EXPERIMENTAL AND COMPUTATIONAL DETERMINATION OF RADIATION DOSE RATES IN THE SLOWPOKE-2 RESEARCH REACTOR AT THE ROYAL MILITARY COLLEGE OF CANADA/COLLÈGE MILITAIRE ROYAL DU CANADA

by

Greg Brian Lamarre

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ABSTRACT

The first SLOWPOKE-2 research reactor designed to use Low Enriched Uranium (LEU) dioxide fuel was commissioned at the Royal Military College of Canada/Collège militaire royal du Canada in September 1985. The reactor is used for teaching, training and research activities, mostly, neutron activation analysis, neutron radiography and radiation effects studies.

Some of these research activities, however, require sound knowledge of the dose rates at the various irradiation sites within the reactor container, as well as at incremental positions within the pool. Accurate measurements of the particle fluxes were attempted before, but unfortunately yielded limited information due to the complexity of the composite radiation field and the capabilities of the instrumentation at that time. The present work aims at yet another attempt to gather sound dose rate data, using improved radiation detectors on one hand, and better computational resources (both hardware and software) on the other hand.

The research focussed on three main classes of particles: neutrons, gamma rays, and charged particles. For the neutrons, the work concentrated on the computer simulation of the reactor core, including the pool, using the latest WIMS-AECL cell code coupled with the ENDF/B-V data library. The results obtained were compared with the previous physical measurements, since little improvement could be obtained by repeating the neutron activation (NA)-based experimental measurements already done in the neutron flux mapping. The WIMS-AECL-generated thermal flux values compared very well with the measured sub-cadmium NA values (within 5%). For the epi-cadmium flux values, the computer simulation proved to be a

more powerful tool than the experimental method, since it allowed a 26-neutron energy group representation of the flux distribution (a significant improvement over the experimental work in which these neutrons are lumped into two groups). This more detailed calculated neutron energy spectrum permitted a much improved neutron dose prediction for the various irradiation sites around the core and in the pool.

The research then concentrated on obtaining data for the gamma radiation, both experimentally and computationally. For the experimental work, CaF_2:Mn thermoluminescent dosimeters (TLDs) were chosen to measure the gamma doses at the various irradiation sites, and at 5-cm increments within the pool at the reactor mid-height. The analytical results were based mostly on the code MICROSHIELD Version 5. In this part of the research, several limitations of this code had to be compensated for, such as the backscattering of gamma photons within the material located beyond the dose point.

The research was completed by including in the model the effects of both energetic electrons and recoil protons, which contribute to the overall dose received by the irradiated samples. Computational models were written based on fundamental physical principles, and calculated dose rates were obtained for the contributions of these charged particles.

The results for the electromagnetic radiation show a significant discrepancy between experimentally-determined and calculated dose rates, with the TLDs always over-predicting the rates. While the present numerical model does not account for all of the effects, the TLD's response is obviously affected by the presence of radiation other than gamma rays. Additional work is needed to sort out the various effects individually. Nevertheless, the data obtained from the present research represent a significant improvement in the quality and the quantity of knowledge on the dose rates in the SLOWPOKE-2 reactor container and pool.

RÉSUMÉ

Le premier réacteur de recherche SLOWPOKE-2 qui utilise du bioxyde d'uranium faiblement enrichi était mis en service au Royal Military College of Canada/Collège militaire royal du Canada en Septembre 1985. Ce réacteur est utilisé pour l'enseignement, l'entraînement, et les activités de recherche telles que principalement, l'analyse par activation neutronique (AAN) et l'étude des effets des radiations.

Ces activités de recherche ont besoin d'une connaissance profonde des taux de dose aux sites d'irradiation dans le vesseau du réacteur de même qu'à une série de positions sélectives dans la piscine. De mesures physiques des flux de particules ont été effectuées dans le passé, mais ces résultats n'ont procuré que de l'information imparfaite à cause de la complexité des radiations et des performances limitées de l'instrumentation. Le présent travail vise à fournir des taux de dose plus précis en utilisant de meilleurs détecteurs de radiation et des ressources informatiques améliorées (logiciels et ordinateurs).

La recherche a mis l'accent sur trois classes de particules: les neutrons, les gammas et les particules chargées. Pour les neutrons, le coeur du réacteur était simulé à l'aide du code WIMS-AECL associé avec la librairie ENDF/B-V. Les résultats obtenus ont été comparés avec les mesures expérimentales du flux neutronique déjà effectuées, étant donné que répéter ces mesures par la technique d'AAN ne pouvait guère améliorer la précision de ces mesures. Les flux de neutrons thermiques obtenus par WIMS-AECL se comparaient très bien avec les résultats expérimentaux pour les neutrons infra-cadmium, l'écart étant inférieur à $\pm$ 5%. Pour les valeurs

de flux épi-cadmium, la simulation informatique était beaucoup plus valable que les méthodes expérimentales, puisque l'on obtenait une représentation en 26 groupes d'énergies de neutrons (une grande amélioration sur les mesures expérimentales où les neutrons n'étaient regroupés que dans 2 différents groupes d'énergie). Cette distribution de flux en 26 groupes a permis des calculs beaucoup plus précis des doses de neutrons aux sites d'irradiation et aux positions dans la piscine.

Une fois les calculs de dose de neutrons complétés, le recherche s'est concentrée sur les doses de gammas, au moyen de mesures expérimentales et de calculs informatiques. Pour la partie expérimentale, les dosimètres thermoluminescents (DTLs) ont été choisis pour mesurer les doses de gamma aux sites d'irradiation et à la sélection de sites distants de 5-cm dans la piscine du réacteur, selon le plan mitoyen du coeur du réacteur. Les résultats analytiques étaient calculés par le code MICROSHIELD Version 5. Dans cette partie du travail, il a fallu modifier ce code pour tenir compte de certains phénomènes, comme, par exemple la rétro-diffusion des gammas dans les matériaux situés au-delà du point de dose.

La recherche était complétée en incluant les effets des électrons énergiques et des protons de recul qui contribuaient aussi à la dose totale reçue par les matériaux irradiés. Des modèles informatiques ont été conçus sur la base des phénomènes physiques et les taux de dose ont pu être ainsi obtenus pour les contributions de ces particules chargées.

Les résultats pour la radiation électromagnétique démontraient un écart significatif entre les taux de dose expérimentaux et calculés; les DTLs produisant des taux plus grands que les mesures expérimentales, et cela de façon consistante. Même si les méthodes numériques ne pouvaient pas tenir compte de tous les effets, la réponse des DTLs était probablement affectée par la présence des autres sources de radiations, notamment les neutrons, et des recherches additionnelles sont nécessaires pour évaluer ces effets variés avec précision. Il reste que l'information obtenue par ces recherches représente une amélioration importante de la qualité et de la quantité des connaissances sur les taux de dose dans le vesseau du réacteur SLOWPOKE-2 en plus que dans la piscine elle-même.

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Dedicated to my wife Megan

whose love and encouragement

helped me to achieve this goal

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LIST OF ACRONYMS

AECL	**Atomic Energy of Canada Limited**
ANS	**American National Standards**
BNL	**Brookhaven National Laboratory**
CANDU	**Canadian Deuterium-Uranium reactor**
DREO	**Defence Research Establishment Ottawa**
ENDF/B-V	**Evaluated Nuclear Data Files (Version 5)**
ICRP	**International Commission on Radiological Protection**
LEU	**Low-Enriched Uranium**
MCNP-4A	**Monte Carlo Napier Program (Version 4-A)**
MS 5	**MICROSHIELD (Version 5)**
NA	**Neutron Activation**
NAA	**Neutron Activation Analysis**
NNDC	**National Nuclear Data Centre**
PMT	**PhotoMultiplier Tube**
RMC	**Royal Military College of Canada**
SLOWPOKE-2	**Safe LOW POwer C(K)ritical Experiment (Version 2)**
SRIM	**Stopping-Power and Ranges In Matter**
TL/TLD	**ThermoLuminescent/ThermoLuminescent Dosimeter**
TRIM	**TRansport of Ions in Matter**
WIMS	**Winfrith Improved Multi-Group Scheme**

NOMENCLATURE

A	Taylor series buildup coefficient
$B(\mu R)$	buildup factor
b,kb,mb	barns, kilo-barns, milli-barns (units of microscopic cross-sections, 1 barn being equal to $10^{-24}cm^2$)
Bc	critical buckling (typical units of cm^{-2})
Bq	becquerel (equal to 1 disintegration-s^{-1})
$\dot{D}$	dose rate (typical units of Gy-h^{-1})
eV, keV, MeV	electron-volt, kilo-electron-volt, mega-electron-volt (units of energy, 1 eV being equal to 1.602×10^{-19} joules)
$\dot{E}$	energy deposition rate (typically in units of J-s^{-1})
$\overline{E}_a$	average energy deposited due to neutron absorption reaction
$\overline{E}_s$	average energy deposited due to neutron scattering reaction
$f_i(E)$	gamma spectrum due to capture, inelastic scattering in isotope i
Gy, μGy	gray, micro-gray, unit of absorbed dose (1 Gy is equal to 1 J-kg^{-1})
I	specific ionization (number of ion pairs created by a proton of average energy $\overline{E}_p$
J_s	particle current density
k	multiplication factor
k_{eff}	effective multiplication factor-lattice eigenvalue
k_{inf}	infinite multiplication factor-lattice eigenvalue
kW_{th}	kilowatts of thermal power
M	mass of specific volume

mk	milli-k (unit of excess reactivity, equal to a reactivity of 0.001)
n	neutron
N_i	atomic density of isotope i (typically units of atoms-cm^{-3})
N(E)	prompt, delayed fission gamma spectrum
p	proton
$\overline{P}$	average reactor power density (typically in units of W-cm^{-3})
R	Roentgen (unit of exposure dose used for γ, x-rays, equal to the deposition of 88 erg per gram of dry air)
rad	radiation absorbed dose (equivalent to 0.01 J-kg^{-1})
$S_{v\gamma p}$, $S_{v\gamma d}$	prompt, delayed gamma intensity (typical units of photons-cm^{-3}-s^{-1})
$S_{\gamma ci}$, $S_{\gamma ai}$, $S_{\gamma si}$	capture, activation, and inelastic scattering gamma intensity in isotope i (typical units of photons-cm^{-3}-s^{-1})
Sv	Sievert, unit of dose equivalent
α_1,α_2	Taylor series buildup coefficients
β	beta particle
γ	gamma photon
ϵ_{rel}	relative efficiency of TLD
λ_i	decay constant of radioisotope i (typical units s^{-1})
μ	linear attenuation coefficient (typical units cm^{-1})
μ_a/ρ	mass absorption coefficient (typical units cm^2-g^{-1})
μ_{sc}	linear scattering coefficient (typical units cm^{-1})
μC, nC	micro-coulomb, nano-coulomb (units of charge)
μs	micro-second (equal to 10^{-6} s)
ν	average number of neutrons liberated for each neutron absorbed in a fission reaction

ξ	average logarithmic energy decrement per collision
Σ	summation symbol
$\Sigma_i(E)$	total macroscopic cross-section of isotope i at energy E (typical units cm^{-1})
$\Sigma_a(E)$	macroscopic absorption cross-section at energy E (typical units cm^{-1})
$\Sigma_s(E)$	macroscopic scattering cross-section at energy E (typical units cm^{-1})
$\Sigma_{is}(E)$	macroscopic inelastic scattering cross-section at energy E (typical units cm^{-1})
$\Sigma_f(E)$	macroscopic fission cross-section at energy E (typical units cm^{-1})
$\sigma_i(E)$	total microscopic cross-section of isotope i at energy E (typical units b)
$\sigma_a(E)$	microscopic absorption cross-section at energy E (typical units b)
σ_c^{th}	microscopic capture cross-section at thermal energies (typical units b)
$\sigma_{inl}(E)$	microscopic inelastic scattering cross-section at energy E (typical units b)
$\phi(E,r)$	particle flux at energy E and radius r (typical units particles-cm^{-2}-s^{-1})
$\chi(r,E'\rightarrow E)$	fission yield spectrum for incident neutron energy E' and outgoing neutron energy E

CHAPTER 1 - INTRODUCTION

The SLOWPOKE-2 or S(afe) LOW PO(wer) K(Critical) E(xperiment) is a research reactor developed by Atomic Encrgy of Canada Ltd (AECL) during the period from 1968-1971. It is a small, light water cooled and moderated, pool type reactor ideally suited for research facilities such as the SLOWPOKE-2 Facility at the Royal Military College of Canada (RMC). The layout of the reactor facility can be seen in Figure 1. The reactor lower assembly can be seen in Figure 2. The maximum thermal power output for the SLOWPOKE-2 is 20 kW_{th}. At half power, the thermal neutron flux measured at a position within the beryllium reflector by a self-powered neutron detector is 5×10^{11} neutrons-cm^{-2}-s^{-1}. It is an inherently safe reactor in terms of power transients, due to its large negative temperature coefficient of reactivity and limited excess reactivity. This combination allows for the reactor to be operated remotely in automatic mode, making it the only type of reactor in Canada licensed for such operation. The height of pool water above the core (approximately 5 metres) provides suitable radiation shielding such that access to the pool is available. This access has enabled researchers at RMC to conduct a great deal of work in the fields of neutron flux measurements by neutron activation (NA), in pool dosimetry experiments, neutron radiography, and radiation processing of high polymers and advanced composite materials. The view from above the SLOWPOKE-2 reactor pool looking down at the reactor assembly can be seen in Figure 3. The access available to researchers of the reactor pool can be clearly seen.

The critical assembly contains the reactor core, comprised of a zirconium alloy cage and 198 fuel elements. The fuel elements are sintered UO_2 enriched to 19.89% in ^{235}U. The

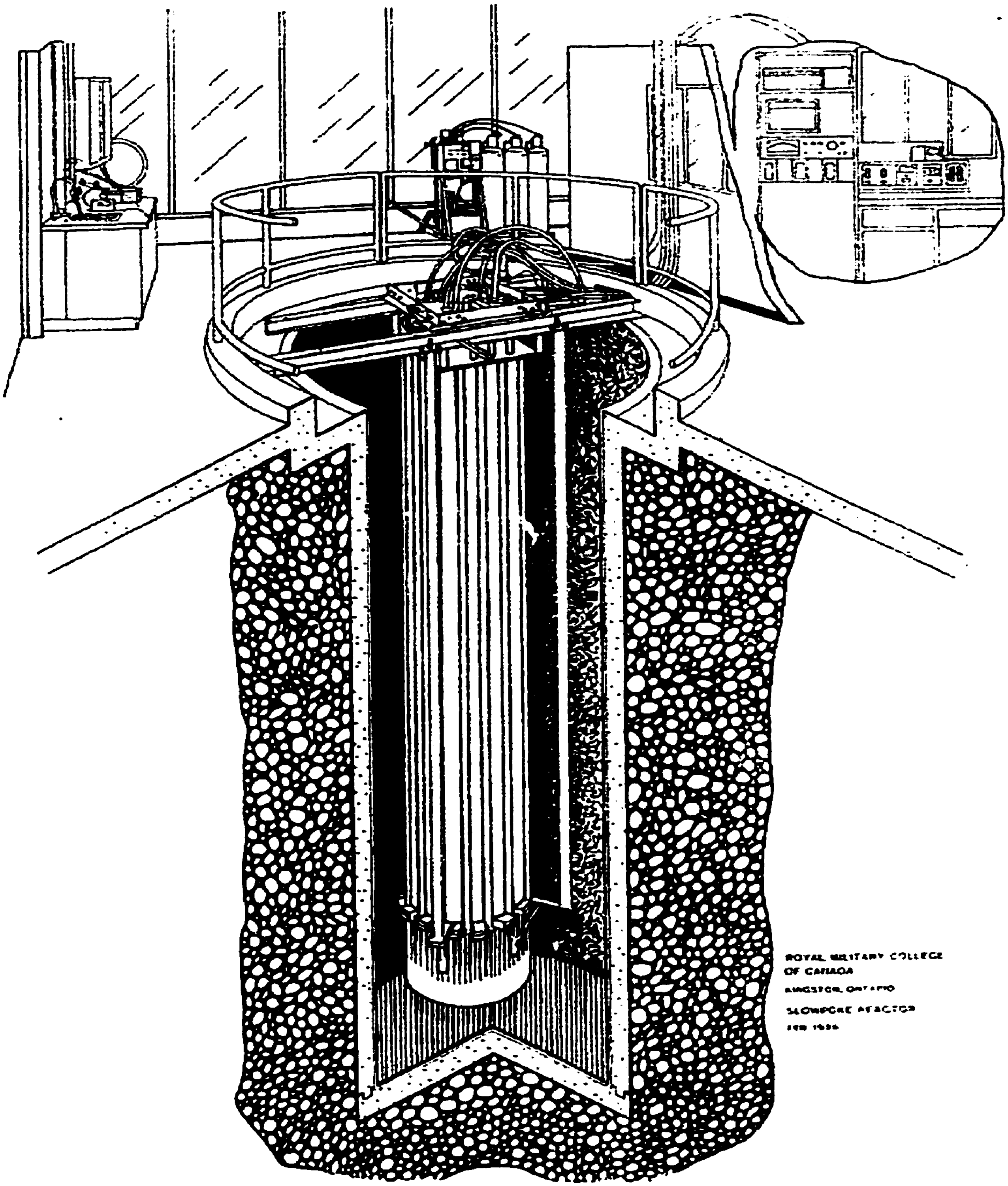


Figure 1: SLOWPOKE-2 Nuclear Reactor

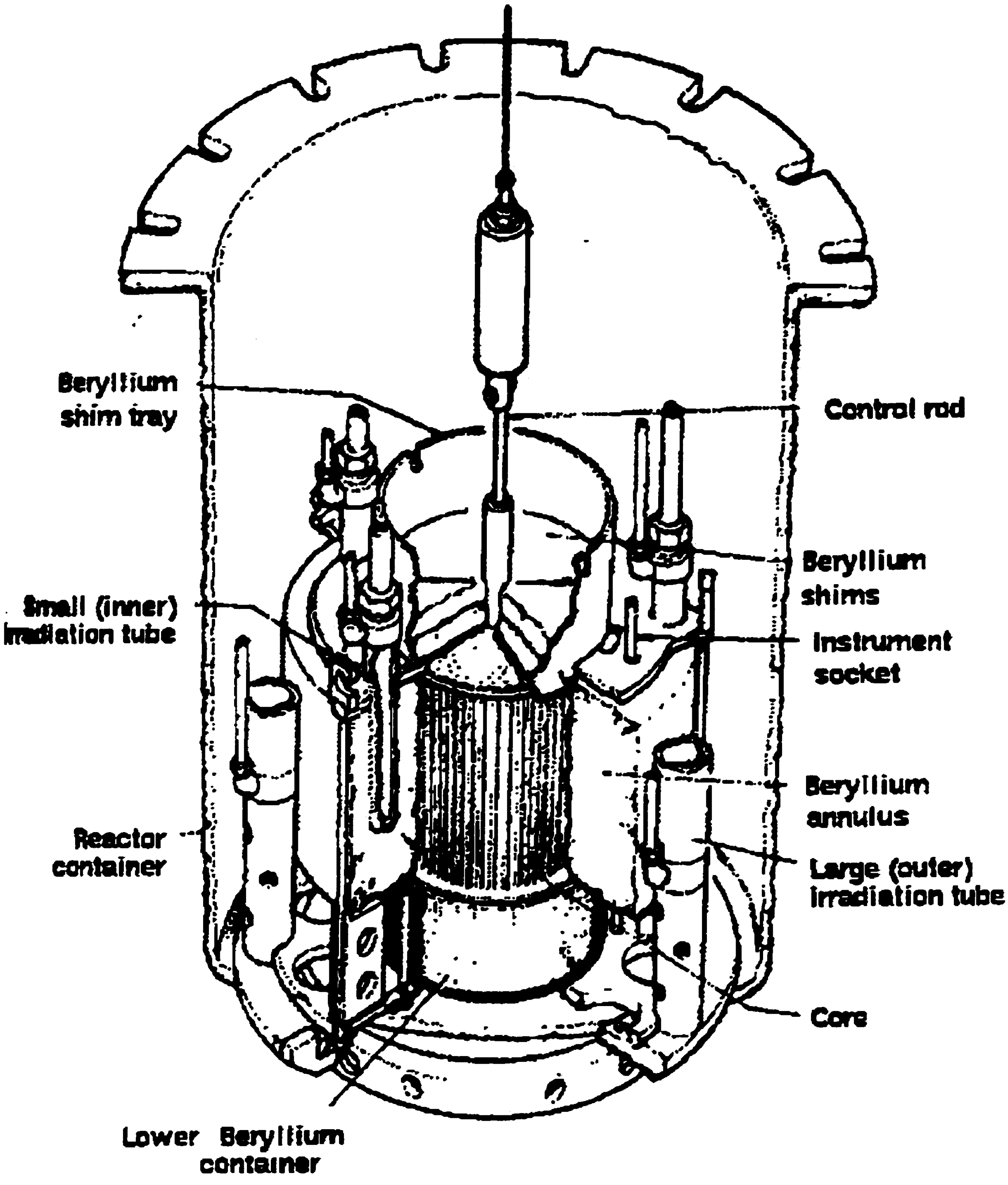


Figure 2: Critical Assembly and Reactor Lower Assembly

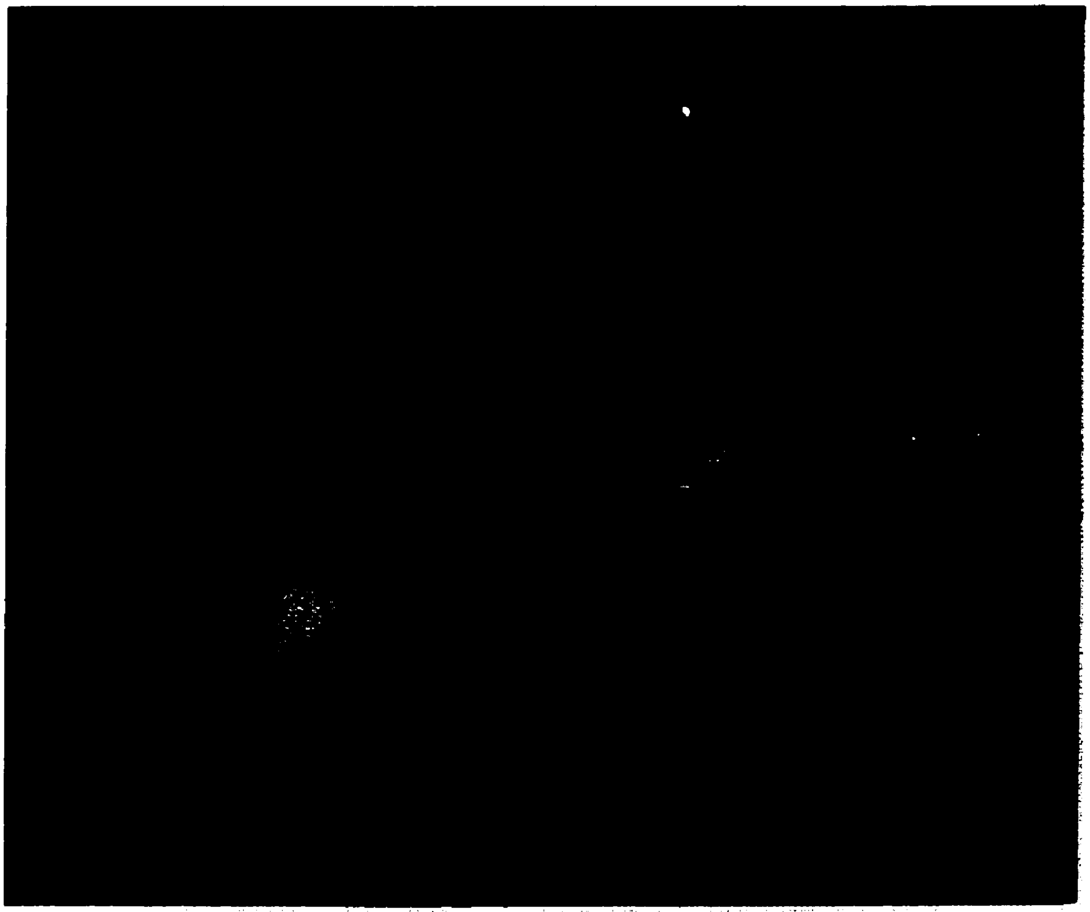


Figure 3: View from above SLOWPOKE-2 Pool

SLOWPOKE-2 at RMC is the first of the SLOWPOKE-2 reactors provided with this Low-Enriched Uranium (LEU) fuel. The fuel elements are clad in Zircalloy-4. The reactor core cylinder is small, only 22 cm high and 22 cm across. Light water (H_2O) is both the moderator and cooling medium for the critical assembly. Surrounding the core is a 10 cm thick beryllium annulus which acts as both a reflector and moderator of neutrons. Neutron reflection above the core is provided by beryllium shims and below the core is provided by a beryllium slab. Between the outer diameter of the beryllium annulus and the reactor vessel container is approximately 9.6 cm of light water (except at one 45° quadrant where a D_2O container was installed). There are a total of nine irradiation sites (5 inner sites within the beryllium reflector, and 4 outer sites within the light water annular region). The reactor vessel container is comprised of aluminum. It is 0.95 cm thick and forms the outer barrier of the reactor assembly. Beyond the reactor container wall is the light water of the pool. The distance from the vessel wall to the pool wall varies as the reactor container is situated off-centre in the pool. The smallest container wall-to-pool wall distance measures approximately 1 metre. A layout of the critical assembly can be seen in Figure 4. The positions of the inner and outer irradiation sites can be clearly seen from this schematic. A mock-up of the RMC reactor container can be seen in Figure 5. It should be noted that all components are full scale with the exception of the height of the reactor container. In Figure 6, a close-up of the reactor lower assembly can be seen.

1.1 PROBLEM DEFINITION

Since its commissioning in 1985, experimental dosimetry techniques around the SLOWPOKE-2 at RMC have consisted solely of thermal and epithermal neutron flux measurements. In particular, up until now, the NA work previously conducted (Andrews, 89) had been the most complete particle flux mapping around the core of the SLOWPOKE-2. This work provided detailed thermal neutron flux mapping as well as some data on thermal-to-epithermal ratios in the irradiation sites and at incremental positions in the pool. An earlier version of WIMS-CITATION (deWit, 89) was used to model the excess reactivity of the SLOWPOKE-2 and to provide thermal/epithermal neutron flux values as a means of comparison to Andrews' work. Recently, Monte Carlo Napier Program (MCNP-4A, Pierre and Bonin, 96 and 99) was used to model the SLOWPOKE-2 resulting in excess reactivity values of a much higher accuracy than the earlier WIMS-CITATION results. None of the previously described

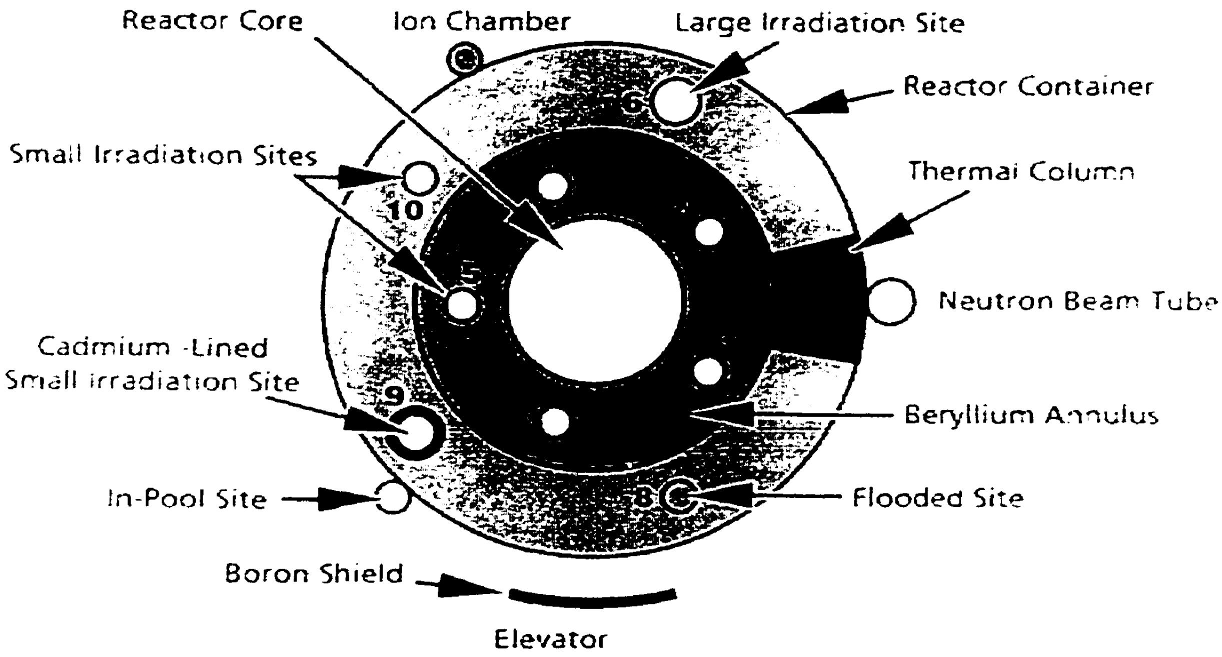


Figure 4: Critical Assembly Layout

techniques provided much accurate information into radiation particle flux distributions as a function of energy. NA cannot provide reliable information on fast neutrons and only limited epithermal data. The accuracy of thermal neutron dose determinations using NA measurements is limited by the fact that results are presented as energy-independent (i.e., one energy group below Cd-cutoff). MCNP, although a powerful tool in terms of reactivity calculations, was not used for flux mapping. The previous attempt at modelling the SLOWPOKE-2 (deWit, 89) was found to produce thermal neutron flux values which differed significantly from those of Andrews. As well, the CITATION output listed neutron flux in only four energy groups.

The emerging polymer-irradiation research around the reactor container of the SLOWPOKE- 2 necessitated a more complete understanding of the radiation particle fluxes present. In particular, radiation doses due to neutrons, gamma rays, and charged particles were considered and a detailed radiation dose rate mapping was conducted. This research would not only form a needed frame of reference for past and future material irradiation experiments around the core of the SLOWPOKE-2 reactor, but would also encourage more refined approaches to particle dose rate mapping in the future as more advanced codes and improved techniques become available.

1.2 APPROACH AND OBJECTIVES

In order to map the energy-dependent neutron fluxes across the reactor assembly effectively, it was decided to use the latest version of WIMS-AECL (Griffiths, 94) in an attempt to model the SLOWPOKE-2 core. WIMS or Winfrith Improved Multigroup Scheme is a multi-group neutron transport code used for reactor lattice-cell calculations. Originally created at the United Kingdom Atomic Energy Establishment, Winfrith, Dorset, WIMS-AECL refers specifically to versions of WIMS developed and maintained independently by AECL. Although used primarily for CANDU analysis, it was found that WIMS-AECL could be adapted to the SLOWPOKE-2 core to provide accurate neutron flux measurements. WIMS-AECL provided energy-dependent regional neutron flux measurements in two dimensions as well as macroscopic absorption and scattering cross-section values necessary for the eventual calculation of neutron radiation dose rates.

It was desired that gamma dose rates be measured experimentally in addition to being calculated analytically. One of the aims of the thesis was to evaluate different dosimetry techniques and select the most appropriate for use in mapping gamma doses at various positions around the SLOWPOKE-2 core. Thermoluminescent dosimetry (TLD) turned out to be the preferred technique following comparison with such dosimeters as gamma bubble detectors and high dose Geiger-Mueller (GM) tubes. The TLD used was CaF_2:Mn (calcium fluoride with manganese activator). The reasons for the selection of CaF_2:Mn TLDs for experimental gamma dosimetry will be discussed in detail in Section 4.2. Following calibration with a reference gamma source, the TLD provided absolute gamma dose values which could easily be converted to gamma dose rates. Similar to the WIMS-AECL calculations, TLD gamma dose measurements were recorded at the inner and outer irradiation sites as well as at incremental positions out into the reactor pool. Due to the fact that WIMS-AECL calculations are done in two dimensions, TLD mapping was done at the reactor mid-plane position only (in the axial direction). It is at this plane that results would be most useful for future material irradiation experiments as well.

MICROSHIELD Version 5 (MS 5, Grove, 98) was used to calculate analytically the dose rate due to gamma rays at a dose point along the cartesian coordinate system. MS 5 was developed by Grove Engineering of Rockville, Maryland as an analytical tool for estimating the flux and exposure rates due to gamma radiation from a shielded source. The SLOWPOKE-2 core was modelled as a right cylinder homogeneous gamma source with the beryllium reflector, light water annular region, reactor vessel and appropriate annular pool regions treated as annular shields[1]. MS 5 allowed only a top shield (in this case, the beryllium shims) and not a bottom

[1] For the purposes of this work, the term 'shield' refers to those materials surrounding the core of the SLOWPOKE-2 (i.e., the beryllium reflector and water annulus, etc.).

shield to be considered in its calculations. MS 5 enabled the core and surrounding shield geometries to be modelled in 3 dimensions. The case outputs from MS 5 provided a photon fluence rate and an energy fluence rate at the designated dose point. The fluence or flux is defined as the time-integrated flux of particles per unit area (Lamarsh, 83). From these, and using ICRP 51 (Grove, 98) quality factor conversions, exposure rates, energy absorption in air rates and effective dose equivalent rates were provided.

Although MS 5 considers buildup in its calculations, it does not account for gamma backscatter from materials immediately beyond the dose point, which had to be considered when comparing analytical with experimental gamma dose values. The relative gamma contribution was calculated numerically at each dose point and then added to the MS 5 output. A rough estimation of the backscattering effect was conducted (Bonin, 96), indicating that the contribution to the gamma flux due to this phenomenon was approximately 9.2% of the incident gamma flux.

The gamma source term to be considered in MS 5 consisted of 25 gamma energy groups of a specific activity (in Bq). These 25 source terms were derived by considering energy-dependent functions for the prompt and delayed fission gamma contributions, functions for the radiative capture gamma sources in the core and shielding materials, and functions for the activation and inelastic scattering gamma sources in the core.

Finally, the contribution to the dose due to recoil protons and stripped electrons had to be considered. Using WIMS-AECL-calculated neutron fluxes and macroscopic scattering cross sections, the recoil proton flux was calculated at the inner and outer irradiation sites as well as at incremental radial distances in the pool. Using proton flux distributions as well as specific ionization values in a volume integral calculation, stripped electron fluxes could then be estimated.

All experimental measurements and analytical calculations were performed at the reactor mid-height and at steady-state half power operation (i.e., at a thermal neutron flux of 5×10^{11} n-cm^{-2}-s^{-1}). This was done as the SLOWPOKE-2 at RMC is operated at steady-state half power during more than 99% of its operating time (Nielsen, 98/99).

CHAPTER 2 - THEORY

2.1 WIMS-AECL MODEL

As mentioned in Section 1.2, it was decided that an attempt to model the SLOWPOKE-2 core would be made using the WIMS-AECL neutron transport code. Although AECL had used WIMS to model CANDU bundles in the past, little in the way of SLOWPOKE-2 modelling had been performed (deWit, 89). Nearly all of the nuclear information required for WIMS-AECL calculations are stored in external libraries. The ENDF/B-V library was used in this set of WIMS calculations and is considered the industry standard. Data contained within such a library would include available nuclides, the energy boundaries applicable to the multi-group cross sections, the cross sections for each nuclide (particle balance, (n,2n), fission, fission yield, P_0 and P_1 scattering, and transport) and fission yield spectra. The current ENDF/B-V library has 89 neutron energy groups (24 fast groups, 23 resonance groups, and 42 thermal groups) for use in WIMS-AECL case evaluations.

WIMS-AECL assumes isotropic scattering; as a result, the following steady-state neutron transport equation can be assumed valid:

$$\phi(r,E) = \iiint \frac{e^{-T(r'\rightarrow r,E)}}{4\pi|r-r'|^2}\, dV' \left[\int \sum\nolimits_s (r',E'\rightarrow E)\Phi(r',E')dE' + \frac{1}{k}\int \chi(r',E'\rightarrow E)\nu \sum\nolimits_f (r',E')\Phi(r',E')dE' \right] \quad (1)$$

where $\phi(r,E)$ is the neutron flux at location r and energy E,

$T(r'\rightarrow r,E)$ is the neutron path length between r′ and r in units of optical thickness,

$$T(r' \to r, E) = |\underline{\delta r}|\int_0^1 \sum(r' + t\underline{\delta r}, E)\,dt$$

Σ	is the total neutron cross section at location r for neutrons of energy E,
$\underline{\delta r}$	= r-r′
$\sum_s(r, E' \to E)$	is the neutron yield cross section for incident neutron energy E' and outgoing neutron energy E for all events besides fission (includes scattering and accounts for (n,2n) neutron emission),
$\nu\sum_f(r, E)$	is the neutron fission-yield cross section,
$\chi(r, E' \to E)$	is the fission yield spectrum for incident neutron energy E′ and outgoing neutron energy E,
k	is the multiplication factor (eigenvalue),

$$k = \frac{\iint \nu \sum_f(r, E)\Phi(r, E)\,dV\,dE}{\iint \sum_a(r, E)\Phi(r, E)\,dV\,dE}, \text{ and}$$

$\sum_a(r, E)$	is the neutron balance cross section, equal to the net absorption cross section for neutrons.

Although Equation 1 describes the energy-dependent characteristics of the neutrons, a multi-group form is required for lattice-cell calculations. The multi-group neutron fluxes and cross sections are therefore represented by the integral of the neutron flux over the discrete energy range. The multi-group neutron fluxes and cross sections are defined as:

$$\Phi_i(r) = \int_{E_{low,i}}^{E_{high,i}} \Phi(r,E)\,dE \tag{2}$$

$$\Sigma_{s,i\rightarrow j}(r) = \frac{\int_{E'_{low,i}}^{E'_{high,i}} \int_{E_{low,j}}^{E_{high,j}} \Sigma(r,E'\rightarrow E)\Phi(r,E')\,dE\,dE'}{\int_{E'_{low,i}}^{E'_{high,i}} \Phi(r,E')\,dE'} \tag{3}$$

$$\nu\Sigma_{f,i}(r) = \frac{\int_{E_{low,i}}^{E_{high,i}} \nu\Sigma_f(r,E)\Phi(r,E)\,dE}{\int_{E_{low,i}}^{E_{high,i}} \Phi(r,E)\,dE} \tag{4}$$

$$\chi_i = \frac{\int_{E_{low,i}}^{E_{high,i}} \iiint\int \chi(r,E'\rightarrow E)\nu\Sigma_f(r,E')\,dE'\,dV\,dE}{\int\iiint \nu\Sigma_f(r,E')\Phi(r,E')\,dE'\,dV\,dE} \tag{5}$$

where $E_{high,i}$ is the high-energy boundary of energy group i, and

$E_{low,i}$ is the low-energy boundary of energy group i.

Now, the steady-state multi-group neutron transport equation can be written as:

$$\Phi_i(r) = \iiint \frac{e^{-T_i(r'\to r)}}{4\pi|r-r'|^2}\, dV' \left[\sum_j \sum\nolimits_{s,j\to i}(r')\Phi_j(r') + \frac{\chi_i}{k}\sum_j \nu \sum\nolimits_{f,j}(r')\Phi_j(r')\right] \tag{6}$$

All calculations within WIMS are carried out in the multi-group forms. Within each energy group, all cross sections are constant. Although up to 89 energy groups may be used in lattice-cell calculations, typically one chooses a number between 2 and 89. In the SLOWPOKE-2 core simulation, a 26 energy group approximation was chosen. The reasons for this choice will be discussed later.

WIMS-AECL calculates collision probabilities for use in the transport code algorithm. A collision probability is defined as the fraction of neutrons emitted from a uniform and isotropic source in one spatial region that undergo their next collision (be it scattering, absorption or fission) in some other region within a lattice cell. The steady-state multi-group collision-probability neutron transport equation for a lattice cell divided into a number of regions is:

$$\Phi_i(m) = \sum_l \frac{P_{i,l,m}}{V_m \sum_{m,i}} \left[\sum_j \sum\nolimits_{l,s,j\to i} \Phi_j(l) + \frac{\chi_i}{k}\sum_j \nu \sum\nolimits_{f,l,j}(l)\Phi_j(l)\right] \tag{7}$$

where $P_{i,l,m}$ is the probability for neutrons born from scattering and fission events in energy group i in region l undergoing their next collision in region m, and

V_m is the volume of region m.

The collision probabilities $P_{l,m}$ are defined as:

$$P_{l,m} = \iiint_l \frac{dr_1}{V_l} \iiint_m \frac{dr_2 \Sigma_m}{4\pi |r_1 - r_2|^2} e^{-T(r_1 \rightarrow r_2)} \tag{8}$$

WIMS-AECL calculations consider neutrons as part of three separate groups:

1. **Fast** - neutrons with energies greater than 10 keV.

Within this energy region, the important processes considered are:

- birth from fission (i.e., the number of neutrons born due to a fission event is ν/k where ν is the fission yield at the incident neutron energy and k is the lattice-cell fission eigenvalue,
- capture and fission reactions,
- (n,2n) reactions,
- elastic and inelastic scattering events,
- neutron transport between fuel pins and moderator, and
- neutron leakage between lattice regions and out of the core.

2. **Resonance** - neutrons with energies between 4 eV and 10 keV.

Within this energy region, the important processes considered are:

- capture and fission reactions with nuclides whose cross sections are dominated by neutron resonances,
- capture reactions in non-fuel nuclides,
- elastic scattering events,
- neutron transport between fuel pins and moderator, and
- neutron leakage between lattice regions and out of the core.

3. **Thermal** - neutrons with energies below 4 eV.

Within this energy region, the important processes considered are:

- capture and fission reactions,
- thermal elastic and inelastic scattering,
- neutron transport between fuel pins and moderator, and
- neutron leakage between lattice regions and out of the core.

The method used by WIMS-AECL to solve the lattice-cell problem consists of at least three separate transport calculations and at least four homogenized-cell calculations in each case. The calculations involved are as follows:

1. material macroscopic cross sections are formed, and thermal cross sections are calculated as required by interpolating within temperature-dependent cross section tables,

2. resonance-shielding calculations are performed,

3. condensation spectra to be applied to the library data to yield energy-dependent data for use in transport calculations are calculated. The material cross sections are condensed using these spectra,

4. the two-dimensional neutron flux distribution for an infinite lattice is solved,

5. the leakage properties of the lattice cell are calculated, and leakage calculations are performed to represent finite-reactor properties, and

6. burnup calculations are performed to model fuel transformation with irradiation and time.

2.1.1 Macroscopic Material Cross Sections

WIMS-AECL calculates the macroscopic cross sections for each material using the microscopic cross sections for each nuclide along with their number densities. This calculation is performed by summing the absorption, fission, fission yield and transport cross sections. The absorption cross section is treated as follows:

$$\sum_a = \sum_{n,\gamma} + \sum_x + \sum_f - \sum_{n,2n} \tag{9}$$

where $\sum_x$ includes all neutron-destroying reactions besides fission, (n,γ) and (n,2n).

The outgoing neutrons from the (n,2n) reactions are represented in the scattering yield cross section. The actual isotopic high-energy scattering cross sections in WIMS-AECL and its libraries are:

$$\sum_{s}(E \rightarrow E') = \sum_{s,elastic}(E \rightarrow E') + \sum_{s,inelastic}(E \rightarrow E') + \sum_{n,2n}(E \rightarrow E') \qquad (10)$$

Thermal neutron cross sections are interpolated according to material temperatures and thermal cross-section data for each nuclide.

The neutron-transport cross-sections are based on the following:

$$\sum_{total} = \sum_{transport}$$

and

$$\sum_{total} = \sum_{a} + \sum_{s} \qquad (11)$$

2.1.2 Resonance Calculations

WIMS-AECL's treatment of the resonance region (4 eV to 10 keV) is based on approximations of the conventional multi-group equations. That is because cross sections for nuclides such as ^{238}U, within this region, can vary by many orders of magnitude over energy ranges that are small compared to the specified energy group widths. It is not practical to use a very fine energy-group structure so that the conventional multi-group transport approximation is adequate without a resonance-shielding treatment. The purpose of the WIMS-AECL resonance treatment is to calculate group-averaged cross sections that account for the fine structure of the flux and cross section characteristics within each energy group.

Resonance treatment is most strongly affected by:

- the composition and size of the fuel. The most dominant nuclide is ^{238}U which is heavily self-shielded because of the relatively small concentrations of other nuclides and the lattice geometry,
- the fuel spacing. The geometric arrangement of the fuel pins has a large influence on the likelihood that neutrons escaping from one fuel element enter another rather than scattering in the moderator,
- the coolant material between the fuel elements. Neutron scattering in the coolant between fuel pins has a significant effect on resonance reaction rates.

2.1.3 Two-Dimensional Neutron-Transport Calculations

Three-dimensional lattice-cell calculations are not currently practical for transport calculations due to the greatly increased computation requirements. However, it is felt that two-dimensional calculations are sufficiently accurate for CANDU and in this case SLOWPOKE-2 modelling purposes. The two-dimensional neutron flux distribution is solved using collision-probability methods. The collision-probability solution is divided into five distinct events:

1. the two-dimensional cell geometry is analysed and processed to generate and store information that allows it to be treated efficiently in the upcoming tracking calculation,
2. a tracking calculation is carried out. Lines are constructed numerically through the cell, and track lengths within each traversed mesh are calculated,
3. the information in the tracking lines is used in a series of numerical integrations carried out for each main transport energy group; region-to-region, region-to-

surface, and surface-to-surface collision probabilities are derived,

4. energy-group-dependent one-dimensional collision probabilities are calculated and then coupled to the two-dimensional collision probabilities from step 3, thereby generating collision probabilities between each pair of subregions in the cell,

5. the neutron flux distribution is solved for each energy group based on the collision probabilities and the regional macroscopic cross sections.

Collision probabilities in two-dimensional cells are calculated by averaging collision probabilities over a series of tracking lines within the lattice-cell geometry. These "tracking lines" represent a series of neutron trajectories, and the calculated collision probabilities quantify the likelihood of neutron collision events, averaged over all trajectories. The tracking calculation is performed by mathematically constructing lines and determining the intersection of those lines within the elements of the lattice-cell, calculating incremental lengths and mesh numbers along each tracking line. WIMS-AECL allows the user to input the number of lines and angles to be used in the tracking calculation or will automatically select an appropriate number.

2.1.4 Diffusion Coefficient Calculations

The leakage characteristics of the lattice cells have a major influence on reactor criticality and reactor power distributions. If the user specifies, WIMS-AECL uses the Benoist-theory diffusion coefficients (Benoist, 68) to account for the leakage properties of the model. The user-inputted axial and radial bucklings are used to specify the reactor environment in which the model exists. Benoist diffusion coefficients will account for "neutron streaming" in very low-density materials.

2.1.5 Leakage Calculations

With WIMS-AECL, the approximation is normally made that the leakage into or out of each lattice cell can be described by an energy-independent critical buckling term. This critical buckling term is calculated internally by WIMS-AECL, using the axial and radial buckling values as a first guess. Via user input, WIMS-AECL can perform a single critical buckling search maintaining the radial-to-axial ratio of the initial guess.

2.1.6 WIMS-AECL Eigenvalues

The equations that yield neutron fluxes and eigenvalues are solved during each lattice-cell calculation. The main transport calculation infinite-lattice eigenvalue in the two-dimensional cell model in the condensed energy-group structure is defined as follows:

$$k_\infty = \frac{\sum_{i \in groups} \sum_{j \in regions} \nu \Sigma_{f,j,i} \Phi_{j,i} V_j}{\sum_{i \in groups} \sum_{j \in regions} \Sigma_{a,j,i} \Phi_{j,i} V_j} \tag{12}$$

The leakage calculation infinite-lattice eigenvalue for the homogenized cell in the condensed energy group structure will differ from that of the main transport k_∞ as a result of end regions specifications. The leakage calculation infinite-lattice eigenvalue is defined as follows:

$$k_\infty = \frac{\sum_{i \in groups} \nu \Sigma_{f,i} \Phi_{\infty,i}}{\sum_{i \in groups} \Sigma_{a,i} \Phi_{\infty,i}} \tag{13}$$

The effective-lattice eigenvalue is defined as follows:

$$k_e = \frac{\sum_{i \in groups} \nu \Sigma_{f,i} \Phi_{e,i}}{\sum_{i \in groups} \Sigma_{a,i} \Phi_{e,i} + D_i B_i J_i} \tag{14}$$

2.2 THERMOLUMINESCENT DOSIMETRY

When certain materials are irradiated, a certain percentage may break chemical bonds and, in some materials, energy may be trapped in meta-stable states. When subsequently heated, some of this trapped energy can be released in the form of light. The phenomenon is known as thermoluminescence (TL). Thermoluminescent materials are insulators with a valence band, a conduction band and defects caused by impurities which act as traps. Instead of promoting the quick recombination of electron-hole pairs, these trapping centres within the band-gap prevent this. When the TL materials are irradiated, electrons are promoted from the valence band to the conduction band and are subsequently trapped by impurities. If the trap energy level is sufficiently lower than the lower level of the conduction band, there is only a small probability per unit time at room temperature that the electron will escape the trap by being thermally excited back to the conduction band. As a result, exposure of the TL material to a continuous source of radiation, although not resulting in a significant yield of prompt scintillation light, leads to a progressive buildup of trapped electrons.

Following radiation exposure, the TL material is placed on a planchet which is progressively heated. At a temperature that is determined by the energy level of the trap, the trapped electrons can pick up sufficient thermal energy so that they are re-excited back to the conduction band. Assuming that this temperature is lower than that required to free the trapped holes, the liberated electron may migrate to a trapped hole, recombine and thereby produce an emission photon. Alternatively, holes may be trapped in an analogous manner. An original hole, created by the incident radiation, may migrate through the crystal until reaching a hole trap with energy somewhat higher than the top of the valence band. If this energy difference is sufficiently large, the hole will not migrate further unless additional thermal energy is added to the crystal. If the trapped holes are released at a lower temperature than the trapped electrons, they may migrate to a trapped electron and their recombination will also result in a photon emission. If the magnitude of the energy difference is approximately 3 to 4 eV, the emitted photons are in the visible range and become the basis of the thermoluminescent dosimeter (TLD) signal. Ideally, one photon is emitted per electron-hole pair formed due to radiation exposure. As a result, the total number of emitted photons can be used as an indication of the original number of electron-hole pairs created by the radiation.

TLD systems produce a signal by using a photo-multiplier tube from which the light yield can be measured as a function of sample temperature. Such a light yield versus temperature plot is known as a "glow curve" and the radiation exposure is directly related to the area under the curve. The glow curve can be characterised by one or more peaks depending on the number of different kinds of traps. The energy dependence of the response of TL detectors is determined by the energy dependence of the mass absorption coefficient μ_e/ρ. It is characterised by the maximum value of the relative efficiency compared to air:

$$\epsilon_{rel} = (\mu_a/\rho)/(\mu_a/\rho)_{air}.$$

If the TLD is heated to a high enough temperature, all the traps can be depleted and the exposure record of the material is erased. In the case of CaF_2:Mn, a pre-irradiation annealing at 400 °C for 1 hour is recommended by the manufacturer. As such, the TLD may be reused many times.

There are a number of TLD materials in use in radiation dosimetry today. Different TL materials have varying properties and applications. Many consist of crystals to which a small concentration of impurity has been added in the form of an activator (for the creation of traps in the forbidden band). In this case, the TLD chosen was calcium-fluoride with manganese activator (CaF_2:Mn).

One undesirable property of calcium-based TLD materials is an over-response to low-energy photons. TLD materials with higher atomic numbers (as compared to the standard LiF dosimeter) over-respond to low-energy X or gamma rays due to enhanced photoelectric interaction probabilities. The over-response of CaF_2:Mn TL material is approximately a factor of 12 at 40 keV. This phenomenon can be seen in Figure 7. This over-response can be corrected by use of an algorithm or an energy-flattening filter. In this work, tin shielding was used as a filter for the TLDs which will be discussed in Section 3.1.

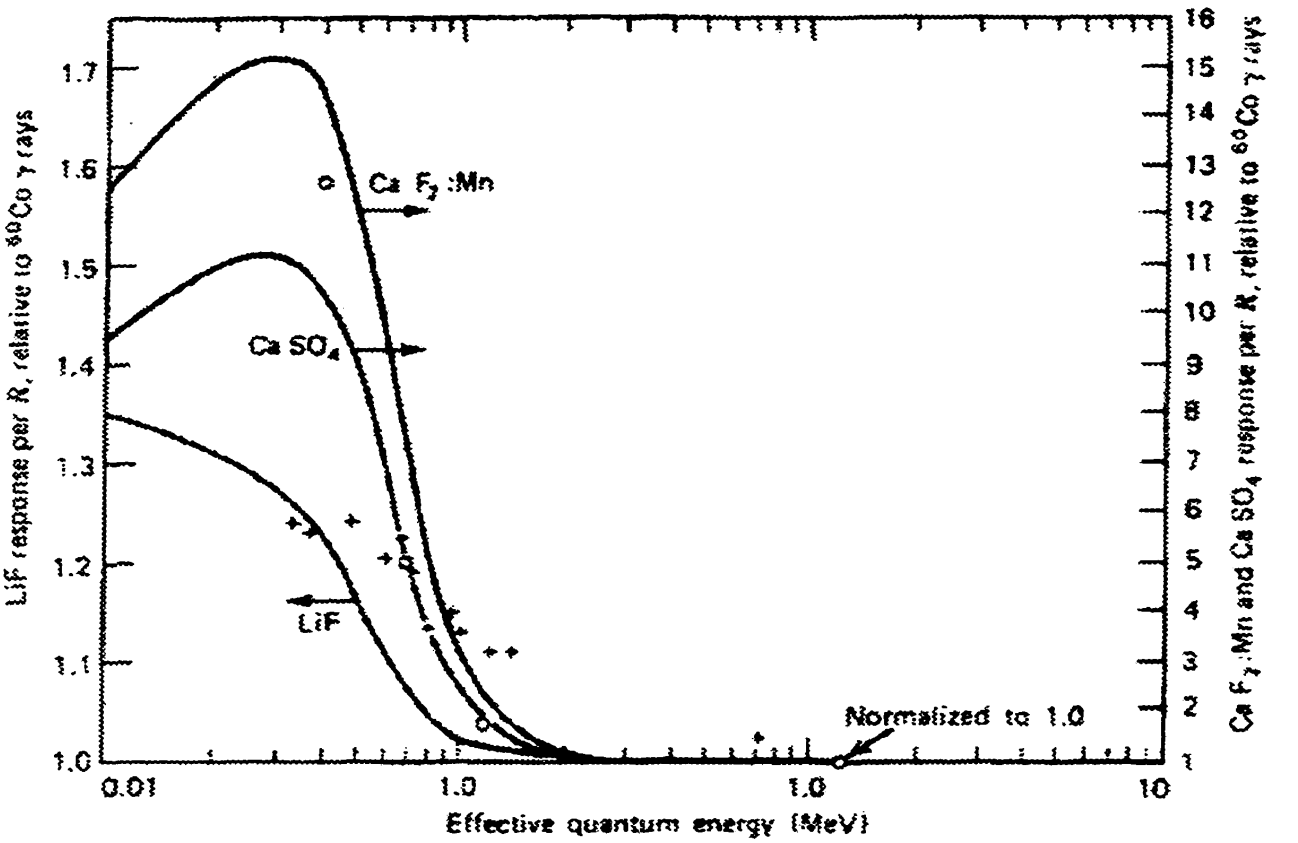


Figure 7 (Knoll, 89)

Photon Energy Dependence of Various TL Materials

Circles are experimental measurements for CaF_2:Mn, crosses are measurements for LiF. The data are represented as the TL response per unit of exposure normalised at 1.25 MeV.

CaF_2:Mn dosimeters also possess a degree of supra-linearity of dose response versus total accumulated dose. CaF_2:Mn dosimeters irradiated with ^{60}Co gamma rays were found to have a supra-linear response above approximately 1000 R accumulated dose. This phenomenon can be seen in Figure 8. As well, the degree of supra-linearity was found to increase with decreasing dosimeter thickness. This observation is explained by sample discolouration at high dose levels, resulting in increased self absorption of TL light within the dosimeter itself. This phenomenon was noted during gamma surveying in and around the SLOWPOKE-2 core.

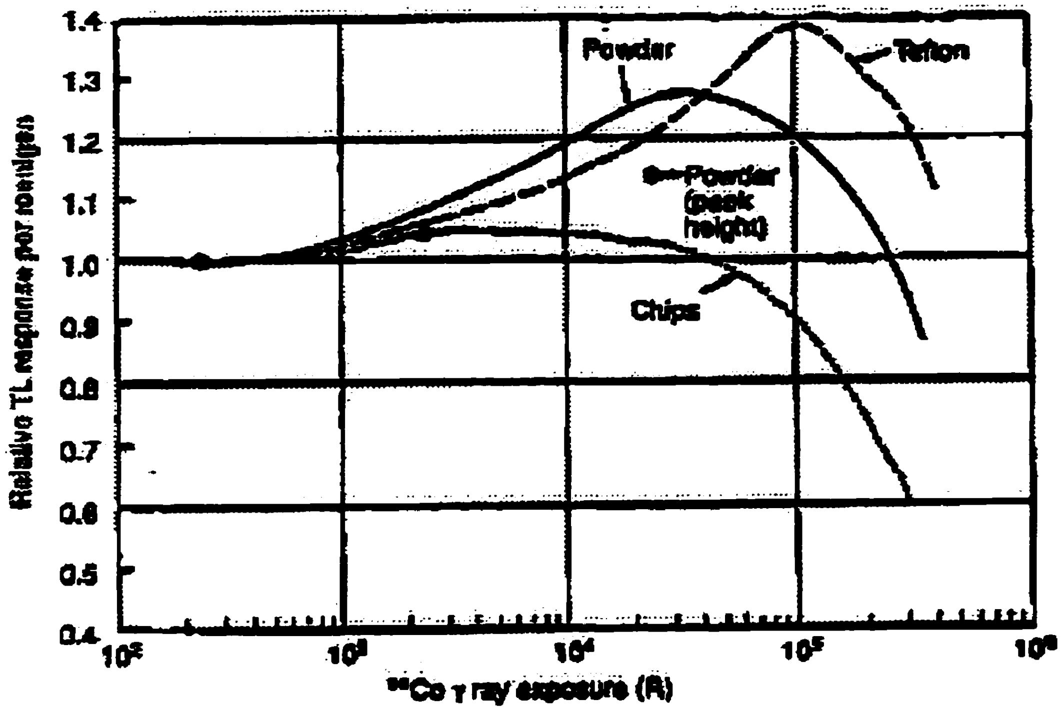


Figure 8 (Harshaw TLD, 98)

Dose-Response Curves for Different CaF_2:Mn Configurations

Dosimeters irradiated with ^{60}Co gamma rays.

