995.Phone: 1-714-964-1552

4.) Ingersoll, D.T.; Roussin, R. W.; Fu, C.Y; White, J.E., DABL-69:a broad-based group neutron/photon cross section library for defense nuclear applications. Oak Ridge National Laboratory, Oak Ridge, Tennessee 37831, June 1989

5.) Oak Ridge National Laboratory, ANISN-ORNL, one-dimensional discrete ordinates transport code system with anisotropic scattering. RSIC computer code collection, CCC-254. Oak Ridge, Tennessee, April,1991

6.) Oak Ridge National Laboratory, MORSE-CG, multidimensional Monte Carlo code system. ORNL version 6174. Oak Ridge, Tennessee, 1989

7.) Peters, George, Thermodynamic properties of high-temperature materials. McGraw-Hill, 1967

8.) Roth, J. Reece, Introduction to Fusion Energy. Charlottesville, Virginia: Ibis Publishing,1988

9.) Siegel,Robert; Howell, John R., Thermal Radiation Heat Transfer. New York, New York: Hemisphere Publishing, 1981

10.) Spenser, Edwin, ANSYS consultant, SSC. Multiple consultations ANSYS program usage, June-July 1995. Phone: 1-714-964-1552

APPENDICES

A. CODE LISTINGS

The following page shows a code listing of the macro that was created for translating MORSE output into ANSYS input, entitled MACRO2.

FIGURE A.1: Listing of MORSE-ANSYS linking code MACRO2

```
C*** This ANSYS macro reads data from the MORSE output, calculates
C*** corresponding heat generations,and calculates values at each model
C*** node by extrapolation. The call is of the form *Use,macro2,argl,
C*** with argl being the overall flux per cm**2
flux=argl *** User Inputted Flux
rho=6.232 *** Correction factor
x=1.0
C*** A text string is used to locate the beginning of the response data.
start=' (detecto'
:BRANCH1
*vre,holdr,gag.ext
(39x,A)
*if,holdr,ne,start,then
*GO:BRANCH1
*endif
C*** Need to skip 4 lines before beginning activity read
*vre,holdr,gag.ext
*vre,holdr,gag.ext
*vre,holdr,gag.ext
*vre,holder,gag.ext
*dim,xpos,array,10
*dim,flx,array,10
C*** The activities are read from the file and then the detector central
C*** points (xpos(i)) and the heat generations (flx(i)) are calculated.
*vre,flx(1),gag.ext
(38x,E10.4)
*do,i,1,10
xpos(i)=x
flx(i)=flx(i)*rho*flux
x=x+2.0
*enddo
C*** The total number of nodes is obtained and a do loop is set up.
*get,numnds,node,,count
*do,i,1,numnds
*get,nd,node,nd,nxth
*get,nx,node,nd,loc,x  *** nx,the position of the present node,is obtained.
qrs=NINT(nx)
*if,qrs,gt,nx,then
qrs=qrs-1
*endif
j=NINT(qrs/2)
*if,j,eq,0,then
j=j+1  *** The number of the detector interval is calculated.
*endif
C*** The heat generation rate is calculated by backward interpolation in
C*** the first interval, as a geometrically decreasing quantity after the
C*** last interval, and by middle interpolation in the other intervals, where
C*** a heat generation rate is available at both ends of the interval.

*if,nx,le,1,then
hetn=flx(j)-(1-nx)*((flx(j+1)-flx(j))/2)
*elseif,nx,ge,19
hetn=flx(10)/((nx+1)-19)**.6
*else
hetn=flx(j)+(nx-xpos(j))*((flx(j+1)-flx(j))/2)
bf,nd,hgen,hetn
*endif
*enddo
```

B.SAMPLE CALCULATIONS

VIEW FACTOR FOR TWO ORTHOGONAL RECTANGULAR PLATES

Viewing the end of the channel opening onto the radiation flux as plate 1, and the septum on one side of the channel as plate 2, the fraction of radiation from plate 1 incident onto plate 2, F12, is calculated as follows:

$$F12=(1/pi*L1)*\{(1/4)*\ln[((1+L1^2+L2^2)^{L1**2+L2**2}*(L1^2)^{L1**2}*(L2^2)^{L2**2}))/$$
$$((1+L1^2)^{L1**2-1}*(1+L2^2)^{L2**2-1}*(L1^2+L2^2)^{L1**2+L2**2})]$$
$$+L1*\tan^{-1}(1/L1)+L2*\tan^{-1}(1/L2)$$
$$-(L1^2+L2^2)^{1/2}*\tan^{-1}(1/(L1^2+L2^2)^{1/2})\},$$

where

L1 = l1/l

L2 = l2/l

l = height of channel opening (plate 1), 7.6 cm.

l1 = distance along channel, the variable x.

l2 = width of channel opening, 0.6 cm.

C.RAW DATA FROM MORSE AND ANSYS

MORSE OUTPUT

responses(detector)

detector	uncol response	fsd uncol	total response	fsd total
1	2.0928E-15	0.02188	6.4185E-15	0.02247
2	8.8356E-16	0.00976	4.6562E-15	0.03463
3	4.8212E-16	0.00755	3.2855E-15	0.03372
4	2.8928E-16	0.00693	2.5472E-15	0.03964
5	1.8699E-16	0.00660	1.9863E-15	0.05537
6	1.2542E-16	0.00637	1.3002E-15	0.05614
7	8.7091E-17	0.00616	9.2803E-16	0.06479
8	6.2130E-17	0.00592	7.7092E-16	0.09638
9	4.4817E-17	0.00579	4.9340E-16	0.08993
10	3.2816E-17	0.00575	3.3135E-16	0.11387

ANSYS NODAL DATA

```
***** POST1 NODAL DEGREE OF FREEDOM LISTING *****

LOAD STEP=      1   SUBSTEP=      1
 TIME=    1.0000        LOAD CASE=   0

THE FOLLOWING DEGREE OF FREEDOM RESULTS ARE IN GLOBAL COORDINATES

  NODE       TEMP
   741    126.12
   742    126.31
   743    126.50
   744    126.64

MAXIMUM VALUES
NODE          185
VALUE      148.30
```

```
LOAD STEP=      1   SUBSTEP=      1
 TIME=    1.0000        LOAD CASE=   0

THE FOLLOWING DEGREE OF FREEDOM RESULTS ARE IN GLOBAL COORDINATES

  NODE       TEMP
   741    129.02
   742    129.65
   743    130.27
   744    130.73

MAXIMUM VALUES
NODE          196
VALUE      626.70
```

VITA

Bruce Munro was born in New York city, New York, on April 30, 1968. He lived in Scotland from the ages of two till four, and then in Spain till he was eleven. During that time he was taught at home, by his parents. His family then moved to Albuquerque, New Mexico, where he attended the Bernalillo County school system, graduating from Highland Highschool in the spring of 1987. He then attended the University of New Mexico in Albuquerque from the fall of 1987 till the spring of 1992, with a one-year absence (1989-1990) at Rensselaer Polytechnic Institute in Troy, New York, graduating with a Bachelor's in Nuclear Engineering. Beginning in the summer of 1992, he entered the Nuclear Engineering Master's program at the University of Tennessee at Knoxville, and will complete his Master's program in August, 1995.