It is important to note that other forms of directly ionizing radiation (mainly electrons and protons) as well as indirectly ionizing radiation (neutrons) can produce a TL response. In the case of this experimental work, it was required that the gamma dose component be separated from the other forms of radiation present around the core of the SLOWPOKE-2. Due to the very short range of electrons and protons in high density, high atomic number materials, shrouding made of higher atomic number materials such as tin of sufficient thickness would absorb either of these particles incident on the TLD. Due to the very short range of protons in most materials, it is unlikely that even the highest energy proton would pass through the polyethylene vial. In the case of incident neutrons, a lithium-bromide (LiBr) encapsulation chamber was used to house one-half of all the TLDs irradiated. The low atomic number Li has a high neutron scattering cross-section σ_{sc} which should reduce any neutron contribution in the measured TLD doses to negligible levels. The assembly of the sample vials will be discussed further in Chapter 3.

2.3 MICROSHIELD VERSION 5

MICROSHIELD 5 (MS 5) is a software package developed by Grove Engineering (Grove, 98) used to analyse shielding and estimate exposure due to gamma radiation. In the past, its primary purpose has been in the designing of shielding around sources, in the assessment of radiation exposure to people and materials due to these source/shielding configurations, and as a teaching tool. In this case, MS 5 was used to model the SLOWPOKE-2 reactor container for the purpose of calculating gamma fluence rates and absorbed dose rates at specific irradiation positions.

MS 5 allows a number of possible geometries to be created as part of the case file. In the case of the SLOWPOKE-2 reactor, a cylindrical source volume with cylindrical side shields and top cladding geometry was chosen as most appropriate. The source may be entered as one or more nuclide activities or as photon energies. When entered by nuclide activity, and after optional decaying, energy groups are created. Otherwise, source photon activities and energies are entered directly. In the case of this work, the source terms were specified using "user defined" photon energies and activities. The development of these gamma source terms representing the SLOWPOKE-2 reactor at steady-state half power operation will be discussed later in Section 2.5. A total of 25 energy groups are permitted with MS 5. The source term (or reactor core) is treated as a homogeneous mixture of uranium, oxygen, zirconium and water defined by their specific densities.

As many as 10 standard shields are possible for each geometry. They are specified in the case file by their thickness in the radial direction away from the source (in the case of the cylindrical source-cylindrical shields configuration). MS 5 treats the cylindrical source volume (SLOWPOKE-2 core) as the first shield in its calculations thus accounting for the absorption of gamma radiation within the core materials. A top cladding can be included in the case file, however, unlike the SLOWPOKE-2 reactor assembly which contains upper and lower beryllium plates, MS 5 does not have the option of entering a bottom clad material. Shield and end clad compositions can be entered either as the appropriate elements with the specific densities or as "custom" materials whose compositions have previously been defined by the user.

MS 5 accounts for contributions due to buildup between the centre of the source volume and the dose point. In MS 5, the buildup factor is calculated from interpolation from tables within ANS 6.4.3. The mean free paths for all materials between the centre of the source volume and the dose point are used, with the buildup characteristics of one of the shields selected by the user. This shield is known as the "Buildup Reference Shield", the selection of which can have a significant impact on the net gamma fluence rate at the dose point. The recommended approach is for the user to select either the last shield before the dose point or the most dominant shield (the one with the most mean-free-paths, which is a measure of the attenuation within that shield). During the process of integration, the buildup factor is calculated by MS 5 based on the materials between the source kernel and dose point, with the line of sight distance between the source kernel and dose point calculated using trigonometric relationships. The attenuation mean free paths are extrapolated from tables and used in the buildup factor calculations.

For the cylindrical source volume-cylindrical shield geometry, MS 5 uses Gauss quadrature for point-kernel numerical integration. For each kernel, MS 5 would calculate the gamma flux at the dose point using the following generalized function:

$$\phi(R) = \frac{S_o B(\mu_o R)}{4\pi R^2} e^{-\mu_o R} \tag{15}$$

where $B(\mu_o R)$ is the buildup factor due to all shields between the source kernel and the dose point.

The user specifies the fineness of the integration mesh in three dimensions. In the case of the SLOWPOKE-2 model, the fineness of the mesh was held constant in the radial, circumferential and axial directions. Prior to compiling the experimental data, the SLOWPOKE-2 model was tested using increasingly finer mesh increments in order to determine the optimum value of integration mesh size. As with any numerical approximation technique, there is a trade-off between coarser mesh increments and faster calculation time yielding more approximate results, and finer mesh increments requiring longer calculation time but providing more accurate results. In the case of the SLOWPOKE-2 model, it was determined that a total of 39 increments in each of the three dimensions produced minimal change in the gamma fluence rate at the dose point. The compilation time of between 15 and 18 minutes per case was deemed acceptable.

MS 5 provides the user with the photon fluence rate and, using the incident gamma energy from the input source terms, the energy fluence rate. The photon fluence rate is then converted to units of exposure rate (mR/h), rate of energy absorption in air (mGy/hr and mrad/hr), and effective dose equivalent rates (mSv/hr) using conversion tables from ICRP 51. The overall effect of the buildup factor on the photon fluence rate at the dose point can be assessed as MS 5 provides results with and without buildup. It is important to note that MS 5 does not take into account gamma backscatter from materials beyond the dose point in its calculations.

2.4 NEUTRON DOSE

The dose rate due to neutrons can be categorized into the following:

1. Prompt Neutrons,
2. Delayed Neutrons, and
3. Photoneutrons.

When an atom of ^{235}U fissions, an average of 2.5 neutrons is given off. The majority of the neutrons released (greater than 99 percent) are emitted within 10^{-14}s of fission. These are known as the *prompt fission neutrons*. It is the prompt fission neutrons that therefore contribute the most to reactor power and reactivity while the reactor is operating. The *delayed fission neutrons* are emitted over a period of several minutes after the moment of fission. These are expelled from the fission fragments, their intensity falling off with time. Most of the neutron-rich fission products undergo beta decay, however, in some cases, the daughter is produced in an excited state with sufficient energy so that neutron emission can occur. It is possible to gather the delayed neutrons into six groups (Lamarsh, 83), each characterized by the average half-life of the precursor, as the neutron emitters are called. The half-lives vary to some extent with the fissile nuclide, however, they are fairly constant for most analytical purposes. The longest lived of the precursors has a half-life of approximately 56 s. In reactor kinetics, the delayed neutrons play an important role in making the overall time constant of the reactor significantly longer than if all fission neutrons were prompt. During transients, the rate of change of the neutron population with time is much slower than if all the neutrons were prompt, making reactor control much easier. When the reactor is shut down, the delayed fission neutrons become important in reactivity, or in the present case, dose rate measurements. Due to the relatively short half-lives of the precursors, the effect due to delayed fission neutrons is only felt in the few minutes following shut-down. All measurements and calculations were performed with the reactor operating at steady state at half power, a situation in which the contribution of the delayed neutrons is negligible. WIMS-AECL accounts for the delayed neutron contribution during reactor operation within its transport code.

Photoneutrons are produced from (γ,n) reactions and in general, provide an almost negligible contribution to the overall neutron population within the reactor assembly (Rockwell, 56). The (γ,n) reaction will occur only if the energy of the incident gamma ray is greater than the binding energy of the neutron in the target nucleus. The relatively low binding energies of beryllium and deuterium (1.6 MeV and 2.2 MeV, respectively) result in these isotopes being strong photoneutron targets. Due to the presence of large volumes of beryllium in the SLOWPOKE-2 assembly, photoneutron production must be considered. The reaction of interest is:

$$^{9}Be + \gamma \rightarrow 2\,^{4}He + {}^{1}n\,.$$

In both ^{2}H and ^{9}Be, the photoneutron cross sections are however quite small in the gamma energy ranges of interest, approximately 0.18 mb at E_{γ}=2.5 MeV for ^{9}Be (NNDC,BNL). When MS 5 incident plus backscattering gamma fluxes at the inner irradiation site are used as a source term, a rough calculation at the inner irradiation site yields a total (γ,n) neutron flux of 8.55 x 10^{5} n-cm^{-2}-s^{-1} at half power reactor operation. The photoneutron production rate per unit volume in this case was calculated using the following:

$$\overline{S}_{pn} = N_{Be}\overline{\sigma}_{(\gamma,n)}\overline{\phi}_{\gamma}$$

where $\overline{S}_{pn}$ is the photoneutron production rate per unit volume (= 4.14 x 10^{6} n-cm^{-3}-s^{-1}), N_{Be} is the atomic density of Be atoms in the beryllium reflector (= 1.23 x 10^{23} atoms-cm^{-3}), $\overline{\sigma}_{(\gamma,n)}$ is the average microscopic cross-section for the photoneutron reaction in Be (= 0.18 mb), at $\overline{E}_{\gamma}$ = 2.5 MeV, and $\overline{\phi}_{\gamma}$ is the MS 5-calculated average incident gamma flux plus backscattering, of energy greater than the photoneutron threshold energy in Be of 1.6 MeV, across the beryllium reflector (= 1.87 x 10^{11} γ-cm^{-2}-s^{-1}).

Using this photoneutron production rate per unit volume, an approximation can be made of the resultant photoneutron flux at a position in the centre of the beryllium annulus. If isotropic scattering and negligible neutron absorption is assumed in an spherical volume of 1 cm^{3}, such that all photoneutrons born within that volume eventually pass through the surface area of the sphere, a resultant total photoneutron flux of 8.55 x 10^{5} n-cm^{-2}-s^{-1} is calculated. When this value is compared with the WIMS-AECL-calculated total neutron flux at the position of the inner irradiation site of 1.13 x 10^{12} n-cm^{-2}-s^{-1}, it can be concluded that the photoneutron effect is indeed negligible for the purposes of this work.

Research performed at AECL in 1969 (Walker and Okazaki, 69) found that the total photoneutron yield in a CANDU reactor is 1.7 x 10^{-3} neutrons/fission due to prompt, delayed, and capture gamma contributions. When compared to the approximately 2.5 neutrons liberated due to ^{235}U fissioning, the effect due to photoneutron production is indeed small. Although the materials present in a CANDU channel differ significantly from the SLOWPOKE-2 core assembly, these AECL results appear to support the previous rough approximation of photoneutron contributions in the SLOWPOKE-2 reactor. As a result, the photoneutron intensity is considered negligible when compared with the prompt and delayed neutron effect with the reactor operating.

Dose rates due to neutrons are a product of the energy-dependent neutron flux spectrum $\phi(E,r)$, the macroscopic absorption and scattering cross-sections ($\Sigma_a(E)$, $\Sigma_s(E)$), and the average energy imparted to the target material $(\overline{E_a}(E), \overline{E_s}(E))$. The neutron flux spectrum and macroscopic cross-sections are provided by WIMS-AECL for each of the 26 neutron energy groups within each of the 98 annular regions. The average energy imparted due to absorption and scattering reactions was calculated using the equation for the average logarithmic energy decrement:

$$\xi = \ln \frac{\overline{E_1}}{\overline{E_2}}$$

where $\overline{E_1}$ is the average energy of the neutron before the collision and $\overline{E_2}$ is the average energy of the neutron after the collision.

Using the average logarithmic energy decrement for the target nuclides at the dose points (inner, outer irradiation sites as well as within the pool), the average energy imparted to the target material could then be calculated.

Theoretically, the total energy deposited in the target is therefore:

$$\dot{E}_{tot} = \int_V \int_E \left[\overline{E_a}(E)\sum_a(E) + \overline{E_s}(E)\sum_s(e)\right]\phi(E,r)\,dE\,dV \tag{16}$$

Practically, this calculation is performed over the 26 neutron energy groups and over a target volume of 1 cm^3 and then the contributions are summed to produce the total energy deposited by neutron interactions at each dose point.

The calculation of the total neutron dose rate is a trivial exercise once the total energy deposited has been calculated. For each of the dose points, the neutron dose rate is simply:

$$\dot{D}_{tot} = \left[\dot{E}_{tot}(Js^{-1}) / M\,(kg)\right](Gy / Jkg^{-1}) \tag{17}$$

where M is the mass of the unit volume target material.

2.5 GAMMA DOSE

The gamma dose received at a dose point can be categorized into two broad groups:

a. fission gamma rays, of which there are two groups (prompt and delayed), and

b. secondary gamma sources, i.e., radiative capture, activation and inelastic scattering in the core as well as in the shielding material.

2.5.1 FISSION GAMMA RAYS

2.5.1.1 Prompt Fission Gamma Rays

Prompt fission gamma rays are normally defined as those emitted within 0.1 μs of the fission event. They are produced in coincidence with the prompt neutrons as well as from very short-lived fission products. The energy range of prompt fission gamma rays is from 300 keV to 10 MeV.

Correlations exist to represent the prompt fission spectrum (Blizard and Abbott, 62). From 0.3 to 1.0 MeV, the following function was used:

$$N(E) \approx 26.8 \exp(-2.3E) \, (\text{MeV})^{-1}. \tag{18}$$

From 1.0-7.0 MeV, the following correlation holds:

$$N(E) \approx 8.0 \exp(-1.1E) \, (\text{MeV})^{-1} \tag{19}$$

where N(E) is the number of prompt gamma photons per MeV and per fission.

Beyond 7 MeV, the prompt fission gamma spectrum falls off quite quickly, and therefore the relative contribution is quite small and is considered negligible in this case.

Using the correlations above, N(E) can be computed at discrete energy increments. To compute the prompt fission gamma intensity for the reactor at power, the following function (Bonin, 98) was used:

$$\overline{S}_{v\gamma p} \cong 3.1e^{10} N(E)\overline{P} \tag{20}$$

where $\overline{S}_{v\gamma p}$ is the prompt gamma photon intensity (in γ-cm^{-3}-s^{-1}), and

$\overline{P}$ is the average reactor power density in W-cm^{-3} (at half power operation).

2.5.1.2 Delayed Fission Gamma Rays

Fission product gamma rays can be grouped in the following two categories: short-lived or saturating and long-lived fission products. With the reactor operating, the short-lived fission product gamma rays have a much greater effect on the overall gamma contribution. Due to the half-lives of the short-lived fission products, the short-lived gamma activity saturates after approximately two hours of operation. The long-lived fission product gamma rays become predominant once the reactor is shut down, and following a relatively lengthy residence time of the fuel. In the case of the SLOWPOKE-2 core, the relatively low thermal neutron flux and resulting low total accumulated fluence (2.6 x 10^{19} n-cm^{-2} as of 10 Aug 98) in comparison to a CANDU core, means that the fuel of the SLOWPOKE-2 is relatively new. When the average fluence of the SLOWPOKE-2 core (~2.6 x 10^{-2} n/kb) is compared to that of a CANDU fuel bundle over its average residence time of 1 year (~1.66 n/kb), it can be seen that, in relative terms, the age of the SLOWPOKE-2 core is approximately 5.7 days. Since the SLOWPOKE-2 reactor is operated intermittently (on average no more than 7 hours per day, 5 days per week), a fraction of the long-lived fission product activity would decay away during these shut down periods, resulting in a even lower relative gamma contribution. In general, the production of long-lived fission product gamma rays is directly related to the term ϕt (the average thermal neutron flux multiplied by the total fuel irradiation time). Due to the fact that both of these terms are relatively small, it is felt that the contribution of long-lived fission product gamma rays can be considered negligible when the reactor is operating.

The delayed fission product gamma rays are represented by the following equation (Blizard and Abbott, 62), which is independent of fuel residence time:

$$N(E) \approx 6.0 \exp(-1.1E)(\text{MeV})^{-1}. \tag{21}$$

Experimentally, the fission product gamma energies have been found to reside in the 0.1-2.8 MeV range (Blizard and Abbott, 62). From this information and the above equation, a series of delayed fission product gamma terms can be computed at discrete energy intervals. Once again, the intensity of delayed fission product gamma rays at discrete energies can be calculated from equation (20) knowing that the reactor is operating at half power.

2.5.2 SECONDARY GAMMA SOURCES

2.5.2.1 Radiative Capture in the Core

Radiative capture reactions (n,γ) refer to the absorption of a neutron within an isotope, thereby producing a compound nucleus in an excited state. This nucleus will decay back to ground state with the emission of one or more gamma photons. The radiative capture gamma source distribution function is as follows (Bonin, 98):

$$S_{\gamma c}(E',r) = \int_E \left[\sum_i \Sigma_i(E)\phi(E,r) f_i(E') \right] dE$$

where $S_{\gamma c}$ is the capture gamma photon intensity (in γ-cm^{-3}-s^{-1}),

E is the neutron energy (in MeV),

E$'$ is the energy of the radiative capture gamma ray (in MeV),

$\Sigma_i(E)$ is the (n,γ) macroscopic cross section for isotope i (in cm^{-1}),

$\phi(E,r)$ is the neutron flux distribution, and

$f_i(E)$ is the photon emission spectrum due to (n,γ) for isotope i.

This reaction is most significant for neutrons in the thermal energy range as $\Sigma_i(E)$ becomes quite small above approximately 10 keV. Radiative capture sources are therefore calculated by considering only the absorption of thermal neutrons. As a result, the radiative capture gamma source distribution function becomes:

$$S_{\gamma ci}(E',r) = N_i \sigma_i^{th} \phi^{th}(r) f_i(E') \qquad (22)$$

where N_i is the number of nuclei of isotope i per cm^3 of the core, and

σ_i^{th} is the thermal neutron microscopic cross section for (n,γ) reactions in isotope i.

By considering the isotopic composition of the SLOWPOKE-2 core, a series of radiative capture gamma source terms can be calculated. Average WIMS-AECL thermal flux values across the fuel pins as well as across the water annular regions within the core assembly were used for these calculations.

2.5.2.2 Activation in the Core

Activation is similar to radiative capture except that the compound nucleus formed is left in an excited state for a period with a measurable half-life. As the activation product decays, it emits beta particles and gamma rays. In order for this (n,γ) reaction to occur, the incident neutron energy must be greater than or equal to a threshold energy. The value of this threshold energy depends on the target nuclide. One of the most significant activation reactions in the SLOWPOKE-2 core is $^{16}O(n,p)^{16}N$ which occurs only for fast neutrons, the threshold energy being approximately 11.0 MeV. If the disappearance of the target isotope is neglected, then the source term for gamma rays due to activation products can be written as (Bonin, 98):

$$S_{\gamma ai}(E') = n(E')\sum_{i'}\phi(r)(1-e^{-\lambda_i t}) \tag{23}$$

where $S_{\gamma ai}$ is the activation photon intensity (in γ-cm^{-3}-s^{-1}),

n(E') is the number of photons emitted of energy E' by the activated product per disintegration,

$\Sigma_{i'}$ is the macroscopic cross-section for (n,γ) capture reactions due to thermal neutrons in stable isotope i', and

λ_i is the decay constant for the radioisotope i formed following neutron capture.

For all calculations, the decay time was taken to be one hour. This was considered to be a good approximation of the operation of the SLOWPOKE-2 reactor in all experimental circumstances. From Equation 23, a series of activation product gamma source terms can be calculated for each radioisotope formed. In many cases, these radioisotopes emit gamma rays of more than one energy.

2.5.2.3 Inelastic Scattering in the Core

Inelastic scattering refers to the event whereby a neutron collides with a target nuclide losing some of its energy in the process. The energy lost by the neutron goes to excite the target nucleus to some level above ground state. Within a very short time ($\sim 10^{-14}$s), the nucleus loses its energy by the emission of one or more gamma rays. In order for inelastic scattering to occur, the incident neutron must possess an energy greater than or equal to the first excited level of the target nuclide. For elements of moderate and high mass number, the minimum excitation energy is in the range of 0.01-1.0 MeV while, with decreasing mass number of the nucleus, there is a general tendency for the minimum excitation energy to increase. For example, ^{16}O has a minimum excitation energy of approximately 6 MeV and in hydrogen, inelastic scattering does not occur at all.

The source term for gamma rays emitted due to inelastic scattering is as follows (Bonin, 98):

$$S_{\gamma is}(E',r) = \int_E \left[\sum_i \Sigma_{is}(E)\phi(E,r) f_i(E') \right] dE \qquad (24)$$

where $S_{\gamma is}$ is the inelastic scattering photon intensity (in γ-cm^{-3}-s^{-1}),

$\Sigma_{is}(E)$ is the macroscopic inelastic scattering cross-section for incident neutrons of energy E in isotope i,

$f_i(E')$ is the energy spectrum of gamma rays emitted through inelastic scattering in isotope i, and

$\phi(E,r)$ is the neutron flux distribution of energies greater than or equal to the minimum excitation energy of isotope i.

Using Equation 24, a series of gamma fluence rates as a function of their emitted energies can be calculated due to inelastic scattering within the core of the SLOWPOKE-2. These calculations are carried out for all isotopes of interest within the core assembly.

2.5.2.4 Radiative Capture in the Shielding

Once again, this reaction is only significant for neutrons of lower energies. Similar to radiative capture gamma contribution calculations in the core, the (n,γ) reaction was considered for all isotopes present within the shielding material. To proceed, a volume integral approximation would be required whereby the beryllium and water annular regions were divided into equi-volume segments. Using Equation 22 with the average thermal neutron flux across a specific volume segment (provided by WIMS-AECL), and using the spectrum of emitted gamma rays due to radiative capture ($f_i(E')$) for each isotope of interest, a series of gamma source terms could be calculated for each volume segment. It was decided to divide both the beryllium and water shields into 16 segments in the azimuthal direction, and 22 segments in the axial direction (each segment being 1 cm thick). Any effects due to the presence of the irradiation sites, neutron flux detector, heavy water annular segment, thermocouples and reactor container wall were neglected and as a result, were not dealt with in this section. As WIMS-AECL performs neutron transport calculations in two dimensions only, it was necessary to consider the variance of this flux distribution in the axial direction. Research performed at École Polytechnique (El Hajjaji et al., 98) provided detailed thermal neutron flux distributions within the core in the axial direction. In all cases, rotational symmetry was assumed and the volume integral approximation was calculated for volume segments 1-8 (i.e., 0-180° in the azimuthal direction). It was assumed that equal gamma contributions due to radiative capture would result from volume segments 9-16. For uniformity and simplicity, the dose point at which all radiative capture gamma contributions from the shielding were measured was one of the inner irradiation sites.

The calculation of gamma flux terms is somewhat complex due to the fact that gamma rays produced far from the dose point will be attenuated as they pass through a certain thickness of beryllium, water and core. A representation of this treatment can be seen in Figures 9 and 10. At position $\vec{r_1}$ (see Figure 9), the resulting gamma flux at the dose point due to radiative capture in isotope i and gamma energy E' is as follows:

$$\phi_{1}{}^{\gamma}{}_{,i,E'} = \frac{S_i(E',r_1)}{4\pi d_1{}^2} e^{-\mu_1 d_1} B(\mu_1 d_1) \tag{25}$$

where d_1 is the distance between the dose point $\vec{p}$ and $\vec{r}_1$,

μ_1 is the linear attenuation coefficient for a gamma ray of energy E' in shield 1 (Be), and

$B(\mu_1 d_1)$ is the buildup factor due to shield 1.

Buildup factors $B(\mu_1 d_1)$ were calculated using a Taylor series expansion:

$$B(\mu_1 d_1) = Ae^{-\alpha_1 \mu_1 d_1} + (1-A)e^{-\alpha_2 \mu_1 d_1} \tag{26}$$

The Taylor coefficients A, α_1 and α_2 were provided by MS 5 for particular elements (and certain compounds, i.e., H_2O) at specific gamma energies. As a result, trace impurities in the beryllium reflector were not taken into account when buildup factors were calculated.

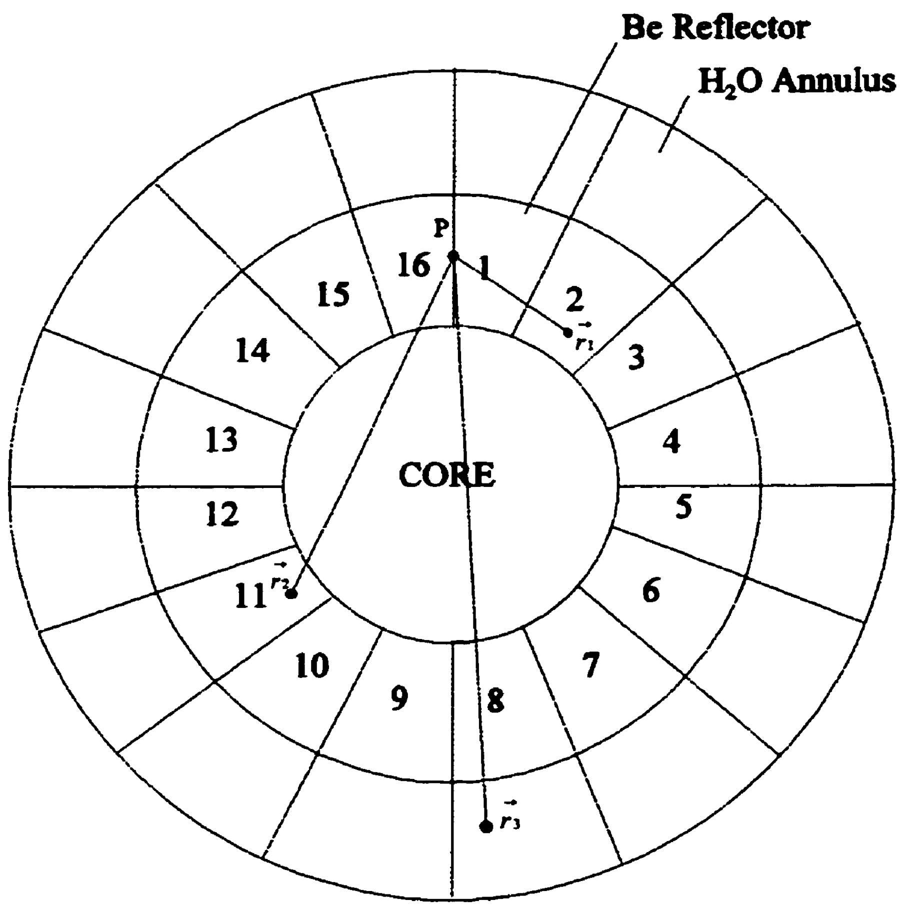


Figure 9: Reactor Container Cross-Section Radiative Capture Schematic

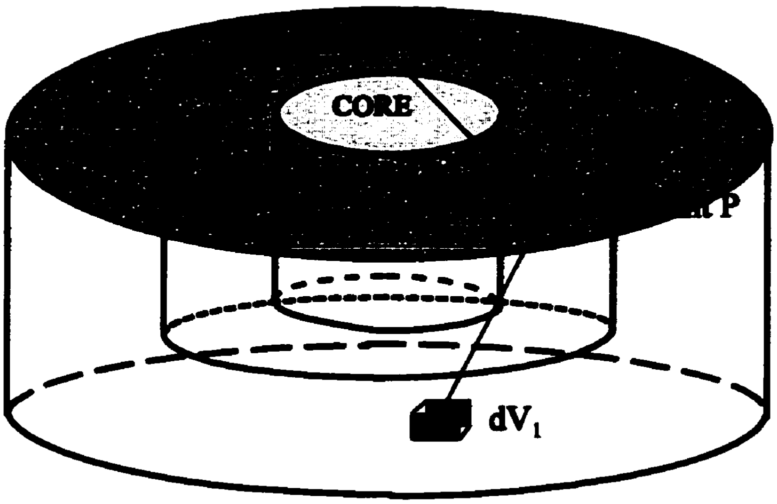


Figure 10: Three-Dimensional Reactor Container Radiative Capture Schematic

At position $\vec{r}_2$ (see Figure 9), the resulting gamma flux at the dose point due to radiative capture in isotope i and gamma energy E' is as follows:

$$\phi^{\gamma}_{2,i,E'} = \frac{S_i(E',r_2)}{4\pi d_2^{\,2}} e^{-\mu_1 d_{21}} e^{-\mu_c d_{2c}} B(\mu_1 d_{21}) B(\mu_c d_{2c}) \tag{27}$$

where d_2 is the distance between the dose point $\vec{P}$ and $\vec{r}_2$, where $d_2=d_{21}+d_{2c}$

d_{21} is the part of d_2 within shield 1 (Be),

d_{2c} is the part of d_2 within the core,

μ_c is the linear attenuation coefficient for a gamma ray of energy E' in the core material (provided by MS 5),

$B(\mu_1 d_{21})$ is the buildup factor due to shield 1, and

$B(\mu_c d_{2c})$ is the buildup factor due to the core.

At position $\vec{r}_3$ (see Figure 9), the gamma flux at the dose point due to radiative capture in isotope i and gamma energy E' is as follows:

$$\phi^{\gamma}_{3,i,E'} = \frac{S_i(E',r_3)}{4\pi d_3^2} e^{-\mu_1 d_{31}} e^{-\mu_2 d_{32}} e^{-\mu_c d_{3c}} B(\mu_1 d_{31}) B(\mu_2 d_{32}) B(\mu_c d_{3c}) \tag{28}$$

where d_3 is the distance between dose point $\vec{P}$ and $\vec{r}_3$, where $d_3=d_{31}+d_{32}+d_{3c}$

d_{31} is the part of d_3 within shield 1 (Be),

d_{32} is the part of d_3 within shield 2 (H_2O),

d_{3c} is the part of d_3 within the core,

μ_2 is the linear attenuation coefficient for a gamma ray of energy E' in H_2O,

$B(\mu_1 d_{31})$ is the buildup factor due to shield 1,

$B(\mu_2 d_{32})$ is the buildup factor due to shield 2, and

$B(\mu_c d_{3c})$ is the buildup factor due to the core.

Once again, rotational and axial symmetry was assumed, and the radiative capture in shielding calculations were carried out for segments 1-8 in the azimuthal direction and 1-11 in the axial direction (with each axial increment representing 1 cm of thickness in the z-direction). As a result, the following summation equation was used for each gamma energy of interest in each isotope i (for both the Be and H_2O annular regions):

$$\phi^{\gamma}{}_{i,E'} = 4\sum_{j=1}^{88}\phi^{\gamma}{}_{j,i,E'} \tag{29}$$

Gamma rays of similar energies could then be summed to produce total flux terms of energy E' at dose point $\vec{P}$, $\phi^{\gamma}{}_{E'}$. All distances were calculated using geometric relationships.

2.5.2.5 Inelastic Scattering in the Shielding

The gamma contribution due to inelastic scattering within the shielding was considered negligible. Inelastic scattering is a more important effect for heavy nuclei as their allowable energy levels are more numerous, have greater widths, and the distance from the ground state to the first energy level is smaller. In general, light nuclei (such as Be, H_2O) have higher thresholds and lower cross sections for inelastic scattering. In this case, beryllium was found to have a relatively high inelastic scattering threshold of approximately 2.5 MeV with an average σ_{inl} of approximately 14 mb (BNL-325, Mughabghab and Garber, 73). After performing a set of rough calculations, it was decided that any gamma rays contributed due to inelastic scattering in beryllium or water would indeed be negligible.

2.5.2.6 Activation in the Shielding

It is important to note that the SLOWPOKE-2 reactor was designed to minimize activation gamma effects in the shielding material. Beryllium does not produce any significant activation products, and the relatively high activation energy of ^{16}O (~11 MeV) results in its contribution being negligible in this case. There may be a certain gamma activation contribution due to the nuclides of the reactor container wall and the trace impurities in the beryllium reflector. For example, the $^{56}Fe\,(n,p)^{56}Mn$ reaction in the beryllium reflector (~0.00241 g-cm^{-3}) has an activation threshold energy of approximately 2.9 MeV and an average cross section of 0.86 mb over the fission spectrum. Another activation reaction of interest, $^{27}Al(n,\alpha)^{24}Na$, has an approximate activation energy of 6.5 MeV with an average cross section over the fission spectrum of 0.35 mb. Neither of these contributions will be significant due to the relatively high activation energies and low neutron cross sections. In the case of the ^{27}Al reaction, the highly thermalized neutron flux at the reactor container wall is such that only a small percentage of the incident neutrons will provide adequate activation energy. There is another activation reaction of interest in the reactor container wall, $^{27}Al(n,\gamma)^{28}Al$. The de-excitation of ^{28}Al results in the emission of a 3.0 MeV beta particle and a 1.78 MeV photon. The half-life of ^{28}Al is 2.3 minutes which means that the activity of this isotope will saturate very quickly. The microscopic cross-section across the thermal energy range is approximately 0.23 b, significantly higher than those of the previously mentioned activation reactions in the shielding. Using Equation 23 and the average thermal neutron flux at the container wall (using WIMS-AECL and accounting for the flux distribution in the axial direction), the resultant gamma production rate due to the $^{27}Al(n,\gamma)^{28}Al$ activation reaction was 2.86×10^{12} γ-s^{-1}. This represents approximately 0.07% of the total gamma production rate in the MS 5 source term (see Annex B).

In a typical core, a relatively small percentage of the absorbing nuclei are transmuted into gamma-emitting unstable nuclei and therefore the induced activities are relatively minor compared to the fission or capture effects. Although important to consider, due to the relatively minor contribution of activation gamma sources in the shielding materials, these effects are considered negligible here.

2.6 GAMMA DOSE CONTRIBUTION DUE TO BACKSCATTER

MS 5-generated gamma flux terms do not take into account effects due to backscattering in materials beyond the dose point. A volume integral calculation is therefore needed to approximate this backscatter effect numerically at the dose points in the pool as well as at the irradiation sites. It is assumed that gamma contributions from beyond 7 half-value layers are negligible (Bonin, 96) so this range will provide the maximum radius for our volume integral calculations.

In order to solve this problem, it is first assumed that a volume element centred at point P, located at a distance r from the dose point S, is selected. A representation of this model can be seen in Figure 11. Compton collisions within the volume element dV are assumed to scatter photons isotropically. A certain percentage of those gammas will be aimed toward dose point S along $\overline{PS}$. The number of Compton collisions occurring within the volume element dV per unit time is given by $\phi\mu_{sc}dV$ where μ_{sc} is the linear Compton scattering coefficient for a given material at a given incident gamma energy, and ϕ is the incident gamma rate at P (in photons-s^{-1}). For simplicity, only the beryllium reflector, water annulus and pool were considered when calculating backscattering (i.e., irradiation sites and the reactor container wall were neglected). This was considered a reasonable approximation given the relatively small volumes of these components. The linear Compton scattering coefficient is the difference between the specific linear attenuation

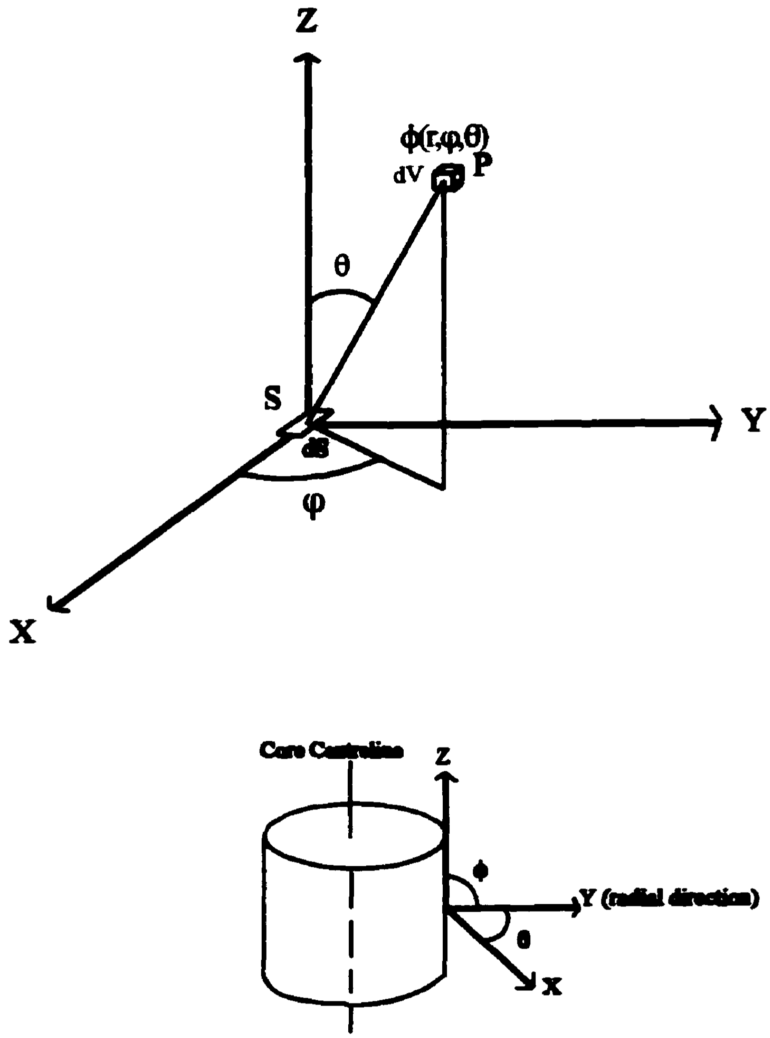


Figure 11: Coordinate System Used for Backscatter Calculations

and linear absorption coefficients.

To determine the number of photons reaching the unit surface area dS at dose point S from dV, the rate of production of Compton scattered photons within dV ($\phi\mu_{sc}dV$) is multiplied by the probability that a scattered photon is correctly aimed at the surface dS. This is represented by the solid angle (dS cosθ)/($4\pi r^2$). The probability that a scattered photon correctly aimed at surface dS actually reaches it is given by the term $e^{-\mu_{atm} r}$ where μ_{atm} is the linear attenuation coefficient for the specific material (either Be or H_2O) for the specific energy of the scattered photon. As a result, the rate of backscattered photons originating in the volume element dV at point P that pass through the elementary surface dS is given by:

$$dndS = (\phi \mu_{sc} dV)(\frac{dS \cos\theta}{4\pi r^2})e^{-\mu_{atm} r} \tag{30}$$

This equation may then be integrated over the hemisphere beyond the dose point in order to evaluate the total contribution due to backscattered gamma rays. The volume element may be written as follows in spherical coordinates:

$$dV = r^2 \sin\theta d\theta d\varphi dr$$

The backscattered gamma current density at dose point S from the hemisphere beyond this point can therefore be expressed using the following volume integral:

$$J_s = \frac{\mu_{sc}}{4\pi} \int_{r=0}^{7HVL} \int_{\phi=0}^{\pi} \int_{\theta=0}^{\pi} \phi \sin\theta \cos\theta d\theta d\varphi e^{-\mu_{atm} r} dr \tag{31}$$

This expression could then be approximated numerically using MS 5-generated values of ϕ at incremental distances in three dimensions from dose point S. Once again, for simplicity, symmetry in the φ and θ directions was assumed and this integral was calculated from $\varphi = 0$ to $\pi/2$ and $\theta = 0$ to $\pi/2$ only. The result produced was then multiplied by four to account for the contributions from the other three identical parts of the forward hemisphere. After trying several combinations of increments, it was decided to use 5 increments in the φ direction (0, 22.5, 45, 67.5, and 90°), 7 increments in the θ direction (0, 15, 30, 45, 60, 75, and 90°), and 7 increments in the r direction. For the dose points at the outer irradiation site and the pool, it was determined that 7 half-value lengths corresponded to approximately 70 cm. This range was calculated based on the highest energy Compton scattered gamma ray. The 7 increments in the r direction therefore corresponded to 10 cm each. Due to the similarity of the μ_{atm} values for beryllium and water, it was assumed that the maximum range of Compton scattered gamma rays to be considered at the inner irradiation site was also 70 cm (i.e., 10 cm increments in the r direction would apply).

A total of 49 MS 5 dose points (7 radial and 7 θ increments) were required to be calculated at each of the 5 φ angles for each of the 5 dose points at which relative gamma backscattering effects were calculated. The resultant incident gamma fluence rates at each of these dose points were then used to approximate the volume integral numerically. Sample calculations revealed that using 7 (r,θ) and 5 φ increments was sufficiently precise without resulting in an overly complex calculation. For example, an increase from 3 to 5 planes in the φ direction resulted in a decrease of 0.1% relative gamma contribution due to backscattering at the first dose point in the pool (31.55 cm in the radial direction). Any further increase in the complexity of the volume integral calculation was deemed unnecessary due to the fact that any uncertainty in the backscattering calculations would be much less than the uncertainty attributable to the MS 5 results.

In the case of the backscattered gamma current density at dose point S situated in the inner irradiation site, Equation 31 had to be modified to account for the fact that backscattered gamma rays would pass through both a certain thickness of beryllium and a certain thickness of water. As a result, the following equation resulted:

$$J_s = \frac{\mu_{sc}}{4\pi} \int_{r=0}^{7HVL} \int_{\phi=0}^{\pi} \int_{\theta=0}^{\pi} \phi \sin\theta \cos\theta d\theta d\phi e^{-(\mu_w r_1 + \mu_{Be} r_2)} dr \tag{32}$$

where μ_w is the linear attenuation coefficient for the average scattered gamma energy in water,

μ_{Be} is the linear attenuation coefficient for the average scattered gamma energy in beryllium,

r_1 is the distance the scattered gamma ray travels in water, and

r_2 is the distance the scattered gamma ray travels in the beryllium reflector.

Once a backscattered gamma distribution in the radial direction is determined, it is important to note that the model as described so far underestimates the contribution of backscattered gamma rays since, at point P, the gamma fluence used is the MS 5-calculated one, which does not account for the photons backscattered to point P from points beyond. The actual incident gamma fluence entering the elemental volume dV will in fact be slightly larger than the values yielded by MS 5. Since the backscattered gamma volume integral approximation at dose point S is directly related to the series of incident gamma fluence values within the volume, this effect can be accounted for simply by squaring the relative backscatter contribution at S and adding it to the original contribution. For example, for a relative backscattered gamma contribution of 15% at a dose point in the pool, a relative contribution to the incident gamma fluence of 15% can be assumed. As a result, the overall effect of this incident gamma contribution is $(0.15)^2$ or 2.25%. This result is added to the calculated backscatter contribution for an overall backscattered gamma contribution of 17.25%.

2.6.1 Compton Scattered Gamma Energy

The average energy of the scattered photon can be computed from the average energy of the recoil electron in a Compton collision. The classic mathematical model that describes the Compton effect gives the following equation for the kinetic energy of the recoil electron in terms of the incident gamma energy $E_{\gamma o}$ and the scattering angle θ (Bonin, 98):

$$E_e = E_{\gamma o} \frac{\alpha(1-\cos\theta)}{1+\alpha(1-\cos\theta)} \tag{33}$$

where $\alpha = E_{\gamma o}/(m_o c^2)$ where $m_o c^2$ is the rest energy of an electron.

If the Compton interaction is assumed to occur isotropically, then the average recoil electron energy across all θ is represented by the following equation:

$$\overline{E_e} = \frac{\int_{\theta=0}^{\pi} E_e(\theta)d\theta}{\int_{\theta=0}^{\pi} d\theta} = \frac{E_{\gamma o}\alpha}{\pi} \int_{\theta=0}^{\pi} \frac{(1-\cos\theta)}{1+\alpha(1-\cos\theta)} d\theta \tag{34}$$

The incident photon energy at a specific dose point was calculated from MS 5 results. It was possible to calculate the average recoil electron energy $\overline{E_e}$ by solving Equation 34 numerically using Simpson's Rule. This was done using a FORTRAN program. Once this integral was solved, the average energy of the backscattered photon was simply the difference between the incident photon energy and the average recoil electron energy. Knowing this value, the linear attenuation coefficients and half-value lengths of the scattered photon in a specific material were easily calculated.

2.7 PROTON DOSE

Unlike photons, light charged particles such as protons have a much smaller and finite range. As a result, their contribution to the dose at a specific point S will come from reactions within a relatively small sphere around the dose point. The radius of this sphere R corresponds to the maximum range of the proton in the specific material. This representation can be seen in Figure 12. Protons produced outside these spheres may be assumed as absorbed prior to reaching the dose point.

The mathematical treatment for proton fluence follows a similar procedure to that of the backscattered photon calculations. Within the elemental volume dV at point P, the rate of proton production is given by $\Sigma_s(E)\phi(E)dV$ where $\Sigma_s(E)$ is the macroscopic scattering cross section for neutrons of energy E in a specific medium, and $\phi(E)$ is the energy-dependent neutron flux. In order to continue with this treatment, one must consider the incident neutron energy required to cause a recoil proton to be ejected. Due to the limited range of protons, only their ranges in water (for the in-pool dose sites) and in air (for the inner and outer irradiation sites) need be considered. One needs to consider the binding energy per nucleon, first of all in water, which will be assumed to be the threshold energy for the incident neutron for the ejection of a recoil proton. The initial and most likely effect of any radiation on water is to split the water molecule, both directly and indirectly, into H atoms and OH radicals:

$$H_2O \rightarrow H^+ + OH^-$$

The OH-H bond strength is approximately 5.16 eV molecule^{-1}. One can therefore make the assumption that an incident neutron with this threshold energy can cause the breaking of this bond, thereby ejecting a proton.

In the case of gamma rays and beta particles, pure water undergoes practically no decomposition at all. For irradiation by light particles, an equilibrium is reached in pure water at low concentrations of hydrogen and hydrogen peroxide such that the water is essentially stable to these types of radiation.

Using an upper energy limit for a recoil proton in water, a maximum range can easily be calculated. This will be the radius of the sphere for the volume integral calculations (Bonin, 98). In the case of water, this maximum range was found to be approximately 0.14 cm. In the case of air, the maximum range was much larger (~10 cm), however, the maximum range of protons in the aluminum shrouding around the irradiation sites (~0.01 cm) led to the assumption that only the recoil protons emitted due to (n,p) reactions in the air of the irradiation site would contribute significantly to the proton dose. Most recoil protons ejected due to (n,p) reactions within the

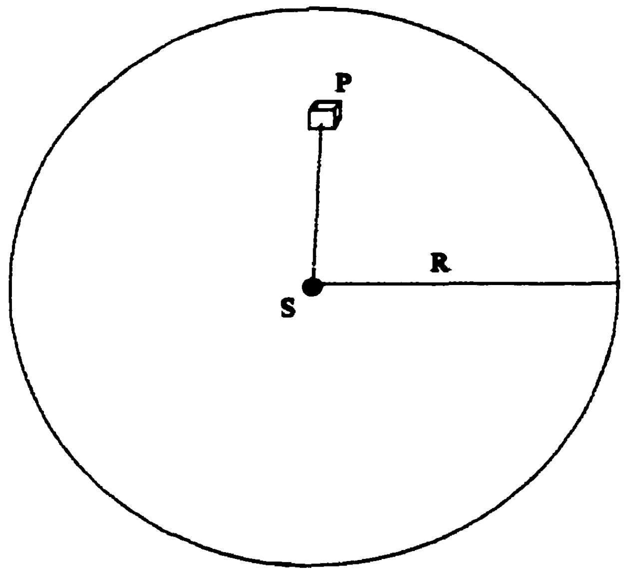


Figure 12: Two-Dimensional Representation of Proton Dose Contribution

aluminum shrouding or farther out in the water or beryllium surrounding the sites would be absorbed prior to reaching the irradiation sites. An approximation was made that the maximum radius of the sphere for volume integral calculations in the inner and outer irradiation sites corresponded to the radius of the cylindrical sites. The binding energies per nucleon for N_2 and O_2 were calculated as 7.23 MeV and 7.71 MeV per nucleon, respectively. Once again, WIMS-AECL provided $\phi(E)$ and $\Sigma_s(E)$ distributions across 26 neutron energy groups and 98 annular regions. Using an average value of 7.5 MeV as the incident neutron threshold in air, and ~5 eV as the incident neutron threshold in water, neutron flux values above these threshold limits in the regions of interest (corresponding to the dose points), could be taken from WIMS-AECL results. In the case of the water sites, 11 of the 26 neutron energy groups were above the threshold value while in the irradiation sites, only the highest neutron energy group possessed sufficient energy to produce a recoil proton. In all cases, one proton is assumed to be ejected isotropically through an (n,p) reaction in the case of N_2 or O_2 molecules, or knocked out isotropically in the case of the H_2O molecule, per collision with a neutron. The energy of the recoil proton is approximated as the energy of the incident neutron minus the binding energy of the proton to the molecule of interest, since the mass of the N_2 or O_2 molecule is much larger than that of a proton.

If the number of protons originating from the volume element dV at point P and passing through the elemental surface dS per unit time is expressed as dpdS, then:

$$dpdS = \sum_{energies} (\sum_s (E)\phi(E,r)dV)(\frac{dS\cos\theta}{4\pi r^2}) \tag{35}$$

The attenuation term is taken as unity since it is assumed that all protons born within a sphere of radius R will reach dose point S if ejected on the proper trajectory. This assumption is an over-prediction of the proton rate as the radius R represents the maximum range due to a neutron at the high energy end of the spectrum giving up all its energy (minus the binding energy) to the recoil proton. This will not necessarily be the case and the proton effect is over-predicted particularly as the distance $\overline{SP}$ approaches R. In the case of the irradiation sites, this over-prediction is compensated for by the under-prediction of proton effects in the axial direction. The radius of the volume sphere is set as the radius of the irradiation site cylinder which under-predicts the effects due to recoil protons formed at distances greater than R in the axial direction.

The resultant volume integral equation for proton current density is therefore:

$$J_s(E') = \frac{\sum_s(E)}{2\pi} \int_{r=0}^{R} \int_{\phi=0}^{2\pi} \int_{\theta=0}^{\pi} \phi(E,r) \sin\theta \cos\theta d\theta d\phi dr \tag{36}$$

where E' is the energy of the recoil proton,

E is the incident neutron energy,

$\Sigma_s(E)$ is the macroscopic scattering cross-section for the material of interest at a specific neutron energy E, and

$\phi(E,r)$ is the neutron flux distribution.

2.8 ELECTRON DOSE

In considering the dose due to electrons, there are typically four sources of electrons:

- electrons stripped by protons,
- electrons ejected due to the Compton effect,
- electrons ejected due to the photoelectric effect, and
- electrons produced by the pair production effect.

The electrons produced due to the latter three effects will not be considered. The energy deposited due to electrons stripped by Compton effect, photoelectric effect and pair production has already been accounted for by MS 5. The range of electrons in the materials of interest is generally much less than that of either the incident or scattered photons; as a result, the dose due to electrons would be deposited into the volume close to the interaction point. The dose rate provided by MS 5 (in Gy-h^{-1}) uses a mass absorption coefficient (μ_a/ρ) which accounts for the combined effects of these three phenomena.

The secondary electrons stripped by the primary "proton-stripped" electrons will not be considered in this work. These delta particles will have minimal effect on the resultant electron dose calculations. The energy deposited in a given volume of absorbing material will be identical whether it is deposited by one primary electron or by many electrons sharing this same energy.

The electron current density as a result of proton stripping is determined using a volume integral calculation similar to the proton current density method. Once again, the electron current density is calculated at the inner and outer irradiation sites as well as at 5 cm increments in the pool. For the irradiation sites, it is important to note that approximately 35 eV is required for each ion pair formed in air. Using the average energy of the incident proton flux at both the inner and outer sites, specific ionization values (ion pairs formed/unit length) could then be calculated. In order to continue with the volume integral calculation, a maximum range of the stripped electrons had to be considered. Due to the limited range of electrons in aluminum, it was assumed that only electrons stripped within the air volume of the irradiation sites would contribute to the dose. For ease of calculation, the maximum radius of the volume integral was set as the radius of the irradiation site (i.e., 1 cm for the inner site and 1.8 cm for the large outer site). This assumption resulted in an under-prediction of the electron dose at all positions above and below the dose point. This under-prediction was offset somewhat by the fact that ion recombination along the electron track was not considered. It is difficult however to assess the relative contributions of these effects.

The electron current density due to proton stripping is given by the following:

$$J_s = \sum_{energies} \frac{I}{2\pi} \int_{r=0}^{R} \int_{\phi=0}^{2\pi} \int_{\theta=0}^{\pi} \phi_p \sin\theta \cos\theta d\theta d\phi e^{-\mu r} dr \tag{37}$$

where I is the number of ion pairs formed by one proton of average energy across the radius R,

ϕ_p is the average proton flux at the dose point, and

μ is the linear attenuation coefficient for an electron in the medium of interest.

In reality, the summation can be neglected as the volume integral was only calculated at the average proton energy at the specific dose point. In each case, the energy of the stripped electron was taken as its rest mass energy.

The mass attenuation coefficient for electrons was calculated as follows (Lamarsh, 83):

$$\frac{\mu}{\rho} \cong \frac{17}{E_{max}^{1.14}} \tag{38}$$

where E_{max} is the maximum electron energy at a specific dose point.

The treatment is similar for the in-pool dose positions. In this case, the maximum radius of the sphere corresponds to the range of rest mass energy electrons in water. In the case of water, approximately 38 eV is required for each ion-pair formed.

The program SRIM 2000 (Cuomo et al., 98) was used to determine the range and stopping power (-dE/dx) of protons in water. SRIM 2000 is a newer version of the program TRIM (Transportation of Ions in Matter, Ziegler, 88) which was first copyrighted by the IBM Corp. in 1984. The program allows the user to input the specific ion and medium of interest and produces among other information, tables of energy-dependent ranges and stopping powers. From the average stopping power of protons in water, the mean energy expended per ion pair formed and the average range of protons in water, the specific ionization I (ion pairs formed by one proton of average energy) could be calculated.

In order to calculate the electron current density due to proton stripping at the in-pool sites, Equation 37 was once again used. The mass attenuation coefficient for electrons in water was calculated using Equation 38. In this case, dr was integrated over the range from r = 0 to r = 0.15 cm which was the maximum range of rest mass energy electrons in water. Once again, ion recombination within the volume integral was not considered.

CHAPTER 3 - EXPERIMENTATION AND EQUIPMENT

TLDs were used to measure gamma doses at various positions around the SLOWPOKE-2 reactor core. Prior to each measurement, the TLDs were first calibrated against a gamma source (in this case, Cs-137) of known activity at a specific distance thereby producing a known gamma dose rate. For the purposes of this experimental work, the calibration centre at Canadian Forces Base (CFB) Kingston was used for the base-lining of all TLDs. The calibration centre provided the TLDs with a known total accumulated gamma dose. Calibration of TLDs took place over a four hour period resulting in a total accumulated gamma dose of 1000 rads. The TLDs were then analysed to determine a calibration factor for each.

3.1 THERMOLUMINESCENT DOSIMETERS

Proper preparation and handling of the TLDs was essential in ensuring accurate, reproducible results. TLDs were only handled with non-metallic tweezers as dirt and oils from the skin can affect their ability to absorb gamma radiation, as will scratching of the ceramic surface. The TLDs were annealed prior to each irradiation (calibration or in-pool). An annealing oven was used to heat the TLDs to 400°C and to maintain that temperature for one hour. This procedure was done to reduce any residual signal present in the TLD structure. The TLDs were calibrated prior to each measurement around the core. The TLDs were always analysed the same day as they were irradiated in order to reduce the effects of fading.

The TLDs were wrapped in tin foil to suppress the low energy gamma over-response characteristics of CaF_2:Mn. Given that a piece of tin is approximately 0.005 cm thick and knowing the linear attenuation coefficients for tin as a function of photon energy, a low-energy gamma dose response relationship between the number of turns of tin and the relative TL response was easily derived. The relationship between filter thicknesses and the resultant low-energy photon response characteristics of the CaF_2:Mn TLD can be seen in Figure 13.

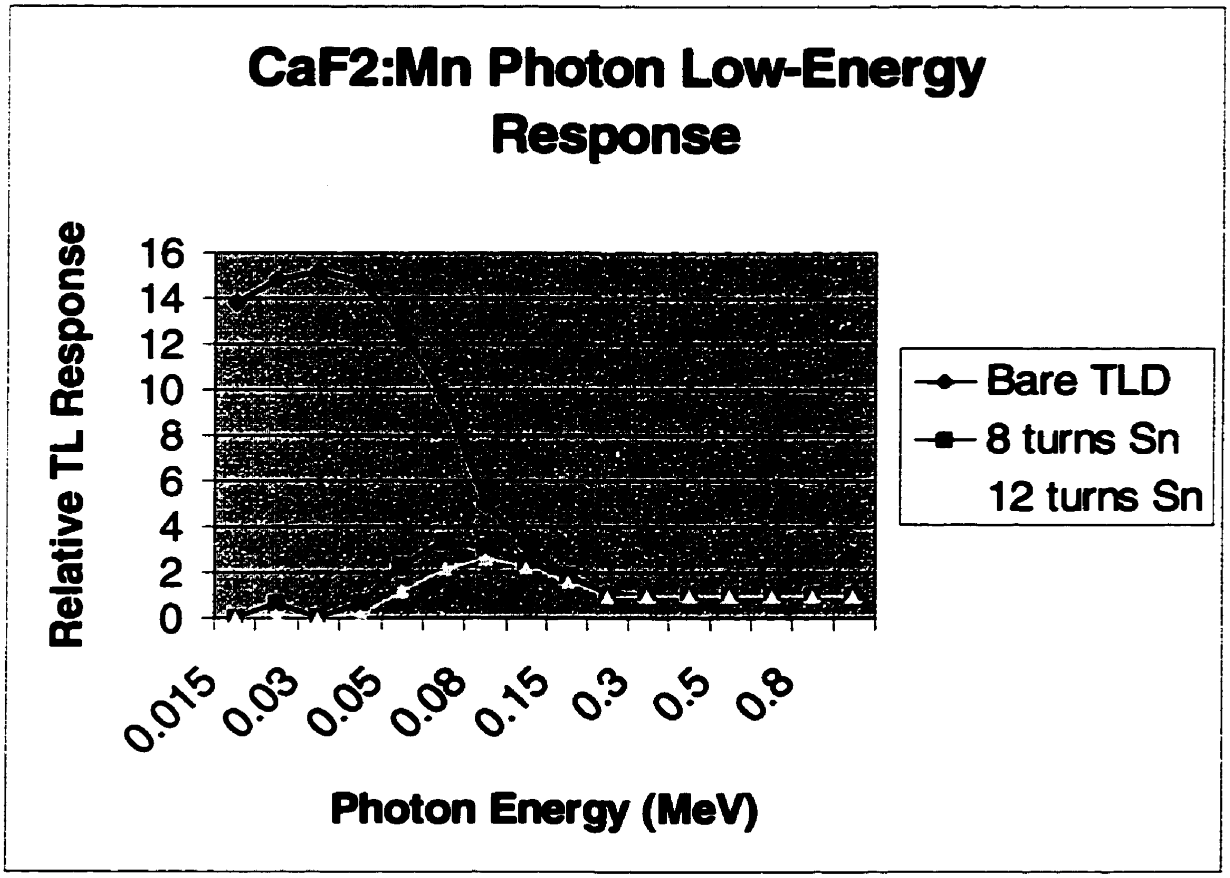


Figure 13: CaF_2:Mn Low-Energy Photon Response. The absolute thicknesses of Sn for 8 and 12 turns is 0.02 and 0.03 cm respectively.

The relative response refers to the TLD response per Roentgen relative to that for ^{60}Co gamma rays. As can be seen, 8 turns of Sn results in fairly good dampening of the over-response characteristics while 12 turns of Sn results in only a slight improvement. For the sake of this experimental work, 8 to 10 turns of tin was used. It was determined that this amount of tin shrouding would result in optimum performance characteristics for the TLDs (i.e., sufficient over-response dampening without excessive over-dampening).

TLDs were shrouded in pairs for ease of handling as well as to maintain consistent positioning during irradiation cycles. Half of the shrouded TLD pairs were then encapsulated in a lithium-bromide (LiBr) shield. The presence of this shield reduced the (n,γ) effect in tin. The fact that only half of the TLD pairs were encapsulated in LiBr allowed for the relative effect due to this shielding to be calculated. Two pairs of shrouded TLDs, one encapsulated in LiBr and one not, were then placed in a 7 cm^3 non-absorbing polyethylene vial. A typical sample assembly can be seen in Figure 14. From left to right can be seen two tin shrouding strips, the LiBr encapsulation material, four CaF_2:Mn TLD chips, and a polyethylene vial.

3.2 SAMPLE HOLDER AND ELEVATOR APPARATUS

As previously mentioned, the sample vials containing the TLDs were irradiated not only at the inner and outer irradiation positions, but also in the pool via the elevator shaft. A schematic of the twin shaft elevator assembly in the reactor pool at RMC can be seen in Figure 15. A sample vial mounting apparatus was constructed of Plexiglas which was found to be relatively resistant to radiation degradation effects. The inner arc of the mounting apparatus was fabricated such that its radius was identical to that of the reactor container. This allowed for a tight fit of the mounting apparatus to the reactor container. The mounting apparatus had three arms extending from the base of the arc at right angles to the surface of the reactor container. Holes of a diameter slightly less than that of the vials were cut at 5 cm increments in these arms for the placement of the sample vials.

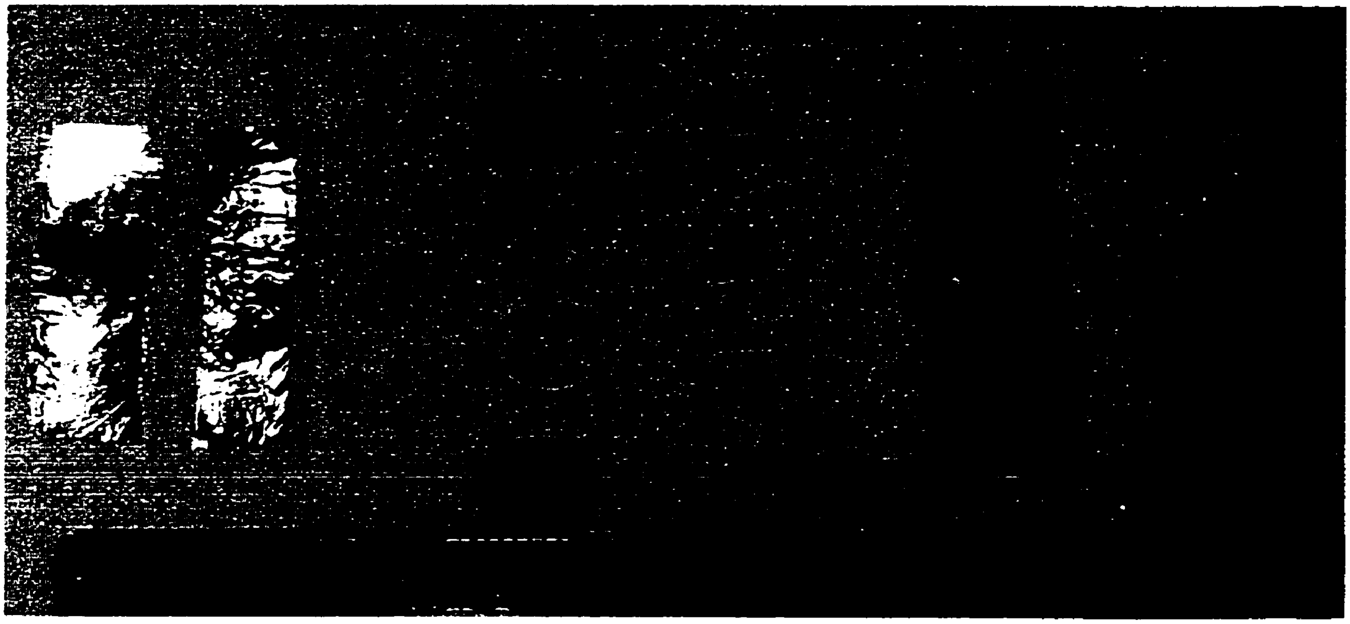


Figure 14: TLD Sample Components
From left to right can be seen two Sn shrouding strips, the LiBr encapsulation material, four CaF_2:Mn TLD chips, and a 7 cm^3 polyethylene vial

The sample holder was attached to the elevator carrier via an L-shaped arm. The height of this arm was adjustable at the elevator carrier such that, when at the bottom of the shaft, the sample holder sat along the plane of the reactor core mid-height. The entire elevator shaft could be swung into place once the sample holder had reached the bottom of the shaft, in line with the reactor core mid-plane. A rope was attached to the elevator carrier for lowering and raising the assembly. A schematic of the elevator carrier with the L-shaped sample holder support system can be seen in Figure 16. Figure 17 is a view of the sample holder resting against the reactor container wall. The sample vials can be seen in each of the three arms. A close-up of the sample holder with vials, the L-shaped bracket, and the elevator carrier can be seen in Figure 18. Figure 19 is a close-up view of the sample holder with arms and sample positions marked (note vials are in position 3).

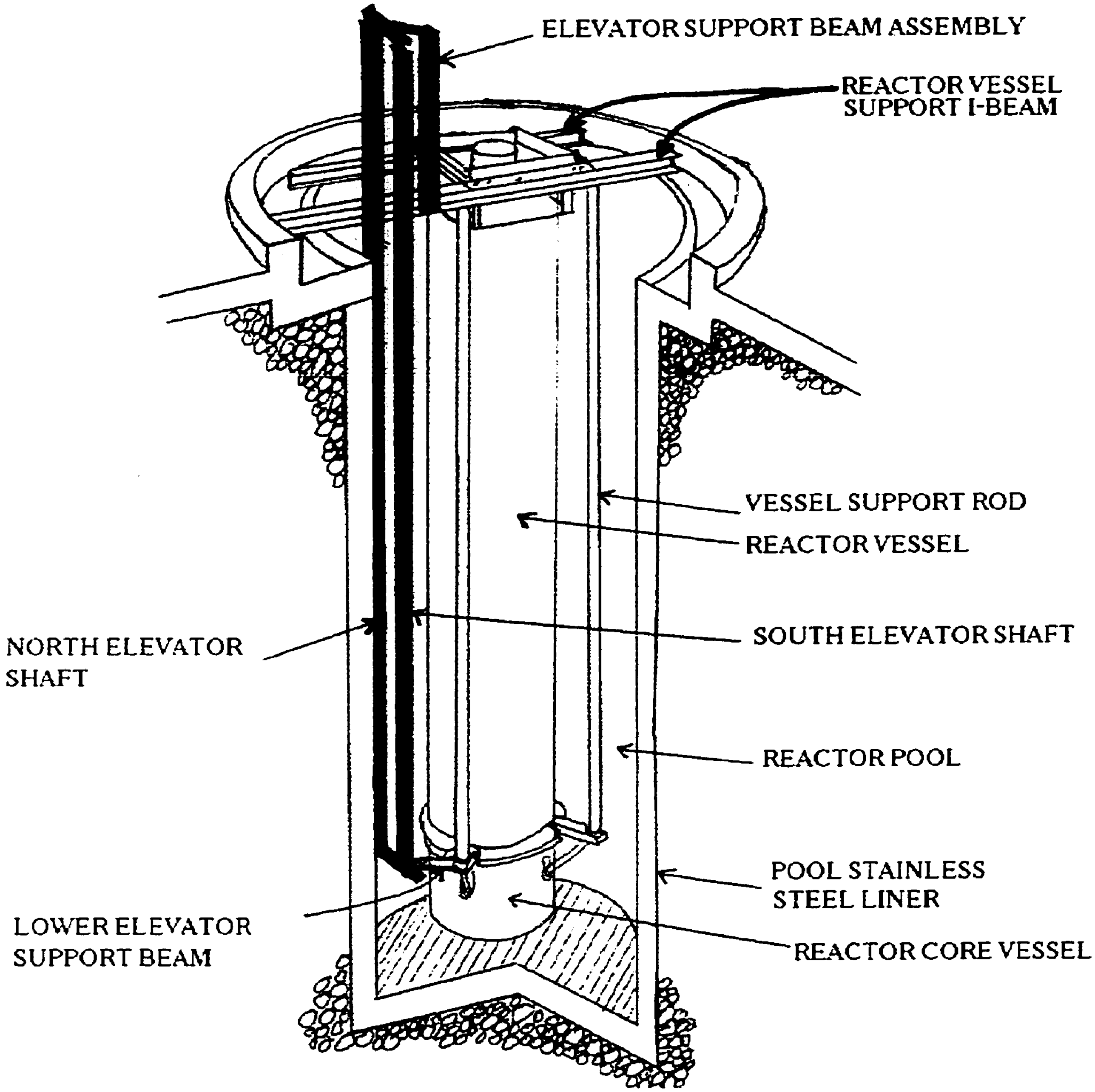


Figure 15: Twin Shaft Elevator Assembly in SLOWPOKE-2 Pool

3.3 TLD READER

The HARSHAW Model 4500 TLD Reader was used to analyse the irradiated TLDs. The Dual Photomultiplier Manual TLD Reader and Workstation read the CaF_2:Mn chips individually. It used a heated planchet which can reach 600 °C. An external PC workstation operating in DOS mode ran the program DREO Production (Harshaw Bicron TLD-REMS) which specified the time-temperature distribution for the planchet, the period of data collection, produced the thermoluminescent glow curve, and analysed the area under the curve. Secondary functions such as annealing and photomultiplier tube (PMT) calibration were also possible using the Defence Research Establishment Ottawa (DREO) software package. A typical glow curve can be seen in Figure 20. The solid light blue line shows the time-temperature distribution of the internal planchet with temperature represented by the y-axis and time by the x-axis. The solid red curve represents the output signal of the PMT (i.e., the glow curve) with thermoluminescent intensity represented by the y-axis and time by the x-axis. The program provided the needed area-under-the-curve in units of charge, which, in the case of this experimental work, was given in micro-coulombs (μC). The value in μC was directly related to the total accumulated gamma dose received (in rads) using the calibration factor.

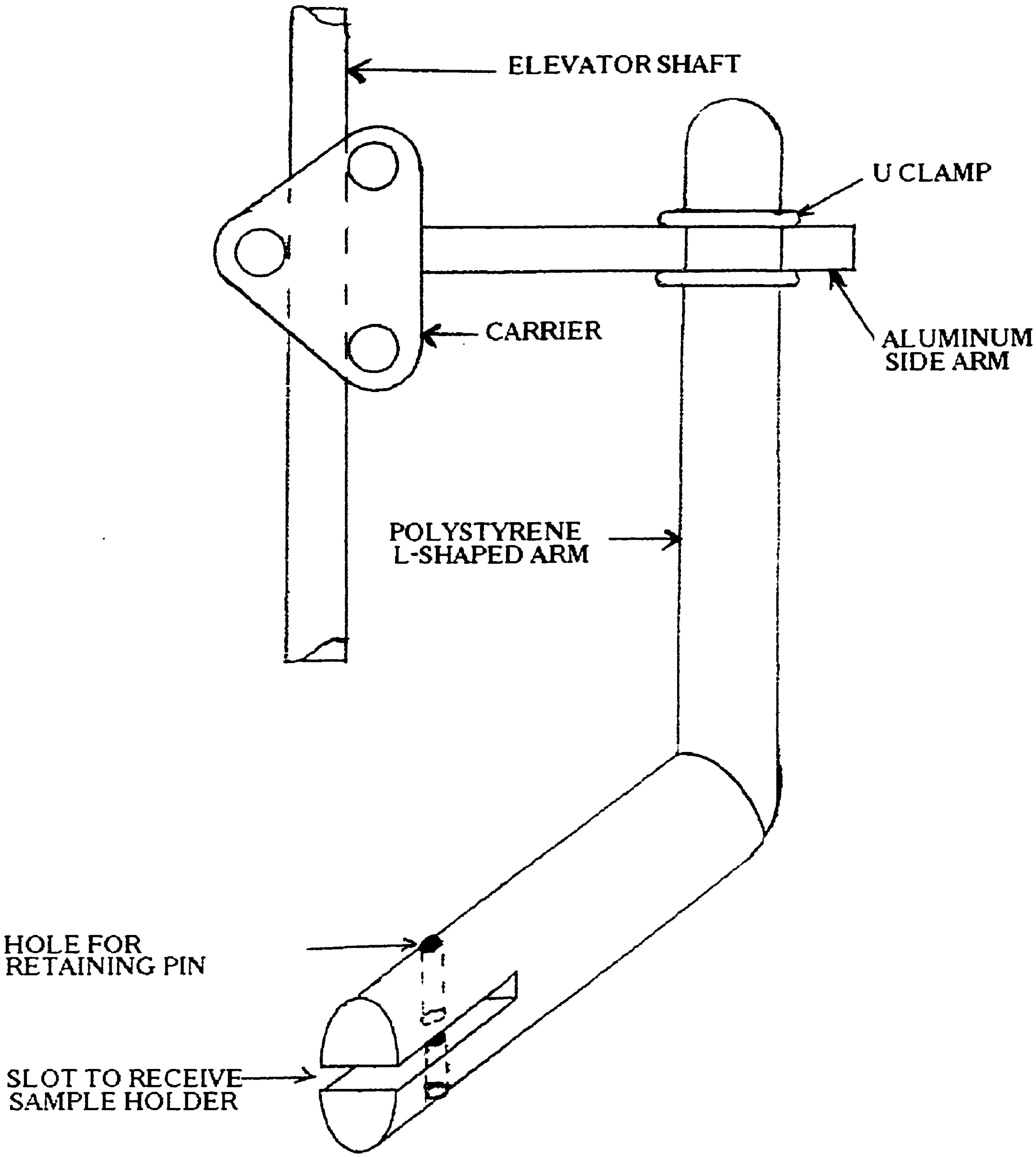


Figure 16: Elevator Carrier with L-Shaped Arm Sample Holder Support System

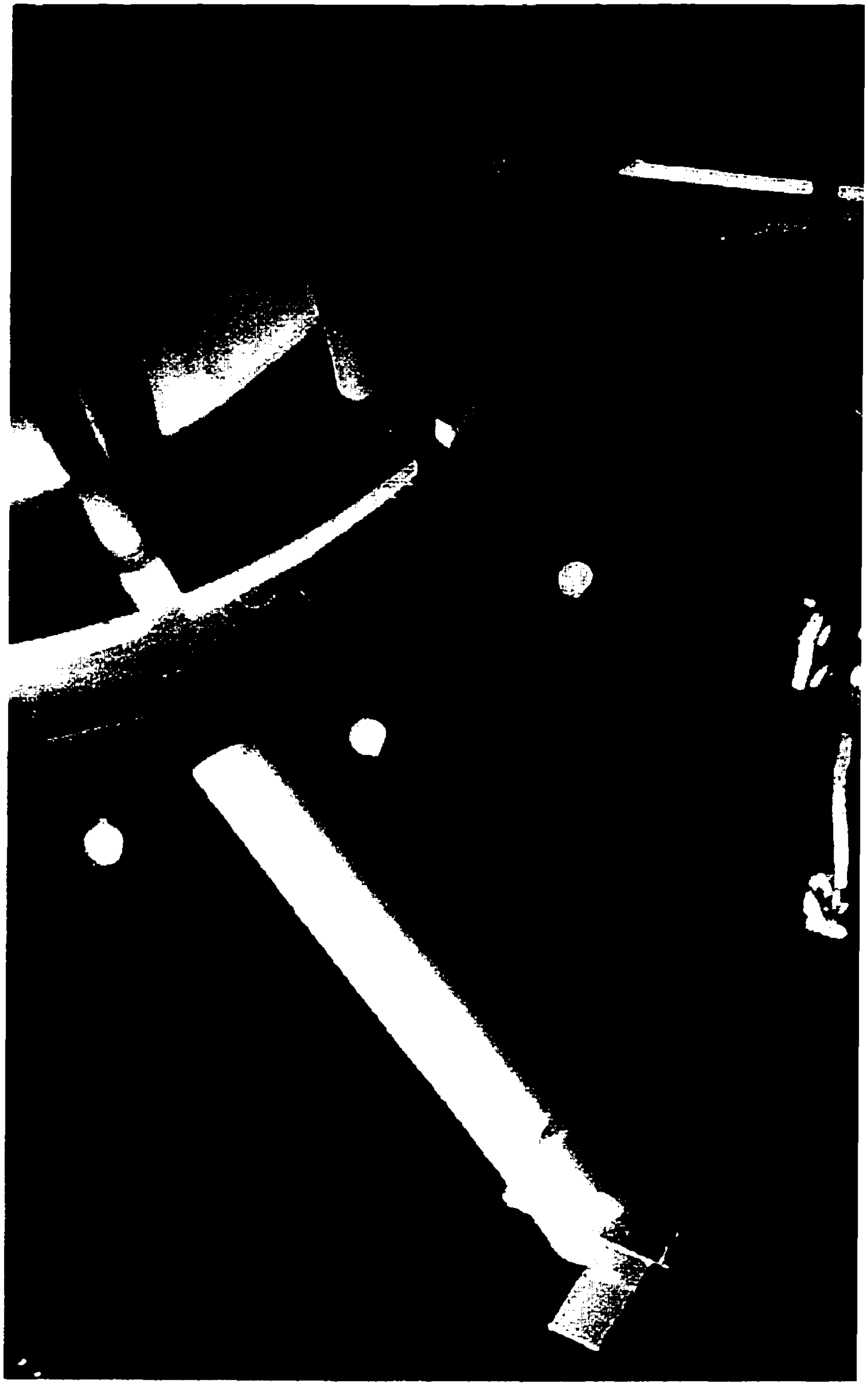
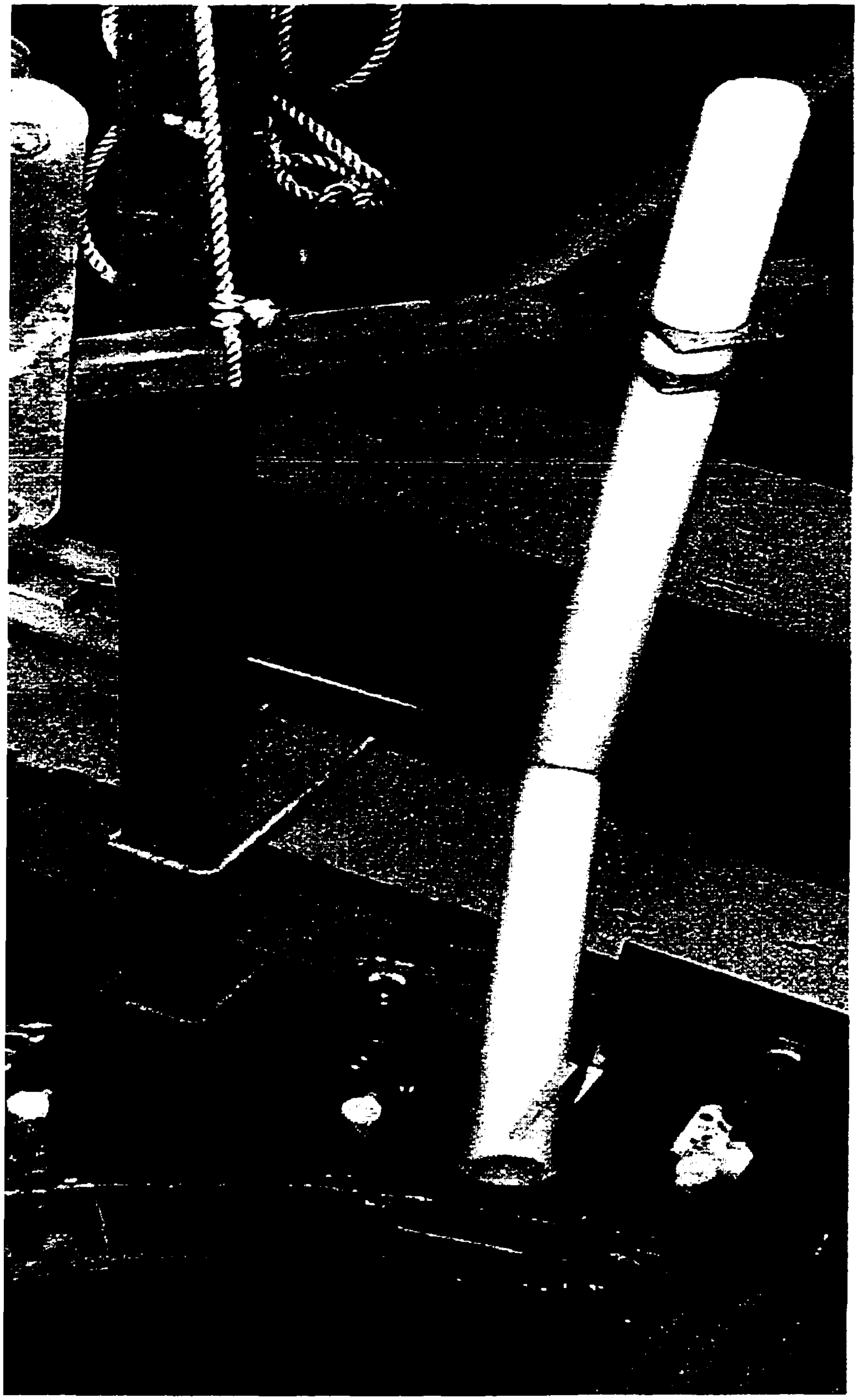


Figure 17: View Looking Down into Pool of Sample Holder A



of Sample Holder with Vials in Elevator Carrier Assembly

Figure 19: Sample Holder with Arms and Vial Positions Marked

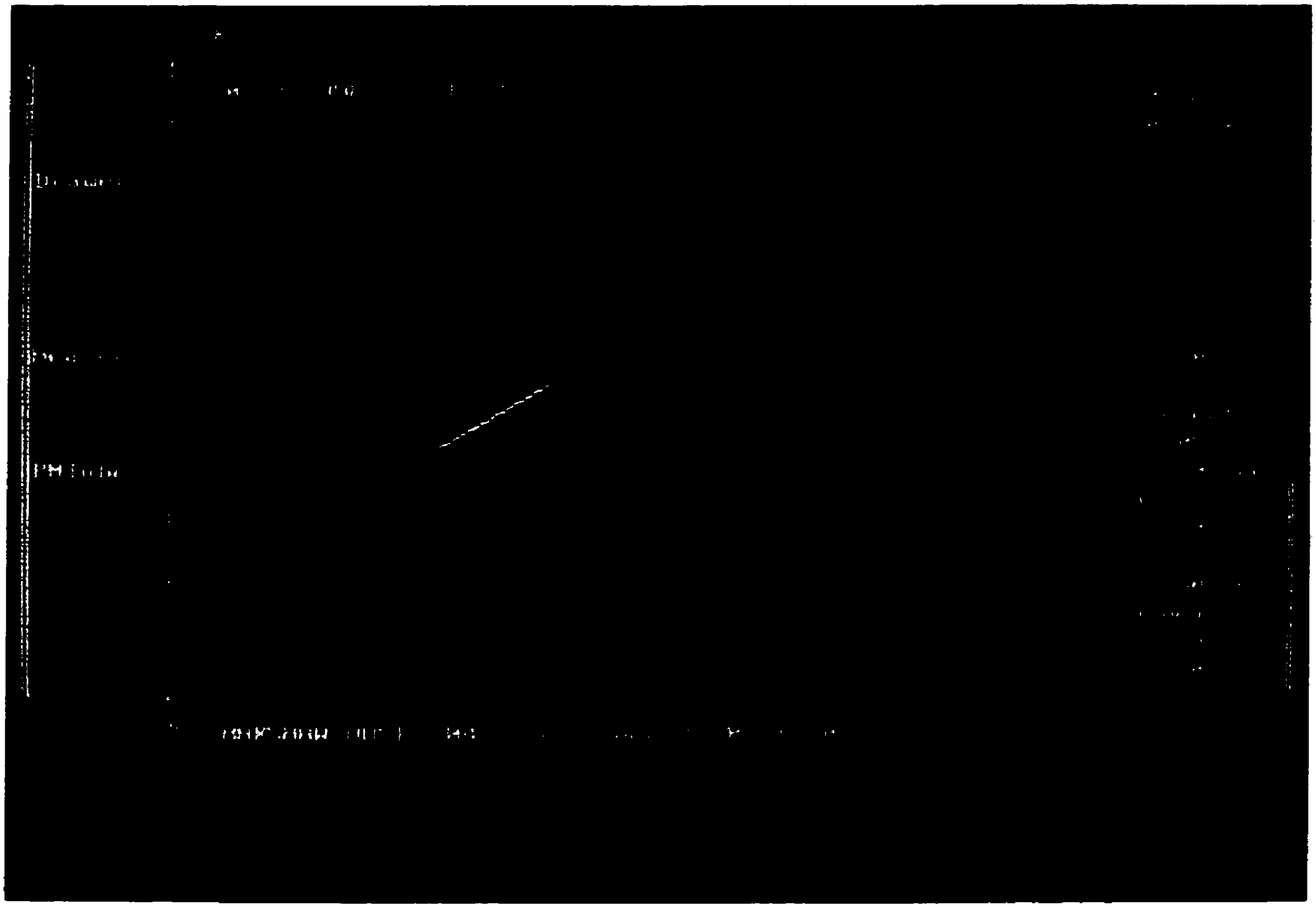


Figure 20: Typical Glow Curve with Time-Temperature Distribution

For the purposes of this experimental procedure, the time-temperature distribution selected by DREO was as follows: the planchet temperature rose to 100°C (the pre-heat set-point) and remained there for 15 seconds, next, the temperature ramped up linearly to 370°C (the maximum set-point) over the next 45 seconds, and then the planchet remained at 370°C for 1 minute at which time the temperature was allowed to fall towards ambient values. The cycle ended once the planchet temperature reached 50°C. The time-temperature profile used was specific to the CaF_2:Mn chips (i.e., other type of TLDs would require a separate profile).

CHAPTER 4 - RESULTS AND DISCUSSION

The presentation of results and subsequent discussion is divided into three sections, the first section being the modelling of the SLOWPOKE-2 core using WIMS-AECL (Annex A), which was used to determine the neutron flux distribution across the reactor vessel and into the pool. These results are compared with the NA values (Andrews, 89). The dose rate due to neutrons at the positions of interest (i.e., the irradiation sites around the core and irradiation positions in the pool) are then calculated. The validity of the WIMS-AECL model will be analysed.

Results from TLD measurements around the core will then be presented in the second section. Presentation of MS 5 gamma dose rate results will allow comparison between experimental and analytical values. The development of the MS 5 source terms will be analysed in detail. The validity of the MS 5-based SLOWPOKE-2 model will be discussed.

The dose rates due to protons and proton-stripped electrons at the positions of interest are then presented in the third section. The analytical techniques used to assess these phenomena will be presented. Once again, the validity of these analytical assumptions will be discussed. An assessment of the uncertainty in the specific particle dose rates will be discussed in Chapter 5.

4.1 WIMS-AECL MODEL

The modelling of the SLOWPOKE-2 core was a long process with many obstacles along the way. One of the first major stumbling blocks encountered was due to the fact that the current version of WIMS-AECL only allowed for the fuel pins to be input into the code in a cluster geometry. In other words, WIMS-AECL only accepts a symmetrical distribution of fuel pins around the central axis of the core. The code required that the fuel pins be grouped into arrays, each array consisting of a certain number of fuel pins spaced evenly around a circle of radius r corresponding to the distance from the centre of the core to the circumference of the circle. This is not in fact the actual geometry of the SLOWPOKE-2 core fuel pin placement as there are fuel pins missing from a symmetrical distribution. The WIMS-AECL Manual (Griffiths, 94) noted that individual placement of each of the 198 fuel pins in a cartesian coordinate system should be possible, however, when that was attempted, the author and the AECL code custodian soon discovered that that option was not supported by this version of the code. As a result, a symmetrical approximation of the fuel pin geometry was required in order to run the WIMS-AECL code. The author attempted to maintain the actual SLOWPOKE-2 fuel pin geometry as much as possible.

WIMS-AECL performs neutron transport calculations in two dimensions only. As a result, certain materials such as the central control rod (which during the experimental phase of this thesis sat approximately 3.5" from the bottom of the core), the top beryllium shims, and the bottom beryllium slab could not be accurately modelled. In the case of this experimental work, the effects due to the beryllium were discounted, and the control rod geometry was reduced to account for the fact that it sat part-way out of the core. It was anticipated that neglecting the effects due to the beryllium would only affect the final neutron flux distribution results in a very minor way. It is important to note once again that the frame of reference for all radiation particle flux results was at the core mid-height. The fact that the neutron flux distribution curve towards the bottom of the core (where the control rod had been removed) would vary from the distribution further up the axis of the core could not be accounted for either by the model. Once again, it was assumed that this effect would have only a minor influence on the neutron flux distributions, particularly as one progressed further from the axis of the core.

The cross-section of the lattice cell was input as a series of annular regions around the central core axis. This geometry lends itself well to the cylindrical SLOWPOKE-2 core assembly. In this case, for instance, the light water coolant within the fuel cage was subdivided into concentric annular regions of a specific thickness. The beryllium reflector, water annulus, container wall, and pool water beyond the reactor container were also treated in a similar fashion. Although WIMS-AECL allowed for sectors within an annular region, it was decided not to include the heavy water thermal column in the model (this thermal column is a small sector-shaped aluminum container filled with D_2O and wedged between the Be reflector and the reactor container wall. Its function is to allow a larger thermal neutron flux to reach the location in the pool where the bottom end of the neutron beam tube is positioned when the neutron radiography system is operating.). The WIMS-AECL output neutron flux distribution is non-directional in nature and it was decided that the effects due to the heavy-water thermal column would reduce the accuracy at the actual irradiation positions. In reality, the experimental irradiation positions are beyond 90° from the position of the thermal column along the circumference of the reactor container. As a result, it was concluded that neutron flux effects due to the heavy water annulus at the irradiation positions would be negligible, as proven by Andrews (Andrews, 89).

Following the input of all geometrical and material specifications related to the

SLOWPOKE-2 lower assembly, values for the radial and axial buckling (in units of cm^{-2}) were required. Numerous runs were required (over the period of a month) each time altering slightly these values of buckling in order to converge towards criticality (i.e., $k_{eff}=1$). WIMS-AECL was able to perform a limited buckling search of its own in the pursuit of a critical system. The top and bottom beryllium geometries were included in the initial axial buckling calculation. A critical bare reactor, infinite slab geometry approximation was used for the initial calculation of radial and axial buckling coefficients. The following equation was used (Lamarsh, 83):

$$B_c^2 = \left(\frac{\pi}{a}\right)^2 \tag{39}$$

where B_c is the critical buckling, and

a is the slab thickness.

The radius r was substituted for a when calculating radial buckling, and the height h substituted for an axial buckling calculation.

An effective multiplication factor (k_{eff}) of 1.00071 was eventually reached after successive runs, which was determined to be acceptable for the purposes of this experimental work. Small changes in the transverse buckling terms between WIMS-AECL runs, it was found, did not significantly affect the neutron flux distribution. It should be noted that the RMC SLOWPOKE-2 reactor is licensed to operate at a relatively small maximum excess reactivity of 4.0 mk.[2]

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[2]The "mk" or "milli-k" unit is used in Canada to express the reactivity of a reactor. 1 mk refers to a relative deviation from criticality of 0.001 or 0.1%. For a ^{235}U-fuelled reactor, 1 mk is equivalent to a reactivity of 15.4 cents or \$0.154.

WIMS-AECL neutron flux values were given in terms of relative absolute flux. As a result, a baseline value was required. In this case, the thermal neutron flux at the position of the self-powered neutron detector used to control the reactor (radius 16.13 cm from central axis of the core) was used to convert all output values from relative to absolute units. As previously mentioned, all work related to this thesis assumes that the reactor is operating at half power (i.e., 5×10^{11} n-cm^{-2}-s^{-1} at the detector position). WIMS-AECL calculated thermal neutron flux values as they vary with distance from the centre of the core can be seen in Table I. These results assume an average coolant temperature of 30°C. An uncertainty in the flux values of ± 5% is assumed.

TABLE I

WIMS-AECL Calculated Thermal Neutron Flux

Steady-State Half Power Operation (10 kW_{th}) at Reactor Mid-Height

POSITION **Distance from Core Centre (cm)**	**THERMAL NEUTRON FLUX (n-cm^{-2}-s^{-1})** **(± 5%)**
15.2 (inner irradiation site)	5.00×10^{11}
23.6 (outer irradiation site)	3.01×10^{11}
31.7	5.57×10^{10}
34.7	2.65×10^{10}
37.7	1.28×10^{10}
40.1	7.35×10^{9}
42.2	4.68×10^{9}
44.9	2.84×10^{9}
47.3	2.05×10^{9}
50.3	1.73×10^{9}

Note: WIMS-AECL thermal neutrons have an upper energy threshold of 0.625 eV.

In order to ascertain the validity of these neutron flux distribution results, it is important to compare these with previous measurements (Andrews, 89). It should be noted that the upper energy threshold of thermal neutrons for Andrews' work was approximately 0.55 eV (the cadmium cut-off). A comparison of WIMS-AECL and Andrews' neutron activation (NA) thermal neutron flux distributions can be seen in Figure 21. The NA values were measured at z = -5 cm (5 cm below reactor mid-height) and at an angular displacement of 90° from the centre of the thermal column.

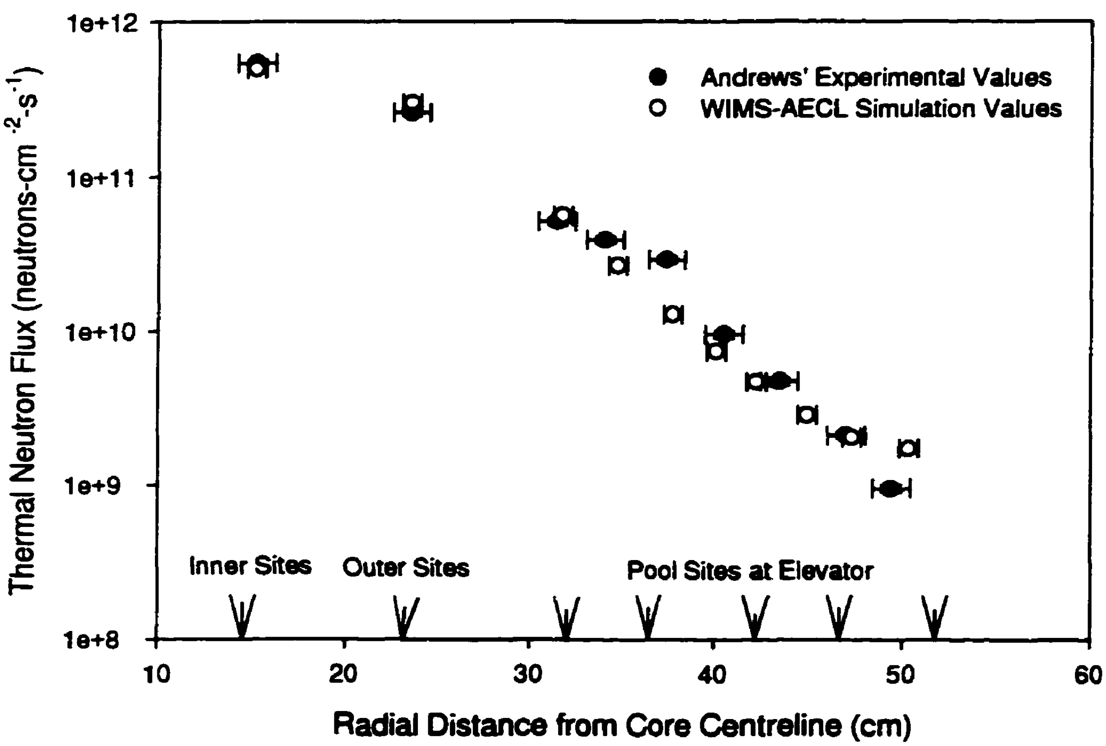

Figure 21: SLOWPOKE-2 Radial Thermal Flux Distribution
Steady-State Half Power Operation (10 kW_{th}) at Reactor Mid-Height

As can be seen in Figure 21, the WIMS-AECL code-generated thermal neutron flux distribution compares well with Andrews' NA-generated values (the variance being no more than $\pm$ 5%). The WIMS-AECL error assessment will be discussed later. A strong correlation between the two sets of data can be seen at the inner and outer irradiation sites. The thermal flux distributions are expected to continue to diverge somewhat beyond 50 cm as Andrews determined that a higher uncertainty ($\pm$ 9%) exists beyond this point due to the lower thermal neutron fluxes.

It is interesting to look at thermal-to-epithermal flux relationships as a means of comparison between experimental NA results and WIMS-AECL code-generated results. Thermal and epithermal neutron flux values from WIMS-AECL calculations as functions of radial distance from the core centre-line can be seen in Table II. For this experimental work, the epithermal energy region has a lower limit of 0.625 eV (coincident with the thermal neutron upper energy limit) and an upper limit of 4.0 eV. This value was chosen for comparative purposes with Andrews' work where gold-cadmium ratios were used. An uncertainty of $\pm$ 5% is assumed on all neutron flux values.

TABLE II

WIMS-AECL Calculated Thermal and Epithermal Neutron Flux

Steady-State Half Power Operation (10 kW_{th}) at Reactor Mid-Height

POSITION Distance from Core Centre (cm)	THERMAL NEUTRON FLUX ($n\text{-}cm^{-2}\text{-}s^{-1}$) ($\pm$ 5%)	EPITHERMAL NEUTRON FLUX ($n\text{-}cm^{-2}\text{-}s^{-1}$) ($\pm$ 5%)	THERMAL/ EPITHERMAL RATIO ($\pm$ 10%)
15.2 (inner irrad site)	5.00×10^{11}	4.24×10^{10}	11.8
23.6 (outer irrad site)	3.01×10^{11}	9.05×10^{9}	33.3
31.7	5.57×10^{10}	8.74×10^{8}	63.7
34.7	2.65×10^{10}	3.88×10^{8}	68.3
37.7	1.28×10^{10}	1.90×10^{8}	67.2
40.1	7.35×10^{9}	1.15×10^{8}	64.1
42.2	4.68×10^{9}	7.72×10^{7}	60.7
44.9	2.84×10^{9}	5.03×10^{7}	56.4
47.3	2.05×10^{9}	3.86×10^{7}	53.1
50.3	1.73×10^{9}	3.41×10^{7}	50.7

A comparison of WIMS-AECL-generated and Andrews' experimental thermal/epithermal ratios can be seen in Figure 22. Once again, Andrews' values were taken at $z = -5$ cm and $\theta = 90°$. It is immediately obvious that the curves differ significantly. Andrews' gold cadmium ratios show a positive slope whereas WIMS-AECL predicts an almost constant ratio from the reactor container wall to approximately 10 cm into the pool, at which point the ratio decreases very gradually in a linear relationship. One of the possible reasons for this discrepancy is the fact that the upper epithermal energy level for the WIMS-AECL calculations was somewhat arbitrarily chosen as 4.0 eV, which was done to coincide with previous comparative research done by Andrews and deWit (Andrews, 89). In fact, it is impossible to specify an accurate upper energy limit for resonance capture in gold. As a result, accurate comparisons between computational and

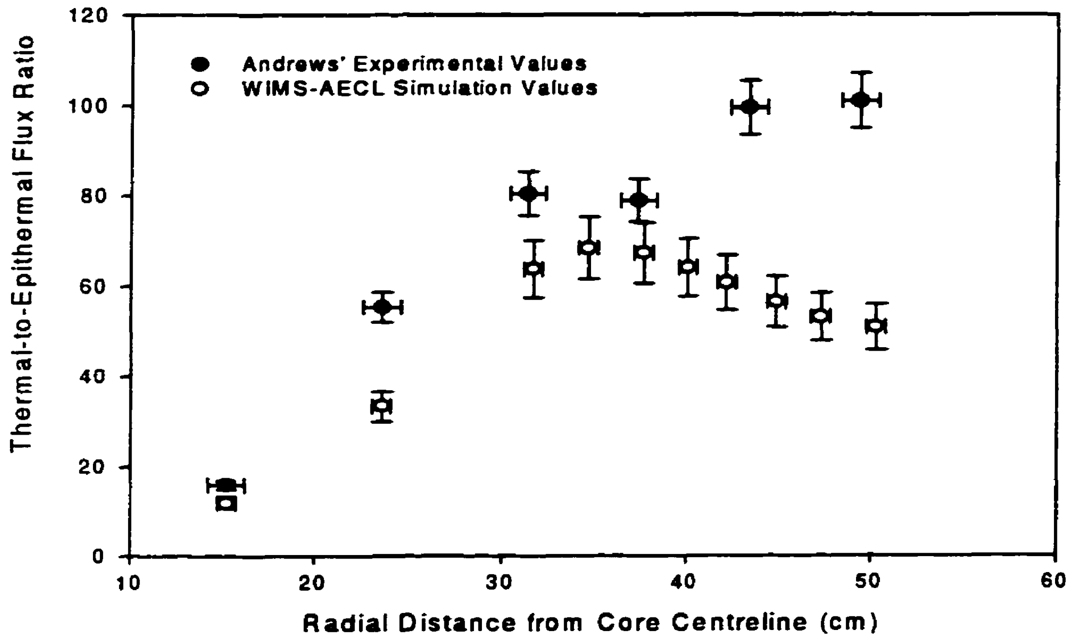

Figure 22: SLOWPOKE-2 Thermal-to-Epithermal Flux Ratios Steady-State Half Power Operation (10 kW_{th}) at Reactor Mid-Height

experimental epithermal neutron fluxes is a near-impossible task. It is generally agreed that the fast and thermal flux distributions approach an almost constant negative slope as one progresses further into the pool with the fast flux diminishing at a larger rate than the thermal neutron flux. This phenomenon results in a thermal-to-epithermal flux ratio more in accordance with the WIMS-AECL predictions.

The neutron dose calculation was a straightforward exercise once the neutron flux distribution was provided by WIMS-AECL. WIMS-AECL calculated the absorption and scattering cross-sections Σ_a and Σ_s for the materials of interest (in this case, air and H_2O) across the 26 neutron energy groups. Using Equations 16 and 17, the dose due to neutrons could easily be calculated. The dose rate due to neutrons at the positions of interest around the SLOWPOKE-2 core can be seen in Table III. A plot of neutron dose rates in the pool around the SLOWPOKE-2 reactor as a function of the radial distance from the core centre-line can be seen in Figure 23. At the inner and outer irradiation sites, the results are given as dose in air while the in-pool neutron dose is given as the dose in water. In both cases, a target volume of 1 cm^3 is assumed. All dose rate values are at the reactor mid-height.

An uncertainty of $\pm$ 10% is assumed for all values of neutron dose rate based on known fundamental errors. The uncertainty on the neutron dose rate values could indeed be much higher due to modelling errors. The degree to which a model predicts the physical realities of the reactor container is due to many factors such as: the model's ability to duplicate all geometric parameters, the model's ability to duplicate all material compositions (such as fuel depletion) and the model's ability to duplicate the kinetics of the physical core (such as resonance effects, neutron streaming, and neutron diffusion in the axial direction). Modelling errors will very likely account for a much higher uncertainty than that already calculated however, it is virtually impossible to predict an absolute uncertainty due to these errors.

TABLE III

Neutron Dose Rates at Positions Around the SLOWPOKE-2

Steady-State Half Power Operation (10 kW_{th}) at Reactor Mid-Height

(Dose Rate in H_2O, except as noted)

Position	Distance from Core Centre-line (cm) (± 0.5 cm)	Neutron Dose Rate (Grays-h^{-1}) (± 10%)
Inner Irrad Site	15.20	1.1×10^2*
Outer Irrad Site	23.60	17*
Pool Site 1	31.70	5.0×10^2
Pool Site 2	34.70	2.7×10^2
Pool Site 3	37.70	1.6×10^2
Pool Site 4	41.10	1.0×10^2
Pool Site 5	42.50	72
Pool Site 6	44.90	54
Pool Site 7	47.30	43
Pool Site 8	50.30	47

Note: * Dose rate in air.

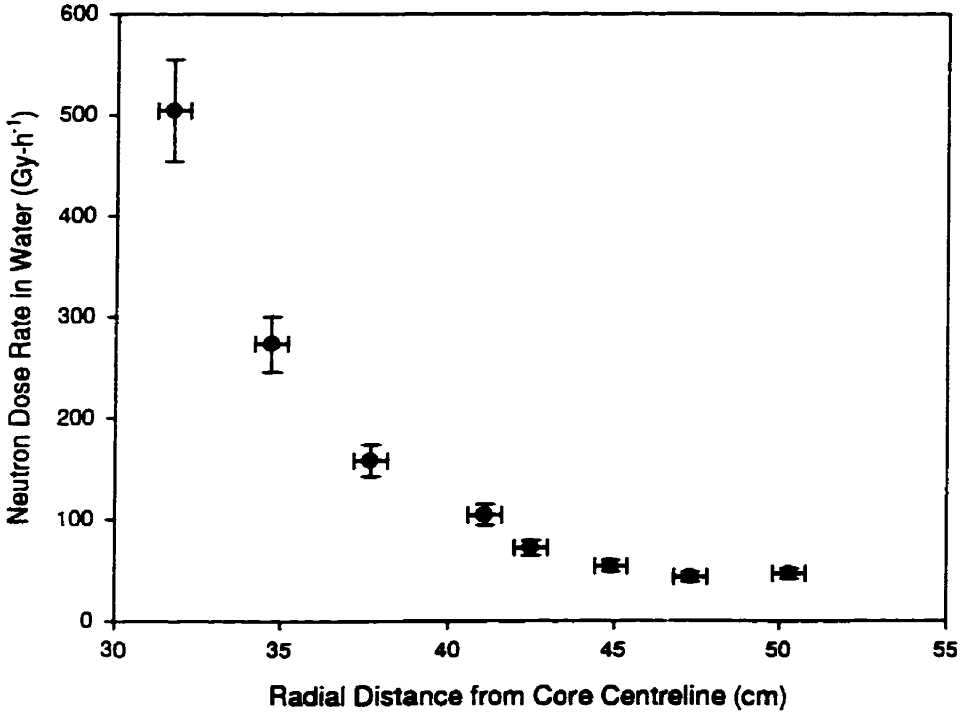

Figure 23: SLOWPOKE-2 In-Pool Neutron Dose Rates
Steady-State Half Power Operation (10 kW_{th}) at Reactor Mid-Height

4.2 THERMOLUMINESCENT DOSIMETRY

CaF_2:Mn TLDs were chosen for gamma dosimetry over other spectroscopic techniques or other TLDs for a number of reasons. Scintillation detectors such as sodium-iodide (NaI) are limited to low gamma dose rate applications due to dead time effects. In addition to this "low dose saturation" characteristic, the NaI detector is bulky and would be impossible to position in the irradiation sites and at the in-pool positions (due to the fact that the pre-amplifier is attached directly to the detector which is in turn hard-wired to a multi-channel analyser). Activation of the detector housing would be another detrimental effect of the use of this type of spectroscopic detector. The usefulness of other experimental dosimeters such as gamma bubble detectors is limited, for the purposes of this work, due to a low saturation dose rate limit and low detection efficiency. When compared with other TLD materials, the CaF_2:Mn possessed a higher sensitivity than most TLDs (approximately 10 times more sensitive than lithium-fluoride, LiF), had a broad useful range (0.1 μGy - 100 Gy) and deeper traps which results in minimal fading (approximately 8% over 24 hours). In addition, the CaF_2:Mn TLDs were readily available from DREO which assisted in its selection as the dosimeter of choice for this work. Like all TLDs, the CaF_2:Mn dosimeters are re-usable, compact, and easily analysed, all of which contributed to its selection.

TLDs were used to map the gamma dose rate at the inner and outer irradiation sites as well as at specific positions in the pool around the SLOWPOKE-2 reactor container. All physical measurements were taken at the reactor mid-height. Following instructions by DREO, an upper accumulated dose limit of 5000 rads was used in order to prevent saturation of the CaF_2:Mn TLDs that were selected for this work. As a result, numerous irradiation cycles were performed prior to arriving at the final irradiation periods.

The frequency of calibration procedures was continually modified as the experimental work progressed. Early in the experimental procedure, calibration of the TLDs following numerous irradiation cycles was attempted. This procedure proved unsuccessful due to the supra-linearity of dose response as a function of total accumulated dose over the lifetime of the dosimeter as discussed in Section 2.2. It was found that results obtained in the past could not be re-produced under similar experimental conditions. This was attributed to this phenomenon of supra-linearity and it was found that following re-calibration of the TLDs, the calibration factors had indeed decreased. Eventually, as the TLDs aged, calibration prior to each irradiation cycle was decided upon which resulted in the most accurate set of results.

As the in-pool irradiations involved raising and lowering of the sample holder with the reactor operating, it was important that the gamma dose received by the TLDs during these steps be accounted for. This operation involved lowering the assembly to the bottom of the pool, swinging the assembly into place and then, following the irradiation cycle, swinging the assembly away from the irradiation position and lifting, using the rope, the assembly to the surface of the pool. The additional radiation dose received by the TLDs due to this entire operation was felt to be minor, as the positioning and removal operation took no more 30 seconds, however, it was desired to have a quantitative assessment of this additional dose. Separate irradiation cycles were conducted under similar experimental conditions whereby the sample holder was lowered into position and then immediately raised to the surface of the pool. These cycles were done at each of the in-pool irradiation positions. These results were then subtracted from each irradiation cycle to provide an "adjusted" dose received. The results of the additional dose received experiment varied from 175.7 rads to 76.4 rads on the left sample holder arm and from 213.7 rads to 113.1

rads on the centre sample holder arm. In general, a higher dose was received by the centre arm than by the left arm. This was not surprising as the centre arm was closer to the reactor core during the lowering and raising of the carrier assembly. Although it was expected that the cumulative dose received by the TLDs would decrease as position numbers increased, this was not always the case. The inherent uncertainty in the TLDs as well as the fact that the cumulative dose received was relatively low were likely factors in the lack of an expected decreasing trend. A further discussion on the uncertainty in the TLD dose response characteristics can be found in Chapter 5.

The centre and left arms of the sample holder were used to conduct TLD measurements (see Figure 19). These arms are separated by a 22.5° angle. The use of two sets of gamma dosimeters equi-distant from the core centre-line provided the author with needed redundancy and a means of comparison for validation of results. The irradiation sites chosen for TLD measurements were inner site #4 and outer site #10. Site #4 was chosen due to its proximity to the in-pool sample holder position. Site #10 is situated almost opposite site #4, on the east side of the reactor container. However, the adjacent outer position, site #9, is cadmium-lined which eliminated its selection as a suitable measurement position.

One of the principal concerns in conducting this gamma dose mapping was in the reproducibility of the results. As discussed in Section 2.5, the gamma dose rates around the SLOWPOKE-2 core during reactor operation are a complex function of not only prompt fission gamma rays but among other sources, delayed gamma rays due to short and long-lived fission products. As a result, it is nearly impossible to reproduce experimental conditions from one irradiation cycle to the next. The recent reactor operating history must be considered in order to determine the importance of fission product activities to the total gamma dose rate at a given position. As discussed in Section 2.5.1, long-lived fission product activities are considered negligible for the purposes of this experimental work due to the relatively young age of the UO_2 fuel in the SLOWPOKE-2 core at RMC. The short-lived fission product activities need to be considered however. Approximately 75% of the total gamma energy is emitted within 30 minutes of the fission event, meaning that short-lived fission product activity predominates when the reactor is operating. The SLOWPOKE-2 reactor is operated intermittently with minimum shut-downs of 16 hours during the week and 64 hours over the weekend. Even over the period of only one night, the short-lived fission product activity is reduced to negligible levels. Initially, the TLD measurements were taken on Monday mornings, shortly after start-up of the reactor where virtually nil short-lived fission product activity is expected. Later, measurements were taken on different days of the week in order to assess the importance of the short-lived fission product effect. It was found that any discrepancies between TLD measurements taken on a Monday and those taken later in the week fell well within the uncertainty attributable to the TLD response, as can be seen in Tables IV and V. TLD measurements were normally taken within the first hour of reactor operation however, when gamma dose rate mapping was performed later in the day, once again relative errors fell within the error bars of the experimental procedure.

The irradiation periods used were shorter than the author would have liked. Once again, the saturation dose of the CaF_2:Mn dosimeter dictated the irradiation period. Irradiation times varied from 3 seconds at the inner irradiation site to 30 seconds in the pool. A longer irradiation period would have decreased the inherent error attributable to the timing of the pneumatic irradiation system and the author's in-pool irradiation timings.

The results of the TLD dose rate measurements at the inner and outer irradiation sites can be seen in Table IV. In-pool TLD dose rate measurements can be seen in Table V (see Figure 19 for sample holder positions; elevator position reference critical assembly can be seen in Figure 4). An average TLD reading (in μC) is given for the pairs of TLDs wrapped in tin shrouding. The total dose and dose rate given are following subtraction of the accumulated dose due to the lowering and raising of the samples. All measurements were taken at the reactor mid-height with the reactor operating at half power. The uncertainty attributable to TLD-calculated dose rates is $\pm$ 18% at the inner irradiation site and at the in-pool sites and $\pm$ 17% at the outer irradiation site (see Chapter 5).

TABLE IV

TLD Measurements at the Irradiation Sites of the SLOWPOKE-2

Steady-State Half Power Operation (10 kW_{th})

Date	Position	Irradiation Period ($\pm$ 0.1)	Average TLD Reading (μC)	Total Dose (Rads)	Dose Rate (Grays-h^{-1})
Mon 13 July	Site 4	3 sec	165.0	2.47×10^3	3.0×10^4
Wed 19 Aug	Site 4	3 sec	101.5	2.43×10^3	2.9×10^4
Mon 13 July	Site 10	5 sec	103.9	1.61×10^3	1.2×10^4
Wed 19 Aug	Site 10	5 sec	74.7	1.74×10^3	1.3×10^4

Note: The conversion 1 rad = 0.01 Gy was used to convert a total dose in rads to a dose in Grays. Dose rates above are in air. Dose rate uncertainties for sites #4 and #10 are $\pm$ 18% and $\pm$ 17%, respectively.

TABLE V

TLD Measurements at the In-Pool Irradiation Positions Around the SLOWPOKE-2 Steady-State Half Power Operation (10 kW_{th}) at Reactor Mid-Height

Date	Sample Holder Position	Irradiation Period (± 1.0)	Average TLD Reading (μC)	Total Dose (Rads)	Dose Rate ($Grays\text{-}h^{-1}$) (± 18%)
Mon 10 Aug	Left 1	30 sec	269.8	4.76×10^3	5.7×10^3
Mon 10 Aug	Centre 1	30 sec	313.9	5.75×10^3	6.9×10^3
Fri 14 Aug	Left 1	20 sec	144.1	3.19×10^3	5.7×10^3
Fri 14 Aug	Centre 1	20 sec	169.7	3.57×10^3	6.4×10^3
Tue 04 Aug	Left 2	30 sec	158.7	3.09×10^3	3.7×10^3
Tue 04 Aug	Centre 2	30 sec	206.1	3.53×10^3	4.2×10^3
Tue 04 Aug	Left 3	30 sec	91.1	1.93×10^3	2.3×10^3
Tue 04 Aug	Centre 3	30 sec	112.4	2.03×10^3	2.4×10^3
Fri 14 Aug	Left 3	30 sec	82.7	1.84×10^3	2.2×10^3
Fri 14 Aug	Centre 3	30 sec	97.6	1.97×10^3	2.4×10^3
Wed 22 Jul	Left 4	30 sec	88.2	1.94×10^3	2.3×10^3
Wed 22 Jul	Centre 4	30 sec	81.8	1.32×10^3	1.6×10^3
Wed 19 Aug	Left 4	20 sec	35.5	786	1.4×10^3
Wed 19 Aug	Centre 4	20 sec	45.0	753	1.4×10^3
Wed 22 Jul	Left 5	30 sec	57.4	946	1.1×10^3
Wed 22 Jul	Centre 5	30 sec	64.8	909	1.1×10^3

Note: The conversion 1 rad = 0.01 Gy was used to convert a total dose in rads to a dose rate in Grays. Dose rates above are in air.

As can be seen from Table V, repeat measurements were performed at some of the in-pool positions in order to assess the accuracy/reproducibility of the results. Time restrictions prevented the taking of repeat measurements at all positions. With the exception of the measurement at sample holder position Left 4 on 22 July 98, all pairs of identical measurements appear to fall within the error bars attributable to TLD uncertainty.

A second trial was conducted in order to assess the accuracy/reproducibility of the TLD gamma dose rate results. Another set of TLD measurements were taken with the reactor at one-quarter power ($\phi_{th} = 2.5 \times 10^{11}$ neutrons-cm^{-2}-s^{-1}). The reactor power was able to stabilize following the change in reactor power prior to the lowering of samples into the reactor pool (approximately 15 minutes elapsed after one-quarter power was reached prior to commencement of the experiment). Since the prompt fission gamma contribution is directly related to the average reactor power and the delayed fission product gamma contribution can be approximated by a direct relationship to average reactor power, a similar direct relationship in TLD measurements is thus expected. Of course, for more accurate results, a longer delay period should have been observed between the change in reactor power and the commencement of the experimental procedure, which would have allowed the fission product gamma activity to reach a saturation level. Such a delay was not possible due to scheduling of the SLOWPOKE-2 Facility. The results of the TLD measurements with the reactor at one-quarter power can be seen in Table VI. Once again, the TLDs of interest were shrouded in similar thicknesses of tin and encapsulated in lithium-bromide. All measurements were conducted at the reactor mid-height. The last column shows the percentage of these dose rate measurements compared to those under identical physical conditions at half power. Comparison was made with the measurement taken at a later date in the case of those sample holder positions where more than one measurement was taken. Once again, an approximately 50% measurement at one-quarter full reactor power compared to identical measurements at one-half power full reactor power was anticipated.

TABLE VI

TLD Measurements at the In-Pool Irradiation Positions Around the SLOWPOKE-2 Steady-State One-Quarter Power Operation (5 kW_{th}) at Reactor Mid-Height

Date	Sample Holder Position	Period (± 1.0)	Average TLD Reading (μC)	Total Dose (Rads)	Dose Rate (Grays/h)	Percent of Half Power Dose Rate
14 Aug 98	Left 1	40 sec	173.1	3.35×10^3	3.0×10^3	53%
14 Aug 98	Centre 2	40 sec	106.6	2.06×10^3	1.9×10^3	44%
14 Aug 98	Left 3	40 sec	65.2	1.23×10^3	1.1×10^3	50%
14 Aug 98	Centre 4	40 sec	51.4	843	7.6×10^2	56%

Note: The conversion 1 rad = 0.01 Gy was used to convert a total dose in rads to a dose in Grays. Dose rates above are in air.

As can be seen, given the inherent uncertainty of the TLDs, the measurements at one-quarter reactor power are approximately one-half the results obtained at one-half reactor power. These results also confirm the fact that the response of CaF_2:Mn TLDs is independent of the gamma dose rate as was attested by the manufacturer.

A plot of TLD-measured gamma dose rates as a function of position from the SLOWPOKE-2 core centre-line can be seen in Figure 24. In the case of positions for which more than two TLD measurements have been taken, the 2nd set of readings were arbitrarily chosen.

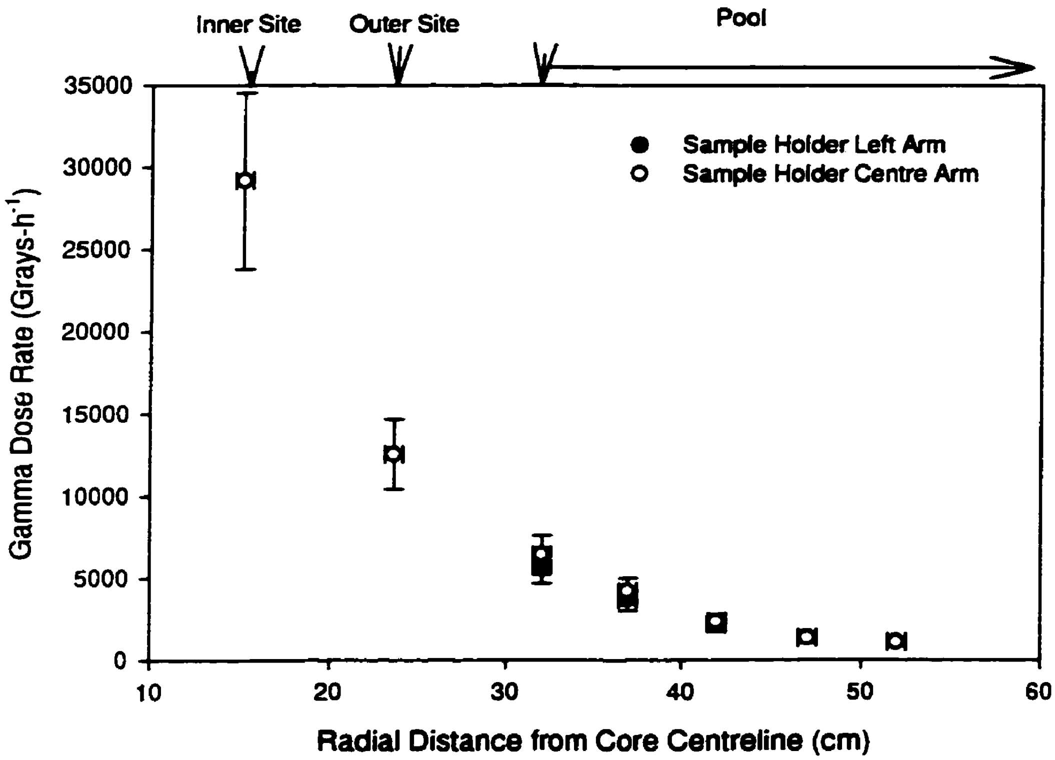

Figure 24: TLD Gamma Dose Rate Measurements Around the SLOWPOKE-2 Steady-State Half Power Operation (10 kW$_{th}$) at Reactor Mid-Height

The effects due to lithium-bromide encapsulation of half of the TLDs during gamma dose rate mapping can be seen in Table VII. It was anticipated that, first of all, all encapsulated TLD dose measurements would be lower than the comparable "un-encapsulated" measurements. This was not always the case. Secondly, it was anticipated that the percentage decrease in dose measurements due to lithium-bromide encapsulation would increase with sample position (i.e., as one progressed further away from the core centre-line). As one progresses further from the centre-line, the neutron flux becomes increasingly "thermalized" resulting in increased scattering of incident neutrons away from the TLDs. Experimentally, quite the opposite was found as the highest percentage decrease in dose response due to lithium-bromide encapsulation occurred at the irradiation sites and in-pool positions close to the reactor container. These results are not easily explained however, the TLD's inherent uncertainty could have been a major factor. It is clear that further study is required into the effects of lithium-bromide encapsulation and radiation particle interaction (i.e., betas, protons, neutrons etc.) on the TLD dose response.

TABLE VII

TLD Dose Response due to Lithium-Bromide Encapsulation

SLOWPOKE-2 Steady-State Half Power Operation (10 kW_{th}) at Reactor Mid-Height

Irradiation Site	**Percentage Difference (%) in Dose Response due to LiBr Encapsulation**
Site 4 (inner irradiation site)	-9.9
Site 10 (outer irradiation site)	-11.0
Sample Holder Position	
Left 1	-16.7
Centre 1	-4.9
Left 2	-5.0
Centre 2	+7.1
Left 3	+1.5
Centre 3	-14.1
Left 4	+3.5
Centre 4	-3.1
Left 5	-6.6
Centre 5	-1.4

Note: In the case of more than one TLD measurement at a particular position, the average percentage change in dose response was used.

4.3 MICROSHIELD VERSION 5 MODEL

In order to attempt to validate the gamma dose rate results from the TLD measurements, MS 5 was used to model the SLOWPOKE-2 reactor container. MS 5 provided gamma dose rates in air at various positions around the modelled core. A sample MS 5 case output can be seen at Annex B. The conditions for using MS 5 are recalled here: in order to construct an appropriate model, all significant gamma source terms were considered (i.e., prompt fission photons, delayed fission product photons, capture gamma sources in the core and shielding, activation gamma sources in the core, inelastic scattering gamma sources in the core). Due to the limitations of the MS 5 software package, certain assumptions were required; for instance, MS 5 required that all gamma source terms originate from the source volume. As a result, (n,γ) sources originating in the shielding material were treated as if they originated from the volume of the core. The author's ability to accurately model the SLOWPOKE-2 reactor container was limited by the modelling capabilities of MS 5 as well. For instance, MS 5 did not permit a shield to be situated directly below the source volume as is the case with the beryllium slab beneath the SLOWPOKE-2 core. Tests were carried out by the author in altering the thickness of the top shield from a 5/8 inch beryllium plate (thickness of Be shim plates following most recent shim, 14 Apr 98) to a 4 inch beryllium plate (thickness of Be bottom shield). It was found that results varied within 5-7% for gamma dose rates at the irradiation sites as well as at positions in the pool. This variance is well within the inherent uncertainty of MS 5 gamma dose rate values of $\pm$ 15% (Grove, 98). The presence of certain elements such as the central cadmium-lined control rod and the irradiation sites could not be accounted for either in the MS 5 model. Due to their small geometries, it was assumed that the omission of these internal structures does not significantly affect the accuracy of gamma dose rates around the core.

4.3.1 Prompt Fission Gamma Rays

The treatment of the prompt fission gamma source term is explained in Section 2.5.1. Using Equations 18 and 19 and integrating across discrete gamma energy levels, a series of terms representing the number of prompt fission photons having an energy equal to E per fission event can be determined. Assuming an average reactor power of 10 kW_{th} (half power), and using Equation 20, a series of prompt gamma production rate terms can then be calculated. The results are found in Table VIII. As previously mentioned, the prompt gamma contribution beyond 7 MeV is considered negligible for the purposes of this analysis. The uncertainty on the prompt gamma production rate terms is $\pm$ 42%. A further explanation of uncertainties is given in Chapter 5.

TABLE VIII

Prompt Fission Gamma Source Terms in the Core of the SLOWPOKE-2

Steady-State Half Power Operation (10 kW_{th})

Average Gamma Energy (MeV)	Gamma Production Rate (γ-s^{-1}) ($\pm$ 42%)
0.65	1.5×10^{15}
1.5	5.0×10^{14}
2.5	1.7×10^{14}
3.5	5.6×10^{13}
4.5	1.9×10^{13}
5.5	6.2×10^{12}
6.5	2.2×10^{12}

4.3.2 Delayed Fission Product Gamma Rays

Similar to the treatment used to calculate the prompt fission gamma source terms, Equation 21 was integrated across discrete energy intervals between 0.1 and 5.0 MeV. Once again, assuming an average reactor power of 10 kW_{th} and using Equation 20, a series of delayed fission product gamma production rate terms can be calculated. The results can be found in Table IX. These gamma production rate values are based on fission product activity saturation which would be achieved had the reactor been operating for a few hours only. The primary contribution during operation is due to short-lived fission products. As discussed in Section 2.5.1.2, the contribution due to long-lived fission product gamma production is assumed negligible in the case of the SLOWPOKE-2 reactor system. The uncertainty on the delayed fission product gamma production rate is $\pm$ 42%, which will be discussed in Chapter 5.

TABLE IX

Delayed Fission Product Gamma Source Terms in the Core of the SLOWPOKE-2 Steady-State Half Power Operation (10 kW_{th})

Average Gamma Energy (MeV)	Gamma Production Rate (γ-s^{-1}) ($\pm$ 42%)
0.25	4.3×10^{14}
0.65	4.6×10^{14}
1.125	2.5×10^{14}
1.575	1.5×10^{14}
2.0	8.4×10^{13}
2.4	5.3×10^{13}
2.7	1.9×10^{13}
2.9	1.6×10^{13}
3.5	4.0×10^{13}
4.5	1.2×10^{13}

4.3.3 Radiative Capture Gamma Rays in the Core

Radiative capture of neutrons in the materials of the core can lead to a significant source of gamma radiation during reactor operations. As a result, it is important to consider this effect in formulating gamma source terms for the MS 5 model. Since the (n,γ) cross-section becomes quite small for most isotopes at neutron energies above 10-20 keV, the assumption is made that only the thermal neutron flux is considered in this analysis. Experiments at other than thermal energies are quite difficult and as a result, little if any data are available. Thermal neutron cross-sections for the (n,γ) reaction are found in references such as BNL-325 (Mughabghab and Garber, 73) and data on the gamma spectra from thermal neutron capture are found in (Blizard and Abbott, 62). The thermal neutron flux distribution $\phi_{th}(r)$ across the core is provided by the

WIMS-AECL output.

A series of radiative capture gamma source terms was calculated using Equation 22. The isotopes present within the volume of the core were each considered, namely, ^{16}O, ^{1}H, ^{238}U, ^{91}Zr, as well as the trace impurities present in the cladding namely, ^{56}Fe, ^{52}Cr, ^{119}Sn, and ^{59}Ni. Radiative capture gamma spectral data are unavailable for ^{235}U. Due to the low atom densities of ^{52}Cr and ^{59}Ni, sample calculations revealed that (n,γ) contributions in these isotopes were negligible. Similarly, radiative capture in the trace isotopes of hydrogen and oxygen (principally, ^{2}H and ^{18}O) was also considered negligible. From the gamma spectral data available, ^{16}O did not appear to be a significant source of capture gammas and is therefore not included in this analysis. The results of (n,γ) capture reaction calculations in the core can be seen in Table X. The uncertainty on the radiative capture gamma source terms is $\pm$ 10%. A further discussion of the uncertainty on the gamma production rate due to radiative capture in the core can be found in Chapter 5. These values correspond to steady-state reactor operation at half power.

As can be seen in Table X, only ^{1}H, ^{238}U and ^{91}Zr contribute significant gamma activities when compared with the contributions due to prompt fission and delayed fission product activities. This series of energy-dependent gamma activities was incorporated into the source array within MS 5.

TABLE X

Radiative Capture Gamma Source Terms in the Core of the SLOWPOKE-2

Steady-State Half Power Operation (10 kW_{th})

Target Isotope	Average Emitted Gamma Energy (MeV)	Gamma Production Rate ($\pm$ 10%)(γ-s^{-1})
^{1}H	2.23	1.2×10^{14}
^{238}U	0.5	2.7×10^{13}
	1.5	1.9×10^{13}
	2.5	9.7×10^{12}
	4.0	3.6×10^{12}
^{91}Zr	4.0	1.9×10^{12}
	6.0	5.8×10^{11}
	8.0	6.7×10^{10}
^{56}Fe	0.5	2.7×10^{10}
	1.5	2.1×10^{10}
	2.5	9.6×10^{9}
	4.0	8.2×10^{9}
	6.0	8.9×10^{9}
	8.0	1.4×10^{10}
	9.5	7.4×10^{8}
^{119}Sn	1.5	1.0×10^{11}
	2.5	4.4×10^{10}
	4.0	9.1×10^{10}
	6.0	2.2×10^{10}
	8.0	2.6×10^{9}
	9.35	2.6×10^{8}

4.3.4 Activation Gamma Sources in the Core

In a typical reactor core, a relatively small percentage of the absorbing nuclei are transmuted into gamma-emitting unstable nuclei. As a result, the induced activities of these compound nuclei are relatively small compared to prompt, delayed or even capture gamma sources. In order to ensure that a complete gamma mapping of the core was conducted, however, all isotopes present in the volume of the SLOWPOKE-2 core were considered for activation activity potentials. It was found that the only significant source of activation gamma rays was due to the $^{16}O(n,p)^{16}N$ reaction in which the ^{16}N radioisotope has a half-life of 7.35 seconds and decays by beta emission accompanied by the emission of 6.13 and 7.10 MeV photons. The energy threshold for this activation reaction to occur is high, approximately 11.0 MeV. Although other isotopes present within the core volume (such as ^{17}O, ^{18}O, ^{2}H, ^{94}Zr, ^{64}Ni and ^{56}Fe among others) will form activated products, these have been determined to be insignificant sources of activation gamma rays due to extremely small activation cross-sections and/or atom densities.

The results of the calculations of the activation gamma sources originating from within the volume of the core can be seen in Table XI. Equation 23 was used and a one hour reactor operating period was assumed, which was fairly typical of actual experimental conditions for TLD measurements. The neutron flux distribution above the activation threshold energy for ^{16}O was approximated using the 1st energy group output of the WIMS-AECL model averaged across the plane of the core. The activation gamma spectral data were obtained from (Rockwell, 56) and (Blizard and Abbott, 62) while activation cross-sections were taken from BNL-325 (Mughabghab and Garber, 73). The uncertainty on the activation gamma production rate is $\pm$ 10% and will be discussed further in Chapter 5. These results are based on steady-state half power operation.

TABLE XI

Activation Gamma Source Terms in the Core of the SLOWPOKE-2

Steady-State Half Power Operation (10 kW_{th})

Activated Isotope	Average Gamma Energy (MeV)	Gamma Production Rate ($\pm$ 10%)(γ-s^{-1})
^{16}O	6.13	1.2×10^{11}
^{16}O	7.10	5.5×10^{11}

4.3.5 Inelastic Scattering Gamma Sources in the Core

Similar to activation gamma sources, the relative contribution to the gamma source terms from inelastic scattering within the core is anticipated to be minimal. The treatment of inelastic scattering phenomena for the present work required that certain assumptions be made. In the case of isotopes for which no gamma spectral data were available, the emitted gamma energy was assumed to be equal to the average excess energy of the neutron flux distribution $\phi(r)$ above the minimum excitation energy of the target isotope. Similar to an activation reaction, a minimum excitation energy exists for an inelastic scattering to occur in a given target isotope. For example, the inelastic scattering threshold energy for ^{235}U is approximately 14 keV and for ^{238}U is approximately 40 keV. For elements of low atomic number, the excitation energy of even the lowest state is large; as discussed in Section 2.5.2, the inelastic scattering threshold energy for ^{16}O is approximately 6.0 MeV and for ^{1}H, inelastic scattering does not occur at all.

Following a detailed assessment of inelastic scattering gamma contributions from the isotopes present within the volume of the core and using Equation 24, it was found that only ^{16}O, ^{91}Zr, ^{56}Fe, ^{235}U and ^{238}U provided significant gamma activity contributions. Many of the trace isotopes present were discounted due to their low inelastic scattering cross-sections σ_{inl}, and/or low atom densities. Limited inelastic scattering gamma spectral data were available (Blizard and Abbott, 62) while σ_{inl} values were taken from BNL-325 (Mughabghab and Garber, 73). The neutron flux distribution above the minimum excitation energy for the target isotope of interest was again approximated across the plane of the core using the WIMS-AECL model output. The results of the inelastic scattering reaction gamma contributions originating from within the core of the SLOWPOKE-2 can be seen in Table XII. Once again, these analytical results assume steady-state reactor operation at half power. The uncertainty on the gamma production rate values is ± 10% and will be discussed further in Chapter 5.

TABLE XII

Inelastic Scattering Gamma Source Terms in the Core of the SLOWPOKE-2 Steady-State Half Power Operation (10 kW_{th})

Target Isotope	Average Emitted Gamma Energy (MeV)	Gamma Production Rate (± 10%)(γ-s^{-1})
^{16}O	6.10	1.1×10^{13}
^{91}Zr	1.61	1.4×10^{12}
^{56}Fe	1.41	2.8×10^{10}
^{235}U	1.00	2.2×10^{12}
^{238}U	1.00	1.1×10^{12}

4.3.6 Radiative Capture Gamma Sources in the Shielding Material

A similar yet more complex treatment than that of the core was required for (n,γ) source terms originating from the shielding materials. As discussed in Section 2.5.2, a volume integral approximation was performed in FORTRAN (Annex C) to determine the radiative capture effects in the surrounding beryllium reflector and water annulus. The beryllium reflector and surrounding water annular region were each divided into 16 equi-volume segments in the azimuthal direction (22.5^0 arcs) and 22 equi-volume segments along the axial direction of the core (1 cm thickness). It is important to note that smaller volume segments would have resulted in a slightly more accurate treatment, however, the fineness of the mesh spacings was decided upon as it provided a reasonably accurate solution within a reasonable amount of time. Radiative capture gamma contributions back to the dose point originating from points beyond the axial dimensions of the core were not included in this analysis. It was assumed that capture gamma rays originating from points far from the reactor mid-plane would have negligible effects at the dose point. Unlike capture gamma rays originating within the volume of the core, an adequate treatment of capture gamma rays within the shielding material required that linear attenuation and buildup factors be included in the calculations to take into account probabilities that an emitted photon at a point in the shielding would reach the dose point.

Capture reactions were considered in all isotopes present within the shielding material. Specifically, the isotopes ^{9}Be and ^{56}Fe in the beryllium reflector and ^{1}H in the water annulus were used as target isotopes. The gamma contribution due to the trace impurity ^{12}C in the beryllium reflector was assumed insignificant due its relative low σ_c^{th} ($\approx$4.5 mb). Once again, small quantities of materials present within the volume of the shielding materials (i.e., the aluminum shrouding of the irradiation sites) were considered insignificant sources of (n,γ) gamma rays for this analysis. Capture gamma contributions from the container wall and beyond in the pool surrounding the reactor container were discounted. Effects due to capture reactions far from the dose point were assumed insignificant.

Similar to the treatment in the core, Equation 22 was used to calculate the gamma source terms at various positions of origin within the shielding material. Once again, (n,γ) reactions are only assumed significant within the thermal range of neutron energies. As a result, the WIMS-AECL-generated thermal neutron flux distribution $\phi^{th}(r)$ across the various thicknesses of shielding material was used. WIMS-AECL performed a 2-dimensional neutron transport calculation only; therefore, a calibration factor was required for volume segments above or below the mid-plane of the reactor core. These data were provided by (El Hajjaji et al., 98) at École Polytechnique de Montréal on a SLOWPOKE-2 reactor where NA measurements using Cu and Al-Au wires provided a detailed axial thermal neutron flux distribution across the core. A detailed knowledge of the axial thermal neutron flux distribution within the pool was provided by Andrews (Andrews, 89). Using these two sets of data, an axial $\phi^{th}(r)$ distribution across the shielding materials was approximated. Once the gamma source terms were calculated, Equations 25,27 and 28 were then used to determine the energy-dependent gamma flux at the dose point. The buildup factors were calculated for each shielding material for each volume segment based on MS 5-generated Taylor coefficients, geometrically-determined distances and using Equation 26.

The production rates of photons emitted by radiative capture in the shielding materials can be seen in Table XIII. As previously discussed, MS 5 treats all gamma source terms as originating from within the core volume. As a result, although these calculations provided an energy-dependent gamma flux at a dose point only (in the case of this work, at the position of the inner irradiation site), this flux was then applied over the entire surface area of the core cylinder in order to be consistent with the MS 5 treatment. This approach overestimates the radiative capture gamma activity at points along the surface of the core above and below the mid-plane. This treatment is considered sufficiently accurate for the purposes of this work given that the overall gamma contribution due to radiative capture in the shielding material is relatively small and given that the MS 5 model treats the core as a homogenized mixture of its constituent elements (i.e., U, Zr, H, O, in addition to the trace elements). The capture gamma terms have been summed for different target isotopes with identical gamma emission energies. The uncertainty on the radiative capture in the shielding gamma production rate terms is $\pm$ 10% and will be discussed further in Chapter 5. Once again, these values are based on steady-state reactor operation at half power.

TABLE XIII

Radiative Capture Gamma Source Terms in the Shielding Material of the SLOWPOKE-2 Steady-State Half Power Operation (10 kW_{th})

Average Emitted Gamma Energy (MeV)	Gamma Production Rate ($\pm$ 10%)(γ-s^{-1})
0.5	1.2×10^{12}
1.5	3.5×10^{11}
2.0	5.2×10^{10}
2.23	4.4×10^{13}
4.0	4.2×10^{12}
6.0	6.2×10^{12}
8.5	2.3×10^{11}

4.3.7 Activation Gamma Sources in the Shielding Materials

The activation gamma source calculations originating from the volume of the core demonstrate that a relatively small percentage of the induced activities around a core is due to activation sources when compared with prompt and delayed fission gamma rays or even capture reactions. In this work, the activation gamma sources within the core represent approximately 0.1% of the total gamma activity (within the core and due to radiative capture in the shielding, see Annex B). While there is an activation gamma contribution due to the trace impurities in the beryllium reflector as well as due to aluminum and the trace elements in the reactor container wall, these effects are considered negligible. The only significant source of activation gammas is due to the $^{16}O(n,p)^{16}N$ reaction. Rough calculations reveal that activation gamma contributions within the shielding due to this reaction are negligible for the purposes of this work (because of the lower fast neutron flux in the shielding materials and the modest cross-section).

4.3.8 Inelastic Scattering Gamma Sources in the Shielding Materials

Similar to the activation gamma contribution, it was assumed that inelastic scattering would not produce significant gamma activity within the shielding materials. Light nuclei such as beryllium and hydrogen and oxygen have higher excitation threshold energies and lower inelastic scattering cross-sections. For instance, due to the relatively high excitation energy for beryllium (~2 MeV), one would expect the gamma contribution due to inelastic scattering to be significantly lower than that of radiative capture in the beryllium reflector. As previously discussed in Section 2.5.2, beryllium has a relatively low inelastic scattering cross-section of 14 mb at approximately 2.5 MeV. As a means of comparison, the inelastic scattering gamma contributions in the core represent approximately 1.5% of the total gamma production rate. Due to the tendency toward lower atomic number elements in the shielding, it is expected that the inelastic scattering contribution is significantly lower. Rough calculations performed by the author revealed this to be the case and inelastic scattering gamma contributions in the shielding were assumed negligible for the purposes of this work.

4.3.9 Gamma Flux Contribution due to Backscattered Photons

MS 5 does not account for backscatter of photons from materials beyond the dose point. As a result, in order to provide as accurate a model as possible, this effect must be accounted for. As discussed in Section 2.6, a volume integral approximation was required in which an array of MS 5-calculated incident gamma flux values at incremental positions within this volume were determined. The integration mesh intervals were discussed in Section 2.6. The fineness of the mesh intervals was decided upon as it provided a sufficiently accurate integral calculation within an acceptable time frame. As it turned out, each of the 49 dose points within each of the 5 planar regions (in the φ direction) required an MS 5 computation time of between 15 and 20 minutes. This turned into a labourious exercise when one considers that this treatment had to be repeated for each dose point around the SLOWPOKE-2 core. FORTRAN programs (Annex D) were written to calculate the backscattered gamma flux at a dose point, which could then be easily given as a relative contribution to the incident flux.

Volume integral calculations were performed for the inner and outer irradiation sites as well as at 10 cm increments in the radial direction in the pool. Equation 32 was numerically approximated to resolve the backscatter contribution at the inner site while Equation 31 was used for all other dose points. This treatment provided a photon backscatter mapping such that interpolations could be performed at various positions around the SLOWPOKE-2 core. In accordance with the rest of this experimental work, all backscatter contributions were calculated at the reactor mid-height. Due to the lack of axial symmetry in the SLOWPOKE-2 reactor (i.e., a 10.16 cm Be plate below, and a 1.59 cm Be shim above), it was necessary to consider this effect as MS 5 does not permit the specification of a lower shield. It was found that, by substituting the lower beryllium plate in place of the MS 5-specified top shield, reductions in the gamma flux values of 1-18% were observed. The highest percentage reductions were found when considering array points at large values of r, and θ (i.e., far from the dose point). In general, gamma flux reductions at points within the volume integral closer to the reactor mid-height fell within a 2-3% range, which is acceptable for the purposes of this work. It is important to note that the larger deviations occur at points which provide the smallest contribution to the backscatter volume integral calculations as they are farthest from the dose point.

4.3.10 Energy of Scattered Photons

As discussed in Section 2.6.1, Equation 34, representing the average energy of the recoil electron due to Compton scattering, was solved numerically using Simpson's Rule. Using the MS 5-generated average incident photon energies at each of the array points and the results of Equation 34, the average energy of the scattered photon could easily be deduced. From this average energy, linear attenuation coefficients in the material of interest could be easily calculated. The calculated results of the gamma backscatter contribution as a function of position from the core centre-line can be seen in Table XIV and in Figure 25. As can be seen, the relative contribution varies from approximately 6.4% at the inner irradiation site to 15.8% at a position in the pool 52 cm from the core centre-line. These results compare reasonably well with previous work carried out by Bonin (Bonin, 96), where using a best-fit curve approximation for the gamma flux distribution, a constant 9.2% relative backscatter contribution was calculated for all dose points around the core. The uncertainty on the backscattered photon flux values in this work is ± 27%, while the ratio of backscattered to incident gamma contributions has an uncertainty of ± 69%. The calculations of uncertainties will be discussed in Chapter 5. All backscatter contributions given are with respect to a dose point at the reactor mid-height.

TABLE XIV

Photon Backscatter Contribution at Dose Points Around the SLOWPOKE-2 Steady-State Half Power Operation (10 kW_{th}) at Reactor Mid-Height

Distance from Core Centre-line (cm)	Ratio of Photon Backscatter Contribution to Incident Flux (± 69%)
15.2 (inner irradiation site)	6.4%
23.6 (outer irradiation site)	10.2%
32.0	13.6%
42.0	15.5%
52.0	15.8%

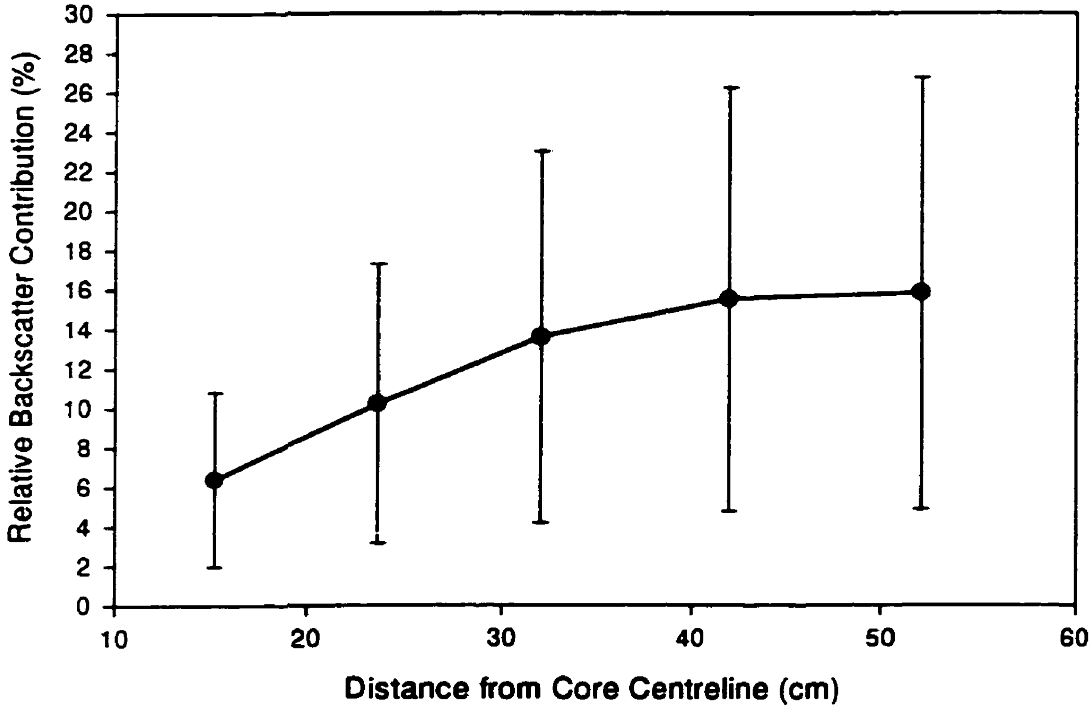

Figure 25: Photon Backscatter Relative Contribution
Steady-State Half Power Operation (10 kW_{th}) at Reactor Mid-Height

4.3.11 Gamma Dose Rates

With all significant gamma source terms accounted for, the MS 5 code was used to calculate gamma dose rates at positions around the core. All calculated dose rates are given at the reactor mid-height and with the reactor at half power. The uncertainty on the gamma dose rates is $\pm$ 58%. The calculated gamma dose rates represent dose in air. The total gamma dose rate includes the backscatter contribution. The dose due to backscatter was calculated by considering the average energies of the incident and scattered photons at each of the dose points.

A dose rate ratio $D_{Scattered\ Photon\ Energy}/D_{Incident\ Photon\ Energy}$ was then calculated which could be applied to the relative backscatter at each position around the core. MS 5 provided a "Dose-to-Flux" conversion table which was used to produce this ratio. Dose rate ratios of 0.46 at the irradiation sites (i.e., in air) and 0.44 for the in-pool positions (i.e., in water) were calculated. These ratios were multiplied by the relative backscatter contributions and the MS 5-calculated direct gamma dose rates to produce backscattered photon dose rate contributions. The MS 5-calculated gamma dose rates at various positions around the SLOWPOKE-2 core can be seen in Table XV and in Figure 26.

TABLE XV

Calculated Gamma Dose Rates in Air Around the SLOWPOKE-2 Steady-State Half Power Operation (10 kW_{th})

Distance from Core Centre-line (cm)	MS 5 Calculated Gamma Dose Rate ($\times 10^3$ Gy-h^{-1})	Backscattered Gamma Dose Rate ($\times 10^3$ Gy-h^{-1})	Total Gamma Dose Rate ($\times 10^3$ Gy-h^{-1}) ($\pm$ 58%)
15.2 (inner irrad site)	19.54	0.57	20
23.6 (outer irrad site)	6.07	0.29	6.4
32.0	2.45	0.15	2.6
37.0	1.55	0.099	1.7
42.0	1.00	0.068	1.1
47.0	0.66	0.045	0.7
52.0	0.45	0.031	0.5

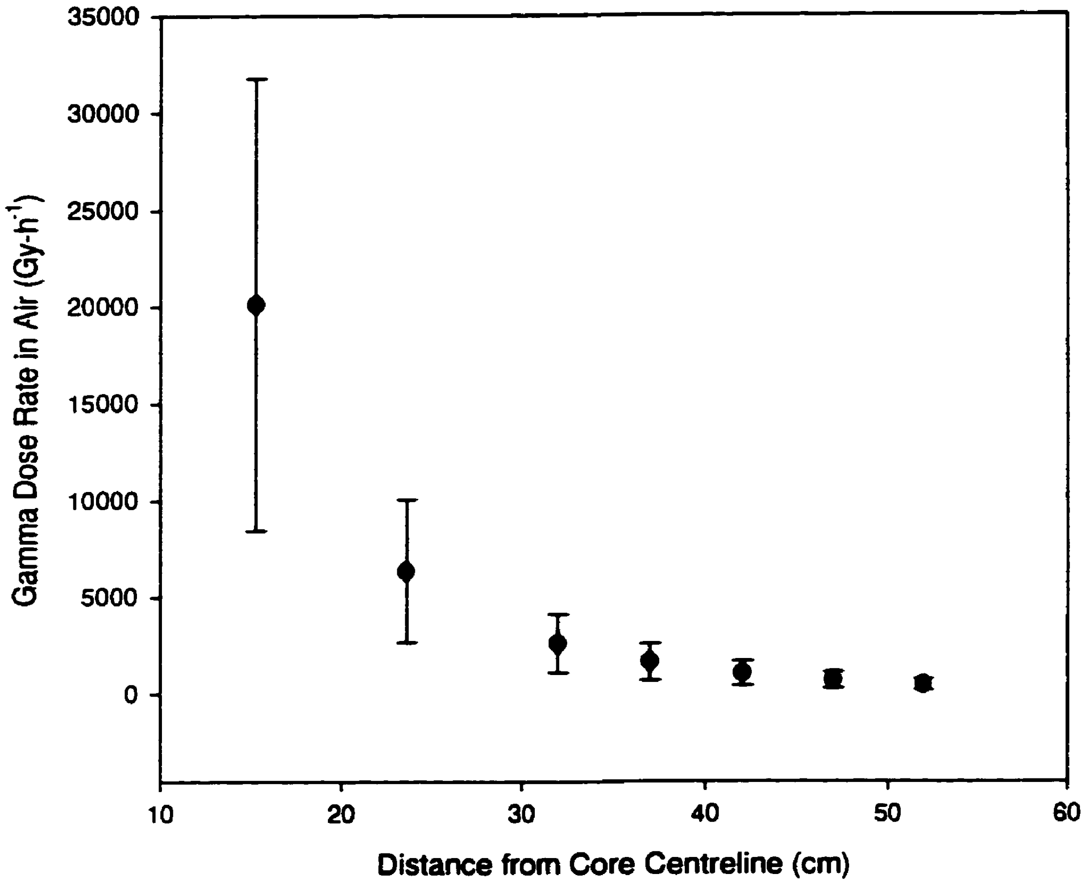

Figure 26: MS 5 Total Gamma Dose Rates Around the SLOWPOKE-2 Steady-State Half Power Operation (10 kW$_{th}$) at Reactor Mid-Height

Previous analytical work on gamma dose rates around the SLOWPOKE-2 core was conducted by Bonin (Bonin, 96) at full reactor power (20 kW$_{th}$) and considering only the prompt and short-lived fission product gamma contributions. These results are 24-46% less than (Bonin, 96). This comparison appears to be consistent with anticipated results as one would expect the gamma dose rate to decrease somewhat linearly with reactor power. In the case of this experimental work, it would appear that the addition of secondary gamma sources such as (n,γ) reactions in the core and shielding, activation gamma sources in the core, and inelastic scattering gamma sources in the core to the gamma source contribution increased the overall gamma dose rate by 10-25%. The percentage difference between these analytical results and those of Bonin increased in the pool as the distance from the core centre-line increased. With the most significant secondary gamma sources originating from positions close to the core, it was not surprising that this trend was observed.

When comparison between MS 5 analytical and TLD experimental gamma dose rates was made, it was quickly realized that significant discrepancies exist between the two. The higher TLD-measured dose rate trend resulted in percentage differences of 31-60% between the experimental and analytical values. Obviously, for the purposes of this work, it would have been preferred to have had improved correlation between experimental and analytical response characteristics. It is suspected that the "true" gamma dose rate curve lies somewhere between the experimental and analytical curves. A plot of MS 5-generated gamma dose rates versus TLD experimental gamma dose rates can be seen in Figures 27 and 28.

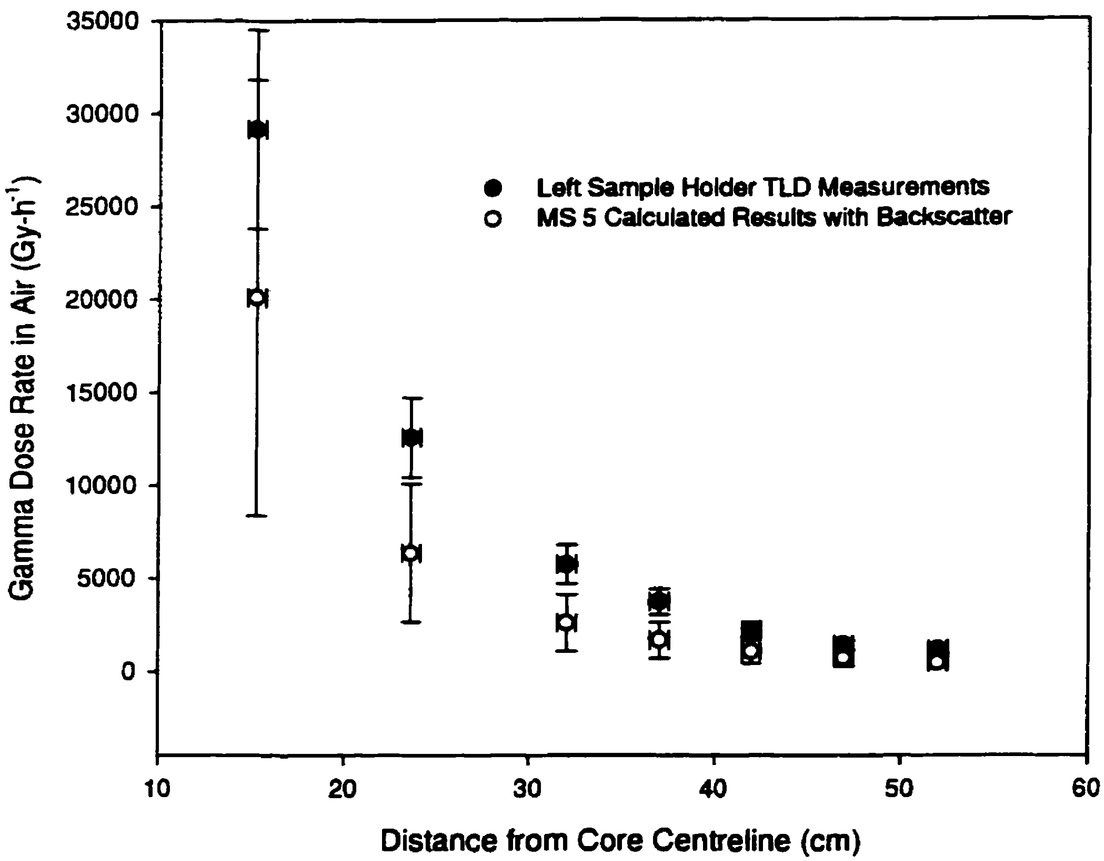

Figure 27: TLD vs MS 5 Gamma Dose Rate Results Around the SLOWPOKE-2 Left Sample Holder, Steady-State Half Power Operation (10 kW_{th}) at Reactor Mid-Height

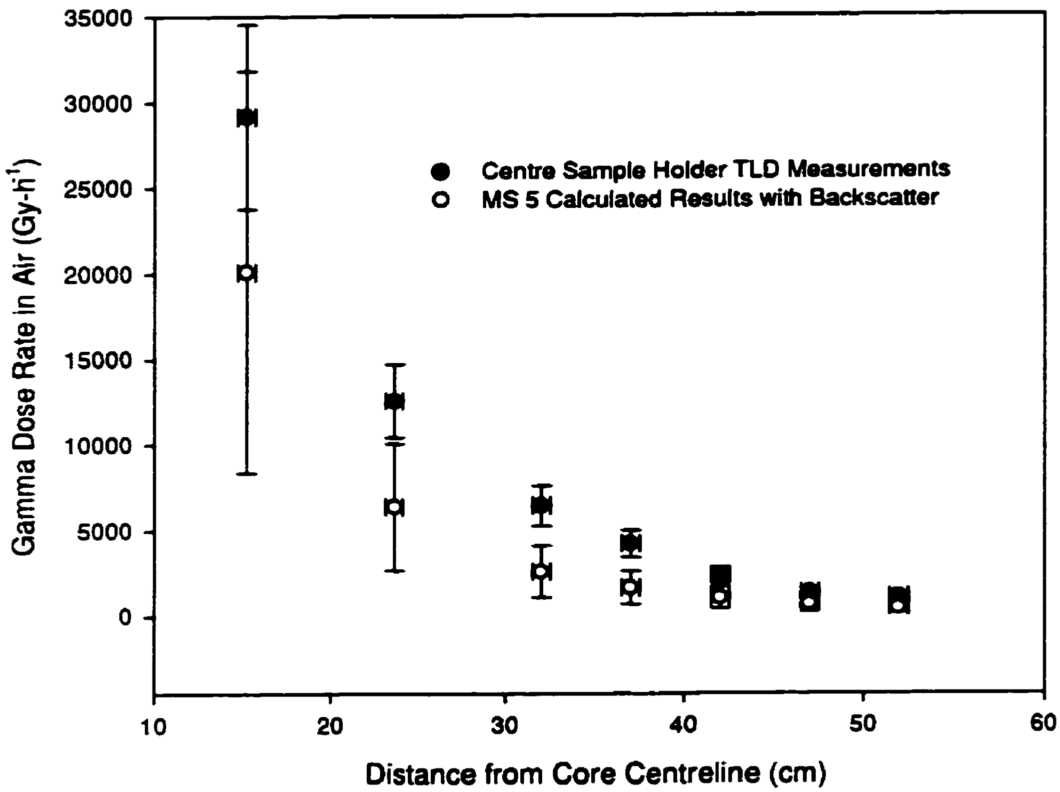

Figure 28: TLD vs MS 5 Gamma Dose Rate Results Around the SLOWPOKE-2 Centre Sample Holder, Steady-State Half Power Operation (10 kW_{th}) at Reactor Mid-Height

The percentage variance between experimental TLD and analytical MS 5 gamma dose rates can be seen in Table XVI. In general, the discrepancy between the experimental and analytical dose rate values is greater than the attributable uncertainties. It is suspected that the TLD over-response at low-energies resulted in an over-estimation of the gamma dose rate. As previously mentioned, the TLDs were calibrated using Cs-137 at one photon energy only (662

TABLE XVI

Percentage Variance Between TLD and MS 5 Gamma Dose Rates

Sample Position	Percentage Variance (%)
Inner Irradiation Site	31.0
Outer Irradiation Site	50.8
Left 1	54.4
Centre 1	59.4
Left 2	54.1
Centre 2	59.5
Left 3	50.0
Centre 3	54.2
Left 4	50.0
Centre 4	50.0
Left 5	54.5
Centre 5	54.5

keV) where the TLD's relative response is 1. As can be seen in Figure 13, from photon energies of 0.05 to 0.15 MeV, the filtered TLDs displayed an over-response characteristic. This over-response characteristic may be somewhat compensated for by the under-response at photon energies less than 0.05 MeV. In order to assess the net effect of tin shielding on the TLD dose response characteristics, an accurate gamma ray spectrum at various irradiation positions is required. By combining the results of Figure 13 (relative TL response of shielded TLDs as a function of photon energy) with the actual photon energy spectrum around the reactor, an analysis of the net effect of over and under-dampening effects could be resolved. Due to the author's inability to accurately predict the photon energy spectrum around the SLOWPOKE-2 core, such a precise exercise is not presently possible. The use of the MS 5 source input values (i.e., prompt fission, delayed fission product photons, etc.) would not benefit this work as all are of energies greater than 150 keV, at which point the relative TL response is 1.

The possibility exists that the TLD response could be partially due to other than gamma radiation. In an attempt to account for secondary electrons produced when the incident photons interact with the atoms of the surrounding materials (i.e., tin shrouding), an "electron equilibrium" experiment was conducted. This involved shielding the tin-shrouded CaF_2:Mn TLDs with Plexiglas. The thickness of the Plexiglas required to approximate electron equilibrium is equal to the range of the maximum energy secondary electrons generated by the primary photons in this material (in this case, approximately 0.67 cm). TLDs were irradiated using a ^{60}Co source at DREO with and without the Plexiglas shielding. Irradiations doses of 1, 5 and 10 rads were used to assess the effect of Plexiglas shielding on the TLD response. The results of the electron equilibrium experiment can be seen in Table XVII. The average ratio of TLD response with and without Plexiglas shielding is 1.01. These results indicate that no change was observed in the dose responses due to the presence of Plexiglas shielding around the TLDs. This would appear to indicate that secondary electrons produced by incident photons within the materials surrounding the TLDs have a negligible effect on the absorbed dose results.

TABLE XVII

Electron Equilibrium Results for CaF_2:Mn TLDs Using a ^{60}Co Source

TLD	Total Dose (Rads)	TLD Output With Plexiglas (nC) A	TLD Output Without Plexiglas (nC) B	Ratio of B/A	Average Ratio
1	1.0	193.7	183.1	0.95	
2	1.0	54.3	58.46	1.08	
3	1.0	93.2	94.53	1.01	
4	1.0	140.9	152.2	1.08	
5	5.0	600.8	576.7	0.96	
6	5.0	578.5	575.5	0.99	
7	5.0	439.2	448.5	1.02	
8	5.0	420.1	419.0	1.00	
9	10.0	1282.0	1264.0	0.99	
10	10.0	889.9	874.5	0.98	
11	10.0	943.2	1057.0	1.12	
12	10.0	872.9	844.1	0.97	**1.01**

Note: nC is a nano-Coulomb (i.e., 10^{-9} C). The above ratios are rounded off to two decimal places.

Due to the thickness of lithium-bromide and tin surrounding the TLDs, it was assumed that heavy charged particle interaction is very unlikely, even at the high energy end. Following sample calculations, it was found that stripped electrons (at rest mass energy) would not be able to reach the surface of the TLDs through the tin shielding either. Certain neutron-induced reactions could result in a TLD response. For example, in the following (n,β) reaction, 90% of decays result in an energetic beta emission of maximum energy 13.0 MeV. Such a decay could very likely result in a TLD response due to the presence of high-energy beta particles.

$$^7_3Li + ^1_0n \rightarrow ^8_3Li \xrightarrow{\beta} ^8_4Be$$

The TLDs could well have been affected by x-rays present due, in large part, to the bremsstrahlung effect with charged particles.

The error attributable to the short irradiation times must also be considered. In particular, the inner and outer irradiation sites with 3 and 5 second irradiation periods were likely sources of additional uncertainty in gamma dose rate measurements due to low counting statistics. When TLD measurements were repeated under identical operational conditions, the percentage variance observed was approximately 2% at the inner site and 8% at the outer site. Given the inherent uncertainty in the TLD response across a broad gamma spectrum of $\pm$ 10%, this degree of variance was assumed reasonably accurate. In-pool irradiation, although measured over longer irradiation periods, had the inherent inaccuracy of a hand-timed measurement. Repeated in-pool TLD measurements displayed variances from 1-15%. Once again, this degree of variance was not considered unreasonable.

Due primarily to the uncertainty in the prompt and delayed fission gamma source equations, the overall error attributable to the MS 5-calculated gamma dose rate distribution is significantly higher than that of the TLD measurements. It is obvious that all potential sources of gamma rays cannot possibly be accounted for in the model. It was assumed that unaccounted-for secondary gamma sources would be almost negligible compared to prompt, delayed fission and capture contributions, however, these effects would lead to an under-estimation of the gamma dose rate distribution.

MS 5 buildup calculations between the source volume and the dose point resulted in significant increases in the dose rate distribution. In particular, gamma dose rate results with buildup were observed to be 150-450% greater than results without buildup. Such a significant buildup effect is not surprising given the relatively large thicknesses of the shielding materials, and the large densities and average atomic numbers of the SLOWPOKE-2 core. It is important to note that the MS 5-generated buildup factors for mixed-element materials are approximations only. As a result, MS 5 buildup approximations could lead to significant errors in gamma dose rate results.

MS 5 results also depended on the user's choice of reference buildup material. It is from the choice of reference buildup material (i.e., one of the shields) that the code is told from which table to retrieve and interpolate buildup factors. The most conservative approach was to select the shield that produced the highest number of mean free paths, which is a measure of the attenuation within the shield. In the case of this experimental work, the beryllium reflector was chosen as reference buildup material for all experimental results. Trials using other shields, such as the equivalent container outer wall-to-dose point thickness in water, resulted in gamma dose rate results 10-32% lower.

As previously mentioned, MS 5 required that all gamma source terms originate from the source volume (i.e., core). As was determined from the (n,γ) calculations in the shielding material, this is not in fact the case with the SLOWPOKE-2. Significant gamma source contributions originate both in the material of the beryllium reflector as well as in the annular light water regions. MS 5 treatment of these secondary gamma sources led to an under-prediction of their effects on the dose rate at all positions of interest around the core. Sample calculations revealed that the overall contribution due to this effect was 1-2% of the total gamma dose rate at dose positions in the pool.

A final point on the accuracy of the MS 5-generated gamma dose rates involves the treatment of the SLOWPOKE-2 core as a homogeneous source volume. Following communications with the developers of MS 5 (Worku, 99), it was felt that the source distribution would be critical to the accuracy of the model. In the case of the SLOWPOKE-2 core, the actual non-concentric positioning of the fuel pins would have an effect on the resultant gamma fluence rates. The developer believed this to be a significant source of error in the present model. A transport code such as MCNP-4A would be required to resolve these effects.

4.4 PROTON DOSE CALCULATIONS

The mathematical treatment for the calculation of proton doses at incremental dose points around the SLOWPOKE-2 core was similar to that of photon backscattering. Unlike photons, however, light charged particles such as protons have finite ranges and as a result, the potential contribution must only be considered in a small sphere surrounding the dose point. This treatment was discussed in detail in Section 2.7. Equation 36 was solved numerically using a volume integral approximation code in FORTRAN for each of the 7 dose points (2 irradiation sites plus 5 in-pool sites). The results of these calculations were 11 energy group proton flux distributions at the in-pool sites and only a single energy group proton flux value at the inner and outer irradiation sites. This variance was due to the higher binding energies of the protons in the air molecules compared to the water molecule.

In order to calculate the resultant dose rate from the energy-dependent proton flux distributions, it was necessary to make certain assumptions. It was assumed that the incident proton transfers all of its energy to the target nuclides within a very short distance. This assumption was considered reasonable except for the case of high energy protons in air. As the proton intensity distribution was in terms of flux, a volume target of radius equal to the range of the average recoil proton in the material of interest was assumed. It was also assumed that the energy of the protons was deposited isotropically and completely within the target volume. The proton dose rate was calculated in air for the irradiation sites and in water for the in-pool positions. The results of calculated proton dose rates at positions around the SLOWPOKE-2 core can be seen in Table XVIII. A plot of the proton dose rate as a function of distance from the core centre-line for in-pool irradiation positions can be seen in Figure 29. The uncertainty on the proton dose rate results is $\pm$ 10%, as will be discussed in detail in Chapter 5. These calculations are based on steady-state half power operation (10 kW_{th}) with the dose points along the reactor mid-height.

TABLE XVIII

Proton Dose Rate Calculations Around the SLOWPOKE-2

Steady-State Half Power Operation (10 kW_{th}) at Reactor Mid-Height

Distance from Core Centre-line (cm)	**Total Proton Flux ($p\text{-}cm^{-2}\text{-}s^{-1}$)**	**Proton Dose Rate ($Gy\text{-}h^{-1}$) (± 10%)**
15.2 (inner irradiation site)	1.2×10^4	14.3*
23.6 (outer irradiation site)	3.8×10^3	2.6*
32.0	2.1×10^8	1.0×10^3 †
37.0	5.4×10^7	3.2×10^2 †
42.0	2.3×10^7	1.5×10^2 †
47.0	1.2×10^7	8.7×10^1 †
50.3	1.1×10^7	9.2×10^1 †

Note: * Dose in air

† Dose in water

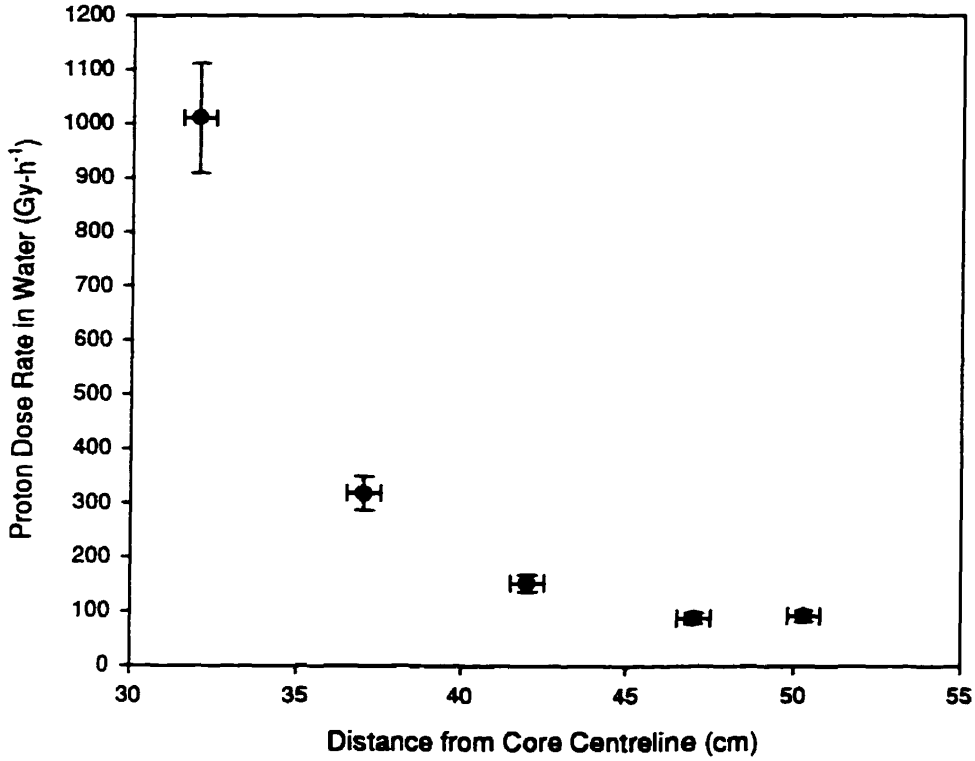

Figure 29: Calculated Proton Dose Rates Around the SLOWPOKE-2 Steady-State Half Power Operation (10 kW_{th}) at Reactor Mid-Height

4.5 ELECTRON DOSE CALCULATIONS

Although the production of electrons around the dose positions of interest typically comes from four different sources, only the electrons produced due to proton stripping were considered for this thesis. The rationale for this assumption is discussed in Section 2.8. A volume integral calculation was performed at each dose point by solving Equation 37 numerically using a FORTRAN code. The volume integral resulted in an electron flux at the dose points of interest. The assumption was made that all the energy of the stripped electrons was deposited in a small volume around the dose point. Since the dose due to electrons is not deposited within a finite range in materials (as a result of the bremsstrahlung effect), this assumption is obviously inconsistent with the physical realities. The result is an over-prediction of the electron dose effect. A target volume of radius equal to the range of an average energy stripped electron was used for conversion of electron flux to a dose rate (in $Gy\text{-}h^{-1}$). It was assumed that the energy was deposited isotropically and completely within this target volume. Electron dose rates are given in air for the irradiation sites and in water for the in-pool positions. For simplicity of electron dose calculations, all stripped electrons were assumed to possess their rest mass energies only (~0.511 MeV). Once again, electron dose rates are based on the SLOWPOKE-2 reactor operating at steady-state at half power (10 kW_{th}). All calculations are performed at the reactor mid-height. Electron dose rate results around the SLOWPOKE-2 core can be seen in Table XIX and Figure 30. The uncertainty on the electron dose rate results is $\pm$ 18% at the irradiation sites and $\pm$ 26% at the in-pool sites, which will be discussed in detail in Chapter 5.

TABLE XIX

Electron Dose Rate Calculations Around the SLOWPOKE-2 Steady-State Half Power Operation (10 kW_{th}) at Reactor Mid-Height

Distance from Core Centre-line (cm)	Total Electron Flux (electrons-cm^{-2}-s^{-1})	Electron Dose Rate (Gy-h^{-1})
15.2 (Inner Irradiation Site)	6.8×10^7	1.7×10^4*
23.6 (Outer Irradiation Site)	3.9×10^7	5.4×10^3*
32.0	1.7×10^{10}	3.2×10^4†
37.0	4.2×10^9	8.2×10^3†
42.0	1.8×10^9	3.5×10^3†
47.0	9.5×10^8	1.8×10^3†
50.3	8.9×10^8	1.7×10^3†

Note: * Dose in air (± 18%)

† Dose in water (± 26%)

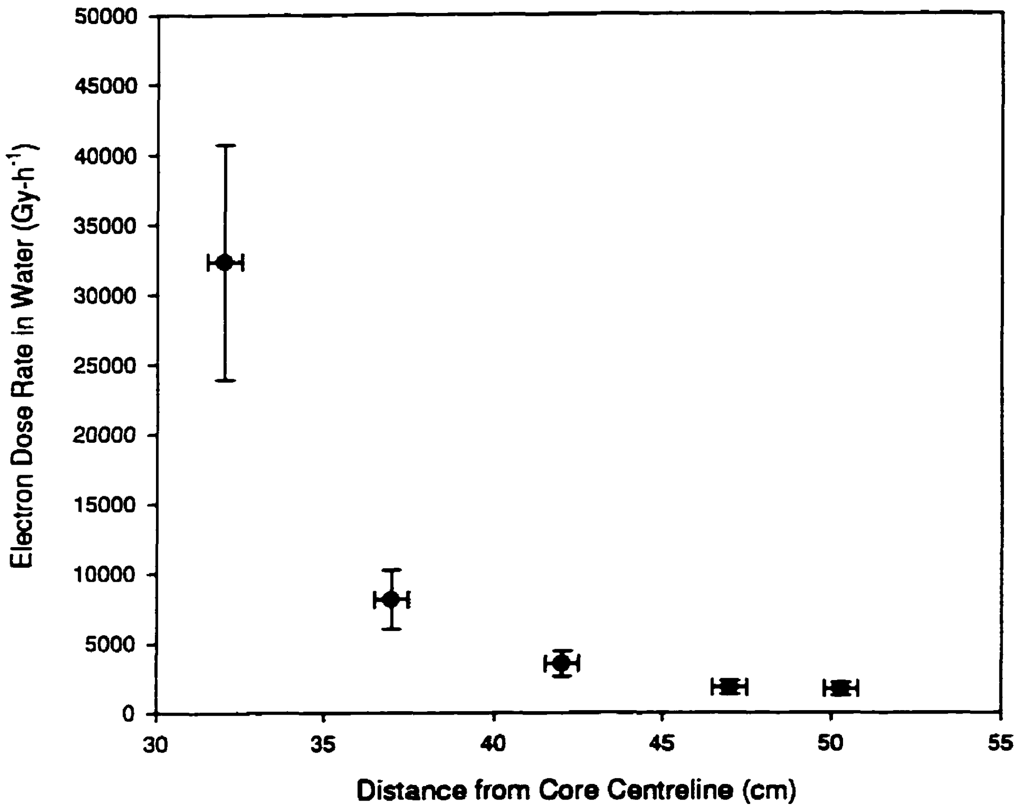

Figure 30: Calculated Electron Dose Rates Around the SLOWPOKE-2 Steady-State Half Power Operation (10 kW_{th}) at Reactor Mid-Height

4.6 TOTAL DOSE RATES

Since the absorbed radiation dose rate (in $Gy\text{-}h^{-1}$) applies to all types of radiation in all absorbing media, a total dose rate can be easily calculated at the irradiation sites and positions in the pool simply by summing the individual particle dose rate contributions. The total dose rates are the summation of neutron, gamma, proton and electron dose rates at positions around the core of the SLOWPOKE-2 reactor core. The total dose rate results can be seen in Table XX and Figure 31. Total dose rate values at the irradiation sites are in air while those given at in-pool positions are in water. The MS 5-calculated in-pool gamma dose rates had to be converted to an equivalent dose in water in order that terms could be summed. MS 5 provided an energy fluence rate (in MeV-cm^{-2}-s^{-1}) and an average gamma energy $\overline{E}_{\gamma}$, at each dose point. Knowing the average gamma energy at the dose point, a mass absorption coefficient in water $(\mu_a/\rho)_{H2O}$ could be extrapolated from tables (Lamarsh, 83). The gamma dose rate in water was then simply calculated using the following equation:

$$\dot{D}_{\gamma} = \phi_{Et}\left(\frac{\mu_a}{\rho}\right)_{H2O}$$

where $\phi_E t$ is the energy fluence rate at the dose point of interest.

Total dose rates are based on steady-state reactor operation at half power (10 kW_{th}) and all measurements are at the reactor mid-height. Total dose rate uncertainties vary from $\pm$ 28% to $\pm$ 40% and will be discussed further in Chapter 5.

TABLE XX

Calculated Total Dose Rates Around the SLOWPOKE-2

Steady-State Half Power Operation (10 kW_{th}) at Reactor Mid-Height

Distance from Core Centre-line (cm)	**Total Dose Rate ($Gy\text{-}h^{-1}$)($\pm$28% to $\pm$40%)**
15.2 (inner irrad site)	3.7×10^{4}*
23.6 (outer irrad site)	1.2×10^{4}*
32.0	3.7×10^{4}†
37.0	1.0×10^{4}†
42.0	4.9×10^{3}†
47.0	2.7×10^{3}†
50.3	2.4×10^{3}†

Note: * Dose in air

† Dose in water

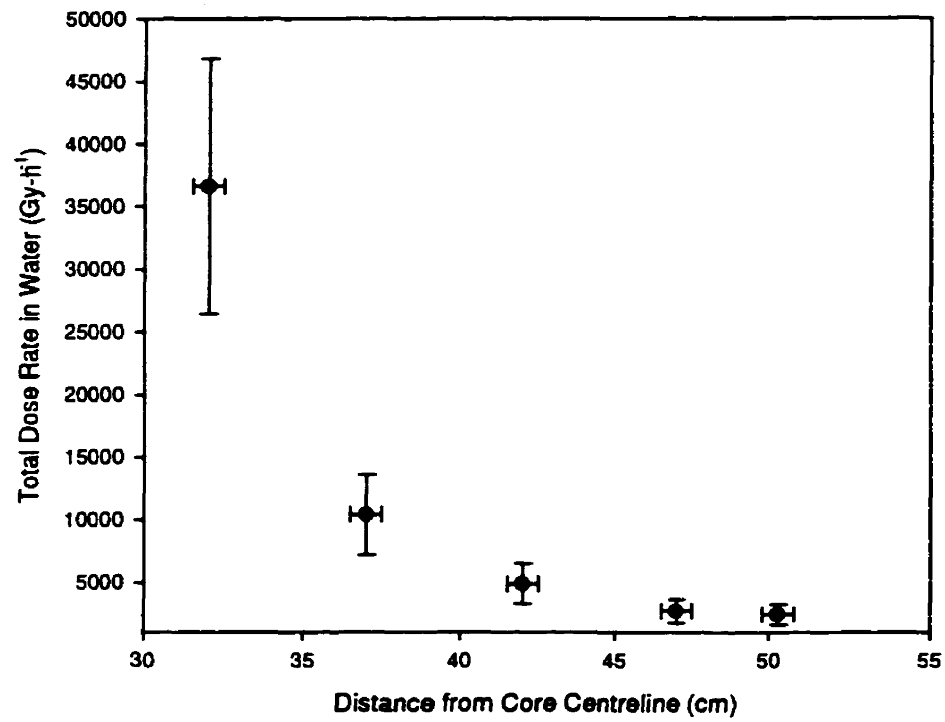

Figure 31: Calculated Total Dose Rates Around the SLOWPOKE-2 Steady-State Half Power Operation (10 kW_{th}) at Reactor Mid-Height

The relative contribution of each component of the total dose rate as a function of distance from the core centre-line can be seen in Figure 32. The electron and MS 5 gamma dose rates are the most significant contributors to the total dose rates. The surprisingly high dose rate contribution due to electrons is believed to be an over-prediction of the actual effect, as discussed in Section 4.5. As discussed in Section 4.3.11, it is believed that the MS 5 gamma dose rate results under-predict the actual gamma effect.

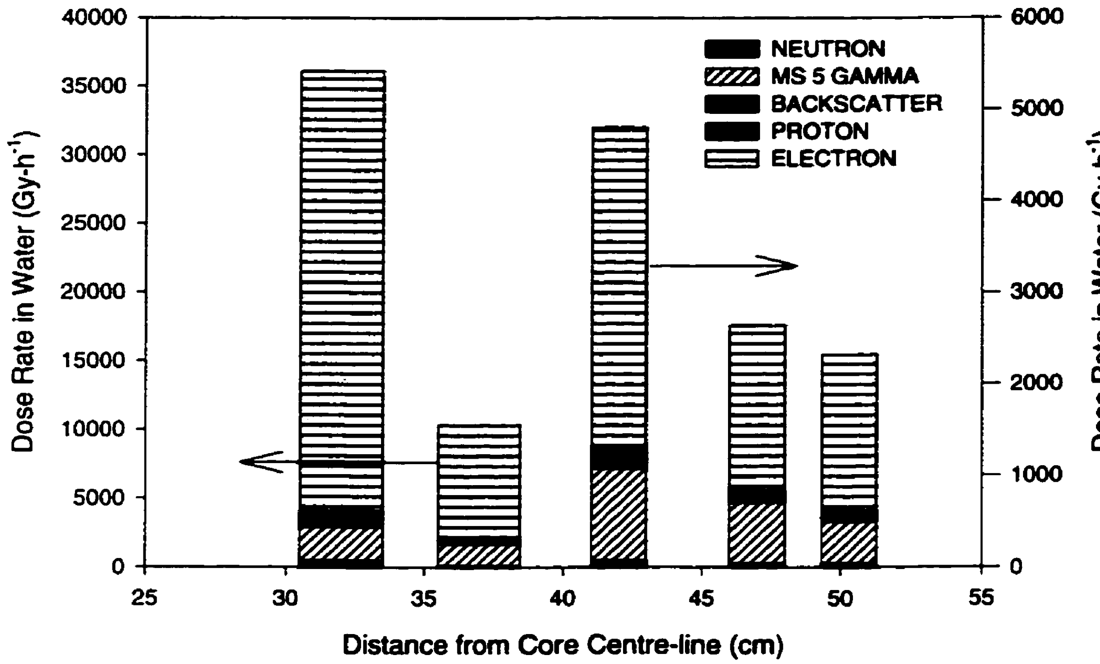

Figure 32: Contributions to Total Dose Rate Distribution Steady-State Half Power Operation (10 kW$_{th}$) at Reactor Mid-Height

CHAPTER 5 - ERROR ANALYSIS

5.1 NEUTRON DOSE RATE

The uncertainty in the neutron dose rate calculations can be traced to two basic sources:

1. errors intrinsic in WIMS-AECL code calculations, and
2. errors due to energy group and annular region approximations.

AECL claims an uncertainty of $\pm$ 5% on WIMS-AECL-generated neutron flux distributions (Edwards, 98). Due to the extensive databases available for absorption and scattering cross-sections, the uncertainty on these WIMS-AECL values is believed to be less than $\pm$ 1%. For the purposes of this thesis, an overall uncertainty on the WIMS-AECL-generated data was assumed to be $\pm$ 5%. In order to calculate neutron dose rates across a 26 energy group distribution, average neutron energies across each group were assumed, which led to certain statistical errors which were difficult to quantify. The neutron energy probability function likely varies across each of the 26 groups. An uncertainty of $\pm$ 5% on the average neutron energy value was considered reasonable for the purposes of this work. The accuracy of the neutron dose rates due to the entire spectrum of neutrons is calculated as:

$$\frac{\Delta \dot{D}_n}{\dot{D}_n} = \frac{\Delta \phi_n}{\phi_n} + \frac{\Delta \overline{E}}{\overline{E}}$$

This represents an uncertainty of $\pm$ 10% on neutron dose rate calculations. As discussed in Section 4.1, errors resulting from assumptions made in creating the WIMS-AECL model will also contribute to the uncertainty of these results. As no model is able to fully duplicate the characteristics of a physical system, the variance between theoretical and actual dose rate distributions may indeed be much greater than the predicted uncertainty.

The uncertainty on the radial distance of the dose points from the core centre-line was also difficult to quantify. WIMS-AECL produced energy-dependent neutron flux distributions across annular regions outward from the core. For the purposes of this work, the calculated neutron flux was seen as corresponding to the centre of the annulus. In reality, the neutron flux distribution across each annular region would not follow a linear curve. As a result, an uncertainty of $\pm$ 0.5 cm on the irradiation position was determined to be reasonable for all radial positions.

5.2 TLD GAMMA DOSE RATE

When considering physical measurements, error propagation generally falls into 3 different categories: random, systematic, and personal errors. Random errors are those which occur unpredictably between successive sets of measurements. Systematic errors are those which are intrinsic to the procedure, equipment, and material used. Personal errors involve any and all human oversights such as poor data collection, and flawed procedural techniques. In the case of TLD measurements around the SLOWPOKE-2 core, attention was paid to minimizing systematic and personal errors. This was done by repeating measurements under nearly identical physical conditions and observing the variance of results. When results fell within a relatively narrow band of probabilities, it was assumed that only uncontrollable random errors were at play. Based on personal communications with DREO personnel (Cousins, 98/99) and literature from the manufacturer (Harshaw-Bicron), it was assumed that an uncertainty of $\pm$ 10% could be assumed on the response characteristics of a typical TLD system across a wide gamma spectrum. As discussed in Section 4.3.11, a certain degree of low-energy gamma under-dampening was anticipated with the tin shrouding due to the very dramatic over-response characteristics of the CaF_2:Mn TLDs. It was assumed that an additional $\pm$ 5% uncertainty on the TLD response across the entire gamma spectrum was reasonable. The cumulative uncertainty on TLD response due to the properties of the dosimeters was therefore $\pm$ 15%.

The dose rate calculations depended a great deal on the accuracy of the irradiation periods. Due to the inherent accuracy in an automated delivery system, an uncertainty on the irradiation periods at the inner and outer irradiation sites of $\pm$ 0.1 second was assumed conservative. The in-pool irradiations being hand-timed and manually set in place, an uncertainty of $\pm$ 1.0 second seemed much more reasonable. As a result, the overall uncertainty of the TLD gamma dose rate measurements was a summation of dosimeter uncertainties and irradiation period uncertainty using the following equation:

$$\frac{\Delta \dot{D}_{\gamma}}{\dot{D}_{\gamma}} = \frac{\Delta \dot{D}osimeter_{\gamma}}{\dot{D}osimeter_{\gamma}} + \frac{\Delta t_{irrad}}{t_{irrad}}$$

Using the irradiation times of 3 seconds and 5 seconds for the inner and outer irradiation sites respectively, and 30 seconds as an average in-pool irradiation time, an uncertainty of $\pm$ 3.4% is calculated due to timing errors for the inner irradiation sites and in-pool positions, while an uncertainty of $\pm$ 2.0% is calculated due to timing errors at the outer irradiation site. As a result, using the above equation, an overall dose rate uncertainty of $\pm$ 18% is found for the inner irradiation site and in-pool positions, and $\pm$ 17% for the outer irradiation site.

5.3 MICROSHIELD 5 DOSE RATE

The uncertainty on the MS 5-calculated dose rates is a complex function of gamma source term approximations, inherent MS 5 calculation approximations and model inconsistencies. As will be discussed, the functions describing gamma source term contributions proved to have the most significant effect on MS 5 dose rate uncertainties. For most applications, Grove Engineering predicted a relative error of 10-15% on calculated exposure rates to be very good (Grove, 98). For more complex geometries in which certain approximations had to be made (such as was the case with the SLOWPOKE-2 reactor container), the uncertainty could very well be greater. As was discussed in Section 4.3.11, MS 5-generated buildup factors no doubt led to significant errors during volume integral calculations. Transport codes such as MCNP-4A are generally required when high atomic number elements are contained in the MS 5 geometry (as is the case in the SLOWPOKE-2 core).

The prompt fission gamma and delayed fission product gamma spectrum functions (Equations 18, 19, and 21) introduce the largest uncertainty to the MS 5 gamma dose rate calculations. An uncertainty of $\pm$ 40% is assumed across the gamma energy range 1.0 to 7.0 MeV. Due to the fact that this range encompasses the majority of gamma source terms and in order to be conservative, this uncertainty is assumed across the entire gamma spectrum. The activity due to the prompt and delayed fission gamma rays was calculated using Equation 20. The uncertainty in the average reactor power $\overline{P}$ can be assumed to be $\pm$ 2% based on (Duke, 96). A total uncertainty of $\pm$ 42% is assumed on the prompt and delayed fission gamma source term activities.

Secondary gamma sources such as radiative capture, activation, and inelastic scattering in the core and shielding materials have significant uncertainties attached to them. Once again, the uncertainty in the WIMS-AECL neutron flux distribution $\phi(E,r)$ will affect the accuracy of all (n,γ) reaction events. The uncertainty attributable to (n,γ) gamma emission spectra $f_i(E')$ is unclear, however, $\pm$ 2-3% would seem reasonable given the inherent accuracy of the radiative capture cross sections $\sigma_{c,i}$. Similar relatively small uncertainties can be assumed for the cross sections of activation and inelastic scattering phenomena. For conservatism and simplicity, the uncertainty of radiative capture, activation and inelastic scattering gamma production rates within the core is assumed to be $\pm$ 10%. The volume integral approximation used in this work to calculate the total radiative capture contributions within the shielding would inevitably result in an uncertainty. This uncertainty is difficult to quantify without having calculated successive volume integrals using incrementally smaller mesh intervals. Due to the excessive work and time required for a single hemispherical (n,γ) shielding calculation, this just was not an option. An uncertainty of $\pm$ 10% would not be unreasonable in this case.

For the sake of simplicity and conservatism, an uncertainty of $\pm$ 42% (as previously discussed) was applied to all MS 5 gamma activity terms. The prompt and delayed fission product gamma rays accounted for approximately 80-85% of the total gamma activity; as a result, the uncertainties attributed to these terms would outweigh all other relative uncertainties. The MS 5 gamma dose rate result uncertainties are a function of the gamma source term and code inaccuracies. If an uncertainty of $\pm$ 15% is assumed on the volume integral calculations performed by MS 5, then the cumulative gamma dose rate uncertainty would be 42% + 15% = 57%.

Uncertainties due to gamma backscatter contributions must also be considered. The backscattered gamma contribution as a function of incident gamma flux at incremental radial positions outward from the core centre-line was presented in Section 4.3.10. An average backscatter contribution of 11.1% from the inner irradiation site to the 52.0 cm position in the pool was simply calculated. In solving the triple integral numerically, only the direct gamma fluxes were used which did not account for the effect of backscattered photons at the positions within the integral. As a result, the incident gamma flux terms are themselves some 11.1% too small. Although no effect would be seen on the percentage contribution from backscattered photons, the total gamma dose at the dose point would be approximately 11.1% too small. The accuracy of the triple integral numerical approximation would lead to an uncertainty which is difficult to quantify without successive calculations using steps of smaller mesh intervals. When the number of planes in the φ-direction was increased from 3 to 5, the resultant variance in the relative backscatter contribution was only 0.1%. As a result, an uncertainty due to the triple integral approximation of 1.0% would appear conservative in this case.

Once again, the relative uncertainty of ± 15% on all MS 5 calculations must be considered as it was with this code that the volume integral was constructed. As a result, an uncertainty of 15% + 11.1% + 1% = 27.1% would appear reasonable on backscattered gamma flux values. The calculation of the uncertainty on the total gamma flux at a dose point is as follows:

$$\Delta\phi_{total} = \Delta\phi_{direct} + \Delta\phi_{backscattered}$$

$$= 57\%\ \phi_{direct} + (27.1\%)(11.1\%)\phi_{direct}$$

$$= 60.0\%\phi_{direct}$$

$$\frac{\Delta\phi_{total}}{\phi_{total}} = 60.0\%\left(\frac{\phi_{direct}}{\phi_{direct} + \phi_{backscattered}}\right)$$

$$= 60.0\%\left(\frac{\phi_{direct}}{111.1\%\ \phi_{direct}}\right) = 54.0\%$$

The error on the total gamma dose is calculated as follows:

$$\Delta\dot{D}_{total} = \Delta\dot{D}_{direct} + \Delta\dot{D}_{backscattered}$$

$$= 57\%\ \dot{D}_{direct} + 84.1\%(11.1\%\ x 45\%)\dot{D}_{direct}$$
$$= 61.2\%\ \dot{D}_{direct}$$

where 84.1% relative error on the backscattered dose is the summation of 57.0% error on the direct flux and 27.1% error on the backscattered flux. The 45% represents the average dose-to-flux conversion factor as discussed in Section 4.3.11.

The relative uncertainty on the MS 5 calculated total gamma dose rate is therefore:

$$\frac{\Delta\dot{D}_{total}}{\dot{D}_{total}} = 61.2\%\left(\frac{\dot{D}_{direct}}{\dot{D}_{total}}\right) = 61.2\%\left(\frac{\dot{D}_{direct}}{\left(11.1\%\ x 45\%\ x\dot{D}_{direct}\right) + \dot{D}_{direct}}\right) = 58.3\%$$

A simplified representation of the above uncertainty calculation is as follows:

$$\frac{\Delta\dot{D}_{tot}}{\dot{D}_{tot}} = \frac{\Delta R}{R} + \frac{\Delta\phi_{ttot}}{\phi_{tot}}$$

where R is the flux-to-dose conversion factor.

Knowing that $\frac{\Delta \phi_{tot}}{\phi_{tot}} = 54.0\%$, we can assume an uncertainty on the flux-to-dose conversion of approximately 4.0%. As a result, $\frac{\Delta \dot{D}_{tot}}{\dot{D}_{tot}} \cong 58.0\%$.

5.4 PROTON DOSE RATE

As the treatment for the calculation of proton dose rates is similar to that of gamma backscatter, the determination of uncertainties will also be similar. Once again, WIMS-AECL $\phi(E,r)$ values are accurate to within $\pm$ 5%. Atomic binding energies and macroscopic scattering cross-sections $\Sigma_s(E)$ have been extensively documented and are believed accurate to within $\pm$ 1.0%. Other potential errors in the treatment of the proton dose rate are due to the triple integral numerical approximation and geometrical assumptions around the dose point (see Section 4.4). Due to the small size of the hemisphere of interest around the dose point and the limited effect of slight $\phi(E,r)$ inconsistencies within such a volume, relative errors of $\pm$ 1% and $\pm$ 2% can be assumed for the latter two sources of error. The cumulative effect of all potential errors on the proton dose rate distribution is calculated as $\pm$ 10%.

5.5 ELECTRON DOSE RATE

The dose rate due to electrons calculated in this thesis considered only electrons stripped by protons. Photoelectric, Compton and pair production effects were accounted for in the mass absorption term μ_a/ρ. As a result, the uncertainty calculation is simplified. The electron dose rate uncertainty is a function of the specific ionization of the materials of interest. Specific ionization was calculated using the stopping power (-dE/dx) and the range of the protons in the material. Extensive research has gone into the formulation of these SRIM tables and as a result, the relative uncertainty on these values is assumed negligible. Some error is introduced through the calculation of the average energy of the proton fluxes which are used in the calculation of specific ionization. An overall uncertainty of ± 2.0% is deemed reasonable.

The greatest degree of error is introduced through the calculation of the triple integral once again. Errors must be considered for the proton flux terms, the stripped electron linear attenuation coefficients, and the overall accuracy of the volume integral calculation. The equation describing the linear attenuation coefficient for electrons is an approximation only, and as a result, an uncertainty of ± 5.0% would appear reasonable. When these variances are applied to the exponential term in the FORTRAN code which calculates the volume integral, an uncertainty of ± 1.0% is observed for the irradiation sites and ± 13.0% for the in-pool irradiation positions.

The proton flux uncertainty as previously calculated is ± 10.0%. When this variance is applied to the triple integral approximation code, an uncertainty of ± 10.0% results for the electron flux at all dose points.

The numerical representation of the triple integral leads to different uncertainties at the irradiation sites as compared to the in-pool positions. As stated in Section 4.5, an under-prediction of the axial contribution of the electron current density is made for purposes of model simplicity. As a result, an uncertainty of ± 5.0% on the electron dose rates at the inner and outer irradiation positions appears reasonable. The triple integral calculations in the pool do not suffer from this approximation error; as a result, a dose rate uncertainty of ± 1.0% was believed acceptable.

The cumulative uncertainty on the electron dose rates is therefore 2+1+10+5%=18% at the irradiation sites and 2+13+10+1%=26% at the in-pool positions.

5.6 TOTAL DOSE RATE

The total dose rate distribution is simply a summation of the dose rates due to neutrons, gamma rays, protons, and electrons. The analytical MS 5-generated gamma dose rate distribution was used in the calculation of the total dose rates. The uncertainty on the total dose rate values is the summation of the absolute uncertainties of each of the particle dose rates. This relationship can be expressed as follows:

$$\dot{D}_T = \dot{D}_n + \dot{D}_\gamma + \dot{D}_p + \dot{D}_e$$

$$\delta \dot{D}_T \cong \delta \dot{D}_n + \delta \dot{D}_\gamma + \delta \dot{D}_p + \delta \dot{D}_e$$

Therefore, the relative uncertainty on the total dose rate is simply $\frac{\delta \dot{D}_T}{\dot{D}_T}$. The resultant relative uncertainty on the total dose rates was 40% at the inner and outer irradiation sites and varied from 28% to 34% at the in-pool irradiation positions.

CHAPTER 6 - CONCLUSIONS AND RECOMMENDATIONS

6.1 CONCLUSIONS

The aim of this work was to produce, both experimentally and analytically, improved dose rate distributions around the core of the SLOWPOKE-2 reactor at RMC, which was accomplished. Although improvements to the present model could and will most certainly be made in the future, the present data represent a significant improvement in the quality and quantity of the knowledge of the dose rates in the SLOWPOKE-2 reactor container and pool.

The ability of WIMS-AECL to model the SLOWPOKE-2 research reactor was confirmed. Although a 26 energy group regional neutron flux output was assumed more than sufficient for the present work, newer versions of the WIMS-AECL code coupled with improved hardware could provide improved neutron energy group resolution. The WIMS-AECL-generated $\phi(E,r)$, $\Sigma_s(E)$, and $\Sigma_a(E)$ data at the dose points of interest facilitated accurate neutron dose rate predictions.

It was determined that, although the CaF_2:Mn TLDs proved that estimations of high dose rate, composite radiation fields are possible, improvements in field characterization are required. The TLDs provided what is believed to be an over-estimation of the gamma dose contribution due to interactions from other particles within the radiation field. This provided an upper limit of gamma dose rate distributions around the SLOWPOKE-2 reactor core.

The analytical model based on the MICROSHIELD Version 5 simulation provided gamma dose rates independent of the complexity of the radiation field. Shortcomings of this analytical technique include the author's inability to account for all possible sources of photons within the volume of the reactor container and beyond. The model constructed attempted to account for all primary and the significant secondary sources of photons. It is believed that the present model under-predicted secondary gamma source term contributions at the dose points. The MS 5 simulation was also limited by its inability to treat secondary gamma contributions as originating from the target material instead of the source volume. It is believed that the overall effect is an under-prediction of the gamma dose rate distribution by MS 5.

Recoil proton dose rate distributions are assumed reasonably accurate for the scope of this work. The stripped electron dose rates are believed to be over-predictions of the actual effect due to the author's inability to account for secondary electromagnetic radiation such as bremsstrahlung. The accuracy of these volume integral approximations was limited also by the regional WIMS-AECL neutron flux resolution. Once again, future versions of WIMS-AECL with greater memory capabilities coupled with an improved hardware suite will allow for a more accurate prediction of not only neutron dose rates but also secondary (n,p) and stripped electron flux distributions.

This work should be viewed as a step forward in the pursuit of a definitive radiation particle dose rate mapping of the SLOWPOKE-2 reactor. It has provided a baseline for future experimental and analytical research into particle dose rate distributions and a reasonably accurate mapping for present irradiation effects work.

6.2 RECOMMENDATIONS

There are a number of areas that should be studied in order to improve the accuracy of dose rate mappings around the SLOWPOKE-2 core. First of all, although the WIMS-AECL two-dimensional neutron transport calculations, coupled with experimental dosimetry measurements across the core and pool in the axial direction, allowed the author to approximate complete three-dimensional energy-dependent neutron flux distributions, the use of a three-dimensional transport code would improve the accuracy of the neutron flux distributions. A code such as MCNP-4A would allow the entire reactor to be modelled in three dimensions; however, the determination of absolute energy-dependent neutron flux distributions across the entire reactor container and into the pool requires excessive computation time. The effects due to the central control rod, the upper beryllium shims, and lower beryllium plate in the neutron flux distribution could be properly ascertained. Ultimately, such a three-dimensional model would result in improved neutron dose rate calculations.

More research is required in the response characteristics of TLDs in a composite radiation field. It is believed that neutrons and high-energy beta particles contributed to the over-response of the TLDs. The use of a calibration source such as Pu-Be with lithium-bromide and tin-shielded CaF_2:Mn dosimeters would provide researchers with much needed information on the neutron response characteristics of a filtered TLD. A similar exercise could be undertaken with a high-energy beta emitter. Unfortunately, this was beyond the scope of the present work. With this data in hand, an improved estimation of the gamma dose contribution around the SLOWPOKE-2 reactor could be made.

An alternative to the approach suggested in the previous paragraph is to investigate the use of another dosimeter for gamma dose rate mapping in a complex radiation field. One of the initial objectives of this work was to determine the most appropriate dosimeter for the present application. Many factors, as discussed in Section 4.2, resulted in the selection of the CaF_2:Mn TLD. This is not to say that this particular dosimeter does not possess certain undesirable response characteristics that hinder its ability to accurately measure gamma doses in a complex field. Investigations into the suitability of other dosimeters for use in complex radiation fields should continue with the aim of improving experimental particle dose rate distribution estimations. Ideally, such a dosimeter should possess a high dose saturation limit, a constant response with accumulated dose, a high relative dose response, a linear response with dose rate, and the ability to shield its response to secondary particle interactions.

The time restrictions on this work precluded more detailed experimental gamma dose rate mapping in the reactor pool. All experimental measurements were taken at the reactor mid-height since this was the most useful plane in terms of high dose radiation effects research. An improved understanding of the gamma dose rate distribution in the SLOWPOKE-2 pool could be attained by irradiating dosimeters at incremental positions above and below the reactor mid-height. At this point, the variation of gamma fluence rates with position in the axial direction is not absolutely known.

The limitations of MS 5 were discussed in Section 4.3.11, including its inability to account for gamma backscatter from points beyond the dose point, as well as its inability to account for buildup accurately (particularly in high atomic number materials). The use of an improved modelling tool for the SLOWPOKE-2 would greatly improve the quality of gamma dose rate estimations. Due to the geometrical limitations of MS 5 case input, structures such as the central control rod, upper and lower beryllium shims/plates, and irradiation sites could not be accounted for. The effects of these model inconsistencies on the gamma fluence rate could very well be significant.

REFERENCES

Andrews, W.S., Thermal Neutron Flux Mapping Around the Reactor Core of the SLOWPOKE-2 at RMC, Royal Military College of Canada/Collège militaire royal du Canada, 1989.

Angelone, M. et al., Gamma and Neutron Dosimetry Using CaF_2:Tm Thermoluminescent Dosimeters for Fusion Reactor Shielding Experiments, Nuclear Science and Engineering: **126**, 176-186, 1997.

ASTM Document, Standard Practice for the Application of Thermoluminescence-Dosimetry (TLD) Systems for Determining Absorbed Dose in Radiation-Hardness Testing of Electronic Devices, E668-78, 1984.

Barry, B.A., Errors in Practical Measurement in Science, Engineering, and Technology, Wiley & Sons, 1978.

Benoist, P., Steaming Effects and Collision Probabilities in Lattices, Nuclear Science and Engineering: **34**, 285-307, 1968.

Blizard, E.P. and Abbott, L.S., Reactor Handbook Vol III Part B (Shielding), Interscience, 1962.

Bonin, H.W., Dose Estimation in the SLOWPOKE-2 Nuclear Reactor Pool and Irradiation Sites, Royal Military College of Canada/Collège militaire royal du Canada, 1998.

Bonin, H.W., Estimation of the Doses from Thermal and Fast Neutron Flux and Gamma Radiation Exposure in the Pool of the SLOWPOKE-2 Nuclear Reactor, RMC-CCE-SL2-96-2, Royal Military College of Canada/Collège militaire royal du Canada, May 1996.

Bonin, H.W., Procedure for Using the Sample Elevator Rig in the Pool of the SLOWPOKE-2 Reactor, RMC-CCE-SL2-96-3, Royal Military College of Canada/Collège militaire royal du Canada, 1996.

Bonin, H.W., Health Physics and Radiation Protection, RMC-CCE-HWB-90-1, Royal Military College of Canada/Collège militaire royal du Canada, 1990.

Burden, R.L. and Faires, J.D., Numerical Analysis, (6th ed.), Brooks/Cole, 1997.

Chang, J.Y., Correlation Study of Energetic Ion Induced Optical Absorption and Thermoluminescence in Crystalline LiF and CaF_2, IEEE Transactions on Nuclear Science, Vol. NS-30, No. 6, 1983.

Choppin, G.R. and Rydberg, J., Nuclear Chemistry Theory and Applications, Pergamon, 1980.

Cousins, T., Personal Communications, Defence Research Establishment Ottawa, April 1998 - January 1999.

Cuomo, G.A. et al., SRIM (The Stopping and Range of Ions in Matter), IBM Corp., 1998.

deWit, R.T., Reactivity Calculations for the LEU Fuelled SLOWPOKE-2 Reactor at RMC, Royal Military College of Canada/Collège militaire royal du Canada, 1989.

Dorschel, B. et al., The Physics of Radiation Protection, Nuclear Technology, Ashford, Kent, 1996.

Duchesne, J.C.D., RMC SLOWPOKE-2 Reactor Thermal Power Determination by Core Water Temperature Measurements, Royal Military College of Canada/Collège militaire royal du Canada, 1994.

Duke, M.J.M., SLOWPOKE-2 Neutron Flux Stability at 2×10^{11} n-cm^{-2}-s^{-1}, University of Alberta, 1996.

Edwards, G., Personal Communications, Atomic Energy of Canada Limited (Chalk River Nuclear Laboratories), August - December 1998.

El Hajjaji, O. et al., Flux Measurements in the LEU Refuelled SLOWPOKE-2 Reactor of École Polytechnique de Montréal, Institut de Génie Nucléaire, École Polytechnique de Montréal, 1998.

Evans, R.D., The Atomic Nucleus, McGraw-Hill, 1955.

Fernandez, J.E. et al., Polarization Effects on Multiple Scattering Gamma Transport, Radiat. Phys. Chem., Vol 41, No 4/5, pp 579-630, 1993.

Friedlander, G. et al., Nuclear and Radiochemistry (3rd ed.), Wiley & Sons, 1981.

Garber, D.I. and Kinsey, R.R., Neutron Cross Sections Volume II, Curves, (3rd ed.), BNL-325, Brookhaven National Laboratory, Upton, New York, 1976.

Gesell, T.F., Environmental Monitoring with Thermoluminescence Dosimetry, IEEE Transactions on Nuclear Science, Vol NS-29, No. 3, 1982.

Glasstone, S. and Sesonske, A., Nuclear Reactor Engineering (4th ed.), Chapman & Hall, 1994.

Gomaa, M.A. et al., Reactor Dosimetry using TLD and Rh Activation Techniques, Atomkernenergie (ATKE) Bd.29, Lfg.3, 1977.

Griffiths, J., WIMS-AECL User's Manual, Atomic Energy of Canada Ltd (Chalk River Nuclear Laboratories), 1994.

Grogan, D. et al., Some Problems Associated with Large Scale TLD Monitoring and their Solutions, Nuclear Instruments and Methods, **175**, 166-168, 1980.

Grove Engineering, Microshield Version 5 User's Manual, Rockville Maryland, 1998.

Harshaw TLD, Thermoluminescent Dosimetry Fact Sheet, Solon Ohio, 1998.

Hayward, E. and Hubbell, J.H., The Albedo of Various Materials for 1-MeV Photons, National Bureau of Standards Report, 1953.

Hornbeck, R.W., Numerical Methods, Quantum, New York, 1975.

Hubbell, J.H. et al., Atomic Form Factors, Incoherent Scattering Functions, and Photon Scattering Cross Sections, Journal of Physical Chemistry, Vol. 4, No. 3, 1975.

Jamil, K., Incore Gamma Dosimetry Using Thermoluminescence Detectors (TLDs) in a Research Reactor, The Nucleus, **32** (1,2), 1995:13-16.

Knoll, G.F., Radiation Detection and Measurement (2^{nd} ed.), Wiley & Sons, 1989.

Lamarsh, J.R., Introduction to Nuclear Engineering (2^{nd} ed.), Addison-Wesley, 1983.

McKeever, S.W.S., Thermoluminescence of Solids, Cambridge University, 1985.

Mughabghab, S.F. and Garber, D.I., Neutron Cross Sections Volume 1, Resonance Parameters, (3^{rd} ed.), BNL-325, Brookhaven National Laboratory, Upton, New York, 1973.

Mukherjee, B. et al., Determination of the Decay Characteristics of Residual Gamma Dose in the Core of a TRIGA Mk-II Reactor by Lithium Fluoride Thermoluminescence Dosimeter Powder, Nuclear Instruments and Methods in Physics Research, **A256**, 610-611, 1987.

National Nuclear Data Centre (NNDC), Brookhaven National Laboratory, Upton New York.

Nielsen, K., Personal Communications, Royal Military College of Canada/Collège militaire royal du Canada, May 1998 - February 1999.

Pagé, D.J.Y.S., Neutron and Gamma Radiation Effects on the Viscoelastic Behaviour of Poly(Aryl Ether Ether Ketone), Royal Military College of Canada/Collège militaire royal du Canada, 1997.

Pierre, J.R.M., Low Enrichment Uranium (LEU)-Fuelled SLOWPOKE-2 Nuclear Reactor Simulation With the Monte-Carlo Based MCNP 4A Code, Royal Military College of Canada/Collège militaire royal du Canada, 1996.

Pierre, J.R.M. and Bonin, H.W., Monte Carlo Simulation of the LEU-Fueled SLOWPOKE-2 Nuclear Reactor Using MCNP 4A, Nuclear Technology, Vol 125, January 1999.

Reactor Manual for the SLOWPOKE-2 Facility, RMC-CCE-SL2-97-3, Royal Military College of Canada/Collège militaire royal du Canada, 1997.

Rockwell, T., Reactor Shielding Design Manual, Van Nostrand, 1956.

Seltzer, S.M., Calculation of Photon Mass Energy-Transfer and Mass Energy-Absorption Coefficients, Radiation Research **136**, 147-170, 1993.

Silva, H., Determination of Absorbed Gamma Doses in a Pool Research Reactor, Kerntechnik 57, No. 5, 1992.

Stephenson R., Introduction to Nuclear Engineering, McGraw-Hill, 1954.

Sunder, S., Calculation of Radiation Dose Rates in a Water Layer in Contact with Used CANDU UO_2 Fuel, Nuclear Technology, Vol. 122, 1998.

Taylor, J.R., An Introduction to Error Analysis, University Science, 1982.

Tsoulfanidis, N., Measurement and Detection of Radiation, Hemisphere, New York, 1983.

Van Dam, H. and Kleiss, E.B.J., Response of Incore Gamma Detectors to Parametric Fluctuations in a Reactor Core, Ann. Nuclear Energy, Vol. 12, No. 4, p. 201-207, 1985.

Viragh, E. and Zsolnay, E.M., Gamma Dose Measurements in the Mixed Radiation Field of Thermal Nuclear Reactors Using TL-Detectors, Nuclear Instruments and Methods, **175**, 253-254, 1980.

Walker, W.H. and Okazaki, A., Prompt and Delayed Photoneutrons in D_2O-Moderated Reactors, CRNL-334, Chalk River Nuclear Laboratories, 1969.

Weast, R.C. et al., CRC Handbook of Chemistry and Physics, (67^{th} ed.), CRC, Boca Raton, Florida, 1986.

Worku, M., Personal Communication, Grove Engineering (MICROSHIELD Version 5), January 1999.

Ziegler, J.F., The Transport of Ions in Matter, TRIM Fact Sheet, IBM, Yorktown, New York, 1988.

ANNEX A

WIMS-AECL SLOWPOKE-2 Model Case File

This Annex contains the WIMS-AECL input file used to model the SLOWPOKE-2 reactor container and pool. In addition, the WIMS-AECL-generated output file with the 26 energy group relative absolute neutron flux values across each of the 98 regions, the Σ_a (E) values across each of the 98 regions, and the resultant lattice k_{eff} are all contained within.

```
 1998-08-28 13:10:22  Starting  0 Prelude        WIMS-AECL Developmental
97-06-05  CPU Time      .100 Secs
 WIMS-AECL 2-5b-4 LD= 97-06-05   USER CHARGE HP 1998-08-28 13:10:22


 PRELUDE DATA
 > Title "SLOWPOKE-2 WIMS-AECL Case"
 > Cell CLUSTER
 > Sequence Pij
 > Scan
     NREGION 150 NGROUP  26
      29 Materials defined,  29 Materials actually used
     ROD TYPES  20 Rod Subdivisions   3 Rod Sectors   1 ANNULUS SECTORS
1 Rods 207
 > Nrods 198 1 0 0 20 3 1 2
 > Preout

 LIBRARY TITLE:
 WIMS-AECL ENDF/B-V-based library (HP 9000 1994 November 5)


 WIMS DATA LIBRARY PARAMETERS
          145 NUCLIDES                      89 ENERGY GROUPS
           47 FISSION GROUPS                24 FAST GROUPS
           23 RESONANCE GROUPS              42 THERMAL GROUPS
           20 FISSILE ISOTOPES              45 FISSION PRODUCT ISOTOPES
          137 P1 NUCLIDES


 1998-08-28 13:10:23  Starting  1 Main Data      WIMS-AECL Developmental
97-06-05  CPU Time      .620 Secs

 SLOWPOKE-2 WIMS-AECL CASE



 MAIN DATA

 > Annulus # 0.169 Gap
 > Annulus # 0.186 Rod
 > Annulus # 0.265 Plate
 > Annulus # 0.565 Coolant
 > Annulus # 0.865 Moderator
 > Annulus # 1.229 Coolant
 > Annulus # 1.331 Clad
 > Annulus # 1.60 Coolant
 > Annulus # 1.88 Coolant
 > Annulus # 2.20 Coolant
 > Annulus # 2.45 Coolant
 > Annulus # 2.70 Coolant
 > Annulus # 3.00 Moderator
 > Annulus # 3.30 Coolant
 > Annulus # 3.65 Coolant
 > Annulus # 3.997 Coolant
 > Annulus # 4.25 Coolant
 > Annulus # 4.52 Coolant
 > Annulus # 4.787 Coolant
 > Annulus # 5.10 Coolant
 > Annulus # 5.471 Coolant
 > Annulus # 5.75 Moderator
 > Annulus # 6.135 Coolant
```

```
> Annulus # 6.445 Coolant
> Annulus # 6.70 Coolant
> Annulus # 6.921 Coolant
> Annulus # 7.193 Coolant
> Annulus # 7.40 Coolant
> Annulus # 7.625 Coolant
> Annulus # 7.911 Coolant
> Annulus # 8.250 Coolant
> Annulus # 8.581 Coolant
> Annulus # 8.861 Moderator
> Annulus # 9.11 Coolant
> Annulus # 9.381 Coolant
> Annulus # 9.556 Coolant
> Annulus # 9.80 Coolant
> Annulus # 10.103 Coolant
> Annulus # 10.38 Coolant
> Annulus # 10.65 Coolant
> Annulus # 11.00 Moderator
> Annulus # 11.90 Reflector
> Annulus # 12.80 Beryl
> Annulus # 13.70 Reflector
> Annulus # 14.60 Beryl
> Annulus # 15.20 Reflector
> Annulus # 16.10 Beryl
> Annulus # 17.00 Reflector
> Annulus # 17.90 Reflector
> Annulus # 18.80 Reflector
> Annulus # 19.70 Reflector
> Annulus # 20.35 Reflector
> Annulus # 21.00 Reflector
> Annulus # 21.30 Coolant
> * Annulus # 22.12 Plate 0.39270 Coolant 5.89048
> * Annulus # 29.48 Therm 0.39270 Coolant 5.89048
> * Annulus # 29.80 Plate 0.39270 Coolant 5.89048
> Annulus # 21.60 Coolant
> Annulus # 21.90 Coolant
> Annulus # 22.20 Coolant
> Annulus # 22.50 Coolant
> Annulus # 22.80 Coolant
> Annulus # 23.10 Coolant
> Annulus # 23.40 Coolant
> Annulus # 23.70 Coolant
> Annulus # 24.00 Coolant
> Annulus # 24.30 Coolant
> Annulus # 24.60 Coolant
> Annulus # 24.90 Coolant
> Annulus # 25.20 Coolant
> Annulus # 25.50 Coolant
> Annulus # 25.80 Coolant
> Annulus # 26.10 Coolant
> Annulus # 26.40 Coolant
> Annulus # 26.70 Coolant
> Annulus # 27.00 Coolant
> Annulus # 27.30 Coolant
> Annulus # 27.60 Coolant
> Annulus # 27.90 Coolant
> Annulus # 28.20 Coolant
> Annulus # 28.50 Coolant
> Annulus # 28.80 Coolant
> Annulus # 29.10 Coolant
> Annulus # 29.40 Coolant
> Annulus # 29.70 Coolant
> Annulus # 30.00 Coolant
> Annulus # 30.30 Coolant
> Annulus # 30.60 Coolant
```

```
> Annulus # 31.55 Vessel
> Annulus # 31.85 Coolant
> Annulus # 32.15 Moderator
> Annulus # 32.45 Coolant
> Annulus # 32.75 Moderator
> Annulus # 33.05 Moderator
> Annulus # 33.35 Moderator
> Annulus # 33.65 Moderator
> Annulus # 33.95 Moderator
> Annulus # 34.25 Moderator
> Annulus # 34.40 Moderator
> Annulus # 34.55 Moderator
> Annulus # 34.85 Coolant
> Annulus # 35.15 Moderator
> Annulus # 35.45 Moderator
> Annulus # 35.75 Moderator
> Annulus # 36.05 Moderator
> Annulus # 36.35 Moderator
> Annulus # 36.65 Moderator
> Annulus # 36.95 Moderator
> Annulus # 37.25 Moderator
> Annulus # 37.55 Moderator
> Annulus # 37.85 Coolant
> Annulus # 38.15 Moderator
> Annulus # 38.45 Moderator
> Annulus # 38.75 Moderator
> Annulus # 39.05 Moderator
> Annulus # 39.35 Moderator
> Annulus # 39.65 Moderator
> Annulus # 39.95 Moderator
> Annulus # 40.25 Coolant
> Annulus # 40.55 Moderator
> Annulus # 40.85 Moderator
> Annulus # 41.15 Moderator
> Annulus # 41.45 Moderator
> Annulus # 41.75 Moderator
> Annulus # 42.05 Moderator
> Annulus # 42.35 Coolant
> Annulus # 42.65 Moderator
> Annulus # 42.95 Moderator
> Annulus # 43.25 Moderator
> Annulus # 43.55 Moderator
> Annulus # 43.85 Moderator
> Annulus # 44.15 Moderator
> Annulus # 44.45 Moderator
> Annulus # 44.75 Moderator
> Annulus # 45.05 Coolant
> Annulus # 45.35 Moderator
> Annulus # 45.65 Moderator
> Annulus # 45.95 Moderator
> Annulus # 46.25 Moderator
> Annulus # 46.55 Moderator
> Annulus # 46.85 Moderator
> Annulus # 47.15 Moderator
> Annulus # 47.45 Coolant
> Annulus # 47.75 Moderator
> Annulus # 48.05 Moderator
> Annulus # 48.35 Moderator
> Annulus # 48.65 Moderator
> Annulus # 48.95 Moderator
> Annulus # 49.25 Moderator
> Annulus # 49.55 Moderator
> Annulus # 49.85 Moderator
> Annulus # 50.15 Coolant
> Annulus # 50.45 Moderator
```

```
> Npijan #
> •
> Array # 1 6 1.88 0.4555987
> Rodsub # # 0.10 Fuel_1
> Rodsub # # 0.19 Fuel_1
> Rodsub # # 0.27 Clad
> *
> Array # 1 2 2.20 0
> Rodsub # # 0.10 Fuel_2
> Rodsub # # 0.19 Fuel_2
> Rodsub # # 0.27 Clad
> •
> Array # 1 6 3.30 0
> Rodsub # # 0.10 Fuel_3
> Rodsub # # 0.19 Fuel_3
> Rodsub # # 0.27 Clad
> •
> Array # 1 12 3.997 15.0D
> Rodsub # # 0.10 Fuel_4
> Rodsub # # 0.19 Fuel_4
> Rodsub # # 0.27 Clad
> *
> Array # 1 12 4.787 25.0D
> Rodsub # # 0.10 Fuel_5
> Rodsub # # 0.19 Fuel_5
> Rodsub # # 0.27 Clad
> *
> Array # 1 6 5.471 0D
> Rodsub # # 0.10 Fuel_6
> Rodsub # # 0.19 Fuel_6
> Rodsub # # 0.27 Clad
> *
> Array # 1 6 5.75 30D
> Rodsub # # 0.10 Fuel_7
> Rodsub # # 0.19 Fuel_7
> Rodsub # # 0.27 Clad
> *
> Array # 1 12 6.135 9.0D
> Rodsub # # 0.10 Fuel_8
> Rodsub # # 0.19 Fuel_8
> Rodsub # # 0.27 Clad
> *
> Array # 1 12 6.921 15.0D
> Rodsub # # 0.10 Fuel_9
> Rodsub # # 0.19 Fuel_9
> Rodsub # # 0.27 Clad
> *
> Array # 1 12 7.193 7.0D
> Rodsub # # 0.10 Fuel_10
> Rodsub # # 0.19 Fuel_10
> Rodsub # # 0.27 Clad
> *
> Array # 1 6 7.625 30.0D
> Rodsub # # 0.10 Fuel_11
> Rodsub # # 0.19 Fuel_11
> Rodsub # # 0.27 Clad
> *
> Array # 1 12 7.911 14.0D
> Rodsub # # 0.10 Fuel_12
> Rodsub # # 0.19 Fuel_12
> Rodsub # # 0.27 Clad
> *
> Array # 1 12 8.581 25.0D
> Rodsub # # 0.10 Fuel_13
> Rodsub # # 0.19 Fuel_13
```

```
 > Rodsub # # 0.27 Clad
 > *
 > Array # 1 6 8.861 0D
 > Rodsub # # 0.10 Fuel_14
 > Rodsub # # 0.19 Fuel_14
 > Rodsub # # 0.27 Clad
 > *
 > Array # 1 12 9.381 6.0D
 > Rodsub # # 0.10 Fuel_15
 > Rodsub # # 0.19 Fuel_15
 > Rodsub # # 0.27 Clad
 > *
 > Array # 1 12 9.556 22.0D
 > Rodsub # # 0.10 Fuel_16
 > Rodsub # # 0.19 Fuel_16
 > Rodsub # # 0.27 Clad
 > *
 > Array # 1 16 10.103 0D
 > Rodsub # # 0.10 Fuel_17
 > Rodsub # # 0.19 Fuel_17
 > Rodsub # # 0.27 Clad
 > •
 > Array # 1 36 10.65 5.0D
 > Rodsub # # 0.10 Fuel_18
 > Rodsub # # 0.19 Fuel_18
 > Rodsub # # 0.27 Clad
 > *
 > Array # 1 5 15.20 0.6288749
 > Rodsub # # 1.00 Gap
 > Rodsub # # 1.17 Al
 > *
 > Array # 1 4 23.60 60.0D
 > Rodsub # # 1.80 Gap
 > Rodsub # # 2.13 Al
 > *
 > Tolerance 1e-6
 > *
 > * Fewgroups 1 2 3 4 5 6 7 8 9 10 11 12 13 14 15 16 17 18 19 20 21 22
$
 > * 23 24 25 26 27 28 29 30 31 32 33 34 35 36 37 38 39 40 41 42 43 44
45 46 $
 > * 47 48 49 50 51 52 53 54 55 56 57 58 59 60 61 62 63 64 65 66 67 68
69 $
 > * 70 71 72 73 74 75 76 77 78 79 80 81 82 83 84 85 86 87 88 89
 > *> Fewgroups 4 8 12 16 20 24 28 32 36 40 44 47 50 53 56 59 62 $
 > 65 68 71 74 77 80 83 86 89
 > Suppress 1 1 1 1 1  1 1 1 1 1  1 0 0 0 0  1
 > *
 > Water Coolant 0.0 303 Cool h-h2o=99.8
 ^ ABOVE MATERIAL DETERMINED TO BE LIGHT WATER WITH A DENSITY OF  .99590
GRAMS PER CUBIC CENTIMETER
 > * Water Therm 0.0 303 Moder d-d2o=99.9
 > Water Moderator 0.0 303 Moder h-h2o=99.8
 ^ ABOVE MATERIAL DETERMINED TO BE LIGHT WATER WITH A DENSITY OF  .99590
GRAMS PER CUBIC CENTIMETER
 > Material Vessel 2.70 303 Moder Al=97.92 Si=0.60 Cu=0.28 Na23=1.00 $
 >   Cr=0.20
 > Material Gap 0.000051 317 Moder N14=78.826 O16=21.174
 > Material Al 2.70 303 Moder Al=97.92 Si=0.60 Cu=0.28 Na23=1.00 $
 >   Cr=0.20
 > Material Plate 2.70 303 Moder Al=97.92 Si=0.60 Cu=0.28 Na23=1.00 $
 >   Cr=0.20
 > Material Reflector 1.85 303 Moder Be9=99.54 Al=0.10 C=0.15 Fe=0.13 $
 >   Si=0.06 Mn=0.02
 > Material zr4 6.3918 316 clad Zr-nat=98.1813 $
```

```
 >   Fe=0.21 Cr=0.1 Ni=0.007 B10=0.00005962
 > Material Clad=zr4
 > Material Beryl=Reflector
 > Material Rod 8.65 303 Clad Cd112=100.0
 > Material Fuel_1 10.4 318 Fuel O16=11.85014 U235=17.533 $
 >   U238=70.61686
 > Material fuel_2 fuel_1
 > Material fuel_3 fuel_1
 > Material fuel_4 fuel_1
 > Material fuel_5 fuel_1
 > Material fuel_6 fuel_1
 > Material fuel_7 fuel_1
 > Material fuel_8 fuel_1
 > Material fuel_9 fuel_1
 > Material fuel_10 fuel_1
 > Material fuel_11 fuel_1
 > Material fuel_12 fuel_1
 > Material fuel_13 fuel_1
 > Material fuel_14 fuel_1
 > Material fuel_15 fuel_1
 > Material fuel_16 fuel_1
 > Material fuel_17 fuel_1
 > Material fuel_18 fuel_1
 > *
 > * Mtrflux
 > * Power 6 5.00000E11 1 1 0.0001
 > Begin
 --- 1262 lines and  837 angles selected for Pij tracking

 1998-08-28 13:10:23  Starting  2 Cross Sections WIMS-AECL Developmental
97-06-05  CPU Time      .880 Secs

 1998-08-28 13:10:25  Starting  3 Resonance      WIMS-AECL Developmental
97-06-05  CPU Time     1.640 Secs

 1998-08-28 13:10:25  Starting  4 Spectrox       WIMS-AECL Developmental
97-06-05  CPU Time     2.050 Secs

0//// ITERATION LIMIT REACHED WITHOUT CONVERGENCE


 1998-08-28 13:10:26  Starting  5 Condensation   WIMS-AECL Developmental
97-06-05  CPU Time     2.370 Secs

 MTR MESH SELECTION IS:     1    1    1    1    1    1    1    1    1
1    1    1
     1    1    1    1    1    1    1    1    1    1    1    1    1    1
 1
     1    1    1    1    1    1    1    1    1    1    1    1    1    1
 1
     1    1    1    1    1    1    1    1    1    1    1    1    1    1
 1
     1    1    1    1    1    1    1    1    1    1    1    1    1    1
 1
     1    1    1    1    1    1    1    1    1    1    1    1    1    1
 1
     1    1    1    1    1    1    1    1    1    1    1    1    1    1
 1
     1    1    1    1    1    1    1    1    1    1    1    1    1    1
 1
     1    1    1    1    1    1    1    1    1    1    1    1    1    1
 1
     1    1    1    1    1    1    1    1    1    1    1    1    1    1
 1
```

```
     1    1    1

 1998-08-28 13:10:26  Starting  8 Pij          WIMS-AECL Developmental
97-06-05  CPU Time     2.670 Secs

 1998-08-31 06:54:11  Starting  7 Perseus      WIMS-AECL Developmental
97-06-05  CPU Time223301.719 Secs

 1998-08-31 06:58:02  Starting 11 Unsmear      WIMS-AECL Developmental
97-06-05  CPU Time223530.422 Secs
1

 1998-08-31 06:58:02  Starting 13 Edit         WIMS-AECL Developmental
97-06-05  CPU Time223530.422 Secs

 SLOWPOKE-2 WIMS-AECL CASE



 EDIT DATA READ BY CHN13

 DEFAULT IS  8 THERMAL GROUPS BELOW 6.2500E-01 EV

 > Benoist 1
 > * Cellav
 > Buckling 1.416803e-3 3.0356974e-3
 > Beeone 1
 > Leakage -6
 > Endcap clad 0.02084 -1 1.2 41
 > Region
 > Begin

            ZONE        REGION                    RADIUS
VOLUME           MATERIAL

              1            1                      .16723
.08786            GAP
              2            2                      .18405
.01856            ROD
              3            3                      .26222
.10960           PLATE
              4            4                      .55908
.76595          COOLANT
              5            5                      .85594
1.31966         MODERATOR
              6            6                     1.21613
2.34467          COOLANT
              7            7                     1.31706
.80324            CLAD
              8            8                     1.58324
2.42533          COOLANT
              9            8                     1.80364
2.34514          COOLANT
             10            9                     1.81986
.18457           FUEL_1
             11            9                     1.86151
.48172           FUEL_1
             12           10                     1.91870
.67921            CLAD
             13           11                     2.16093
3.10452          COOLANT
             14           11                     2.39509
3.35157          COOLANT
             15           12                     2.39917
```

.06152		FUEL_2	
	16	12	2.40980
.16057		FUEL_2	
	17	13	2.42471
.22640		CLAD	
	18	14	2.67172
3.95484		COOLANT	
	19	15	2.96858
5.26017		MODERATOR	
	20	16	3.23305
5.15280		COOLANT	
	21	17	3.24213
.18457		FUEL_3	
	22	17	3.26569
.48172		FUEL_3	
	23	18	3.29862
.67921		CLAD	
	24	19	3.61177
6.79823		COOLANT	
	25	19	3.90140
6.83630		COOLANT	
	26	19	4.10339
5.07961		COOLANT	
	27	20	4.11768
.36914		FUEL_4	
	28	20	4.15475
.96344		FUEL_4	
	29	21	4.20647
1.35841		CLAD	
	30	22	4.47259
7.25603		COOLANT	
	31	22	4.69198
6.31653		COOLANT	
	32	23	4.70448
.36913		FUEL_5	
	33	23	4.73696
.96344		FUEL_5	
	34	24	4.78239
1.35842		CLAD	
	35	25	5.04658
8.15783		COOLANT	
	36	25	5.39409
11.39837		COOLANT	
	37	26	5.39953
.18457		FUEL_6	
	38	26	5.41371
.48171		FUEL_6	
	39	27	5.43364
.67922		CLAD	
	40	28	5.67111
8.28444		MODERATOR	
	41	29	5.67629
.18457		FUEL_7	
	42	29	5.68978
.48172		FUEL_7	
	43	30	5.70874
.67922		CLAD	
	44	31	6.03569
12.06310		COOLANT	
	45	32	6.04542
.36914		FUEL_8	
	46	32	6.07073
.96342		FUEL_8	
	47	33	6.10624
1.35844		CLAD	

	48	34	6.37749
10.63820		COOLANT	
	49	34	6.62695
10.19177		COOLANT	
	50	34	6.81742
8.04482		COOLANT	
	51	35	6.82604
.36914		FUEL_9	
	52	35	6.84846
.96342		FUEL_9	
	53	36	6.87996
1.35842		CLAD	
	54	37	7.08774
9.11783		COOLANT	
	55	37	7.26651
8.06168		COOLANT	
	56	38	7.27459
.36911		FUEL_10	
	57	38	7.29564
.96345		FUEL_10	
	58	39	7.32522
1.35841		CLAD	
	59	40	7.53103
9.60587		COOLANT	
	60	41	7.53493
.18455		FUEL_11	
	61	41	7.54510
.48172		FUEL_11	
	62	42	7.55941
.67922		CLAD	
	63	43	7.80093
11.65454		COOLANT	
	64	44	7.80846
.36913		FUEL_12	
	65	44	7.82807
.96343		FUEL_12	
	66	45	7.85564
1.35841		CLAD	
	67	46	8.16358
15.49756		COOLANT	
	68	46	8.46603
15.80085		COOLANT	
	69	47	8.47296
.36911		FUEL_13	
	70	47	8.49104
.96349		FUEL_13	
	71	48	8.51647
1.35842		CLAD	
	72	49	8.75604
13.00012		MODERATOR	
	73	50	8.99110
13.10543		COOLANT	
	74	51	8.99437
.18456		FUEL_14	
	75	51	9.00288
.48167		FUEL_14	
	76	52	9.01488
.67920		CLAD	
	77	53	9.25436
13.74468		COOLANT	
	78	53	9.39332
8.14046		COOLANT	
	79	54	9.39957
.36913		FUEL_15	
	80	54	9.41587

.96346		FUEL_15	
	81	55	9.43880
1.35841		CLAD	
	82	56	9.65387
12.90042		COOLANT	
	83	57	9.65996
.36918		FUEL_16	
	84	57	9.67582
.96339		FUEL_16	
	85	58	9.69814
1.35844		CLAD	
	86	59	9.96873
16.71883		COOLANT	
	87	60	9.97659
.49216		FUEL_17	
	88	60	9.99706
1.28457		FUEL_17	
	89	61	10.02585
1.81121		CLAD	
	90	62	10.27127
15.64913		COOLANT	
	91	62	10.47763
13.45167		COOLANT	
	92	63	10.49444
1.10744		FUEL_18	
	93	63	10.53818
2.89031		FUEL_18	
	94	64	10.59955
4.07528		CLAD	
	95	65	10.88478
19.25113		MODERATOR	
	96	66	11.00000
7.92206		CLAD	
	97	67	11.90000
64.74813		REFLECTOR	
	98	68	12.80000
69.83765		BERYL	
	99	69	13.70000
74.92686		REFLECTOR	
	100	70	14.55698
76.07612		BERYL	
	101	71	15.08886
49.53617		REFLECTOR	
	102	72	15.90028
78.99689		BERYL	
	103	73	16.05675
15.70796		GAP	
	104	74	16.11408
5.79466		AL	
	105	75	17.00000
92.16264		REFLECTOR	
	106	75	17.90000
98.67732		REFLECTOR	
	107	75	18.80000
103.76675		REFLECTOR	
	108	75	19.70000
108.85632		REFLECTOR	
	109	75	20.35000
81.78350		REFLECTOR	
	110	75	21.00000
84.43803		REFLECTOR	
	111	76	21.30000
39.86671		COOLANT	
	112	76	21.59640
39.94396		COOLANT	

	113	76	21.87901
38.59966		COOLANT	
	114	76	22.15504
38.18600		COOLANT	
	115	76	22.42712
38.10717		COOLANT	
	116	76	22.69671
38.21612		COOLANT	
	117	76	22.96483
38.46250		COOLANT	
	118	76	23.23233
38.82273		COOLANT	
	119	76	23.49994
39.28820		COOLANT	
	120	76	23.76832
39.85522		COOLANT	
	121	76	24.03818
40.52910		COOLANT	
	122	76	24.31022
41.32074		COOLANT	
	123	76	24.58529
42.25259		COOLANT	
	124	76	24.86447
43.37112		COOLANT	
	125	76	25.14952
44.78771		COOLANT	
	126	77	25.40587
40.71503		GAP	
	127	78	25.50776
16.29733		AL	
	128	79	25.80000
47.10516		COOLANT	
	129	79	26.10000
48.91481		COOLANT	
	130	79	26.40000
49.47989		COOLANT	
	131	79	26.70000
50.04573		COOLANT	
	132	79	27.00000
50.61101		COOLANT	
	133	79	27.30000
51.17646		COOLANT	
	134	79	27.60000
51.74213		COOLANT	
	135	79	27.90000
52.30739		COOLANT	
	136	79	28.20000
52.87325		COOLANT	
	137	79	28.50000
53.43832		COOLANT	
	138	79	28.80000
54.00378		COOLANT	
	139	79	29.10000
54.56964		COOLANT	
	140	79	29.40000
55.13491		COOLANT	
	141	79	29.70000
55.70056		COOLANT	
	142	79	30.00000
56.26583		COOLANT	
	143	79	30.30000
56.83130		COOLANT	
	144	79	30.60000
57.39715		COOLANT	
	145	80	31.55000

185.48723 VESSEL'
146 81 31.85000
59.75334 COOLANT
147 82 32.15000
60.31861 MODERATOR
148 83 32.45000
60.88406 COOLANT
149 84 32.75000
61.44936 MODERATOR
150 84 33.05000
62.01499 MODERATOR
151 84 33.34999
62.58028 MODERATOR
152 84 33.65000
63.14670 MODERATOR
153 84 33.95000
63.71120 MODERATOR
154 84 34.25000
64.27686 MODERATOR
155 84 34.40000
32.35088 MODERATOR
156 84 34.55000
32.49125 MODERATOR
157 85 34.84999
65.40780 COOLANT
158 86 35.15000
65.97421 MODERATOR
159 86 35.45000
66.53870 MODERATOR
160 86 35.75000
67.10438 MODERATOR
161 86 36.05000
67.66963 MODERATOR
162 86 36.34999
68.23531 MODERATOR
163 86 36.65000
68.80172 MODERATOR
164 86 36.95000
69.36621 MODERATOR
165 86 37.25000
69.93150 MODERATOR
166 86 37.55000
70.49715 MODERATOR
167 87 37.84999
71.06281 COOLANT
168 88 38.15000
71.62885 MODERATOR
169 88 38.45000
72.19373 MODERATOR
170 88 38.75000
72.75901 MODERATOR
171 88 39.05000
73.32465 MODERATOR
172 88 39.34999
73.88993 MODERATOR
173 88 39.65000
74.45673 MODERATOR
174 88 39.95000
75.02085 MODERATOR
175 89 40.25000
75.58652 COOLANT
176 90 40.55000
76.15216 MODERATOR
177 90 40.84999
76.71745 MODERATOR

178	90	41.15000	77.28387	MODERATOR
179	90	41.45000	77.84835	MODERATOR
180	90	41.75000	78.41403	MODERATOR
181	90	42.05000	78.97929	MODERATOR
182	91	42.34999	79.54496	COOLANT
183	92	42.65000	80.11137	MODERATOR
184	92	42.95000	80.67587	MODERATOR
185	92	43.25000	81.24154	MODERATOR
186	92	43.55000	81.80679	MODERATOR
187	92	43.84999	82.37247	MODERATOR
188	92	44.15000	82.93887	MODERATOR
189	92	44.45000	83.50338	MODERATOR
190	92	44.75000	84.06867	MODERATOR
191	93	45.05000	84.63431	COOLANT
192	94	45.34999	85.19958	MODERATOR
193	94	45.65000	85.76640	MODERATOR
194	94	45.95000	86.33089	MODERATOR
195	94	46.25000	86.89618	MODERATOR
196	94	46.55000	87.46144	MODERATOR
197	94	46.84999	88.02749	MODERATOR
198	94	47.15000	88.59351	MODERATOR
199	95	47.45000	89.15880	COOLANT
200	96	47.75000	89.72331	MODERATOR
201	96	48.05000	90.28934	MODERATOR
202	96	48.35000	90.85461	MODERATOR
203	96	48.65000	91.42142	MODERATOR
204	96	48.95000	91.98516	MODERATOR
205	96	49.25000	92.55121	MODERATOR
206	96	49.55000	93.11647	MODERATOR
207	96	49.85000	93.68251	MODERATOR
208	97	50.15000	94.24856	COOLANT
209	98	50.45000	94.81306	MODERATOR

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 FLUX SCALE FACTOR 7.682884E+01
1FEW-GROUP REGIONAL AND CELL EDIT - CROSS-SECTIONS, INTEGRATED AND
AVERAGED FLUXES, TOTAL EVENTS


0REGION  1     MATERIAL GAP         VOLUME  8.785713E-02

                    CROSS-SECTIONS                              FLUXES
                      REACTIONS
 GROUP   DIFFUSION   ABSORPTION  NU*FISSION            RIF           RAF
        TRANSPORT   ABSORPTION  NU*FISSION

    1    1.41755E+05  5.13322E-07   .00000E+00      2.14806E-04
2.44494E-03        5.05111E-10  1.10265E-10   .00000E+00
    2    1.02864E+05  2.81158E-07   .00000E+00      7.41031E-04
8.43450E-03        2.40133E-09  2.08347E-10   .00000E+00
    3    7.59244E+04  6.05228E-08   .00000E+00      6.17387E-04
7.02717E-03        2.71054E-09  3.73660E-11   .00000E+00
    4    4.70167E+04  1.33476E-08   .00000E+00      3.72666E-04
4.24172E-03        2.64208E-09  4.97420E-12   .00000E+00
    5    3.93658E+04  2.66520E-09   .00000E+00      2.49531E-04
2.84019E-03        2.11293E-09  6.65052E-13   .00000E+00
    6    2.80739E+04  3.59990E-09   .00000E+00      3.52007E-04
4.00659E-03        4.17953E-09  1.26719E-12   .00000E+00
    7    2.25878E+04  9.38580E-09   .00000E+00      3.07270E-04
3.49738E-03        4.53444E-09  2.88398E-12   .00000E+00
    8    2.01120E+04  2.55719E-08   .00000E+00      2.98171E-04
3.39382E-03        4.94183E-09  7.62480E-12   .00000E+00
    9    1.89538E+04  5.23663E-08   .00000E+00      1.46929E-04
1.67237E-03        2.58399E-09  7.69415E-12   .00000E+00
   10    1.87168E+04  8.63328E-08   .00000E+00      1.45978E-04
1.66154E-03        2.59977E-09  1.26027E-11   .00000E+00
   11    1.85665E+04  1.42363E-07   .00000E+00      1.43663E-04
1.63519E-03        2.57924E-09  2.04523E-11   .00000E+00
   12    1.84522E+04  2.18329E-07   .00000E+00      1.03552E-04
1.17865E-03        1.87065E-09  2.26085E-11   .00000E+00
   13    1.83667E+04  3.07848E-07   .00000E+00      9.55483E-05
1.08754E-03        1.73409E-09  2.94144E-11   .00000E+00
   14    1.82499E+04  4.20257E-07   .00000E+00      9.18103E-05
1.04500E-03        1.67691E-09  3.85839E-11   .00000E+00
   15    1.81808E+04  4.94670E-07   .00000E+00      1.10316E-05
1.25563E-04        2.02258E-10  5.45702E-12   .00000E+00
   16    1.81627E+04  5.12788E-07   .00000E+00      1.13496E-05
1.29182E-04        2.08294E-10  5.81991E-12   .00000E+00
   17    1.81308E+04  5.34075E-07   .00000E+00      1.42185E-05
1.61837E-04        2.61407E-10  7.59375E-12   .00000E+00
   18    1.80596E+04  6.01494E-07   .00000E+00      6.10038E-05
6.94352E-04        1.12597E-09  3.66934E-11   .00000E+00
   19    1.79037E+04  7.66370E-07   .00000E+00      9.96893E-05
1.13468E-03        1.85603E-09  7.63989E-11   .00000E+00
   20    1.77278E+04  9.34163E-07   .00000E+00      4.00321E-05
4.55650E-04        7.52717E-10  3.73965E-11   .00000E+00
   21    1.75125E+04  1.12938E-06   .00000E+00      9.60418E-05
1.09316E-03        1.82807E-09  1.08468E-10   .00000E+00
   22    1.70165E+04  1.60405E-06   .00000E+00      5.28894E-04
6.01994E-03        1.03604E-08  8.48371E-10   .00000E+00
   23    1.65119E+04  2.08502E-06   .00000E+00      5.62286E-04
6.40000E-03        1.13512E-08  1.17238E-09   .00000E+00
   24    1.59381E+04  2.64136E-06   .00000E+00      5.59849E-04
6.37227E-03        1.17088E-08  1.47876E-09   .00000E+00
   25    1.50606E+04  3.51301E-06   .00000E+00      4.44860E-04
5.06345E-03        9.84598E-09  1.56280E-09   .00000E+00
   26    1.27683E+04  5.89029E-06   .00000E+00      2.41749E-04
2.75161E-03        6.31116E-09  1.42397E-09   .00000E+00
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 THERMAL 1.58810E+04  2.60688E-06   .00000E+00      2.57340E-03
2.92907E-02       5.40143E-08  6.70854E-09   .00000E+00

  TOTAL  2.35107E+04  1.10952E-06   .00000E+00      6.55135E-03
7.45683E-02       9.28847E-08  7.26886E-09   .00000E+00


0REGION  2     MATERIAL ROD          VOLUME  1.856439E-02

                       CROSS-SECTIONS                         FLUXES
                         REACTIONS
 GROUP   DIFFUSION     ABSORPTION   NU*FISSION          RIF          RAF
        TRANSPORT     ABSORPTION   NU*FISSION

    1    1.88411E+00  1.63923E-03   .00000E+00      4.74048E-05
2.55353E-03       8.38677E-06  7.77075E-08   .00000E+00
    2    1.51321E+00  4.22684E-03   .00000E+00      1.53875E-04
8.28874E-03       3.38960E-05  6.50407E-07   .00000E+00
    3    1.10709E+00  5.30195E-03   .00000E+00      1.28158E-04
6.90342E-03       3.85869E-05  6.79485E-07   .00000E+00
    4    9.90868E-01  5.21266E-03   .00000E+00      7.81255E-05
4.20835E-03       2.62818E-05  4.07242E-07   .00000E+00
    5    9.89607E-01  6.22389E-03   .00000E+00      5.25322E-05
2.82973E-03       1.76946E-05  3.26954E-07   .00000E+00
    6    1.06502E+00  1.49950E-02   .00000E+00      7.46576E-05
4.02155E-03       2.33667E-05  1.11949E-06   .00000E+00
    7    8.99228E-01  3.52846E-02   .00000E+00      6.54346E-05
3.52474E-03       2.42558E-05  2.30883E-06   .00000E+00
    8    9.38336E-01  9.02201E-02   .00000E+00      6.44833E-05
3.47349E-03       2.29070E-05  5.81769E-06   .00000E+00
    9    6.88249E-01  2.92774E-01   .00000E+00      3.26966E-05
1.76126E-03       1.58357E-05  9.57274E-06   .00000E+00
   10    1.95032E+00  3.64505E-03   .00000E+00      3.09158E-05
1.66533E-03       5.28387E-06  1.12690E-07   .00000E+00
   11    1.89148E+00  4.54311E-03   .00000E+00      3.04312E-05
1.63922E-03       5.36286E-06  1.38252E-07   .00000E+00
   12    1.86091E+00  6.89370E-03   .00000E+00      2.19550E-05
1.18264E-03       3.93266E-06  1.51351E-07   .00000E+00
   13    1.83613E+00  9.65830E-03   .00000E+00      2.03074E-05
1.09389E-03       3.68662E-06  1.96135E-07   .00000E+00
   14    1.79822E+00  1.31529E-02   .00000E+00      1.95360E-05
1.05234E-03       3.62135E-06  2.56954E-07   .00000E+00
   15    1.77780E+00  1.54883E-02   .00000E+00      2.34747E-06
1.26450E-04       4.40145E-07  3.63584E-08   .00000E+00
   16    1.77291E+00  1.60562E-02   .00000E+00      2.41755E-06
1.30225E-04       4.54535E-07  3.88167E-08   .00000E+00
   17    1.76234E+00  1.67224E-02   .00000E+00      3.02609E-06
1.63005E-04       5.72365E-07  5.06037E-08   .00000E+00
   18    1.74283E+00  1.88245E-02   .00000E+00      1.30234E-05
7.01527E-04       2.49086E-06  2.45160E-07   .00000E+00
   19    1.69767E+00  2.39423E-02   .00000E+00      2.13021E-05
1.14747E-03       4.18260E-06  5.10020E-07   .00000E+00
   20    1.65282E+00  2.91880E-02   .00000E+00      8.58735E-06
4.62571E-04       1.73186E-06  2.50648E-07   .00000E+00
   21    1.60650E+00  3.50535E-02   .00000E+00      2.07048E-05
1.11530E-03       4.29604E-06  7.25776E-07   .00000E+00
   22    1.49752E+00  4.98980E-02   .00000E+00      1.14253E-04
6.15439E-03       2.54314E-05  5.70098E-06   .00000E+00
   23    1.40088E+00  6.51494E-02   .00000E+00      1.22020E-04
6.57282E-03       2.90343E-05  7.94956E-06   .00000E+00
   24    1.30459E+00  8.25157E-02   .00000E+00      1.21983E-04
6.57080E-03       3.11677E-05  1.00655E-05   .00000E+00
   25    1.17743E+00  1.09716E-01   .00000E+00      9.74938E-05
5.25166E-03       2.76007E-05  1.06966E-05   .00000E+00
   26    9.30478E-01  1.82963E-01   .00000E+00      5.35253E-05
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2.88322E-03       1.91748E-05  9.79312E-06   .00000E+00

 THERMAL 1.30854E+00  8.16123E-02   .00000E+00       5.59869E-04
3.01582E-02       1.42619E-04  4.56922E-05   .00000E+00

  TOTAL  1.23017E+00  4.84437E-02   .00000E+00       1.40120E-03
7.54777E-02       3.79676E-04  6.78791E-05   .00000E+00


0REGION  3     MATERIAL PLATE       VOLUME  1.095991E-01

                       CROSS-SECTIONS                              FLUXES
                         REACTIONS
 GROUP   DIFFUSION     ABSORPTION   NU*FISSION            RIF           RAF
        TRANSPORT    ABSORPTION   NU*FISSION

    1    3.95965E+00  2.27382E-03   .00000E+00       2.66630E-04
2.43277E-03       2.24456E-05  6.06267E-07   .00000E+00
    2    3.10175E+00  5.71386E-05   .00000E+00       8.89944E-04
8.11999E-03       9.56389E-05  5.08502E-08   .00000E+00
    3    2.20964E+00  1.67482E-05   .00000E+00       7.60940E-04
6.94294E-03       1.14791E-04  1.27444E-08   .00000E+00
    4    1.62440E+00  5.00338E-05   .00000E+00       4.65227E-04
4.24481E-03       9.54662E-05  2.32771E-08   .00000E+00
    5    9.26281E-01  1.39174E-04   .00000E+00       3.10788E-04
2.83568E-03       1.11841E-04  4.32537E-08   .00000E+00
    6    3.33061E+00  2.80616E-04   .00000E+00       4.39093E-04
4.00635E-03       4.39451E-05  1.23216E-07   .00000E+00
    7    3.02145E+00  5.19427E-04   .00000E+00       3.83079E-04
3.49527E-03       4.22622E-05  1.98981E-07   .00000E+00
    8    4.08502E+00  2.21816E-04   .00000E+00       3.72205E-04
3.39606E-03       3.03716E-05  8.25612E-08   .00000E+00
    9    4.09110E+00  2.38056E-04   .00000E+00       1.83024E-04
1.66994E-03       1.49123E-05  4.35699E-08   .00000E+00
   10    4.08233E+00  3.84755E-04   .00000E+00       1.82184E-04
1.66228E-03       1.48758E-05  7.00964E-08   .00000E+00
   11    4.06966E+00  6.34997E-04   .00000E+00       1.79319E-04
1.63613E-03       1.46875E-05  1.13867E-07   .00000E+00
   12    4.05320E+00  9.73709E-04   .00000E+00       1.29272E-04
1.17950E-03       1.06312E-05  1.25873E-07   .00000E+00
   13    4.03627E+00  1.37216E-03   .00000E+00       1.19291E-04
1.08843E-03       9.85160E-06  1.63687E-07   .00000E+00
   14    4.01077E+00  1.87212E-03   .00000E+00       1.14656E-04
1.04614E-03       9.52899E-06  2.14650E-07   .00000E+00
   15    4.00087E+00  2.20303E-03   .00000E+00       1.37783E-05
1.25715E-04       1.14794E-06  3.03540E-08   .00000E+00
   16    3.99705E+00  2.28364E-03   .00000E+00       1.41717E-05
1.29305E-04       1.18185E-06  3.23630E-08   .00000E+00
   17    3.98901E+00  2.37847E-03   .00000E+00       1.77517E-05
1.61970E-04       1.48339E-06  4.22220E-08   .00000E+00
   18    3.97288E+00  2.67940E-03   .00000E+00       7.61929E-05
6.95197E-04       6.39276E-06  2.04152E-07   .00000E+00
   19    3.93557E+00  3.41570E-03   .00000E+00       1.24450E-04
1.13550E-03       1.05406E-05  4.25082E-07   .00000E+00
   20    3.89960E+00  4.16532E-03   .00000E+00       5.00248E-05
4.56434E-04       4.27606E-06  2.08369E-07   .00000E+00
   21    3.85636E+00  5.03787E-03   .00000E+00       1.20171E-04
1.09646E-03       1.03873E-05  6.05406E-07   .00000E+00
   22    3.75376E+00  7.15826E-03   .00000E+00       6.61867E-04
6.03899E-03       5.87737E-05  4.73782E-06   .00000E+00
   23    3.65252E+00  9.30090E-03   .00000E+00       7.03970E-04
6.42314E-03       6.42451E-05  6.54755E-06   .00000E+00
   24    3.54298E+00  1.17702E-02   .00000E+00       7.00286E-04
6.38952E-03       6.58848E-05  8.24253E-06   .00000E+00
   25    3.37502E+00  1.56477E-02   .00000E+00       5.56214E-04
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5.07498E-03        5.49344E-05  8.70349E-06   .00000E+00
   26    2.94383E+00  2.62317E-02   .00000E+00        3.01892E-04
2.75451E-03        3.41836E-05  7.91915E-06   .00000E+00

 THERMAL 3.53848E+00  1.16157E-02   .00000E+00        3.21887E-03
2.93695E-02        3.03226E-04  3.73894E-05   .00000E+00

  TOTAL  2.87096E+00  4.86349E-03   .00000E+00        8.13642E-03
7.42380E-02        9.44681E-04  3.95714E-05   .00000E+00


0REGION  4      MATERIAL COOLANT      VOLUME  7.659541E-01

                         CROSS-SECTIONS                          FLUXES
                           REACTIONS
 GROUP   DIFFUSION     ABSORPTION    NU*FISSION           RIF           RAF
        TRANSPORT     ABSORPTION   NU*FISSION

    1    2.70642E+00  2.17943E-03   .00000E+00        1.85611E-03
2.42327E-03        2.28606E-04  4.04526E-06   .00000E+00
    2    2.01031E+00  1.01023E-05   .00000E+00        6.15868E-03
8.04053E-03        1.02118E-03  6.22170E-08   .00000E+00
    3    1.31913E+00  3.55934E-06   .00000E+00        5.25441E-03
6.85996E-03        1.32775E-03  1.87022E-08   .00000E+00
    4    7.92566E-01  3.59323E-06   .00000E+00        3.22361E-03
4.20861E-03        1.35577E-03  1.15832E-08   .00000E+00
    5    6.54618E-01  6.63753E-06   .00000E+00        2.16044E-03
2.82058E-03        1.10010E-03  1.43400E-08   .00000E+00
    6    5.56182E-01  1.94203E-05   .00000E+00        3.06448E-03
4.00086E-03        1.83661E-03  5.95132E-08   .00000E+00
    7    5.63543E-01  6.29493E-05   .00000E+00        2.67828E-03
3.49666E-03        1.58419E-03  1.68596E-07   .00000E+00
    8    5.77544E-01  1.71416E-04   .00000E+00        2.60657E-03
3.40304E-03        1.50440E-03  4.46808E-07   .00000E+00
    9    5.80367E-01  3.53087E-04   .00000E+00        1.28755E-03
1.68098E-03        7.39506E-04  4.54619E-07   .00000E+00
   10    5.80777E-01  5.82085E-04   .00000E+00        1.27475E-03
1.66426E-03        7.31633E-04  7.42011E-07   .00000E+00
   11    5.80114E-01  9.59742E-04   .00000E+00        1.25514E-03
1.63866E-03        7.21204E-04  1.20461E-06   .00000E+00
   12    5.78579E-01  1.47308E-03   .00000E+00        9.05064E-04
1.18162E-03        5.21429E-04  1.33323E-06   .00000E+00
   13    5.30274E-01  2.07713E-03   .00000E+00        8.34919E-04
1.09004E-03        5.24835E-04  1.73423E-06   .00000E+00
   14    4.84734E-01  2.83531E-03   .00000E+00        8.02451E-04
1.04765E-03        5.51815E-04  2.27520E-06   .00000E+00
   15    4.56194E-01  3.34008E-03   .00000E+00        9.64642E-05
1.25940E-04        7.04848E-05  3.22198E-07   .00000E+00
   16    4.50003E-01  3.46267E-03   .00000E+00        9.92151E-05
1.29531E-04        7.34921E-05  3.43549E-07   .00000E+00
   17    4.42700E-01  3.60538E-03   .00000E+00        1.24265E-04
1.62236E-04        9.35662E-05  4.48024E-07   .00000E+00
   18    4.19829E-01  4.05921E-03   .00000E+00        5.33444E-04
6.96443E-04        4.23540E-04  2.16536E-06   .00000E+00
   19    3.66366E-01  5.17002E-03   .00000E+00        8.71469E-04
1.13776E-03        7.92896E-04  4.50551E-06   .00000E+00
   20    3.18292E-01  6.29875E-03   .00000E+00        3.50446E-04
4.57529E-04        3.67007E-04  2.20737E-06   .00000E+00
   21    2.76329E-01  7.56452E-03   .00000E+00        8.42268E-04
1.09963E-03        1.01602E-03  6.37135E-06   .00000E+00
   22    2.07862E-01  1.07793E-02   .00000E+00        4.63939E-03
6.05701E-03        7.43985E-03  5.00092E-05   .00000E+00
   23    1.64184E-01  1.40641E-02   .00000E+00        4.93262E-03
6.43984E-03        1.00144E-02  6.93729E-05   .00000E+00
   24    1.30440E-01  1.78189E-02   .00000E+00        4.90532E-03
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6.40420E-03        1.25353E-02  8.74075E-05   .00000E+00
   25    1.01321E-01  2.37240E-02   .00000E+00       3.89726E-03
5.08812E-03        1.28215E-02  9.24586E-05   .00000E+00
   26    7.46679E-02  3.97569E-02   .00000E+00       2.12134E-03
2.76955E-03        9.47013E-03  8.43381E-05   .00000E+00

 THERMAL 1.38091E-01  1.75828E-02   .00000E+00       2.25601E-02
2.94536E-02        5.44571E-02  3.96670E-04   .00000E+00

  TOTAL  2.74809E-01  7.26576E-03   .00000E+00       5.67760E-02
7.41245E-02        6.88672E-02  4.12521E-04   .00000E+00


0REGION  5     MATERIAL MODERATOR   VOLUME  1.319656E+00

                     CROSS-SECTIONS                              FLUXES
                       REACTIONS
 GROUP    DIFFUSION    ABSORPTION   NU*FISSION          RIF           RAF
        TRANSPORT    ABSORPTION   NU*FISSION

    1    2.74822E+00  2.21855E-03   .00000E+00       3.24658E-03
2.46017E-03        3.93779E-04  7.20270E-06   .00000E+00
    2    2.00863E+00  9.66120E-06   .00000E+00       1.07498E-02
8.14588E-03        1.78393E-03  1.03856E-07   .00000E+00
    3    1.31914E+00  3.55843E-06   .00000E+00       9.15241E-03
6.93545E-03        2.31272E-03  3.25682E-08   .00000E+00
    4    7.92343E-01  3.59826E-06   .00000E+00       5.60076E-03
4.24410E-03        2.35620E-03  2.01530E-08   .00000E+00
    5    6.53939E-01  6.66158E-06   .00000E+00       3.74331E-03
2.83658E-03        1.90808E-03  2.49363E-08   .00000E+00
    6    5.56006E-01  1.95488E-05   .00000E+00       5.29766E-03
4.01443E-03        3.17602E-03  1.03563E-07   .00000E+00
    7    5.63650E-01  6.32557E-05   .00000E+00       4.62335E-03
3.50345E-03        2.73418E-03  2.92454E-07   .00000E+00
    8    5.77572E-01  1.72185E-04   .00000E+00       4.49436E-03
3.40571E-03        2.59382E-03  7.73861E-07   .00000E+00
    9    5.80369E-01  3.53483E-04   .00000E+00       2.22252E-03
1.68417E-03        1.27650E-03  7.85625E-07   .00000E+00
   10    5.80777E-01  5.82735E-04   .00000E+00       2.19479E-03
1.66315E-03        1.25968E-03  1.27898E-06   .00000E+00
   11    5.80112E-01  9.60720E-04   .00000E+00       2.15894E-03
1.63599E-03        1.24053E-03  2.07414E-06   .00000E+00
   12    5.78577E-01  1.47362E-03   .00000E+00       1.55617E-03
1.17923E-03        8.96552E-04  2.29321E-06   .00000E+00
   13    5.30259E-01  2.07768E-03   .00000E+00       1.43609E-03
1.08823E-03        9.02758E-04  2.98373E-06   .00000E+00
   14    4.84689E-01  2.83608E-03   .00000E+00       1.38012E-03
1.04582E-03        9.49146E-04  3.91413E-06   .00000E+00
   15    4.56193E-01  3.34009E-03   .00000E+00       1.65815E-04
1.25650E-04        1.21158E-04  5.53835E-07   .00000E+00
   16    4.50003E-01  3.46266E-03   .00000E+00       1.70565E-04
1.29249E-04        1.26343E-04  5.90609E-07   .00000E+00
   17    4.42698E-01  3.60542E-03   .00000E+00       2.13642E-04
1.61892E-04        1.60864E-04  7.70270E-07   .00000E+00
   18    4.19805E-01  4.05969E-03   .00000E+00       9.17031E-04
6.94901E-04        7.28141E-04  3.72287E-06   .00000E+00
   19    3.66218E-01  5.17320E-03   .00000E+00       1.49666E-03
1.13413E-03        1.36227E-03  7.74255E-06   .00000E+00
   20    3.18091E-01  6.30426E-03   .00000E+00       5.99950E-04
4.54626E-04        6.28699E-04  3.78224E-06   .00000E+00
   21    2.74446E-01  7.62670E-03   .00000E+00       1.43400E-03
1.08665E-03        1.74170E-03  1.09367E-05   .00000E+00
   22    2.07048E-01  1.08270E-02   .00000E+00       7.84011E-03
5.94102E-03        1.26220E-02  8.48848E-05   .00000E+00
   23    1.64120E-01  1.40700E-02   .00000E+00       8.31004E-03
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6.29712E-03        1.68780E-02  1.16922E-04    .00000E+00
   24     1.30412E-01  1.78227E-02    .00000E+00        8.23651E-03
6.24140E-03        2.10525E-02  1.46797E-04    .00000E+00
   25     1.01309E-01  2.37277E-02    .00000E+00        6.53164E-03
4.94950E-03        2.14908E-02  1.54981E-04    .00000E+00
   26     7.46682E-02  3.97555E-02    .00000E+00        3.55409E-03
2.69319E-03        1.58661E-02  1.41295E-04    .00000E+00

 THERMAL 1.38230E-01  1.75602E-02    .00000E+00        3.80030E-02
2.87976E-02        9.16421E-02  6.67340E-04    .00000E+00

  TOTAL  2.78325E-01  7.13947E-03    .00000E+00        9.73269E-02
7.37517E-02        1.16563E-01  6.94862E-04    .00000E+00


0REGION  6     MATERIAL COOLANT     VOLUME  2.344669E+00

                      CROSS-SECTIONS                               FLUXES
                        REACTIONS
 GROUP  DIFFUSION     ABSORPTION   NU*FISSION             RIF            RAF
       TRANSPORT     ABSORPTION   NU*FISSION

    1     2.70642E+00  2.17943E-03    .00000E+00        5.94236E-03
2.53441E-03        7.31884E-04  1.29510E-05    .00000E+00
    2     2.01031E+00  1.01023E-05    .00000E+00        1.97659E-02
8.43015E-03        3.27743E-03  1.99682E-07    .00000E+00
    3     1.31913E+00  3.55934E-06    .00000E+00        1.68054E-02
7.16749E-03        4.24658E-03  5.98161E-08    .00000E+00
    4     7.92566E-01  3.59323E-06    .00000E+00        1.01974E-02
4.34920E-03        4.28879E-03  3.66417E-08    .00000E+00
    5     6.54618E-01  6.63753E-06    .00000E+00        6.75657E-03
2.88167E-03        3.44046E-03  4.48469E-08    .00000E+00
    6     5.56182E-01  1.94203E-05    .00000E+00        9.49486E-03
4.04955E-03        5.69050E-03  1.84393E-07    .00000E+00
    7     5.63543E-01  6.29493E-05    .00000E+00        8.23835E-03
3.51365E-03        4.87295E-03  5.18599E-07    .00000E+00
    8     5.77544E-01  1.71416E-04    .00000E+00        7.98049E-03
3.40367E-03        4.60599E-03  1.36798E-06    .00000E+00
    9     5.80367E-01  3.53087E-04    .00000E+00        3.93757E-03
1.67937E-03        2.26154E-03  1.39030E-06    .00000E+00
   10     5.80777E-01  5.82085E-04    .00000E+00        3.88201E-03
1.65568E-03        2.22806E-03  2.25966E-06    .00000E+00
   11     5.80114E-01  9.59742E-04    .00000E+00        3.81253E-03
1.62604E-03        2.19068E-03  3.65904E-06    .00000E+00
   12     5.78579E-01  1.47308E-03    .00000E+00        2.74490E-03
1.17070E-03        1.58140E-03  4.04345E-06    .00000E+00
   13     5.30274E-01  2.07713E-03    .00000E+00        2.53783E-03
1.08238E-03        1.59530E-03  5.27141E-06    .00000E+00
   14     4.84734E-01  2.83531E-03    .00000E+00        2.43933E-03
1.04037E-03        1.67743E-03  6.91626E-06    .00000E+00
   15     4.56194E-01  3.34008E-03    .00000E+00        2.92374E-04
1.24697E-04        2.13633E-04  9.76550E-07    .00000E+00
   16     4.50003E-01  3.46267E-03    .00000E+00        3.00914E-04
1.28339E-04        2.22897E-04  1.04196E-06    .00000E+00
   17     4.42700E-01  3.60538E-03    .00000E+00        3.77107E-04
1.60836E-04        2.83945E-04  1.35961E-06    .00000E+00
   18     4.19829E-01  4.05921E-03    .00000E+00        1.61864E-03
6.90348E-04        1.28516E-03  6.57038E-06    .00000E+00
   19     3.66366E-01  5.17002E-03    .00000E+00        2.63791E-03
1.12507E-03        2.40007E-03  1.36380E-05    .00000E+00
   20     3.18292E-01  6.29875E-03    .00000E+00        1.05598E-03
4.50374E-04        1.10588E-03  6.65134E-06    .00000E+00
   21     2.76329E-01  7.56452E-03    .00000E+00        2.51454E-03
1.07245E-03        3.03327E-03  1.90213E-05    .00000E+00
   22     2.07862E-01  1.07793E-02    .00000E+00        1.33846E-02
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5.70852E-03        2.14639E-02  1.44276E-04   .00000E+00
   23    1.64184E-01  1.40641E-02   .00000E+00        1.40024E-02
5.97201E-03        2.84282E-02  1.96931E-04   .00000E+00
   24    1.30440E-01  1.78189E-02   .00000E+00        1.37941E-02
5.88320E-03        3.52503E-02  2.45797E-04   .00000E+00
   25    1.01321E-01  2.37240E-02   .00000E+00        1.08857E-02
4.64273E-03        3.58124E-02  2.58251E-04   .00000E+00
   26    7.46679E-02  3.97569E-02   .00000E+00        5.90714E-03
2.51939E-03        2.63707E-02  2.34849E-04   .00000E+00

 THERMAL 1.39045E-01  1.74412E-02   .00000E+00        6.41823E-02
2.73737E-02        1.53865E-01  1.11942E-03   .00000E+00

  TOTAL  2.87583E-01  6.81973E-03   .00000E+00        1.71307E-01
7.30623E-02        1.98559E-01  1.16827E-03   .00000E+00


0REGION  7     MATERIAL CLAD        VOLUME  8.032376E-01

                   CROSS-SECTIONS                            FLUXES
                     REACTIONS
 GROUP  DIFFUSION   ABSORPTION  NU*FISSION          RIF            RAF
        TRANSPORT  ABSORPTION  NU*FISSION

    1    3.16321E+00 -5.37083E-05   .00000E+00        2.08685E-03
2.59804E-03        2.19908E-04 -1.12081E-07   .00000E+00
    2    2.49217E+00  3.74237E-04   .00000E+00        7.01856E-03
8.73784E-03        9.38748E-04  2.62660E-06   .00000E+00
    3    1.67902E+00  5.04506E-04   .00000E+00        5.98988E-03
7.45718E-03        1.18916E-03  3.02193E-06   .00000E+00
    4    1.12257E+00  5.15706E-04   .00000E+00        3.59338E-03
4.47363E-03        1.06701E-03  1.85313E-06   .00000E+00
    5    9.21946E-01  6.62955E-04   .00000E+00        2.35278E-03
2.92912E-03        8.50656E-04  1.55979E-06   .00000E+00
    6    9.25362E-01  1.34549E-03   .00000E+00        3.27566E-03
4.07808E-03        1.17996E-03  4.40737E-06   .00000E+00
    7    8.79884E-01  6.78327E-03   .00000E+00        2.82625E-03
3.51857E-03        1.07069E-03  1.91712E-05   .00000E+00
    8    1.34604E+00  1.24927E-02   .00000E+00        2.72649E-03
3.39438E-03        6.75187E-04  3.40612E-05   .00000E+00
    9    1.59365E+00  1.42310E-04   .00000E+00        1.34499E-03
1.67446E-03        2.81321E-04  1.91404E-07   .00000E+00
   10    1.57942E+00  2.24967E-04   .00000E+00        1.32331E-03
1.64747E-03        2.79283E-04  2.97702E-07   .00000E+00
   11    1.57349E+00  3.85370E-04   .00000E+00        1.29727E-03
1.61506E-03        2.74819E-04  4.99930E-07   .00000E+00
   12    1.57020E+00  6.03067E-04   .00000E+00        9.33175E-04
1.16177E-03        1.98101E-04  5.62767E-07   .00000E+00
   13    1.57409E+00  8.53588E-04   .00000E+00        8.64862E-04
1.07672E-03        1.83145E-04  7.38235E-07   .00000E+00
   14    1.56863E+00  1.16803E-03   .00000E+00        8.31439E-04
1.03511E-03        1.76681E-04  9.71147E-07   .00000E+00
   15    1.56887E+00  1.37786E-03   .00000E+00        9.93992E-05
1.23748E-04        2.11191E-05  1.36958E-07   .00000E+00
   16    1.56929E+00  1.42882E-03   .00000E+00        1.02343E-04
1.27413E-04        2.17387E-05  1.46230E-07   .00000E+00
   17    1.56537E+00  1.48846E-03   .00000E+00        1.28359E-04
1.59803E-04        2.73332E-05  1.91058E-07   .00000E+00
   18    1.56347E+00  1.67631E-03   .00000E+00        5.50740E-04
6.85650E-04        1.17419E-04  9.23210E-07   .00000E+00
   19    1.55973E+00  2.13456E-03   .00000E+00        8.95899E-04
1.11536E-03        1.91464E-04  1.91235E-06   .00000E+00
   20    1.55524E+00  2.60384E-03   .00000E+00        3.57096E-04
4.44571E-04        7.65360E-05  9.29821E-07   .00000E+00
   21    1.55192E+00  3.12774E-03   .00000E+00        8.45543E-04
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1.05267E-03       1.81612E-04  2.64464E-06   .00000E+00
   22     1.54000E+00  4.45409E-03   .00000E+00      4.42584E-03
5.51000E-03       9.57975E-04  1.97131E-05   .00000E+00
   23     1.52869E+00  5.81622E-03   .00000E+00      4.58886E-03
5.71295E-03       1.00061E-03  2.66898E-05   .00000E+00
   24     1.51588E+00  7.36893E-03   .00000E+00      4.49328E-03
5.59396E-03       9.88047E-04  3.31107E-05   .00000E+00
   25     1.49532E+00  9.79734E-03   .00000E+00      3.52731E-03
4.39137E-03       7.86303E-04  3.45583E-05   .00000E+00
   26     1.43463E+00  1.63396E-02   .00000E+00      1.90497E-03
2.37161E-03       4.42614E-04  3.11265E-05   .00000E+00

 THERMAL 1.51626E+00  7.16225E-03   .00000E+00      2.10388E-02
2.61925E-02       4.62516E-03  1.50685E-04   .00000E+00

  TOTAL  1.45263E+00  3.80123E-03   .00000E+00      5.83845E-02
7.26865E-02       1.33974E-02  2.21933E-04   .00000E+00


0REGION  8     MATERIAL COOLANT     VOLUME  4.770473E+00

                   CROSS-SECTIONS                            FLUXES
                     REACTIONS
 GROUP   DIFFUSION   ABSORPTION   NU*FISSION          RIF           RAF
       TRANSPORT    ABSORPTION   NU*FISSION

    1     2.70642E+00  2.17943E-03   .00000E+00      1.31889E-02
2.76469E-03       1.62439E-03  2.87442E-05   .00000E+00
    2     2.01031E+00  1.01023E-05   .00000E+00      4.38597E-02
9.19400E-03       7.27248E-03  4.43085E-07   .00000E+00
    3     1.31913E+00  3.55934E-06   .00000E+00      3.68401E-02
7.72253E-03       9.30919E-03  1.31127E-07   .00000E+00
    4     7.92566E-01  3.59323E-06   .00000E+00      2.18626E-02
4.58291E-03       9.19488E-03  7.85575E-08   .00000E+00
    5     6.54618E-01  6.63753E-06   .00000E+00      1.41836E-02
2.97321E-03       7.22233E-03  9.41441E-08   .00000E+00
    6     5.56182E-01  1.94203E-05   .00000E+00      1.96088E-02
4.11045E-03       1.17520E-02  3.80809E-07   .00000E+00
    7     5.63543E-01  6.29493E-05   .00000E+00      1.68443E-02
3.53095E-03       9.96334E-03  1.06034E-06   .00000E+00
    8     5.77544E-01  1.71416E-04   .00000E+00      1.62164E-02
3.39932E-03       9.35939E-03  2.77975E-06   .00000E+00
    9     5.80367E-01  3.53087E-04   .00000E+00      7.96441E-03
1.66952E-03       4.57435E-03  2.81213E-06   .00000E+00
   10     5.80777E-01  5.82085E-04   .00000E+00      7.81693E-03
1.63861E-03       4.48648E-03  4.55011E-06   .00000E+00
   11     5.80114E-01  9.59742E-04   .00000E+00      7.64728E-03
1.60305E-03       4.39413E-03  7.33942E-06   .00000E+00
   12     5.78579E-01  1.47308E-03   .00000E+00      5.49464E-03
1.15180E-03       3.16559E-03  8.09402E-06   .00000E+00
   13     5.30274E-01  2.07713E-03   .00000E+00      5.11061E-03
1.07130E-03       3.21256E-03  1.06154E-05   .00000E+00
   14     4.84734E-01  2.83531E-03   .00000E+00      4.91201E-03
1.02967E-03       3.37780E-03  1.39271E-05   .00000E+00
   15     4.56194E-01  3.34008E-03   .00000E+00      5.84635E-04
1.22553E-04       4.27183E-04  1.95273E-06   .00000E+00
   16     4.50003E-01  3.46267E-03   .00000E+00      6.02934E-04
1.26389E-04       4.46615E-04  2.08776E-06   .00000E+00
   17     4.42700E-01  3.60538E-03   .00000E+00      7.56591E-04
1.58599E-04       5.69679E-04  2.72780E-06   .00000E+00
   18     4.19829E-01  4.05921E-03   .00000E+00      3.24797E-03
6.80849E-04       2.57880E-03  1.31842E-05   .00000E+00
   19     3.66366E-01  5.17002E-03   .00000E+00      5.26776E-03
1.10424E-03       4.79281E-03  2.72344E-05   .00000E+00
   20     3.18292E-01  6.29875E-03   .00000E+00      2.08904E-03
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4.37911E-04       2.18776E-03  1.31584E-05   .00000E+00
   21    2.76329E-01  7.56452E-03   .00000E+00       4.91848E-03
1.03102E-03       5.93312E-03  3.72059E-05   .00000E+00
   22    2.07862E-01  1.07793E-02   .00000E+00       2.51060E-02
5.26279E-03       4.02606E-02  2.70624E-04   .00000E+00
   23    1.64184E-01  1.40641E-02   .00000E+00       2.56534E-02
5.37754E-03       5.20826E-02  3.60792E-04   .00000E+00
   24    1.30440E-01  1.78189E-02   .00000E+00       2.48382E-02
5.20665E-03       6.34729E-02  4.42589E-04   .00000E+00
   25    1.01321E-01  2.37240E-02   .00000E+00       1.92733E-02
4.04012E-03       6.34065E-02  4.57239E-04   .00000E+00
   26    7.46679E-02  3.97569E-02   .00000E+00       1.03011E-02
2.15934E-03       4.59861E-02  4.09538E-04   .00000E+00

 THERMAL 1.40762E-01  1.71854E-02   .00000E+00       1.17447E-01
2.46196E-02       2.78122E-01  2.01838E-03   .00000E+00

  TOTAL  3.09200E-01  6.15760E-03   .00000E+00       3.44190E-01
7.21500E-02       3.71054E-01  2.11938E-03   .00000E+00


0REGION  9     MATERIAL FUEL_1      VOLUME  6.662875E-01

                     CROSS-SECTIONS                            FLUXES
                       REACTIONS
 GROUP  DIFFUSION    ABSORPTION   NU*FISSION           RIF            RAF
       TRANSPORT    ABSORPTION   NU*FISSION

    1    2.32885E+00  1.78377E-02  5.28855E-02       2.89428E-03
4.34389E-03       4.14265E-04  5.16274E-05  1.53066E-04
    2    1.91875E+00  1.57905E-02  3.97823E-02       9.57893E-03
1.43766E-02       1.66410E-03  1.51256E-04  3.81072E-04
    3    1.18870E+00  8.57947E-03  1.48435E-02       7.57690E-03
1.13718E-02       2.12470E-03  6.50058E-05  1.12468E-04
    4    8.16399E-01  9.08466E-03  1.44654E-02       3.91168E-03
5.87086E-03       1.59713E-03  3.55363E-05  5.65841E-05
    5    8.11678E-01  1.23757E-02  1.74729E-02       2.20662E-03
3.31181E-03       9.06197E-04  2.73085E-05  3.85561E-05
    6    7.06212E-01  2.23040E-02  2.45211E-02       2.79481E-03
4.19459E-03       1.31915E-03  6.23352E-05  6.85317E-05
    7    6.33971E-01  4.98772E-02  5.54151E-02       2.33040E-03
3.49759E-03       1.22529E-03  1.16234E-04  1.29139E-04
    8    5.71165E-01  1.29766E-01  1.54831E-01       2.20104E-03
3.30344E-03       1.28453E-03  2.85620E-04  3.40790E-04
    9    4.83010E-01  2.42135E-01  2.40387E-01       1.05688E-03
1.58623E-03       7.29373E-04  2.55908E-04  2.54062E-04
   10    3.98856E-01  3.85543E-01  4.87313E-01       1.00653E-03
1.51066E-03       8.41182E-04  3.88061E-04  4.90495E-04
   11    3.78054E-01  4.94895E-01  5.51727E-01       9.63439E-04
1.44598E-03       8.49472E-04  4.76801E-04  5.31555E-04
   12    3.77531E-01  5.09213E-01  1.61345E-01       6.89516E-04
1.03486E-03       6.08793E-04  3.51110E-04  1.11250E-04
   13    6.68706E-01  1.23913E-01  2.00238E-01       6.92835E-04
1.03984E-03       3.45361E-04  8.58510E-05  1.38732E-04
   14    6.13009E-01  1.58972E-01  2.55781E-01       6.61375E-04
9.92627E-04       3.59633E-04  1.05140E-04  1.69167E-04
   15    3.30150E-01  6.22014E-01  1.20522E+00       7.21269E-05
1.08252E-04       7.28224E-05  4.48639E-05  8.69290E-05
   16    4.05488E-01  4.34493E-01  8.86229E-01       7.67869E-05
1.15246E-04       6.31230E-05  3.33634E-05  6.80508E-05
   17    4.60036E-01  3.35121E-01  7.02402E-01       9.81538E-05
1.47315E-04       7.11204E-05  3.28934E-05  6.89435E-05
   18    4.74286E-01  3.11396E-01  6.58316E-01       4.23401E-04
6.35462E-04       2.97571E-04  1.31845E-04  2.78731E-04
   19    3.62015E-01  5.26401E-01  1.08735E+00       6.57975E-04
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9.87523E-04       6.05844E-04  3.46358E-04  7.15447E-04
   20    2.31685E-01  1.04238E+00  2.03659E+00       2.35928E-04
3.54093E-04       3.39437E-04  2.45927E-04  4.80489E-04
   21    2.29393E-01  1.05628E+00  2.06603E+00       5.49188E-04
8.24251E-04       7.98032E-04  5.80096E-04  1.13464E-03
   22    1.89976E-01  1.35500E+00  2.75817E+00       2.62266E-03
3.93623E-03       4.60173E-03  3.55371E-03  7.23373E-03
   23    1.45443E-01  1.89028E+00  3.87453E+00       2.41995E-03
3.63199E-03       5.54618E-03  4.57438E-03  9.37619E-03
   24    1.14148E-01  2.51666E+00  5.16504E+00       2.09441E-03
3.14340E-03       6.11604E-03  5.27090E-03  1.08177E-02
   25    8.57929E-02  3.47833E+00  7.14022E+00       1.38792E-03
2.08306E-03       5.39251E-03  4.82764E-03  9.91004E-03
   26    5.39375E-02  5.75904E+00  1.18196E+01       5.41308E-04
8.12424E-04       3.34528E-03  3.11742E-03  6.39807E-03

 THERMAL 1.30982E-01  2.14252E+00  4.38337E+00       1.05093E-02
1.57730E-02       2.67451E-02  2.25164E-02  4.60663E-02

  TOTAL  3.99377E-01  5.06929E-01  9.95967E-01       4.97450E-02
7.46600E-02       4.15189E-02  2.52172E-02  4.95444E-02


0REGION 10     MATERIAL CLAD        VOLUME  6.792079E-01

                     CROSS-SECTIONS                          FLUXES
                       REACTIONS
 GROUP   DIFFUSION    ABSORPTION   NU*FISSION          RIF          RAF
        TRANSPORT    ABSORPTION  NU*FISSION

    1    3.16321E+00 -5.37083E-05   .00000E+00       2.34888E-03
3.45826E-03       2.47520E-04 -1.26154E-07   .00000E+00
    2    2.49217E+00  3.74237E-04   .00000E+00       7.82032E-03
1.15139E-02       1.04599E-03  2.92665E-06   .00000E+00
    3    1.67902E+00  5.04506E-04   .00000E+00       6.37241E-03
9.38212E-03       1.26511E-03  3.21492E-06   .00000E+00
    4    1.12257E+00  5.15706E-04   .00000E+00       3.51576E-03
5.17626E-03       1.04396E-03  1.81310E-06   .00000E+00
    5    9.21946E-01  6.62955E-04   .00000E+00       2.12785E-03
3.13284E-03       7.69332E-04  1.41067E-06   .00000E+00
    6    9.25362E-01  1.34549E-03   .00000E+00       2.82017E-03
4.15214E-03       1.01588E-03  3.79450E-06   .00000E+00
    7    8.79884E-01  6.78327E-03   .00000E+00       2.38813E-03
3.51605E-03       9.04713E-04  1.61993E-05   .00000E+00
    8    1.34604E+00  1.24927E-02   .00000E+00       2.27788E-03
3.35373E-03       5.64092E-04  2.84568E-05   .00000E+00
    9    1.59365E+00  1.42310E-04   .00000E+00       1.10898E-03
1.63276E-03       2.31958E-04  1.57819E-07   .00000E+00
   10    1.57942E+00  2.24967E-04   .00000E+00       1.07479E-03
1.58241E-03       2.26832E-04  2.41792E-07   .00000E+00
   11    1.57349E+00  3.85370E-04   .00000E+00       1.04185E-03
1.53391E-03       2.20708E-04  4.01496E-07   .00000E+00
   12    1.57020E+00  6.03067E-04   .00000E+00       7.47186E-04
1.10008E-03       1.58618E-04  4.50603E-07   .00000E+00
   13    1.57409E+00  8.53588E-04   .00000E+00       7.17017E-04
1.05567E-03       1.51837E-04  6.12037E-07   .00000E+00
   14    1.56863E+00  1.16803E-03   .00000E+00       6.87105E-04
1.01163E-03       1.46010E-04  8.02559E-07   .00000E+00
   15    1.56887E+00  1.37786E-03   .00000E+00       7.89640E-05
1.16259E-04       1.67773E-05  1.08801E-07   .00000E+00
   16    1.56929E+00  1.42882E-03   .00000E+00       8.23966E-05
1.21313E-04       1.75019E-05  1.17730E-07   .00000E+00
   17    1.56537E+00  1.48846E-03   .00000E+00       1.04179E-04
1.53383E-04       2.21841E-05  1.55066E-07   .00000E+00
   18    1.56347E+00  1.67631E-03   .00000E+00       4.47975E-04
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6.59555E-04       9.55089E-05  7.50944E-07   .00000E+00
   19     1.55973E+00  2.13456E-03   .00000E+00        7.13813E-04
1.05095E-03       1.52550E-04  1.52368E-06   .00000E+00
   20     1.55524E+00  2.60384E-03   .00000E+00        2.72300E-04
4.00908E-04       5.83618E-05  7.09026E-07   .00000E+00
   21     1.55192E+00  3.12774E-03   .00000E+00        6.35779E-04
9.36059E-04       1.36557E-04  1.98855E-06   .00000E+00
   22     1.54000E+00  4.45409E-03   .00000E+00        3.14029E-03
4.62345E-03       6.79716E-04  1.39871E-05   .00000E+00
   23     1.52869E+00  5.81622E-03   .00000E+00        3.08744E-03
4.54565E-03       6.73223E-04  1.79572E-05   .00000E+00
   24     1.51588E+00  7.36893E-03   .00000E+00        2.87797E-03
4.23724E-03       6.32849E-04  2.12075E-05   .00000E+00
   25     1.49532E+00  9.79734E-03   .00000E+00        2.13319E-03
3.14070E-03       4.75526E-04  2.08996E-05   .00000E+00
   26     1.43463E+00  1.63396E-02   .00000E+00        1.06707E-03
1.57105E-03       2.47930E-04  1.74355E-05   .00000E+00

 THERMAL 1.51882E+00  6.87172E-03   .00000E+00        1.39278E-02
2.05060E-02       3.05671E-03  9.57082E-05   .00000E+00

  TOTAL  1.47869E+00  3.16357E-03   .00000E+00        4.96897E-02
7.31583E-02       1.12012E-02  1.57197E-04   .00000E+00


0REGION 11     MATERIAL COOLANT     VOLUME  6.456087E+00

                     CROSS-SECTIONS                           FLUXES
                       REACTIONS
 GROUP   DIFFUSION    ABSORPTION   NU*FISSION          RIF            RAF
        TRANSPORT    ABSORPTION  NU*FISSION

    1     2.70642E+00  2.17943E-03   .00000E+00        1.79063E-02
2.77356E-03       2.20542E-03  3.90256E-05   .00000E+00
    2     2.01031E+00  1.01023E-05   .00000E+00        5.94285E-02
9.20504E-03       9.85397E-03  6.00366E-07   .00000E+00
    3     1.31913E+00  3.55934E-06   .00000E+00        4.98568E-02
7.72245E-03       1.25984E-02  1.77457E-07   .00000E+00
    4     7.92566E-01  3.59323E-06   .00000E+00        2.96402E-02
4.59105E-03       1.24659E-02  1.06504E-07   .00000E+00
    5     6.54618E-01  6.63753E-06   .00000E+00        1.92479E-02
2.98135E-03       9.80106E-03  1.27758E-07   .00000E+00
    6     5.56182E-01  1.94203E-05   .00000E+00        2.66122E-02
4.12203E-03       1.59493E-02  5.16818E-07   .00000E+00
    7     5.63543E-01  6.29493E-05   .00000E+00        2.28404E-02
3.53781E-03       1.35100E-02  1.43779E-06   .00000E+00
    8     5.77544E-01  1.71416E-04   .00000E+00        2.19658E-02
3.40233E-03       1.26777E-02  3.76528E-06   .00000E+00
    9     5.80367E-01  3.53087E-04   .00000E+00        1.07742E-02
1.66884E-03       6.18816E-03  3.80423E-06   .00000E+00
   10     5.80777E-01  5.82085E-04   .00000E+00        1.05631E-02
1.63614E-03       6.06261E-03  6.14860E-06   .00000E+00
   11     5.80114E-01  9.59742E-04   .00000E+00        1.03225E-02
1.59889E-03       5.93133E-03  9.90697E-06   .00000E+00
   12     5.78579E-01  1.47308E-03   .00000E+00        7.41018E-03
1.14778E-03       4.26918E-03  1.09158E-05   .00000E+00
   13     5.30274E-01  2.07713E-03   .00000E+00        6.88935E-03
1.06711E-03       4.33068E-03  1.43101E-05   .00000E+00
   14     4.84734E-01  2.83531E-03   .00000E+00        6.61902E-03
1.02524E-03       4.55164E-03  1.87670E-05   .00000E+00
   15     4.56194E-01  3.34008E-03   .00000E+00        7.87479E-04
1.21975E-04       5.75398E-04  2.63024E-06   .00000E+00
   16     4.50003E-01  3.46267E-03   .00000E+00        8.12187E-04
1.25802E-04       6.01616E-04  2.81234E-06   .00000E+00
   17     4.42700E-01  3.60538E-03   .00000E+00        1.01907E-03
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1.57847E-04       7.67316E-04  3.67415E-06   .00000E+00
   18    4.19829E-01  4.05921E-03   .00000E+00      4.37437E-03
6.77557E-04       3.47313E-03  1.77564E-05   .00000E+00
   19    3.66366E-01  5.17002E-03   .00000E+00      7.08926E-03
1.09807E-03       6.45008E-03  3.66516E-05   .00000E+00
   20    3.18292E-01  6.29875E-03   .00000E+00      2.80678E-03
4.34749E-04       2.93941E-03  1.76792E-05   .00000E+00
   21    2.76329E-01  7.56452E-03   .00000E+00      6.55911E-03
1.01596E-03       7.91220E-03  4.96165E-05   .00000E+00
   22    2.07862E-01  1.07793E-02   .00000E+00      3.27032E-02
5.06548E-03       5.24436E-02  3.52516E-04   .00000E+00
   23    1.64184E-01  1.40641E-02   .00000E+00      3.31266E-02
5.13107E-03       6.72551E-02  4.65897E-04   .00000E+00
   24    1.30440E-01  1.78189E-02   .00000E+00      3.19331E-02
4.94620E-03       8.16035E-02  5.69012E-04   .00000E+00
   25    1.01321E-01  2.37240E-02   .00000E+00      2.47019E-02
3.82613E-03       8.12658E-02  5.86027E-04   .00000E+00
   26    7.46679E-02  3.97569E-02   .00000E+00      1.31727E-02
2.04035E-03       5.88057E-02  5.23706E-04   .00000E+00

 THERMAL 1.41347E-01  1.71021E-02   .00000E+00      1.52093E-01
2.35580E-02       3.58676E-01  2.60110E-03   .00000E+00

  TOTAL  3.15909E-01  5.96214E-03   .00000E+00      4.59162E-01
7.11208E-02       4.84488E-01  2.73759E-03   .00000E+00


0REGION 12     MATERIAL FUEL_2      VOLUME  2.220972E-01

                    CROSS-SECTIONS                             FLUXES
                      REACTIONS
 GROUP  DIFFUSION    ABSORPTION   NU*FISSION          RIF           RAF
       TRANSPORT    ABSORPTION   NU*FISSION

    1    2.32885E+00  1.78377E-02  5.28855E-02      9.89075E-04
4.45334E-03       1.41568E-04  1.76428E-05  5.23077E-05
    2    1.91875E+00  1.57905E-02  3.97823E-02      3.27602E-03
1.47504E-02       5.69125E-04  5.17300E-05  1.30327E-04
    3    1.18870E+00  8.57947E-03  1.48435E-02      2.58721E-03
1.16490E-02       7.25500E-04  2.21969E-05  3.84033E-05
    4    8.16399E-01  9.08466E-03  1.44654E-02      1.32524E-03
5.96692E-03       5.41090E-04  1.20393E-05  1.91701E-05
    5    8.11678E-01  1.23757E-02  1.74729E-02      7.41077E-04
3.33672E-03       3.04340E-04  9.17135E-06  1.29488E-05
    6    7.06212E-01  2.23040E-02  2.45211E-02      9.33029E-04
4.20099E-03       4.40392E-04  2.08102E-05  2.28789E-05
    7    6.33971E-01  4.98772E-02  5.54151E-02      7.76163E-04
3.49470E-03       4.08096E-04  3.87129E-05  4.30112E-05
    8    5.71165E-01  1.29766E-01  1.54831E-01      7.31421E-04
3.29325E-03       4.26859E-04  9.49135E-05  1.13247E-04
    9    4.83010E-01  2.42135E-01  2.40387E-01      3.50410E-04
1.57773E-03       2.41823E-04  8.48463E-05  8.42340E-05
   10    3.98856E-01  3.85543E-01  4.87313E-01      3.32664E-04
1.49783E-03       2.78015E-04  1.28256E-04  1.62111E-04
   11    3.78054E-01  4.94895E-01  5.51727E-01      3.17678E-04
1.43036E-03       2.80100E-04  1.57217E-04  1.75272E-04
   12    3.77531E-01  5.09213E-01  1.61345E-01      2.27204E-04
1.02299E-03       2.00605E-04  1.15695E-04  3.66582E-05
   13    6.68706E-01  1.23913E-01  2.00238E-01      2.29795E-04
1.03466E-03       1.14547E-04  2.84745E-05  4.60138E-05
   14    6.13009E-01  1.58972E-01  2.55781E-01      2.19203E-04
9.86968E-04       1.19195E-04  3.48471E-05  5.60680E-05
   15    3.30150E-01  6.22014E-01  1.20522E+00      2.37424E-05
1.06901E-04       2.39713E-05  1.47681E-05  2.86149E-05
   16    4.05488E-01  4.34493E-01  8.86229E-01      2.53308E-05
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1.14053E-04       2.08233E-05  1.10061E-05  2.24489E-05
   17     4.60036E-01  3.35121E-01  7.02402E-01        3.24266E-05
1.46002E-04       2.34957E-05  1.08668E-05  2.27765E-05
   18     4.74286E-01  3.11396E-01  6.58316E-01        1.39935E-04
6.30061E-04       9.83478E-05  4.35751E-05  9.21214E-05
   19     3.62015E-01  5.26401E-01  1.08735E+00        2.16777E-04
9.76044E-04       1.99602E-04  1.14111E-04  2.35712E-04
   20     2.31685E-01  1.04238E+00  2.03659E+00        7.72161E-05
3.47668E-04       1.11093E-04  8.04886E-05  1.57258E-04
   21     2.29393E-01  1.05628E+00  2.06603E+00        1.79179E-04
8.06757E-04       2.60367E-04  1.89263E-04  3.70189E-04
   22     1.89976E-01  1.35500E+00  2.75817E+00        8.48099E-04
3.81859E-03       1.48808E-03  1.14918E-03  2.33920E-03
   23     1.45443E-01  1.89028E+00  3.87453E+00        7.81137E-04
3.51709E-03       1.79025E-03  1.47657E-03  3.02654E-03
   24     1.14148E-01  2.51666E+00  5.16504E+00        6.77497E-04
3.05045E-03       1.97841E-03  1.70503E-03  3.49930E-03
   25     8.57929E-02  3.47833E+00  7.14022E+00        4.51294E-04
2.03196E-03       1.75342E-03  1.56975E-03  3.22234E-03
   26     5.39375E-02  5.75904E+00  1.18196E+01        1.77403E-04
7.98761E-04       1.09635E-03  1.02167E-03  2.09683E-03

 THERMAL 1.30935E-01  2.14342E+00  4.38519E+00        3.40860E-03
1.53473E-02       8.67757E-03  7.30605E-03  1.49474E-02

  TOTAL  4.07423E-01  4.92182E-01  9.66385E-01        1.66662E-02
7.50402E-02       1.36355E-02  8.20282E-03  1.61060E-02


0REGION 13      MATERIAL CLAD         VOLUME  2.264031E-01

                     CROSS-SECTIONS                                  FLUXES
                       REACTIONS
 GROUP   DIFFUSION     ABSORPTION   NU*FISSION            RIF            RAF
        TRANSPORT     ABSORPTION   NU*FISSION

    1     3.16321E+00 -5.37083E-05   .00000E+00        8.17106E-04
3.60908E-03       8.61051E-05 -4.38854E-08   .00000E+00
    2     2.49217E+00  3.74237E-04   .00000E+00        2.71912E-03
1.20101E-02       3.63689E-04  1.01759E-06   .00000E+00
    3     1.67902E+00  5.04506E-04   .00000E+00        2.20214E-03
9.72665E-03       4.37188E-04  1.11100E-06   .00000E+00
    4     1.12257E+00  5.15706E-04   .00000E+00        1.19789E-03
5.29098E-03       3.55701E-04  6.17762E-07   .00000E+00
    5     9.21946E-01  6.62955E-04   .00000E+00        7.15879E-04
3.16197E-03       2.58829E-04  4.74595E-07   .00000E+00
    6     9.25362E-01  1.34549E-03   .00000E+00        9.41705E-04
4.15942E-03       3.39220E-04  1.26705E-06   .00000E+00
    7     8.79884E-01  6.78327E-03   .00000E+00        7.95381E-04
3.51312E-03       3.01321E-04  5.39528E-06   .00000E+00
    8     1.34604E+00  1.24927E-02   .00000E+00        7.56957E-04
3.34341E-03       1.87452E-04  9.45641E-06   .00000E+00
    9     1.59365E+00  1.42310E-04   .00000E+00        3.67680E-04
1.62401E-03       7.69050E-05  5.23243E-08   .00000E+00
   10     1.57942E+00  2.24967E-04   .00000E+00        3.55224E-04
1.56899E-03       7.49694E-05  7.99137E-08   .00000E+00
   11     1.57349E+00  3.85370E-04   .00000E+00        3.43537E-04
1.51737E-03       7.27761E-05  1.32389E-07   .00000E+00
   12     1.57020E+00  6.03067E-04   .00000E+00        2.46214E-04
1.08750E-03       5.22682E-05  1.48484E-07   .00000E+00
   13     1.57409E+00  8.53588E-04   .00000E+00        2.37839E-04
1.05051E-03       5.03653E-05  2.03017E-07   .00000E+00
   14     1.56863E+00  1.16803E-03   .00000E+00        2.27743E-04
1.00592E-03       4.83954E-05  2.66011E-07   .00000E+00
   15     1.56887E+00  1.37786E-03   .00000E+00        2.59920E-05
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1.14804E-04       5.52245E-06  3.58133E-08   .00000E+00
   16    1.56929E+00  1.42882E-03   .00000E+00        2.71831E-05
1.20065E-04       5.77396E-06  3.88398E-08   .00000E+00
   17    1.56537E+00  1.48846E-03   .00000E+00        3.44199E-05
1.52029E-04       7.32947E-06  5.12327E-08   .00000E+00
   18    1.56347E+00  1.67631E-03   .00000E+00        1.48071E-04
6.54017E-04       3.15691E-05  2.48213E-07   .00000E+00
   19    1.55973E+00  2.13456E-03   .00000E+00        2.35221E-04
1.03895E-03       5.02695E-05  5.02092E-07   .00000E+00
   20    1.55524E+00  2.60384E-03   .00000E+00        8.91639E-05
3.93828E-04       1.91104E-05  2.32169E-07   .00000E+00
   21    1.55192E+00  3.12774E-03   .00000E+00        2.07591E-04
9.16908E-04       4.45879E-05  6.49291E-07   .00000E+00
   22    1.54000E+00  4.45409E-03   .00000E+00        1.01694E-03
4.49174E-03       2.20118E-04  4.52955E-06   .00000E+00
   23    1.52869E+00  5.81622E-03   .00000E+00        9.98626E-04
4.41083E-03       2.17753E-04  5.80823E-06   .00000E+00
   24    1.51588E+00  7.36893E-03   .00000E+00        9.33313E-04
4.12235E-03       2.05230E-04  6.87752E-06   .00000E+00
   25    1.49532E+00  9.79734E-03   .00000E+00        6.95617E-04
3.07247E-03       1.55066E-04  6.81520E-06   .00000E+00
   26    1.43463E+00  1.63396E-02   .00000E+00        3.50686E-04
1.54894E-03       8.14809E-05  5.73008E-06   .00000E+00

 THERMAL 1.51875E+00  6.87940E-03   .00000E+00        4.52716E-03
1.99960E-02       9.93615E-04  3.11441E-05   .00000E+00

  TOTAL  1.48371E+00  3.09795E-03   .00000E+00        1.66873E-02
7.37059E-02       3.74900E-03  5.16962E-05   .00000E+00


0REGION 14     MATERIAL COOLANT     VOLUME  3.954837E+00

                        CROSS-SECTIONS                              FLUXES
                          REACTIONS
 GROUP   DIFFUSION     ABSORPTION    NU*FISSION          RIF             RAF
        TRANSPORT     ABSORPTION    NU*FISSION

    1    2.70642E+00  2.17943E-03   .00000E+00        1.06122E-02
2.68336E-03       1.30704E-03  2.31286E-05   .00000E+00
    2    2.01031E+00  1.01023E-05   .00000E+00        3.51803E-02
8.89550E-03       5.83332E-03  3.55403E-07   .00000E+00
    3    1.31913E+00  3.55934E-06   .00000E+00        2.96162E-02
7.48861E-03       7.48377E-03  1.05414E-07   .00000E+00
    4    7.92566E-01  3.59323E-06   .00000E+00        1.77921E-02
4.49882E-03       7.48291E-03  6.39311E-08   .00000E+00
    5    6.54618E-01  6.63753E-06   .00000E+00        1.16687E-02
2.95048E-03       5.94171E-03  7.74511E-08   .00000E+00
    6    5.56182E-01  1.94203E-05   .00000E+00        1.62419E-02
4.10686E-03       9.73419E-03  3.15424E-07   .00000E+00
    7    5.63543E-01  6.29493E-05   .00000E+00        1.39854E-02
3.53629E-03       8.27232E-03  8.80374E-07   .00000E+00
    8    5.77544E-01  1.71416E-04   .00000E+00        1.34673E-02
3.40526E-03       7.77271E-03  2.30850E-06   .00000E+00
    9    5.80367E-01  3.53087E-04   .00000E+00        6.61387E-03
1.67235E-03       3.79867E-03  2.33527E-06   .00000E+00
   10    5.80777E-01  5.82085E-04   .00000E+00        6.49042E-03
1.64113E-03       3.72514E-03  3.77797E-06   .00000E+00
   11    5.80114E-01  9.59742E-04   .00000E+00        6.34699E-03
1.60487E-03       3.64698E-03  6.09147E-06   .00000E+00
   12    5.78579E-01  1.47308E-03   .00000E+00        4.55644E-03
1.15212E-03       2.62507E-03  6.71198E-06   .00000E+00
   13    5.30274E-01  2.07713E-03   .00000E+00        4.22032E-03
1.06713E-03       2.65291E-03  8.76614E-06   .00000E+00
   14    4.84734E-01  2.83531E-03   .00000E+00        4.05333E-03
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1.02490E-03       2.78732E-03  1.14925E-05   .00000E+00
   15    4.56194E-01  3.34008E-03   .00000E+00        4.83968E-04
1.22374E-04       3.53627E-04  1.61649E-06   .00000E+00
   16    4.50003E-01  3.46267E-03   .00000E+00        4.98570E-04
1.26066E-04       3.69309E-04  1.72638E-06   .00000E+00
   17    4.42700E-01  3.60538E-03   .00000E+00        6.25053E-04
1.58048E-04       4.70637E-04  2.25355E-06   .00000E+00
   18    4.19829E-01  4.05921E-03   .00000E+00        2.68232E-03
6.78239E-04       2.12969E-03  1.08881E-05   .00000E+00
   19    3.66366E-01  5.17002E-03   .00000E+00        4.35540E-03
1.10128E-03       3.96271E-03  2.25175E-05   .00000E+00
   20    3.18292E-01  6.29875E-03   .00000E+00        1.73024E-03
4.37500E-04       1.81200E-03  1.08984E-05   .00000E+00
   21    2.76329E-01  7.56452E-03   .00000E+00        4.03450E-03
1.02014E-03       4.86678E-03  3.05190E-05   .00000E+00
   22    2.07862E-01  1.07793E-02   .00000E+00        2.01354E-02
5.09135E-03       3.22897E-02  2.17045E-04   .00000E+00
   23    1.64184E-01  1.40641E-02   .00000E+00        2.05558E-02
5.19765E-03       4.17333E-02  2.89100E-04   .00000E+00
   24    1.30440E-01  1.78189E-02   .00000E+00        1.99988E-02
5.05680E-03       5.11061E-02  3.56357E-04   .00000E+00
   25    1.01321E-01  2.37240E-02   .00000E+00        1.56553E-02
3.95851E-03       5.15037E-02  3.71406E-04   .00000E+00
   26    7.46679E-02  3.97569E-02   .00000E+00        8.45868E-03
2.13882E-03       3.77614E-02  3.36291E-04   .00000E+00

 THERMAL 1.40606E-01  1.72151E-02   .00000E+00        9.49242E-02
2.40020E-02       2.25036E-01  1.63413E-03   .00000E+00

  TOTAL  3.09708E-01  6.13094E-03   .00000E+00        2.80060E-01
7.08145E-02       3.01423E-01  1.71703E-03   .00000E+00


0REGION 15     MATERIAL MODERATOR    VOLUME  5.260170E+00

                    CROSS-SECTIONS                          FLUXES
                      REACTIONS
 GROUP   DIFFUSION    ABSORPTION   NU*FISSION         RIF          RAF
       TRANSPORT    ABSORPTION   NU*FISSION

    1    2.74822E+00  2.21855E-03   .00000E+00        1.40978E-02
2.68010E-03       1.70993E-03  3.12767E-05   .00000E+00
    2    2.00863E+00  9.66120E-06   .00000E+00        4.67060E-02
8.87918E-03       7.75090E-03  4.51236E-07   .00000E+00
    3    1.31914E+00  3.55843E-06   .00000E+00        3.93289E-02
7.47674E-03       9.93802E-03  1.39949E-07   .00000E+00
    4    7.92343E-01  3.59826E-06   .00000E+00        2.36494E-02
4.49593E-03       9.94914E-03  8.50967E-08   .00000E+00
    5    6.53939E-01  6.66158E-06   .00000E+00        1.55126E-02
2.94907E-03       7.90726E-03  1.03338E-07   .00000E+00
    6    5.56006E-01  1.95488E-05   .00000E+00        2.15858E-02
4.10364E-03       1.29410E-02  4.21976E-07   .00000E+00
    7    5.63650E-01  6.32557E-05   .00000E+00        1.85905E-02
3.53421E-03       1.09941E-02  1.17596E-06   .00000E+00
    8    5.77572E-01  1.72185E-04   .00000E+00        1.78916E-02
3.40133E-03       1.03257E-02  3.08066E-06   .00000E+00
    9    5.80369E-01  3.53483E-04   .00000E+00        8.79024E-03
1.67109E-03       5.04865E-03  3.10720E-06   .00000E+00
   10    5.80777E-01  5.82735E-04   .00000E+00        8.61849E-03
1.63844E-03       4.94653E-03  5.02230E-06   .00000E+00
   11    5.80112E-01  9.60720E-04   .00000E+00        8.42266E-03
1.60122E-03       4.83968E-03  8.09183E-06   .00000E+00
   12    5.78577E-01  1.47362E-03   .00000E+00        6.04492E-03
1.14919E-03       3.48264E-03  8.90793E-06   .00000E+00
   13    5.30259E-01  2.07768E-03   .00000E+00        5.59687E-03
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1.06401E-03        3.51833E-03  1.16285E-05   .00000E+00
   14    4.84689E-01  2.83608E-03   .00000E+00         5.37366E-03
1.02158E-03        3.69561E-03  1.52402E-05   .00000E+00
   15    4.56193E-01  3.34009E-03   .00000E+00         6.41604E-04
1.21974E-04        4.68810E-04  2.14302E-06   .00000E+00
   16    4.50003E-01  3.46266E-03   .00000E+00         6.60909E-04
1.25644E-04        4.89559E-04  2.28851E-06   .00000E+00
   17    4.42698E-01  3.60542E-03   .00000E+00         8.28506E-04
1.57506E-04        6.23831E-04  2.98711E-06   .00000E+00
   18    4.19805E-01  4.05969E-03   .00000E+00         3.55469E-03
6.75776E-04        2.82250E-03  1.44310E-05   .00000E+00
   19    3.66218E-01  5.17320E-03   .00000E+00         5.76797E-03
1.09654E-03        5.25003E-03  2.98389E-05   .00000E+00
   20    3.18091E-01  6.30426E-03   .00000E+00         2.28514E-03
4.34424E-04        2.39465E-03  1.44061E-05   .00000E+00
   21    2.74446E-01  7.62670E-03   .00000E+00         5.29843E-03
1.00727E-03        6.43531E-03  4.04095E-05   .00000E+00
   22    2.07048E-01  1.08270E-02   .00000E+00         2.63180E-02
5.00327E-03        4.23702E-02  2.84945E-04   .00000E+00
   23    1.64120E-01  1.40700E-02   .00000E+00         2.68685E-02
5.10792E-03        5.45709E-02  3.78039E-04   .00000E+00
   24    1.30412E-01  1.78227E-02   .00000E+00         2.61212E-02
4.96585E-03        6.67657E-02  4.65552E-04   .00000E+00
   25    1.01309E-01  2.37277E-02   .00000E+00         2.04560E-02
3.88884E-03        6.73056E-02  4.85372E-04   .00000E+00
   26    7.46682E-02  3.97555E-02   .00000E+00         1.10591E-02
2.10242E-03        4.93700E-02  4.39660E-04   .00000E+00

 THERMAL 1.40566E-01  1.72195E-02   .00000E+00         1.24174E-01
2.36065E-02        2.94462E-01  2.13822E-03   .00000E+00

  TOTAL  3.11574E-01  6.07671E-03   .00000E+00         3.70070E-01
7.03532E-02        3.95915E-01  2.24881E-03   .00000E+00


0REGION 16     MATERIAL COOLANT     VOLUME  5.152802E+00

                    CROSS-SECTIONS                                 FLUXES
                      REACTIONS
 GROUP    DIFFUSION    ABSORPTION   NU*FISSION           RIF             RAF
         TRANSPORT    ABSORPTION   NU*FISSION

    1    2.70642E+00  2.17943E-03   .00000E+00         1.38753E-02
2.69277E-03        1.70894E-03  3.02403E-05   .00000E+00
    2    2.01031E+00  1.01023E-05   .00000E+00         4.60985E-02
8.94630E-03        7.64370E-03  4.65703E-07   .00000E+00
    3    1.31913E+00  3.55934E-06   .00000E+00         3.88166E-02
7.53310E-03        9.80861E-03  1.38161E-07   .00000E+00
    4    7.92566E-01  3.59323E-06   .00000E+00         2.33036E-02
4.52250E-03        9.80090E-03  8.37351E-08   .00000E+00
    5    6.54618E-01  6.63753E-06   .00000E+00         1.52511E-02
2.95977E-03        7.76592E-03  1.01230E-07   .00000E+00
    6    5.56182E-01  1.94203E-05   .00000E+00         2.11802E-02
4.11042E-03        1.26938E-02  4.11326E-07   .00000E+00
    7    5.63543E-01  6.29493E-05   .00000E+00         1.81978E-02
3.53163E-03        1.07639E-02  1.14554E-06   .00000E+00
    8    5.77544E-01  1.71416E-04   .00000E+00         1.74877E-02
3.39383E-03        1.00932E-02  2.99768E-06   .00000E+00
    9    5.80367E-01  3.53087E-04   .00000E+00         8.57481E-03
1.66411E-03        4.92494E-03  3.02765E-06   .00000E+00
   10    5.80777E-01  5.82085E-04   .00000E+00         8.40298E-03
1.63076E-03        4.82284E-03  4.89124E-06   .00000E+00
   11    5.80114E-01  9.59742E-04   .00000E+00         8.20548E-03
1.59243E-03        4.71487E-03  7.87514E-06   .00000E+00
   12    5.78579E-01  1.47308E-03   .00000E+00         5.88451E-03
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1.14200E-03       3.39021E-03  8.66833E-06   .00000E+00
   13     5.30274E-01  2.07713E-03   .00000E+00       5.45242E-03
1.05815E-03       3.42742E-03  1.13254E-05   .00000E+00
   14     4.84734E-01  2.83531E-03   .00000E+00       5.23484E-03
1.01592E-03       3.59980E-03  1.48424E-05   .00000E+00
   15     4.56194E-01  3.34008E-03   .00000E+00       6.24208E-04
1.21139E-04       4.56098E-04  2.08490E-06   .00000E+00
   16     4.50003E-01  3.46267E-03   .00000E+00       6.43167E-04
1.24819E-04       4.76416E-04  2.22707E-06   .00000E+00
   17     4.42700E-01  3.60538E-03   .00000E+00       8.06450E-04
1.56507E-04       6.07220E-04  2.90756E-06   .00000E+00
   18     4.19829E-01  4.05921E-03   .00000E+00       3.46008E-03
6.71495E-04       2.74721E-03  1.40452E-05   .00000E+00
   19     3.66366E-01  5.17002E-03   .00000E+00       5.61017E-03
1.08876E-03       5.10435E-03  2.90047E-05   .00000E+00
   20     3.18292E-01  6.29875E-03   .00000E+00       2.22264E-03
4.31345E-04       2.32767E-03  1.39998E-05   .00000E+00
   21     2.76329E-01  7.56452E-03   .00000E+00       5.14214E-03
9.97931E-04       6.20292E-03  3.88978E-05   .00000E+00
   22     2.07862E-01  1.07793E-02   .00000E+00       2.50026E-02
4.85224E-03       4.00949E-02  2.69510E-04   .00000E+00
   23     1.64184E-01  1.40641E-02   .00000E+00       2.52567E-02
4.90155E-03       5.12773E-02  3.55213E-04   .00000E+00
   24     1.30440E-01  1.78189E-02   .00000E+00       2.44251E-02
4.74016E-03       6.24172E-02  4.35228E-04   .00000E+00
   25     1.01321E-01  2.37240E-02   .00000E+00       1.90172E-02
3.69064E-03       6.25639E-02  4.51163E-04   .00000E+00
   26     7.46679E-02  3.97569E-02   .00000E+00       1.02192E-02
1.98324E-03       4.56209E-02  4.06285E-04   .00000E+00

 THERMAL 1.41379E-01  1.71033E-02   .00000E+00       1.16896E-01
2.26859E-02       2.75609E-01  1.99930E-03   .00000E+00

  TOTAL  3.18527E-01  5.87837E-03   .00000E+00       3.58396E-01
6.95535E-02       3.75055E-01  2.10678E-03   .00000E+00


0REGION 17     MATERIAL FUEL_3      VOLUME  6.662885E-01

                    CROSS-SECTIONS                           FLUXES
                      REACTIONS
 GROUP  DIFFUSION    ABSORPTION   NU*FISSION          RIF            RAF
        TRANSPORT   ABSORPTION   NU*FISSION

    1     2.32885E+00  1.78377E-02  5.28855E-02       2.86990E-03
4.30730E-03       4.10775E-04  5.11925E-05  1.51776E-04
    2     1.91875E+00  1.57905E-02  3.97823E-02       9.49780E-03
1.42548E-02       1.65000E-03  1.49975E-04  3.77844E-04
    3     1.18870E+00  8.57947E-03  1.48435E-02       7.50523E-03
1.12642E-02       2.10460E-03  6.43909E-05  1.11404E-04
    4     8.16399E-01  9.08466E-03  1.44654E-02       3.88848E-03
5.83604E-03       1.58766E-03  3.53256E-05  5.62486E-05
    5     8.11678E-01  1.23757E-02  1.74729E-02       2.20173E-03
3.30447E-03       9.04190E-04  2.72480E-05  3.84707E-05
    6     7.06212E-01  2.23040E-02  2.45211E-02       2.79185E-03
4.19015E-03       1.31776E-03  6.22692E-05  6.84592E-05
    7     6.33971E-01  4.98772E-02  5.54151E-02       2.32325E-03
3.48685E-03       1.22153E-03  1.15877E-04  1.28743E-04
    8     5.71165E-01  1.29766E-01  1.54831E-01       2.18524E-03
3.27971E-03       1.27531E-03  2.83569E-04  3.38343E-04
    9     4.83010E-01  2.42135E-01  2.40387E-01       1.04562E-03
1.56932E-03       7.21598E-04  2.53180E-04  2.51353E-04
   10     3.98856E-01  3.85543E-01  4.87313E-01       9.91747E-04
1.48846E-03       8.28826E-04  3.82361E-04  4.83291E-04
   11     3.78054E-01  4.94895E-01  5.51727E-01       9.45830E-04
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1.41955E-03       8.33946E-04  4.68086E-04  5.21839E-04
   12    3.77531E-01  5.09213E-01  1.61345E-01       6.75419E-04
1.01370E-03       5.96347E-04  3.43932E-04  1.08975E-04
   13    6.68706E-01  1.23913E-01  2.00238E-01       6.80853E-04
1.02186E-03       3.39388E-04  8.43662E-05  1.36333E-04
   14    6.13009E-01  1.58972E-01  2.55781E-01       6.49163E-04
9.74297E-04       3.52992E-04  1.03199E-04  1.66044E-04
   15    3.30150E-01  6.22014E-01  1.20522E+00       7.04420E-05
1.05723E-04       7.11213E-05  4.38159E-05  8.48983E-05
   16    4.05488E-01  4.34493E-01  8.86229E-01       7.50820E-05
1.12687E-04       6.17215E-05  3.26226E-05  6.65399E-05
   17    4.60036E-01  3.35121E-01  7.02402E-01       9.60511E-05
1.44158E-04       6.95968E-05  3.21887E-05  6.74665E-05
   18    4.74286E-01  3.11396E-01  6.58316E-01       4.14274E-04
6.21764E-04       2.91157E-04  1.29003E-04  2.72723E-04
   19    3.62015E-01  5.26401E-01  1.08735E+00       6.41513E-04
9.62815E-04       5.90687E-04  3.37693E-04  6.97548E-04
   20    2.31685E-01  1.04238E+00  2.03659E+00       2.28317E-04
3.42670E-04       3.28487E-04  2.37993E-04  4.64989E-04
   21    2.29393E-01  1.05628E+00  2.06603E+00       5.22668E-04
7.84447E-04       7.59495E-04  5.52083E-04  1.07985E-03
   22    1.89976E-01  1.35500E+00  2.75817E+00       2.37862E-03
3.56995E-03       4.17354E-03  3.22303E-03  6.56062E-03
   23    1.45443E-01  1.89028E+00  3.87453E+00       2.16548E-03
3.25007E-03       4.96297E-03  4.09336E-03  8.39024E-03
   24    1.14148E-01  2.51666E+00  5.16504E+00       1.87143E-03
2.80873E-03       5.46490E-03  4.70974E-03  9.66600E-03
   25    8.57929E-02  3.47833E+00  7.14022E+00       1.24648E-03
1.87078E-03       4.84297E-03  4.33567E-03  8.90013E-03
   26    5.39375E-02  5.75904E+00  1.18196E+01       4.90869E-04
7.36722E-04       3.03357E-03  2.82694E-03  5.80190E-03

 THERMAL 1.31715E-01  2.12841E+00  4.35408E+00       9.54537E-03
1.43262E-02       2.41566E-02  2.03165E-02  4.15613E-02

  TOTAL  4.16318E-01  4.74252E-01  9.28564E-01       4.84533E-02
7.27213E-02       3.87951E-02  2.29791E-02  4.49920E-02


0REGION 18     MATERIAL CLAD        VOLUME  6.792112E-01

                     CROSS-SECTIONS                             FLUXES
                       REACTIONS
 GROUP   DIFFUSION    ABSORPTION   NU*FISSION          RIF           RAF
        TRANSPORT    ABSORPTION   NU*FISSION

    1    3.16321E+00 -5.37083E-05   .00000E+00       2.34283E-03
3.44934E-03       2.46883E-04 -1.25829E-07   .00000E+00
    2    2.49217E+00  3.74237E-04   .00000E+00       7.81690E-03
1.15088E-02       1.04553E-03  2.92537E-06   .00000E+00
    3    1.67902E+00  5.04506E-04   .00000E+00       6.37535E-03
9.38640E-03       1.26569E-03  3.21640E-06   .00000E+00
    4    1.12257E+00  5.15706E-04   .00000E+00       3.52125E-03
5.18433E-03       1.04560E-03  1.81593E-06   .00000E+00
    5    9.21946E-01  6.62955E-04   .00000E+00       2.13006E-03
3.13608E-03       7.70131E-04  1.41213E-06   .00000E+00
    6    9.25362E-01  1.34549E-03   .00000E+00       2.81923E-03
4.15074E-03       1.01554E-03  3.79324E-06   .00000E+00
    7    8.79884E-01  6.78327E-03   .00000E+00       2.38133E-03
3.50602E-03       9.02138E-04  1.61532E-05   .00000E+00
    8    1.34604E+00  1.24927E-02   .00000E+00       2.26246E-03
3.33101E-03       5.60273E-04  2.82641E-05   .00000E+00
    9    1.59365E+00  1.42310E-04   .00000E+00       1.09788E-03
1.61641E-03       2.29637E-04  1.56239E-07   .00000E+00
   10    1.57942E+00  2.24967E-04   .00000E+00       1.05992E-03
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1.56051E-03       2.23694E-04  2.38447E-07   .00000E+00
   11    1.57349E+00  3.85370E-04   .00000E+00       1.02385E-03
1.50741E-03       2.16896E-04  3.94561E-07   .00000E+00
   12    1.57020E+00  6.03067E-04   .00000E+00       7.32706E-04
1.07876E-03       1.55545E-04  4.41871E-07   .00000E+00
   13    1.57409E+00  8.53588E-04   .00000E+00       7.05015E-04
1.03799E-03       1.49295E-04  6.01792E-07   .00000E+00
   14    1.56863E+00  1.16803E-03   .00000E+00       6.74793E-04
9.93496E-04       1.43394E-04  7.88180E-07   .00000E+00
   15    1.56887E+00  1.37786E-03   .00000E+00       7.71989E-05
1.13660E-04       1.64023E-05  1.06369E-07   .00000E+00
   16    1.56929E+00  1.42882E-03   .00000E+00       8.06450E-05
1.18733E-04       1.71298E-05  1.15227E-07   .00000E+00
   17    1.56537E+00  1.48846E-03   .00000E+00       1.02035E-04
1.50226E-04       2.17277E-05  1.51875E-07   .00000E+00
   18    1.56347E+00  1.67631E-03   .00000E+00       4.38685E-04
6.45874E-04       9.35283E-05  7.35371E-07   .00000E+00
   19    1.55973E+00  2.13456E-03   .00000E+00       6.96758E-04
1.02583E-03       1.48906E-04  1.48727E-06   .00000E+00
   20    1.55524E+00  2.60384E-03   .00000E+00       2.63971E-04
3.88643E-04       5.65766E-05  6.87338E-07   .00000E+00
   21    1.55192E+00  3.12774E-03   .00000E+00       6.06316E-04
8.92677E-04       1.30229E-04  1.89640E-06   .00000E+00
   22    1.54000E+00  4.45409E-03   .00000E+00       2.85728E-03
4.20676E-03       6.18459E-04  1.27266E-05   .00000E+00
   23    1.52869E+00  5.81622E-03   .00000E+00       2.77491E-03
4.08549E-03       6.05075E-04  1.61395E-05   .00000E+00
   24    1.51588E+00  7.36893E-03   .00000E+00       2.58540E-03
3.80648E-03       5.68516E-04  1.90517E-05   .00000E+00
   25    1.49532E+00  9.79734E-03   .00000E+00       1.92803E-03
2.83863E-03       4.29792E-04  1.88895E-05   .00000E+00
   26    1.43463E+00  1.63396E-02   .00000E+00       9.74878E-04
1.43531E-03       2.26510E-04  1.59291E-05   .00000E+00

 THERMAL 1.51907E+00  6.84194E-03   .00000E+00       1.26875E-02
1.86798E-02       2.78406E-03  8.68074E-05   .00000E+00

  TOTAL  1.47755E+00  3.06213E-03   .00000E+00       4.83297E-02
7.11556E-02       1.09031E-02  1.47992E-04   .00000E+00


0REGION 19     MATERIAL COOLANT     VOLUME  1.871414E+01

                   CROSS-SECTIONS                              FLUXES
                     REACTIONS
 GROUP  DIFFUSION    ABSORPTION   NU*FISSION          RIF           RAF
        TRANSPORT   ABSORPTION  NU*FISSION

    1    2.70642E+00  2.17943E-03   .00000E+00       5.20640E-02
2.78206E-03       6.41240E-03  1.13470E-04   .00000E+00
    2    2.01031E+00  1.01023E-05   .00000E+00       1.73542E-01
9.27333E-03       2.87754E-02  1.75318E-06   .00000E+00
    3    1.31913E+00  3.55934E-06   .00000E+00       1.45623E-01
7.78145E-03       3.67977E-02  5.18322E-07   .00000E+00
    4    7.92566E-01  3.59323E-06   .00000E+00       8.65044E-02
4.62241E-03       3.63816E-02  3.10830E-07   .00000E+00
    5    6.54618E-01  6.63753E-06   .00000E+00       5.60068E-02
2.99275E-03       2.85188E-02  3.71747E-07   .00000E+00
    6    5.56182E-01  1.94203E-05   .00000E+00       7.71252E-02
4.12122E-03       4.62230E-02  1.49780E-06   .00000E+00
    7    5.63543E-01  6.29493E-05   .00000E+00       6.58726E-02
3.51994E-03       3.89634E-02  4.14664E-06   .00000E+00
    8    5.77544E-01  1.71416E-04   .00000E+00       6.30358E-02
3.36835E-03       3.63815E-02  1.08053E-05   .00000E+00
    9    5.80367E-01  3.53087E-04   .00000E+00       3.07994E-02
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1.64578E-03       1.76896E-02  1.08749E-05   .00000E+00
   10    5.80777E-01  5.82085E-04   .00000E+00       3.00863E-02
1.60768E-03       1.72679E-02  1.75128E-05   .00000E+00
   11    5.80114E-01  9.59742E-04   .00000E+00       2.92922E-02
1.56524E-03       1.68313E-02  2.81129E-05   .00000E+00
   12    5.78579E-01  1.47308E-03   .00000E+00       2.09701E-02
1.12055E-03       1.20814E-02  3.08906E-05   .00000E+00
   13    5.30274E-01  2.07713E-03   .00000E+00       1.95095E-02
1.04250E-03       1.22638E-02  4.05237E-05   .00000E+00
   14    4.84734E-01  2.83531E-03   .00000E+00       1.87297E-02
1.00083E-03       1.28797E-02  5.31044E-05   .00000E+00
   15    4.56194E-01  3.34008E-03   .00000E+00       2.22232E-03
1.18751E-04       1.62381E-03  7.42272E-06   .00000E+00
   16    4.50003E-01  3.46267E-03   .00000E+00       2.29303E-03
1.22529E-04       1.69853E-03  7.93999E-06   .00000E+00
   17    4.42700E-01  3.60538E-03   .00000E+00       2.87778E-03
1.53776E-04       2.16684E-03  1.03755E-05   .00000E+00
   18    4.19829E-01  4.05921E-03   .00000E+00       1.23476E-02
6.59799E-04       9.80364E-03  5.01213E-05   .00000E+00
   19    3.66366E-01  5.17002E-03   .00000E+00       1.99517E-02
1.06613E-03       1.81528E-02  1.03151E-04   .00000E+00
   20    3.18292E-01  6.29875E-03   .00000E+00       7.85375E-03
4.19669E-04       8.22489E-03  4.94688E-05   .00000E+00
   21    2.76329E-01  7.56452E-03   .00000E+00       1.79952E-02
9.61585E-04       2.17075E-02  1.36125E-04   .00000E+00
   22    2.07862E-01  1.07793E-02   .00000E+00       8.40254E-02
4.48994E-03       1.34745E-01  9.05732E-04   .00000E+00
   23    1.64184E-01  1.40641E-02   .00000E+00       8.31203E-02
4.44158E-03       1.68754E-01  1.16901E-03   .00000E+00
   24    1.30440E-01  1.78189E-02   .00000E+00       7.93553E-02
4.24039E-03       2.02789E-01  1.41402E-03   .00000E+00
   25    1.01321E-01  2.37240E-02   .00000E+00       6.11040E-02
3.26513E-03       2.01024E-01  1.44963E-03   .00000E+00
   26    7.46679E-02  3.97569E-02   .00000E+00       3.25576E-02
1.73973E-03       1.45344E-01  1.29439E-03   .00000E+00

 THERMAL 1.42832E-01  1.68968E-02   .00000E+00       3.85963E-01
2.06242E-02       9.00742E-01  6.52153E-03   .00000E+00

  TOTAL  3.36331E-01  5.42119E-03   .00000E+00       1.27487E+00
6.81231E-02       1.26350E+00  6.91129E-03   .00000E+00


0REGION 20     MATERIAL FUEL_4      VOLUME  1.332585E+00

                    CROSS-SECTIONS                              FLUXES
                      REACTIONS
 GROUP   DIFFUSION    ABSORPTION   NU*FISSION             RIF            RAF
        TRANSPORT    ABSORPTION   NU*FISSION

    1    2.32885E+00  1.78377E-02  5.28855E-02       5.45886E-03
4.09645E-03       7.81338E-04  9.73737E-05  2.88695E-04
    2    1.91875E+00  1.57905E-02  3.97823E-02       1.81607E-02
1.36282E-02       3.15497E-03  2.86768E-04  7.22476E-04
    3    1.18870E+00  8.57947E-03  1.48435E-02       1.44856E-02
1.08703E-02       4.06202E-03  1.24278E-04  2.15017E-04
    4    8.16399E-01  9.08466E-03  1.44654E-02       7.61160E-03
5.71191E-03       3.10780E-03  6.91489E-05  1.10105E-04
    5    8.11678E-01  1.23757E-02  1.74729E-02       4.36112E-03
3.27268E-03       1.79099E-03  5.39720E-05  7.62015E-05
    6    7.06212E-01  2.23040E-02  2.45211E-02       5.56360E-03
4.17504E-03       2.62603E-03  1.24090E-04  1.36425E-04
    7    6.33971E-01  4.98772E-02  5.54151E-02       4.62600E-03
3.47145E-03       2.43229E-03  2.30732E-04  2.56350E-04
    8    5.71165E-01  1.29766E-01  1.54831E-01       4.34356E-03
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3.25950E-03       2.53491E-03  5.63646E-04  6.72519E-04
    9    4.83010E-01  2.42135E-01  2.40387E-01       2.07539E-03
1.55742E-03       1.43226E-03  5.02523E-04  4.98897E-04
   10    3.98856E-01  3.85543E-01  4.87313E-01       1.96730E-03
1.47631E-03       1.64412E-03  7.58481E-04  9.58692E-04
   11    3.78054E-01  4.94895E-01  5.51727E-01       1.87430E-03
1.40652E-03       1.65259E-03  9.27583E-04  1.03410E-03
   12    3.77531E-01  5.09213E-01  1.61345E-01       1.33676E-03
1.00313E-03       1.18026E-03  6.80695E-04  2.15679E-04
   13    6.68706E-01  1.23913E-01  2.00238E-01       1.34475E-03
1.00913E-03       6.70326E-04  1.66632E-04  2.69270E-04
   14    6.13009E-01  1.58972E-01  2.55781E-01       1.28241E-03
9.62349E-04       6.97331E-04  2.03867E-04  3.28017E-04
   15    3.30150E-01  6.22014E-01  1.20522E+00       1.39336E-04
1.04560E-04       1.40679E-04  8.66687E-05  1.67930E-04
   16    4.05488E-01  4.34493E-01  8.86229E-01       1.48419E-04
1.11377E-04       1.22009E-04  6.44872E-05  1.31534E-04
   17    4.60036E-01  3.35121E-01  7.02402E-01       1.89787E-04
1.42420E-04       1.37516E-04  6.36014E-05  1.33306E-04
   18    4.74286E-01  3.11396E-01  6.58316E-01       8.18253E-04
6.14034E-04       5.75077E-04  2.54800E-04  5.38669E-04
   19    3.62015E-01  5.26401E-01  1.08735E+00       1.26651E-03
9.50415E-04       1.16616E-03  6.66691E-04  1.37714E-03
   20    2.31685E-01  1.04238E+00  2.03659E+00       4.50602E-04
3.38142E-04       6.48297E-04  4.69699E-04  9.17694E-04
   21    2.29393E-01  1.05628E+00  2.06603E+00       1.02220E-03
7.67080E-04       1.48537E-03  1.07973E-03  2.11190E-03
   22    1.89976E-01  1.35500E+00  2.75817E+00       4.48173E-03
3.36318E-03       7.86367E-03  6.07275E-03  1.23613E-02
   23    1.45443E-01  1.89028E+00  3.87453E+00       4.00637E-03
3.00647E-03       9.18201E-03  7.57315E-03  1.55228E-02
   24    1.14148E-01  2.51666E+00  5.16504E+00       3.42161E-03
2.56765E-03       9.99170E-03  8.61101E-03  1.76727E-02
   25    8.57929E-02  3.47833E+00  7.14022E+00       2.25364E-03
1.69118E-03       8.75613E-03  7.83892E-03  1.60915E-02
   26    5.39375E-02  5.75904E+00  1.18196E+01       8.78640E-04
6.59350E-04       5.42998E-03  5.06012E-03  1.03852E-02

 THERMAL 1.33123E-01  2.10176E+00  4.29892E+00       1.77813E-02
1.33435E-02       4.45233E-02  3.73721E-02  7.64403E-02

  TOTAL  4.25706E-01  4.55614E-01  8.89121E-01       9.35691E-02
7.02162E-02       7.32658E-02  4.26314E-02  8.31942E-02


0REGION 21     MATERIAL CLAD        VOLUME  1.358412E+00

                     CROSS-SECTIONS                           FLUXES
                       REACTIONS
 GROUP   DIFFUSION   ABSORPTION  NU*FISSION          RIF           RAF
        TRANSPORT   ABSORPTION  NU*FISSION

    1    3.16321E+00 -5.37083E-05   .00000E+00       4.54483E-03
3.34569E-03       4.78925E-04 -2.44095E-07   .00000E+00
    2    2.49217E+00  3.74237E-04   .00000E+00       1.52116E-02
1.11980E-02       2.03458E-03  5.69273E-06   .00000E+00
    3    1.67902E+00  5.04506E-04   .00000E+00       1.24717E-02
9.18112E-03       2.47600E-03  6.29207E-06   .00000E+00
    4    1.12257E+00  5.15706E-04   .00000E+00       6.95688E-03
5.12133E-03       2.06577E-03  3.58771E-06   .00000E+00
    5    9.21946E-01  6.62955E-04   .00000E+00       4.23936E-03
3.12082E-03       1.53276E-03  2.81050E-06   .00000E+00
    6    9.25362E-01  1.34549E-03   .00000E+00       5.62532E-03
4.14110E-03       2.02635E-03  7.56881E-06   .00000E+00
    7    8.79884E-01  6.78327E-03   .00000E+00       4.74230E-03
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3.49106E-03        1.79656E-03  3.21683E-05   .00000E+00
    8    1.34604E+00  1.24927E-02   .00000E+00        4.49654E-03
3.31014E-03        1.11352E-03  5.61737E-05   .00000E+00
    9    1.59365E+00  1.42310E-04   .00000E+00        2.17845E-03
1.60368E-03        4.55652E-04  3.10015E-07   .00000E+00
   10    1.57942E+00  2.24967E-04   .00000E+00        2.10160E-03
1.54710E-03        4.43539E-04  4.72790E-07   .00000E+00
   11    1.57349E+00  3.85370E-04   .00000E+00        2.02772E-03
1.49271E-03        4.29559E-04  7.81422E-07   .00000E+00
   12    1.57020E+00  6.03067E-04   .00000E+00        1.44921E-03
1.06684E-03        3.07649E-04  8.73971E-07   .00000E+00
   13    1.57409E+00  8.53588E-04   .00000E+00        1.39192E-03
1.02467E-03        2.94755E-04  1.18812E-06   .00000E+00
   14    1.56863E+00  1.16803E-03   .00000E+00        1.33250E-03
9.80924E-04        2.83156E-04  1.55640E-06   .00000E+00
   15    1.56887E+00  1.37786E-03   .00000E+00        1.52605E-04
1.12340E-04        3.24235E-05  2.10268E-07   .00000E+00
   16    1.56929E+00  1.42882E-03   .00000E+00        1.59317E-04
1.17282E-04        3.38407E-05  2.27636E-07   .00000E+00
   17    1.56537E+00  1.48846E-03   .00000E+00        2.01492E-04
1.48329E-04        4.29063E-05  2.99913E-07   .00000E+00
   18    1.56347E+00  1.67631E-03   .00000E+00        8.65978E-04
6.37493E-04        1.84628E-04  1.45165E-06   .00000E+00
   19    1.55973E+00  2.13456E-03   .00000E+00        1.37456E-03
1.01189E-03        2.93760E-04  2.93408E-06   .00000E+00
   20    1.55524E+00  2.60384E-03   .00000E+00        5.20419E-04
3.83108E-04        1.11541E-04  1.35509E-06   .00000E+00
   21    1.55192E+00  3.12774E-03   .00000E+00        1.18421E-03
8.71764E-04        2.54355E-04  3.70392E-06   .00000E+00
   22    1.54000E+00  4.45409E-03   .00000E+00        5.37031E-03
3.95338E-03        1.16241E-03  2.39198E-05   .00000E+00
   23    1.52869E+00  5.81622E-03   .00000E+00        5.11769E-03
3.76741E-03        1.11592E-03  2.97656E-05   .00000E+00
   24    1.51588E+00  7.36893E-03   .00000E+00        4.70985E-03
3.46718E-03        1.03567E-03  3.47066E-05   .00000E+00
   25    1.49532E+00  9.79734E-03   .00000E+00        3.47167E-03
2.55568E-03        7.73898E-04  3.40131E-05   .00000E+00
   26    1.43463E+00  1.63396E-02   .00000E+00        1.73689E-03
1.27862E-03        4.03561E-04  2.83801E-05   .00000E+00

 THERMAL 1.51978E+00  6.76067E-03   .00000E+00        2.34856E-02
1.72890E-02        5.15111E-03  1.58778E-04   .00000E+00

  TOTAL  1.47338E+00  2.99248E-03   .00000E+00        9.36349E-02
6.89297E-02        2.11837E-02  2.80200E-04   .00000E+00


0REGION 22     MATERIAL COOLANT     VOLUME  1.357256E+01

                        CROSS-SECTIONS                              FLUXES
                          REACTIONS
 GROUP   DIFFUSION     ABSORPTION    NU*FISSION          RIF              RAF
       TRANSPORT     ABSORPTION    NU*FISSION

    1    2.70642E+00  2.17943E-03   .00000E+00        3.74308E-02
2.75783E-03        4.61013E-03  8.15778E-05   .00000E+00
    2    2.01031E+00  1.01023E-05   .00000E+00        1.25124E-01
9.21886E-03        2.07470E-02  1.26404E-06   .00000E+00
    3    1.31913E+00  3.55934E-06   .00000E+00        1.05046E-01
7.73956E-03        2.65441E-02  3.73893E-07   .00000E+00
    4    7.92566E-01  3.59323E-06   .00000E+00        6.23934E-02
4.59703E-03        2.62411E-02  2.24194E-07   .00000E+00
    5    6.54618E-01  6.63753E-06   .00000E+00        4.03600E-02
2.97365E-03        2.05514E-02  2.67891E-07   .00000E+00
    6    5.56182E-01  1.94203E-05   .00000E+00        5.55235E-02
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4.09087E-03       3.32766E-02  1.07829E-06   .00000E+00
   7    5.63543E-01  6.29493E-05   .00000E+00       4.73689E-02
3.49005E-03       2.80185E-02  2.98184E-06   .00000E+00
   8    5.77544E-01  1.71416E-04   .00000E+00       4.52734E-02
3.33566E-03       2.61298E-02  7.76058E-06   .00000E+00
   9    5.80367E-01  3.53087E-04   .00000E+00       2.20976E-02
1.62811E-03       1.26918E-02  7.80239E-06   .00000E+00
  10    5.80777E-01  5.82085E-04   .00000E+00       2.15625E-02
1.58868E-03       1.23756E-02  1.25512E-05   .00000E+00
  11    5.80114E-01  9.59742E-04   .00000E+00       2.09693E-02
1.54498E-03       1.20490E-02  2.01252E-05   .00000E+00
  12    5.78579E-01  1.47308E-03   .00000E+00       1.49992E-02
1.10511E-03       8.64138E-03  2.20949E-05   .00000E+00
  13    5.30274E-01  2.07713E-03   .00000E+00       1.39602E-02
1.02856E-03       8.77545E-03  2.89971E-05   .00000E+00
  14    4.84734E-01  2.83531E-03   .00000E+00       1.33992E-02
9.87231E-04       9.21415E-03  3.79910E-05   .00000E+00
  15    4.56194E-01  3.34008E-03   .00000E+00       1.58828E-03
1.17021E-04       1.16053E-03  5.30497E-06   .00000E+00
  16    4.50003E-01  3.46267E-03   .00000E+00       1.63913E-03
1.20768E-04       1.21416E-03  5.67575E-06   .00000E+00
  17    4.42700E-01  3.60538E-03   .00000E+00       2.05737E-03
1.51583E-04       1.54911E-03  7.41760E-06   .00000E+00
  18    4.19829E-01  4.05921E-03   .00000E+00       8.82637E-03
6.50310E-04       7.00790E-03  3.58281E-05   .00000E+00
  19    3.66366E-01  5.17002E-03   .00000E+00       1.42480E-02
1.04976E-03       1.29634E-02  7.36623E-05   .00000E+00
  20    3.18292E-01  6.29875E-03   .00000E+00       5.59744E-03
4.12408E-04       5.86195E-03  3.52569E-05   .00000E+00
  21    2.76329E-01  7.56452E-03   .00000E+00       1.27404E-02
9.38691E-04       1.53687E-02  9.63752E-05   .00000E+00
  22    2.07862E-01  1.07793E-02   .00000E+00       5.80947E-02
4.28031E-03       9.31622E-02  6.26218E-04   .00000E+00
  23    1.64184E-01  1.40641E-02   .00000E+00       5.69325E-02
4.19468E-03       1.15587E-01  8.00706E-04   .00000E+00
  24    1.30440E-01  1.78189E-02   .00000E+00       5.41307E-02
3.98824E-03       1.38329E-01  9.64549E-04   .00000E+00
  25    1.01321E-01  2.37240E-02   .00000E+00       4.15905E-02
3.06431E-03       1.36827E-01  9.86693E-04   .00000E+00
  26    7.46679E-02  3.97569E-02   .00000E+00       2.21480E-02
1.63182E-03       9.88731E-02  8.80533E-04   .00000E+00

 THERMAL 1.43433E-01  1.68147E-02   .00000E+00       2.65482E-01
1.95602E-02       6.16972E-01  4.46399E-03   .00000E+00

  TOTAL  3.43712E-01  5.24064E-03   .00000E+00       9.05101E-01
6.66861E-02       8.77770E-01  4.74331E-03   .00000E+00


0REGION 23     MATERIAL FUEL_5      VOLUME  1.332578E+00

                     CROSS-SECTIONS                            FLUXES
                       REACTIONS
 GROUP   DIFFUSION    ABSORPTION   NU*FISSION          RIF           RAF
       TRANSPORT    ABSORPTION   NU*FISSION

   1    2.32885E+00  1.78377E-02  5.28855E-02       5.41712E-03
4.06514E-03       7.75364E-04  9.66291E-05  2.86487E-04
   2    1.91875E+00  1.57905E-02  3.97823E-02       1.80256E-02
1.35268E-02       3.13148E-03  2.84633E-04  7.17099E-04
   3    1.18870E+00  8.57947E-03  1.48435E-02       1.43532E-02
1.07710E-02       4.02490E-03  1.23143E-04  2.13052E-04
   4    8.16399E-01  9.08466E-03  1.44654E-02       7.53329E-03
5.65317E-03       3.07582E-03  6.84374E-05  1.08972E-04
   5    8.11678E-01  1.23757E-02  1.74729E-02       4.31128E-03
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3.23529E-03        1.77052E-03  5.33552E-05  7.53307E-05
    6    7.06212E-01  2.23040E-02  2.45211E-02        5.49658E-03
4.12477E-03        2.59440E-03  1.22595E-04  1.34782E-04
    7    6.33971E-01  4.98772E-02  5.54151E-02        4.56921E-03
3.42885E-03        2.40243E-03  2.27900E-04  2.53204E-04
    8    5.71165E-01  1.29766E-01  1.54831E-01        4.28601E-03
3.21633E-03        2.50133E-03  5.56178E-04  6.63609E-04
    9    4.83010E-01  2.42135E-01  2.40387E-01        2.04563E-03
1.53509E-03        1.41172E-03  4.95319E-04  4.91744E-04
   10    3.98856E-01  3.85543E-01  4.87313E-01        1.93602E-03
1.45284E-03        1.61798E-03  7.46419E-04  9.43446E-04
   11    3.78054E-01  4.94895E-01  5.51727E-01        1.84193E-03
1.38223E-03        1.62404E-03  9.11561E-04  1.01624E-03
   12    3.77531E-01  5.09213E-01  1.61345E-01        1.31287E-03
9.85207E-04        1.15917E-03  6.68528E-04  2.11824E-04
   13    6.68706E-01  1.23913E-01  2.00238E-01        1.32429E-03
9.93779E-04        6.60125E-04  1.64096E-04  2.65173E-04
   14    6.13009E-01  1.58972E-01  2.55781E-01        1.26224E-03
9.47215E-04        6.86362E-04  2.00661E-04  3.22857E-04
   15    3.30150E-01  6.22014E-01  1.20522E+00        1.36701E-04
1.02584E-04        1.38019E-04  8.50299E-05  1.64755E-04
   16    4.05488E-01  4.34493E-01  8.86229E-01        1.45748E-04
1.09373E-04        1.19812E-04  6.33263E-05  1.29166E-04
   17    4.60036E-01  3.35121E-01  7.02402E-01        1.86485E-04
1.39943E-04        1.35124E-04  6.24950E-05  1.30988E-04
   18    4.74286E-01  3.11396E-01  6.58316E-01        8.04066E-04
6.03391E-04        5.65107E-04  2.50383E-04  5.29330E-04
   19    3.62015E-01  5.26401E-01  1.08735E+00        1.24230E-03
9.32252E-04        1.14387E-03  6.53947E-04  1.35081E-03
   20    2.31685E-01  1.04238E+00  2.03659E+00        4.40247E-04
3.30373E-04        6.33399E-04  4.58905E-04  8.96605E-04
   21    2.29393E-01  1.05628E+00  2.06603E+00        9.93657E-04
7.45665E-04        1.44390E-03  1.04958E-03  2.05293E-03
   22    1.89976E-01  1.35500E+00  2.75817E+00        4.28679E-03
3.21691E-03        7.52162E-03  5.80860E-03  1.18237E-02
   23    1.45443E-01  1.89028E+00  3.87453E+00        3.81242E-03
2.86093E-03        8.73750E-03  7.20653E-03  1.47713E-02
   24    1.14148E-01  2.51666E+00  5.16504E+00        3.25435E-03
2.44214E-03        9.50328E-03  8.19007E-03  1.68088E-02
   25    8.57929E-02  3.47833E+00  7.14022E+00        2.14847E-03
1.61227E-03        8.34752E-03  7.47311E-03  1.53406E-02
   26    5.39375E-02  5.75904E+00  1.18196E+01        8.41376E-04
6.31390E-04        5.19970E-03  4.84552E-03  9.94477E-03

 THERMAL 1.33390E-01  2.09677E+00  4.28856E+00        1.70196E-02
1.27719E-02        4.25308E-02  3.56863E-02  7.29895E-02

  TOTAL  4.32422E-01  4.44168E-01  8.65661E-01        9.20079E-02
6.90450E-02        7.09245E-02  4.08670E-02  7.96476E-02


0REGION 24     MATERIAL CLAD        VOLUME  1.358418E+00

                    CROSS-SECTIONS                            FLUXES
                      REACTIONS
 GROUP   DIFFUSION   ABSORPTION   NU*FISSION         RIF           RAF
        TRANSPORT   ABSORPTION  NU*FISSION

    1    3.16321E+00 -5.37083E-05  .00000E+00        4.48966E-03
3.30506E-03        4.73112E-04 -2.41132E-07  .00000E+00
    2    2.49217E+00  3.74237E-04  .00000E+00        1.50682E-02
1.10925E-02        2.01541E-03  5.63909E-06  .00000E+00
    3    1.67902E+00  5.04506E-04  .00000E+00        1.23664E-02
9.10355E-03        2.45509E-03  6.23894E-06  .00000E+00
    4    1.12257E+00  5.15706E-04  .00000E+00        6.89182E-03
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5.07342E-03       2.04645E-03  3.55416E-06   .00000E+00
     5    9.21946E-01  6.62955E-04   .00000E+00       4.19209E-03
3.08601E-03       1.51567E-03  2.77917E-06   .00000E+00
     6    9.25362E-01  1.34549E-03   .00000E+00       5.55781E-03
4.09139E-03       2.00203E-03  7.47798E-06   .00000E+00
     7    8.79884E-01  6.78327E-03   .00000E+00       4.68441E-03
3.44843E-03       1.77463E-03  3.17756E-05   .00000E+00
     8    1.34604E+00  1.24927E-02   .00000E+00       4.43800E-03
3.26703E-03       1.09902E-03  5.54424E-05   .00000E+00
     9    1.59365E+00  1.42310E-04   .00000E+00       2.14813E-03
1.58135E-03       4.49310E-04  3.05700E-07   .00000E+00
    10    1.57942E+00  2.24967E-04   .00000E+00       2.06943E-03
1.52341E-03       4.36750E-04  4.65554E-07   .00000E+00
    11    1.57349E+00  3.85370E-04   .00000E+00       1.99424E-03
1.46806E-03       4.22465E-04  7.68518E-07   .00000E+00
    12    1.57020E+00  6.03067E-04   .00000E+00       1.42445E-03
1.04861E-03       3.02394E-04  8.59042E-07   .00000E+00
    13    1.57409E+00  8.53588E-04   .00000E+00       1.37124E-03
1.00944E-03       2.90377E-04  1.17048E-06   .00000E+00
    14    1.56863E+00  1.16803E-03   .00000E+00       1.31204E-03
9.65861E-04       2.78809E-04  1.53251E-06   .00000E+00
    15    1.56887E+00  1.37786E-03   .00000E+00       1.49835E-04
1.10301E-04       3.18350E-05  2.06451E-07   .00000E+00
    16    1.56929E+00  1.42882E-03   .00000E+00       1.56556E-04
1.15249E-04       3.32542E-05  2.23691E-07   .00000E+00
    17    1.56537E+00  1.48846E-03   .00000E+00       1.98109E-04
1.45838E-04       4.21860E-05  2.94878E-07   .00000E+00
    18    1.56347E+00  1.67631E-03   .00000E+00       8.51467E-04
6.26808E-04       1.81534E-04  1.42732E-06   .00000E+00
    19    1.55973E+00  2.13456E-03   .00000E+00       1.34939E-03
9.93351E-04       2.88380E-04  2.88034E-06   .00000E+00
    20    1.55524E+00  2.60384E-03   .00000E+00       5.09142E-04
3.74805E-04       1.09124E-04  1.32572E-06   .00000E+00
    21    1.55192E+00  3.12774E-03   .00000E+00       1.15294E-03
8.48735E-04       2.47637E-04  3.60609E-06   .00000E+00
    22    1.54000E+00  4.45409E-03   .00000E+00       5.14823E-03
3.78987E-03       1.11434E-03  2.29307E-05   .00000E+00
    23    1.52869E+00  5.81622E-03   .00000E+00       4.88444E-03
3.59569E-03       1.06506E-03  2.84090E-05   .00000E+00
    24    1.51588E+00  7.36893E-03   .00000E+00       4.49654E-03
3.31013E-03       9.88764E-04  3.31347E-05   .00000E+00
    25    1.49532E+00  9.79734E-03   .00000E+00       3.32471E-03
2.44749E-03       7.41139E-04  3.25733E-05   .00000E+00
    26    1.43463E+00  1.63396E-02   .00000E+00       1.67245E-03
1.23117E-03       3.88588E-04  2.73272E-05   .00000E+00

 THERMAL 1.51984E+00  6.75251E-03   .00000E+00       2.25378E-02
1.65912E-02       4.94303E-03  1.52187E-04   .00000E+00

  TOTAL  1.47326E+00  2.96085E-03   .00000E+00       9.19018E-02
6.76536E-02       2.07934E-02  2.72107E-04   .00000E+00


0REGION 25     MATERIAL COOLANT     VOLUME  1.955620E+01

                     CROSS-SECTIONS                               FLUXES
                        REACTIONS
 GROUP   DIFFUSION    ABSORPTION   NU*FISSION          RIF            RAF
        TRANSPORT    ABSORPTION   NU*FISSION

     1    2.70642E+00  2.17943E-03   .00000E+00       5.21477E-02
2.66656E-03       6.42271E-03  1.13652E-04   .00000E+00
     2    2.01031E+00  1.01023E-05   .00000E+00       1.74703E-01
8.93341E-03       2.89680E-02  1.76491E-06   .00000E+00
     3    1.31913E+00  3.55934E-06   .00000E+00       1.47124E-01
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7.52315E-03        3.71771E-02  5.23666E-07   .00000E+00
    4    7.92566E-01  3.59323E-06   .00000E+00       8.78957E-02
4.49452E-03        3.69667E-02  3.15830E-07   .00000E+00
    5    6.54618E-01  6.63753E-06   .00000E+00       5.71156E-02
2.92059E-03        2.90834E-02  3.79107E-07   .00000E+00
    6    5.56182E-01  1.94203E-05   .00000E+00       7.88475E-02
4.03184E-03        4.72552E-02  1.53125E-06   .00000E+00
    7    5.63543E-01  6.29493E-05   .00000E+00       6.74237E-02
3.44769E-03        3.98808E-02  4.24428E-06   .00000E+00
    8    5.77544E-01  1.71416E-04   .00000E+00       6.45208E-02
3.29925E-03        3.72386E-02  1.10599E-05   .00000E+00
    9    5.80367E-01  3.53087E-04   .00000E+00       3.15243E-02
1.61199E-03        1.81060E-02  1.11308E-05   .00000E+00
   10    5.80777E-01  5.82085E-04   .00000E+00       3.07790E-02
1.57387E-03        1.76654E-02  1.79160E-05   .00000E+00
   11    5.80114E-01  9.59742E-04   .00000E+00       2.99443E-02
1.53119E-03        1.72060E-02  2.87388E-05   .00000E+00
   12    5.78579E-01  1.47308E-03   .00000E+00       2.14203E-02
1.09532E-03        1.23408E-02  3.15538E-05   .00000E+00
   13    5.30274E-01  2.07713E-03   .00000E+00       1.99034E-02
1.01775E-03        1.25114E-02  4.13418E-05   .00000E+00
   14    4.84734E-01  2.83531E-03   .00000E+00       1.90991E-02
9.76625E-04        1.31337E-02  5.41518E-05   .00000E+00
   15    4.56194E-01  3.34008E-03   .00000E+00       2.26729E-03
1.15937E-04        1.65667E-03  7.57294E-06   .00000E+00
   16    4.50003E-01  3.46267E-03   .00000E+00       2.33868E-03
1.19588E-04        1.73234E-03  8.09807E-06   .00000E+00
   17    4.42700E-01  3.60538E-03   .00000E+00       2.93438E-03
1.50048E-04        2.20945E-03  1.05796E-05   .00000E+00
   18    4.19829E-01  4.05921E-03   .00000E+00       1.25867E-02
6.43617E-04        9.99351E-03  5.10920E-05   .00000E+00
   19    3.66366E-01  5.17002E-03   .00000E+00       2.03318E-02
1.03966E-03        1.84986E-02  1.05116E-04   .00000E+00
   20    3.18292E-01  6.29875E-03   .00000E+00       7.99545E-03
4.08845E-04        8.37329E-03  5.03614E-05   .00000E+00
   21    2.76329E-01  7.56452E-03   .00000E+00       1.81609E-02
9.28651E-04        2.19073E-02  1.37378E-04   .00000E+00
   22    2.07862E-01  1.07793E-02   .00000E+00       8.26693E-02
4.22727E-03        1.32571E-01  8.91114E-04   .00000E+00
   23    1.64184E-01  1.40641E-02   .00000E+00       8.12385E-02
4.15411E-03        1.64934E-01  1.14255E-03   .00000E+00
   24    1.30440E-01  1.78189E-02   .00000E+00       7.75001E-02
3.96294E-03        1.98048E-01  1.38097E-03   .00000E+00
   25    1.01321E-01  2.37240E-02   .00000E+00       5.97901E-02
3.05735E-03        1.96701E-01  1.41846E-03   .00000E+00
   26    7.46679E-02  3.97569E-02   .00000E+00       3.19647E-02
1.63451E-03        1.42697E-01  1.27082E-03   .00000E+00

 THERMAL 1.43200E-01  1.68491E-02   .00000E+00       3.79651E-01
1.94133E-02        8.83730E-01  6.39676E-03   .00000E+00

  TOTAL  3.41033E-01  5.29735E-03   .00000E+00       1.28223E+00
6.55663E-02        1.25328E+00  6.79241E-03   .00000E+00


0REGION 26      MATERIAL FUEL_6      VOLUME  6.662828E-01

                    CROSS-SECTIONS                              FLUXES
                      REACTIONS
 GROUP   DIFFUSION    ABSORPTION   NU*FISSION           RIF           RAF
        TRANSPORT    ABSORPTION   NU*FISSION

    1    2.32885E+00  1.78377E-02  5.28855E-02       2.65685E-03
3.98757E-03        3.80280E-04  4.73922E-05  1.40509E-04
    2    1.91875E+00  1.57905E-02  3.97823E-02       8.85321E-03
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1.32875E-02       1.53802E-03  1.39797E-04  3.52201E-04
    3    1.18870E+00  8.57947E-03  1.48435E-02        7.05682E-03
1.05913E-02       1.97886E-03  6.05437E-05  1.04748E-04
    4    8.16399E-01  9.08466E-03  1.44654E-02        3.70677E-03
5.56336E-03       1.51346E-03  3.36748E-05  5.36200E-05
    5    8.11678E-01  1.23757E-02  1.74729E-02        2.12118E-03
3.18361E-03       8.71110E-04  2.62511E-05  3.70633E-05
    6    7.06212E-01  2.23040E-02  2.45211E-02        2.70546E-03
4.06053E-03       1.27698E-03  6.03425E-05  6.63409E-05
    7    6.33971E-01  4.98772E-02  5.54151E-02        2.25144E-03
3.37911E-03       1.18378E-03  1.12296E-04  1.24764E-04
    8    5.71165E-01  1.29766E-01  1.54831E-01        2.11280E-03
3.17103E-03       1.23304E-03  2.74169E-04  3.27128E-04
    9    4.83010E-01  2.42135E-01  2.40387E-01        1.00877E-03
1.51402E-03       6.96167E-04  2.44258E-04  2.42495E-04
   10    3.98856E-01  3.85543E-01  4.87313E-01        9.54179E-04
1.43209E-03       7.97430E-04  3.67877E-04  4.64983E-04
   11    3.78054E-01  4.94895E-01  5.51727E-01        9.07415E-04
1.36191E-03       8.00075E-04  4.49075E-04  5.00645E-04
   12    3.77531E-01  5.09213E-01  1.61345E-01        6.46734E-04
9.70661E-04       5.71020E-04  3.29325E-04  1.04347E-04
   13    6.68706E-01  1.23913E-01  2.00238E-01        6.52971E-04
9.80021E-04       3.25490E-04  8.09113E-05  1.30750E-04
   14    6.13009E-01  1.58972E-01  2.55781E-01        6.22167E-04
9.33789E-04       3.38313E-04  9.89072E-05  1.59139E-04
   15    3.30150E-01  6.22014E-01  1.20522E+00        6.73001E-05
1.01008E-04       6.79490E-05  4.18616E-05  8.11116E-05
   16    4.05488E-01  4.34493E-01  8.86229E-01        7.17757E-05
1.07726E-04       5.90035E-05  3.11861E-05  6.36097E-05
   17    4.60036E-01  3.35121E-01  7.02402E-01        9.18560E-05
1.37863E-04       6.65571E-05  3.07828E-05  6.45198E-05
   18    4.74286E-01  3.11396E-01  6.58316E-01        3.96036E-04
5.94396E-04       2.78339E-04  1.23324E-04  2.60717E-04
   19    3.62015E-01  5.26401E-01  1.08735E+00        6.11308E-04
9.17490E-04       5.62875E-04  3.21793E-04  6.64705E-04
   20    2.31685E-01  1.04238E+00  2.03659E+00        2.16093E-04
3.24327E-04       3.10901E-04  2.25251E-04  4.40094E-04
   21    2.29393E-01  1.05628E+00  2.06603E+00        4.86094E-04
7.29561E-04       7.06349E-04  5.13451E-04  1.00429E-03
   22    1.89976E-01  1.35500E+00  2.75817E+00        2.08920E-03
3.13561E-03       3.66573E-03  2.83087E-03  5.76237E-03
   23    1.45443E-01  1.89028E+00  3.87453E+00        1.85785E-03
2.78837E-03       4.25791E-03  3.51184E-03  7.19829E-03
   24    1.14148E-01  2.51666E+00  5.16504E+00        1.58637E-03
2.38093E-03       4.63249E-03  3.99235E-03  8.19368E-03
   25    8.57929E-02  3.47833E+00  7.14022E+00        1.04867E-03
1.57391E-03       4.07442E-03  3.64762E-03  7.48773E-03
   26    5.39375E-02  5.75904E+00  1.18196E+01        4.11538E-04
6.17662E-04       2.54330E-03  2.37006E-03  4.86423E-03

 THERMAL 1.33422E-01  2.09618E+00  4.28733E+00        8.30712E-03
1.24679E-02       2.07540E-02  1.74133E-02  3.56154E-02

  TOTAL  4.33737E-01  4.41798E-01  8.60662E-01        4.51909E-02
6.78253E-02       3.47298E-02  1.99652E-02  3.88941E-02


0REGION 27     MATERIAL CLAD        VOLUME  6.792185E-01

                        CROSS-SECTIONS                               FLUXES
                          REACTIONS
 GROUP   DIFFUSION    ABSORPTION   NU*FISSION          RIF          RAF
       TRANSPORT    ABSORPTION   NU*FISSION

    1    3.16321E+00 -5.37083E-05   .00000E+00       2.20432E-03
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3.24538E-03       2.32287E-04 -1.18390E-07   .00000E+00
    2    2.49217E+00  3.74237E-04   .00000E+00       7.41386E-03
1.09153E-02       9.91621E-04  2.77454E-06   .00000E+00
    3    1.67902E+00  5.04506E-04   .00000E+00       6.08859E-03
8.96411E-03       1.20876E-03  3.07173E-06   .00000E+00
    4    1.12257E+00  5.15706E-04   .00000E+00       3.39407E-03
4.99702E-03       1.00783E-03  1.75034E-06   .00000E+00
    5    9.21946E-01  6.62955E-04   .00000E+00       2.06315E-03
3.03754E-03       7.45941E-04  1.36778E-06   .00000E+00
    6    9.25362E-01  1.34549E-03   .00000E+00       2.73552E-03
4.02746E-03       9.85389E-04  3.68062E-06   .00000E+00
    7    8.79884E-01  6.78327E-03   .00000E+00       2.30808E-03
3.39814E-03       8.74389E-04  1.56563E-05   .00000E+00
    8    1.34604E+00  1.24927E-02   .00000E+00       2.18758E-03
3.22073E-03       5.41731E-04  2.73287E-05   .00000E+00
    9    1.59365E+00  1.42310E-04   .00000E+00       1.05929E-03
1.55957E-03       2.21564E-04  1.50747E-07   .00000E+00
   10    1.57942E+00  2.24967E-04   .00000E+00       1.01982E-03
1.50145E-03       2.15230E-04  2.29425E-07   .00000E+00
   11    1.57349E+00  3.85370E-04   .00000E+00       9.82301E-04
1.44622E-03       2.08094E-04  3.78549E-07   .00000E+00
   12    1.57020E+00  6.03067E-04   .00000E+00       7.01586E-04
1.03293E-03       1.48938E-04  4.23104E-07   .00000E+00
   13    1.57409E+00  8.53588E-04   .00000E+00       6.76146E-04
9.95476E-04       1.43182E-04  5.77150E-07   .00000E+00
   14    1.56863E+00  1.16803E-03   .00000E+00       6.46728E-04
9.52165E-04       1.37430E-04  7.55399E-07   .00000E+00
   15    1.56887E+00  1.37786E-03   .00000E+00       7.37526E-05
1.08585E-04       1.56700E-05  1.01621E-07   .00000E+00
   16    1.56929E+00  1.42882E-03   .00000E+00       7.70923E-05
1.13502E-04       1.63752E-05  1.10151E-07   .00000E+00
   17    1.56537E+00  1.48846E-03   .00000E+00       9.75797E-05
1.43665E-04       2.07789E-05  1.45244E-07   .00000E+00
   18    1.56347E+00  1.67631E-03   .00000E+00       4.19378E-04
6.17442E-04       8.94120E-05  7.03007E-07   .00000E+00
   19    1.55973E+00  2.13456E-03   .00000E+00       6.63943E-04
9.77510E-04       1.41893E-04  1.41723E-06   .00000E+00
   20    1.55524E+00  2.60384E-03   .00000E+00       2.49812E-04
3.67794E-04       5.35420E-05  6.50471E-07   .00000E+00
   21    1.55192E+00  3.12774E-03   .00000E+00       5.63746E-04
8.29992E-04       1.21086E-04  1.76325E-06   .00000E+00
   22    1.54000E+00  4.45409E-03   .00000E+00       2.50944E-03
3.69460E-03       5.43169E-04  1.11773E-05   .00000E+00
   23    1.52869E+00  5.81622E-03   .00000E+00       2.38159E-03
3.50637E-03       5.19311E-04  1.38519E-05   .00000E+00
   24    1.51588E+00  7.36893E-03   .00000E+00       2.19339E-03
3.22929E-03       4.82315E-04  1.61630E-05   .00000E+00
   25    1.49532E+00  9.79734E-03   .00000E+00       1.62426E-03
2.39137E-03       3.62077E-04  1.59134E-05   .00000E+00
   26    1.43463E+00  1.63396E-02   .00000E+00       8.18873E-04
1.20561E-03       1.90263E-04  1.33801E-05   .00000E+00

 THERMAL 1.51983E+00  6.75294E-03   .00000E+00       1.10051E-02
1.62025E-02       2.41366E-03  7.43166E-05   .00000E+00

  TOTAL  1.47298E+00  2.95440E-03   .00000E+00       4.51539E-02
6.64792E-02       1.02183E-02  1.33403E-04   .00000E+00


0REGION 28     MATERIAL MODERATOR   VOLUME  8.284443E+00

                     CROSS-SECTIONS                          FLUXES
                       REACTIONS
 GROUP   DIFFUSION    ABSORPTION   NU*FISSION           RIF           RAF
        TRANSPORT    ABSORPTION   NU*FISSION
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```
   1    2.74822E+00  2.21855E-03   .00000E+00       2.17197E-02
2.62175E-03       2.63439E-03  4.81863E-05   .00000E+00
   2    2.00863E+00  9.66120E-06   .00000E+00       7.28247E-02
8.79054E-03       1.20853E-02  7.03574E-07   .00000E+00
   3    1.31914E+00  3.55843E-06   .00000E+00       6.14152E-02
7.41332E-03       1.55190E-02  2.18542E-07   .00000E+00
   4    7.92343E-01  3.59826E-06   .00000E+00       3.67586E-02
4.43707E-03       1.54641E-02  1.32267E-07   .00000E+00
   5    6.53939E-01  6.66158E-06   .00000E+00       2.39006E-02
2.88500E-03       1.21829E-02  1.59216E-07   .00000E+00
   6    5.56006E-01  1.95488E-05   .00000E+00       3.30061E-02
3.98410E-03       1.97876E-02  6.45228E-07   .00000E+00
   7    5.63650E-01  6.32557E-05   .00000E+00       2.82619E-02
3.41144E-03       1.67136E-02  1.78773E-06   .00000E+00
   8    5.77572E-01  1.72185E-04   .00000E+00       2.70573E-02
3.26603E-03       1.56155E-02  4.65885E-06   .00000E+00
   9    5.80369E-01  3.53483E-04   .00000E+00       1.32369E-02
1.59780E-03       7.60258E-03  4.67902E-06   .00000E+00
  10    5.80777E-01  5.82735E-04   .00000E+00       1.29172E-02
1.55922E-03       7.41378E-03  7.52733E-06   .00000E+00
  11    5.80112E-01  9.60720E-04   .00000E+00       1.25635E-02
1.51651E-03       7.21898E-03  1.20700E-05   .00000E+00
  12    5.78577E-01  1.47362E-03   .00000E+00       8.98774E-03
1.08489E-03       5.17807E-03  1.32445E-05   .00000E+00
  13    5.30259E-01  2.07768E-03   .00000E+00       8.34867E-03
1.00775E-03       5.24818E-03  1.73459E-05   .00000E+00
  14    4.84689E-01  2.83608E-03   .00000E+00       8.00961E-03
9.66826E-04       5.50842E-03  2.27159E-05   .00000E+00
  15    4.56193E-01  3.34009E-03   .00000E+00       9.51035E-04
1.14798E-04       6.94906E-04  3.17654E-06   .00000E+00
  16    4.50003E-01  3.46266E-03   .00000E+00       9.80866E-04
1.18399E-04       7.26562E-04  3.39641E-06   .00000E+00
  17    4.42698E-01  3.60542E-03   .00000E+00       1.23058E-03
1.48542E-04       9.26578E-04  4.43677E-06   .00000E+00
  18    4.19805E-01  4.05969E-03   .00000E+00       5.27746E-03
6.37033E-04       4.19041E-03  2.14249E-05   .00000E+00
  19    3.66218E-01  5.17320E-03   .00000E+00       8.51961E-03
1.02839E-03       7.75458E-03  4.40737E-05   .00000E+00
  20    3.18091E-01  6.30426E-03   .00000E+00       3.34080E-03
4.03262E-04       3.50089E-03  2.10613E-05   .00000E+00
  21    2.74446E-01  7.62670E-03   .00000E+00       7.55384E-03
9.11810E-04       9.17466E-03  5.76109E-05   .00000E+00
  22    2.07048E-01  1.08270E-02   .00000E+00       3.44216E-02
4.15497E-03       5.54164E-02  3.72682E-04   .00000E+00
  23    1.64120E-01  1.40700E-02   .00000E+00       3.38748E-02
4.08896E-03       6.88008E-02  4.76617E-04   .00000E+00
  24    1.30412E-01  1.78227E-02   .00000E+00       3.22728E-02
3.89559E-03       8.24891E-02  5.75190E-04   .00000E+00
  25    1.01309E-01  2.37277E-02   .00000E+00       2.48784E-02
3.00302E-03       8.18564E-02  5.90306E-04   .00000E+00
  26    7.46682E-02  3.97555E-02   .00000E+00       1.32897E-02
1.60418E-03       5.93278E-02  5.28339E-04   .00000E+00

 THERMAL 1.43128E-01  1.68565E-02   .00000E+00       1.58152E-01
1.90902E-02       3.68321E-01  2.66588E-03   .00000E+00

  TOTAL  3.41343E-01  5.28826E-03   .00000E+00       5.35599E-01
6.46512E-02       5.23032E-01  2.83239E-03   .00000E+00


0REGION 29     MATERIAL FUEL_7      VOLUME  6.662909E-01

                      CROSS-SECTIONS                           FLUXES
                        REACTIONS
 GROUP   DIFFUSION    ABSORPTION   NU*FISSION            RIF            RAF
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	TRANSPORT	ABSORPTION	NU*FISSION					
1	2.32885E+00	1.78377E-02	5.28855E-02	2.62257E-03	3.93607E-03	3.75374E-04	4.67807E-05	1.38696E-04
2	1.91875E+00	1.57905E-02	3.97823E-02	8.74553E-03	1.31257E-02	1.51931E-03	1.38097E-04	3.47918E-04
3	1.18870E+00	8.57947E-03	1.48435E-02	6.97313E-03	1.04656E-02	1.95539E-03	5.98258E-05	1.03506E-04
4	8.16399E-01	9.08466E-03	1.44654E-02	3.66774E-03	5.50471E-03	1.49753E-03	3.33202E-05	5.30554E-05
5	8.11678E-01	1.23757E-02	1.74729E-02	2.10238E-03	3.15534E-03	8.63387E-04	2.60184E-05	3.67347E-05
6	7.06212E-01	2.23040E-02	2.45211E-02	2.68577E-03	4.03092E-03	1.26769E-03	5.99032E-05	6.58579E-05
7	6.33971E-01	4.98772E-02	5.54151E-02	2.23710E-03	3.35754E-03	1.17623E-03	1.11580E-04	1.23969E-04
8	5.71165E-01	1.29766E-01	1.54831E-01	2.10057E-03	3.15263E-03	1.22590E-03	2.72582E-04	3.25234E-04
9	4.83010E-01	2.42135E-01	2.40387E-01	1.00336E-03	1.50589E-03	6.92437E-04	2.42949E-04	2.41196E-04
10	3.98856E-01	3.85543E-01	4.87313E-01	9.49542E-04	1.42512E-03	7.93554E-04	3.66089E-04	4.62724E-04
11	3.78054E-01	4.94895E-01	5.51727E-01	9.03271E-04	1.35567E-03	7.96422E-04	4.47024E-04	4.98359E-04
12	3.77531E-01	5.09213E-01	1.61345E-01	6.43795E-04	9.66238E-04	5.68425E-04	3.27829E-04	1.03873E-04
13	6.68706E-01	1.23913E-01	2.00238E-01	6.49176E-04	9.74313E-04	3.23598E-04	8.04411E-05	1.29990E-04
14	6.13009E-01	1.58972E-01	2.55781E-01	6.18608E-04	9.28436E-04	3.36378E-04	9.83414E-05	1.58228E-04
15	3.30150E-01	6.22014E-01	1.20522E+00	6.70085E-05	1.00569E-04	6.76547E-05	4.16802E-05	8.07602E-05
16	4.05488E-01	4.34493E-01	8.86229E-01	7.14352E-05	1.07213E-04	5.87236E-05	3.10381E-05	6.33080E-05
17	4.60036E-01	3.35121E-01	7.02402E-01	9.13936E-05	1.37168E-04	6.62220E-05	3.06279E-05	6.41950E-05
18	4.74286E-01	3.11396E-01	6.58316E-01	3.94012E-04	5.91351E-04	2.76916E-04	1.22694E-04	2.59384E-04
19	3.62015E-01	5.26401E-01	1.08735E+00	6.08666E-04	9.13514E-04	5.60442E-04	3.20402E-04	6.61832E-04
20	2.31685E-01	1.04238E+00	2.03659E+00	2.15512E-04	3.23451E-04	3.10065E-04	2.24646E-04	4.38911E-04
21	2.29393E-01	1.05628E+00	2.06603E+00	4.84722E-04	7.27494E-04	7.04356E-04	5.12002E-04	1.00145E-03
22	1.89976E-01	1.35500E+00	2.75817E+00	2.08163E-03	3.12420E-03	3.65244E-03	2.82061E-03	5.74147E-03
23	1.45443E-01	1.89028E+00	3.87453E+00	1.85225E-03	2.77993E-03	4.24508E-03	3.50126E-03	7.17659E-03
24	1.14148E-01	2.51666E+00	5.16504E+00	1.58231E-03	2.37481E-03	4.62064E-03	3.98214E-03	8.17272E-03
25	8.57929E-02	3.47833E+00	7.14022E+00	1.04572E-03	1.56946E-03	4.06296E-03	3.63736E-03	7.46666E-03
26	5.39375E-02	5.75904E+00	1.18196E+01	4.09719E-04	6.14925E-04	2.53206E-03	2.35959E-03	4.84273E-03
THERMAL	1.33419E-01	2.09624E+00	4.28745E+00	8.28052E-03	1.24278E-02	2.06880E-02	1.73580E-02	3.55024E-02
TOTAL	4.32301E-01	4.44012E-01	8.65031E-01	4.48069E-02	6.72483E-02	3.45492E-02	1.98948E-02	3.87594E-02

0REGION 30 MATERIAL CLAD VOLUME 6.792215E-01

CROSS-SECTIONS FLUXES

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                      REACTIONS
 GROUP   DIFFUSION    ABSORPTION   NU*FISSION          RIF           RAF
        TRANSPORT    ABSORPTION   NU*FISSION

    1   3.16321E+00 -5.37083E-05   .00000E+00      2.16933E-03
3.19385E-03      2.28600E-04 -1.16511E-07   .00000E+00
    2   2.49217E+00  3.74237E-04   .00000E+00      7.30453E-03
1.07543E-02      9.76998E-04  2.73363E-06   .00000E+00
    3   1.67902E+00  5.04506E-04   .00000E+00      6.00412E-03
8.83970E-03      1.19199E-03  3.02911E-06   .00000E+00
    4   1.12257E+00  5.15706E-04   .00000E+00      3.35486E-03
4.93928E-03      9.96188E-04  1.73012E-06   .00000E+00
    5   9.21946E-01  6.62955E-04   .00000E+00      2.04420E-03
3.00963E-03      7.39090E-04  1.35521E-06   .00000E+00
    6   9.25362E-01  1.34549E-03   .00000E+00      2.71550E-03
3.99796E-03      9.78176E-04  3.65368E-06   .00000E+00
    7   8.79884E-01  6.78327E-03   .00000E+00      2.29340E-03
3.37651E-03      8.68827E-04  1.55567E-05   .00000E+00
    8   1.34604E+00  1.24927E-02   .00000E+00      2.17502E-03
3.20222E-03      5.38620E-04  2.71718E-05   .00000E+00
    9   1.59365E+00  1.42310E-04   .00000E+00      1.05371E-03
1.55135E-03      2.20397E-04  1.49953E-07   .00000E+00
   10   1.57942E+00  2.24967E-04   .00000E+00      1.01502E-03
1.49438E-03      2.14218E-04  2.28346E-07   .00000E+00
   11   1.57349E+00  3.85370E-04   .00000E+00      9.78021E-04
1.43992E-03      2.07187E-04  3.76900E-07   .00000E+00
   12   1.57020E+00  6.03067E-04   .00000E+00      6.98548E-04
1.02845E-03      1.48293E-04  4.21271E-07   .00000E+00
   13   1.57409E+00  8.53588E-04   .00000E+00      6.72241E-04
9.89723E-04      1.42355E-04  5.73817E-07   .00000E+00
   14   1.56863E+00  1.16803E-03   .00000E+00      6.43065E-04
9.46767E-04      1.36651E-04  7.51120E-07   .00000E+00
   15   1.56887E+00  1.37786E-03   .00000E+00      7.34517E-05
1.08141E-04      1.56061E-05  1.01206E-07   .00000E+00
   16   1.56929E+00  1.42882E-03   .00000E+00      7.67399E-05
1.12982E-04      1.63003E-05  1.09648E-07   .00000E+00
   17   1.56537E+00  1.48846E-03   .00000E+00      9.71000E-05
1.42958E-04      2.06767E-05  1.44530E-07   .00000E+00
   18   1.56347E+00  1.67631E-03   .00000E+00      4.17277E-04
6.14346E-04      8.89641E-05  6.99485E-07   .00000E+00
   19   1.55973E+00  2.13456E-03   .00000E+00      6.61176E-04
9.73433E-04      1.41301E-04  1.41132E-06   .00000E+00
   20   1.55524E+00  2.60384E-03   .00000E+00      2.49212E-04
3.66908E-04      5.34134E-05  6.48908E-07   .00000E+00
   21   1.55192E+00  3.12774E-03   .00000E+00      5.62257E-04
8.27797E-04      1.20766E-04  1.75860E-06   .00000E+00
   22   1.54000E+00  4.45409E-03   .00000E+00      2.49999E-03
3.68067E-03      5.41124E-04  1.11352E-05   .00000E+00
   23   1.52869E+00  5.81622E-03   .00000E+00      2.37324E-03
3.49406E-03      5.17490E-04  1.38033E-05   .00000E+00
   24   1.51588E+00  7.36893E-03   .00000E+00      2.18570E-03
3.21795E-03      4.80623E-04  1.61063E-05   .00000E+00
   25   1.49532E+00  9.79734E-03   .00000E+00      1.61720E-03
2.38096E-03      3.60503E-04  1.58443E-05   .00000E+00
   26   1.43463E+00  1.63396E-02   .00000E+00      8.13529E-04
1.19774E-03      1.89021E-04  1.32928E-05   .00000E+00

 THERMAL 1.51986E+00  6.75046E-03   .00000E+00      1.09623E-02
1.61395E-02      2.40424E-03  7.40006E-05   .00000E+00

  TOTAL  1.47198E+00  2.96481E-03   .00000E+00      4.47484E-02
6.58820E-02      1.01334E-02  1.32671E-04   .00000E+00


0REGION 31     MATERIAL COOLANT     VOLUME  1.206310E+01
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                   CROSS-SECTIONS                               FLUXES
                     REACTIONS
 GROUP    DIFFUSION    ABSORPTION   NU*FISSION            RIF          RAF
        TRANSPORT    ABSORPTION  NU*FISSION

    1     2.70642E+00  2.17943E-03   .00000E+00       3.13607E-02
2.59972E-03        3.86250E-03  6.83484E-05   .00000E+00
    2     2.01031E+00  1.01023E-05   .00000E+00       1.05518E-01
8.74716E-03        1.74961E-02  1.06598E-06   .00000E+00
    3     1.31913E+00  3.55934E-06   .00000E+00       8.89952E-02
7.37747E-03        2.24883E-02  3.16764E-07   .00000E+00
    4     7.92566E-01  3.59323E-06   .00000E+00       5.32127E-02
4.41119E-03        2.23799E-02  1.91205E-07   .00000E+00
    5     6.54618E-01  6.63753E-06   .00000E+00       3.45464E-02
2.86381E-03        1.75911E-02  2.29303E-07   .00000E+00
    6     5.56182E-01  1.94203E-05   .00000E+00       4.76865E-02
3.95309E-03        2.85797E-02  9.26088E-07   .00000E+00
    7     5.63543E-01  6.29493E-05   .00000E+00       4.08057E-02
3.38269E-03        2.41364E-02  2.56869E-06   .00000E+00
    8     5.77544E-01  1.71416E-04   .00000E+00       3.90661E-02
3.23848E-03        2.25473E-02  6.69655E-06   .00000E+00
    9     5.80367E-01  3.53087E-04   .00000E+00       1.90899E-02
1.58250E-03        1.09643E-02  6.74038E-06   .00000E+00
   10     5.80777E-01  5.82085E-04   .00000E+00       1.86306E-02
1.54443E-03        1.06929E-02  1.08446E-05   .00000E+00
   11     5.80114E-01  9.59742E-04   .00000E+00       1.81177E-02
1.50191E-03        1.04104E-02  1.73883E-05   .00000E+00
   12     5.78579E-01  1.47308E-03   .00000E+00       1.29583E-02
1.07421E-03        7.46556E-03  1.90885E-05   .00000E+00
   13     5.30274E-01  2.07713E-03   .00000E+00       1.20520E-02
9.99080E-04        7.57596E-03  2.50336E-05   .00000E+00
   14     4.84734E-01  2.83531E-03   .00000E+00       1.15633E-02
9.58566E-04        7.95163E-03  3.27855E-05   .00000E+00
   15     4.56194E-01  3.34008E-03   .00000E+00       1.37103E-03
1.13655E-04        1.00179E-03  4.57935E-06   .00000E+00
   16     4.50003E-01  3.46267E-03   .00000E+00       1.41466E-03
1.17272E-04        1.04789E-03  4.89850E-06   .00000E+00
   17     4.42700E-01  3.60538E-03   .00000E+00       1.77538E-03
1.47174E-04        1.33678E-03  6.40092E-06   .00000E+00
   18     4.19829E-01  4.05921E-03   .00000E+00       7.61487E-03
6.31253E-04        6.04600E-03  3.09103E-05   .00000E+00
   19     3.66366E-01  5.17002E-03   .00000E+00       1.22885E-02
1.01869E-03        1.11806E-02  6.35319E-05   .00000E+00
   20     3.18292E-01  6.29875E-03   .00000E+00       4.82177E-03
3.99712E-04        5.04963E-03  3.03711E-05   .00000E+00
   21     2.76329E-01  7.56452E-03   .00000E+00       1.09211E-02
9.05331E-04        1.31740E-02  8.26128E-05   .00000E+00
   22     2.07862E-01  1.07793E-02   .00000E+00       4.93395E-02
4.09012E-03        7.91221E-02  5.31843E-04   .00000E+00
   23     1.64184E-01  1.40641E-02   .00000E+00       4.82861E-02
4.00279E-03        9.80324E-02  6.79100E-04   .00000E+00
   24     1.30440E-01  1.78189E-02   .00000E+00       4.59004E-02
3.80502E-03        1.17296E-01  8.17894E-04   .00000E+00
   25     1.01321E-01  2.37240E-02   .00000E+00       3.52725E-02
2.92400E-03        1.16042E-01  8.36805E-04   .00000E+00
   26     7.46679E-02  3.97569E-02   .00000E+00       1.87722E-02
1.55616E-03        8.38029E-02  7.46323E-04   .00000E+00

 THERMAL 1.43595E-01  1.67928E-02   .00000E+00       2.25602E-01
1.87018E-02        5.23700E-01  3.78848E-03   .00000E+00

  TOTAL  3.44086E-01  5.22115E-03   .00000E+00       7.71381E-01
6.39455E-02        7.47274E-01  4.02749E-03   .00000E+00
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0REGION 32     MATERIAL FUEL_8      VOLUME  1.332561E+00

                        CROSS-SECTIONS                              FLUXES
                          REACTIONS
 GROUP   DIFFUSION    ABSORPTION   NU*FISSION          RIF           RAF
        TRANSPORT    ABSORPTION   NU*FISSION

    1    2.32885E+00  1.78377E-02  5.28855E-02      5.18176E-03
3.88858E-03       7.41677E-04  9.24309E-05  2.74040E-04
    2    1.91875E+00  1.57905E-02  3.97823E-02      1.73296E-02
1.30047E-02       3.01057E-03  2.73643E-04  6.89410E-04
    3    1.18870E+00  8.57947E-03  1.48435E-02      1.38333E-02
1.03810E-02       3.87910E-03  1.18682E-04  2.05334E-04
    4    8.16399E-01  9.08466E-03  1.44654E-02      7.27336E-03
5.45818E-03       2.96969E-03  6.60760E-05  1.05212E-04
    5    8.11678E-01  1.23757E-02  1.74729E-02      4.16349E-03
3.12443E-03       1.70983E-03  5.15261E-05  7.27483E-05
    6    7.06212E-01  2.23040E-02  2.45211E-02      5.31555E-03
3.98898E-03       2.50895E-03  1.18558E-04  1.30343E-04
    7    6.33971E-01  4.98772E-02  5.54151E-02      4.42755E-03
3.32258E-03       2.32794E-03  2.20834E-04  2.45353E-04
    8    5.71165E-01  1.29766E-01  1.54831E-01      4.15692E-03
3.11950E-03       2.42599E-03  5.39426E-04  6.43622E-04
    9    4.83010E-01  2.42135E-01  2.40387E-01      1.98458E-03
1.48930E-03       1.36959E-03  4.80535E-04  4.77067E-04
   10    3.98856E-01  3.85543E-01  4.87313E-01      1.87703E-03
1.40859E-03       1.56868E-03  7.23677E-04  9.14702E-04
   11    3.78054E-01  4.94895E-01  5.51727E-01      1.78473E-03
1.33933E-03       1.57362E-03  8.83256E-04  9.84686E-04
   12    3.77531E-01  5.09213E-01  1.61345E-01      1.27184E-03
9.54434E-04       1.12295E-03  6.47638E-04  2.05205E-04
   13    6.68706E-01  1.23913E-01  2.00238E-01      1.28468E-03
9.64068E-04       6.40381E-04  1.59188E-04  2.57242E-04
   14    6.13009E-01  1.58972E-01  2.55781E-01      1.22401E-03
9.18540E-04       6.65575E-04  1.94583E-04  3.13079E-04
   15    3.30150E-01  6.22014E-01  1.20522E+00      1.32341E-04
9.93132E-05       1.33617E-04  8.23179E-05  1.59500E-04
   16    4.05488E-01  4.34493E-01  8.86229E-01      1.41164E-04
1.05934E-04       1.16044E-04  6.13348E-05  1.25104E-04
   17    4.60036E-01  3.35121E-01  7.02402E-01      1.80683E-04
1.35591E-04       1.30920E-04  6.05507E-05  1.26912E-04
   18    4.74286E-01  3.11396E-01  6.58316E-01      7.79023E-04
5.84606E-04       5.47506E-04  2.42584E-04  5.12843E-04
   19    3.62015E-01  5.26401E-01  1.08735E+00      1.20243E-03
9.02347E-04       1.10717E-03  6.32961E-04  1.30746E-03
   20    2.31685E-01  1.04238E+00  2.03659E+00      4.25176E-04
3.19067E-04       6.11715E-04  4.43195E-04  8.65910E-04
   21    2.29393E-01  1.05628E+00  2.06603E+00      9.56546E-04
7.17825E-04       1.38997E-03  1.01038E-03  1.97626E-03
   22    1.89976E-01  1.35500E+00  2.75817E+00      4.09449E-03
3.07265E-03       7.18422E-03  5.54804E-03  1.12933E-02
   23    1.45443E-01  1.89028E+00  3.87453E+00      3.63694E-03
2.72929E-03       8.33533E-03  6.87482E-03  1.40914E-02
   24    1.14148E-01  2.51666E+00  5.16504E+00      3.10687E-03
2.33150E-03       9.07262E-03  7.81893E-03  1.60471E-02
   25    8.57929E-02  3.47833E+00  7.14022E+00      2.05438E-03
1.54168E-03       7.98193E-03  7.14581E-03  1.46687E-02
   26    5.39375E-02  5.75904E+00  1.18196E+01      8.05910E-04
6.04783E-04       4.98051E-03  4.64127E-03  9.52556E-03

 THERMAL 1.33476E-01  2.09519E+00  4.28526E+00      1.62827E-02
1.22191E-02       4.06635E-02  3.41154E-02  6.97758E-02

  TOTAL  4.33756E-01  4.41552E-01  8.60014E-01      8.86243E-02
6.65068E-02       6.81061E-02  3.91322E-02  7.62182E-02
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0REGION 33 MATERIAL CLAD VOLUME 1.358440E+00

	CROSS-SECTIONS	REACTIONS		FLUXES	
GROUP	DIFFUSION	ABSORPTION	NU*FISSION	RIF	RAF
	TRANSPORT	ABSORPTION	NU*FISSION		
1	3.16321E+00	-5.37083E-05	.00000E+00	4.31271E-03	
3.17475E-03	4.54465E-04	-2.31628E-07	.00000E+00		
2	2.49217E+00	3.74237E-04	.00000E+00	1.45506E-02	
1.07113E-02	1.94618E-03	5.44538E-06	.00000E+00		
3	1.67902E+00	5.04506E-04	.00000E+00	1.19539E-02	
8.79971E-03	2.37319E-03	6.03081E-06	.00000E+00		
4	1.12257E+00	5.15706E-04	.00000E+00	6.66477E-03	
4.90620E-03	1.97903E-03	3.43706E-06	.00000E+00		
5	9.21946E-01	6.62955E-04	.00000E+00	4.05079E-03	
2.98194E-03	1.46458E-03	2.68549E-06	.00000E+00		
6	9.25362E-01	1.34549E-03	.00000E+00	5.37503E-03	
3.95677E-03	1.93619E-03	7.23204E-06	.00000E+00		
7	8.79884E-01	6.78327E-03	.00000E+00	4.53884E-03	
3.34121E-03	1.71948E-03	3.07881E-05	.00000E+00		
8	1.34604E+00	1.24927E-02	.00000E+00	4.30399E-03	
3.16833E-03	1.06584E-03	5.37683E-05	.00000E+00		
9	1.59365E+00	1.42310E-04	.00000E+00	2.08385E-03	
1.53400E-03	4.35864E-04	2.96551E-07	.00000E+00		
10	1.57942E+00	2.24967E-04	.00000E+00	2.00619E-03	
1.47683E-03	4.23403E-04	4.51327E-07	.00000E+00		
11	1.57349E+00	3.85370E-04	.00000E+00	1.93213E-03	
1.42232E-03	4.09309E-04	7.44586E-07	.00000E+00		
12	1.57020E+00	6.03067E-04	.00000E+00	1.37982E-03	
1.01574E-03	2.92919E-04	8.32124E-07	.00000E+00		
13	1.57409E+00	8.53588E-04	.00000E+00	1.33025E-03	
9.79249E-04	2.81697E-04	1.13549E-06	.00000E+00		
14	1.56863E+00	1.16803E-03	.00000E+00	1.27233E-03	
9.36608E-04	2.70369E-04	1.48612E-06	.00000E+00		
15	1.56887E+00	1.37786E-03	.00000E+00	1.45043E-04	
1.06772E-04	3.08170E-05	1.99849E-07	.00000E+00		
16	1.56929E+00	1.42882E-03	.00000E+00	1.51631E-04	
1.11621E-04	3.22079E-05	2.16653E-07	.00000E+00		
17	1.56537E+00	1.48846E-03	.00000E+00	1.91941E-04	
1.41295E-04	4.08725E-05	2.85697E-07	.00000E+00		
18	1.56347E+00	1.67631E-03	.00000E+00	8.24941E-04	
6.07271E-04	1.75879E-04	1.38286E-06	.00000E+00		
19	1.55973E+00	2.13456E-03	.00000E+00	1.30608E-03	
9.61457E-04	2.79125E-04	2.78791E-06	.00000E+00		
20	1.55524E+00	2.60384E-03	.00000E+00	4.91709E-04	
3.61966E-04	1.05388E-04	1.28033E-06	.00000E+00		
21	1.55192E+00	3.12774E-03	.00000E+00	1.10991E-03	
8.17050E-04	2.38396E-04	3.47152E-06	.00000E+00		
22	1.54000E+00	4.45409E-03	.00000E+00	4.91807E-03	
3.62038E-03	1.06452E-03	2.19055E-05	.00000E+00		
23	1.52869E+00	5.81622E-03	.00000E+00	4.66021E-03	
3.43056E-03	1.01617E-03	2.71048E-05	.00000E+00		
24	1.51588E+00	7.36893E-03	.00000E+00	4.29247E-03	
3.15985E-03	9.43890E-04	3.16309E-05	.00000E+00		
25	1.49532E+00	9.79734E-03	.00000E+00	3.17796E-03	
2.33942E-03	7.08424E-04	3.11355E-05	.00000E+00		
26	1.43463E+00	1.63396E-02	.00000E+00	1.60043E-03	
1.17814E-03	3.71856E-04	2.61505E-05	.00000E+00		
THERMAL	1.51988E+00	6.74807E-03	.00000E+00	2.15568E-02	
1.58688E-02	4.72777E-03	1.45467E-04	.00000E+00		
TOTAL	1.47267E+00	2.95235E-03	.00000E+00	8.86256E-02	

6.52407E-02 2.00600E-02 2.61654E-04 .00000E+00

0REGION 34 MATERIAL COOLANT VOLUME 2.887479E+01

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                    CROSS-SECTIONS                                  FLUXES
                      REACTIONS
GROUP    DIFFUSION    ABSORPTION   NU*FISSION           RIF              RAF
        TRANSPORT    ABSORPTION   NU*FISSION

   1    2.70642E+00  2.17943E-03   .00000E+00      7.28280E-02
2.52220E-03      8.96978E-03  1.58723E-04   .00000E+00
   2    2.01031E+00  1.01023E-05   .00000E+00      2.46107E-01
8.52325E-03      4.08076E-02  2.48626E-06   .00000E+00
   3    1.31913E+00  3.55934E-06   .00000E+00      2.07808E-01
7.19687E-03      5.25113E-02  7.39660E-07   .00000E+00
   4    7.92566E-01  3.59323E-06   .00000E+00      1.24619E-01
4.31585E-03      5.24118E-02  4.47786E-07   .00000E+00
   5    6.54618E-01  6.63753E-06   .00000E+00      8.10788E-02
2.80794E-03      4.12855E-02  5.38163E-07   .00000E+00
   6    5.56182E-01  1.94203E-05   .00000E+00      1.12149E-01
3.88397E-03      6.72136E-02  2.17797E-06   .00000E+00
   7    5.63543E-01  6.29493E-05   .00000E+00      9.60727E-02
3.32722E-03      5.68266E-02  6.04772E-06   .00000E+00
   8    5.77544E-01  1.71416E-04   .00000E+00      9.20588E-02
3.18821E-03      5.31324E-02  1.57804E-05   .00000E+00
   9    5.80367E-01  3.53087E-04   .00000E+00      4.49891E-02
1.55808E-03      2.58395E-02  1.58851E-05   .00000E+00
  10    5.80777E-01  5.82085E-04   .00000E+00      4.39308E-02
1.52142E-03      2.52138E-02  2.55715E-05   .00000E+00
  11    5.80114E-01  9.59742E-04   .00000E+00      4.27361E-02
1.48005E-03      2.45561E-02  4.10156E-05   .00000E+00
  12    5.78579E-01  1.47308E-03   .00000E+00      3.05655E-02
1.05855E-03      1.76095E-02  4.50254E-05   .00000E+00
  13    5.30274E-01  2.07713E-03   .00000E+00      2.84075E-02
9.83818E-04      1.78571E-02  5.90061E-05   .00000E+00
  14    4.84734E-01  2.83531E-03   .00000E+00      2.72564E-02
9.43952E-04      1.87432E-02  7.72804E-05   .00000E+00
  15    4.56194E-01  3.34008E-03   .00000E+00      3.23397E-03
1.12000E-04      2.36301E-03  1.08017E-05   .00000E+00
  16    4.50003E-01  3.46267E-03   .00000E+00      3.33604E-03
1.15535E-04      2.47113E-03  1.15516E-05   .00000E+00
  17    4.42700E-01  3.60538E-03   .00000E+00      4.18598E-03
1.44970E-04      3.15185E-03  1.50920E-05   .00000E+00
  18    4.19829E-01  4.05921E-03   .00000E+00      1.79539E-02
6.21785E-04      1.42549E-02  7.28786E-05   .00000E+00
  19    3.66366E-01  5.17002E-03   .00000E+00      2.89893E-02
1.00396E-03      2.63755E-02  1.49875E-04   .00000E+00
  20    3.18292E-01  6.29875E-03   .00000E+00      1.13971E-02
3.94709E-04      1.19357E-02  7.17877E-05   .00000E+00
  21    2.76329E-01  7.56452E-03   .00000E+00      2.58440E-02
8.95037E-04      3.11754E-02  1.95497E-04   .00000E+00
  22    2.07862E-01  1.07793E-02   .00000E+00      1.16442E-01
4.03264E-03      1.86729E-01  1.25516E-03   .00000E+00
  23    1.64184E-01  1.40641E-02   .00000E+00      1.13904E-01
3.94475E-03      2.31252E-01  1.60196E-03   .00000E+00
  24    1.30440E-01  1.78189E-02   .00000E+00      1.08449E-01
3.75584E-03      2.77137E-01  1.93244E-03   .00000E+00
  25    1.01321E-01  2.37240E-02   .00000E+00      8.35519E-02
2.89359E-03      2.74874E-01  1.98218E-03   .00000E+00
  26    7.46679E-02  3.97569E-02   .00000E+00      4.46394E-02
1.54597E-03      1.99280E-01  1.77473E-03   .00000E+00

 THERMAL 1.43481E-01  1.68105E-02   .00000E+00      5.33217E-01
1.84665E-02      1.23876E+00  8.96363E-03   .00000E+00
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  TOTAL  3.42509E-01  5.25490E-03   .00000E+00      1.81253E+00
6.27722E-02       1.76398E+00  9.52468E-03   .00000E+00


0REGION 35     MATERIAL FUEL_9      VOLUME  1.332556E+00

                        CROSS-SECTIONS                              FLUXES
                          REACTIONS
 GROUP  DIFFUSION     ABSORPTION   NU*FISSION             RIF            RAF
       TRANSPORT     ABSORPTION   NU*FISSION

    1   2.32885E+00  1.78377E-02  5.28855E-02      4.96153E-03
3.72332E-03       7.10154E-04  8.85024E-05  2.62393E-04
    2   1.91875E+00  1.57905E-02  3.97823E-02      1.66913E-02
1.25258E-02       2.89969E-03  2.63564E-04  6.64018E-04
    3   1.18870E+00  8.57947E-03  1.48435E-02      1.33782E-02
1.00395E-02       3.75148E-03  1.14778E-04  1.98579E-04
    4   8.16399E-01  9.08466E-03  1.44654E-02      7.06745E-03
5.30368E-03       2.88562E-03  6.42054E-05  1.02234E-04
    5   8.11678E-01  1.23757E-02  1.74729E-02      4.05577E-03
3.04360E-03       1.66559E-03  5.01930E-05  7.08661E-05
    6   7.06212E-01  2.23040E-02  2.45211E-02      5.19033E-03
3.89502E-03       2.44984E-03  1.15765E-04  1.27272E-04
    7   6.33971E-01  4.98772E-02  5.54151E-02      4.32899E-03
3.24864E-03       2.27612E-03  2.15918E-04  2.39892E-04
    8   5.71165E-01  1.29766E-01  1.54831E-01      4.06827E-03
3.05298E-03       2.37425E-03  5.27922E-04  6.29895E-04
    9   4.83010E-01  2.42135E-01  2.40387E-01      1.94245E-03
1.45769E-03       1.34052E-03  4.70335E-04  4.66941E-04
   10   3.98856E-01  3.85543E-01  4.87313E-01      1.83762E-03
1.37902E-03       1.53574E-03  7.08481E-04  8.95494E-04
   11   3.78054E-01  4.94895E-01  5.51727E-01      1.74732E-03
1.31126E-03       1.54063E-03  8.64740E-04  9.64043E-04
   12   3.77531E-01  5.09213E-01  1.61345E-01      1.24507E-03
9.34346E-04       1.09931E-03  6.34004E-04  2.00885E-04
   13   6.68706E-01  1.23913E-01  2.00238E-01      1.25744E-03
9.43630E-04       6.26803E-04  1.55813E-04  2.51787E-04
   14   6.13009E-01  1.58972E-01  2.55781E-01      1.19800E-03
8.99021E-04       6.51429E-04  1.90448E-04  3.06425E-04
   15   3.30150E-01  6.22014E-01  1.20522E+00      1.29503E-04
9.71838E-05       1.30752E-04  8.05526E-05  1.56080E-04
   16   4.05488E-01  4.34493E-01  8.86229E-01      1.38133E-04
1.03660E-04       1.13553E-04  6.00177E-05  1.22417E-04
   17   4.60036E-01  3.35121E-01  7.02402E-01      1.76800E-04
1.32678E-04       1.28106E-04  5.92494E-05  1.24185E-04
   18   4.74286E-01  3.11396E-01  6.58316E-01      7.62192E-04
5.71978E-04       5.35677E-04  2.37343E-04  5.01763E-04
   19   3.62015E-01  5.26401E-01  1.08735E+00      1.17614E-03
8.82623E-04       1.08296E-03  6.19123E-04  1.27888E-03
   20   2.31685E-01  1.04238E+00  2.03659E+00      4.15951E-04
3.12146E-04       5.98443E-04  4.33579E-04  8.47123E-04
   21   2.29393E-01  1.05628E+00  2.06603E+00      9.35980E-04
7.02395E-04       1.36008E-03  9.88656E-04  1.93377E-03
   22   1.89976E-01  1.35500E+00  2.75817E+00      3.98828E-03
2.99296E-03       6.99787E-03  5.40413E-03  1.10003E-02
   23   1.45443E-01  1.89028E+00  3.87453E+00      3.52983E-03
2.64892E-03       8.08986E-03  6.67236E-03  1.36764E-02
   24   1.14148E-01  2.51666E+00  5.16504E+00      3.00731E-03
2.25680E-03       8.78189E-03  7.56837E-03  1.55329E-02
   25   8.57929E-02  3.47833E+00  7.14022E+00      1.98322E-03
1.48828E-03       7.70545E-03  6.89830E-03  1.41606E-02
   26   5.39375E-02  5.75904E+00  1.18196E+01      7.76447E-04
5.82675E-04       4.79844E-03  4.47159E-03  9.17733E-03

 THERMAL 1.33732E-01  2.09042E+00  4.27539E+00      1.58132E-02
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 1.18668E-02       3.94150E-02  3.30561E-02  6.76074E-02

   TOTAL  4.33435E-01  4.41425E-01  8.59321E-01      8.59895E-02
 6.45297E-02       6.61303E-02  3.79579E-02  7.38926E-02


 0REGION 36     MATERIAL CLAD        VOLUME  1.358417E+00

                        CROSS-SECTIONS                            FLUXES
                          REACTIONS
  GROUP   DIFFUSION    ABSORPTION   NU*FISSION            RIF           RAF
         TRANSPORT    ABSORPTION   NU*FISSION

     1    3.16321E+00 -5.37083E-05   .00000E+00      4.16024E-03
 3.06257E-03       4.38399E-04 -2.23439E-07   .00000E+00
     2    2.49217E+00  3.74237E-04   .00000E+00      1.41077E-02
 1.03854E-02       1.88694E-03  5.27963E-06   .00000E+00
     3    1.67902E+00  5.04506E-04   .00000E+00      1.16062E-02
 8.54389E-03       2.30415E-03  5.85538E-06   .00000E+00
     4    1.12257E+00  5.15706E-04   .00000E+00      6.49058E-03
 4.77805E-03       1.92730E-03  3.34723E-06   .00000E+00
     5    9.21946E-01  6.62955E-04   .00000E+00      3.94997E-03
 2.90777E-03       1.42813E-03  2.61865E-06   .00000E+00
     6    9.25362E-01  1.34549E-03   .00000E+00      5.24925E-03
 3.86424E-03       1.89088E-03  7.06281E-06   .00000E+00
     7    8.79884E-01  6.78327E-03   .00000E+00      4.43728E-03
 3.26651E-03       1.68101E-03  3.00993E-05   .00000E+00
     8    1.34604E+00  1.24927E-02   .00000E+00      4.21123E-03
 3.10010E-03       1.04287E-03  5.26095E-05   .00000E+00
     9    1.59365E+00  1.42310E-04   .00000E+00      2.03889E-03
 1.50093E-03       4.26460E-04  2.90153E-07   .00000E+00
    10    1.57942E+00  2.24967E-04   .00000E+00      1.96312E-03
 1.44515E-03       4.14313E-04  4.41637E-07   .00000E+00
    11    1.57349E+00  3.85370E-04   .00000E+00      1.89056E-03
 1.39174E-03       4.00503E-04  7.28566E-07   .00000E+00
    12    1.57020E+00  6.03067E-04   .00000E+00      1.34996E-03
 9.93778E-04       2.86581E-04  8.14119E-07   .00000E+00
    13    1.57409E+00  8.53588E-04   .00000E+00      1.30171E-03
 9.58259E-04       2.75654E-04  1.11113E-06   .00000E+00
    14    1.56863E+00  1.16803E-03   .00000E+00      1.24495E-03
 9.16470E-04       2.64552E-04  1.45414E-06   .00000E+00
    15    1.56887E+00  1.37786E-03   .00000E+00      1.41853E-04
 1.04425E-04       3.01391E-05  1.95453E-07   .00000E+00
    16    1.56929E+00  1.42882E-03   .00000E+00      1.48303E-04
 1.09173E-04       3.15010E-05  2.11898E-07   .00000E+00
    17    1.56537E+00  1.48846E-03   .00000E+00      1.87739E-04
 1.38204E-04       3.99777E-05  2.79442E-07   .00000E+00
    18    1.56347E+00  1.67631E-03   .00000E+00      8.06822E-04
 5.93943E-04       1.72016E-04  1.35248E-06   .00000E+00
    19    1.55973E+00  2.13456E-03   .00000E+00      1.27686E-03
 9.39963E-04       2.72881E-04  2.72554E-06   .00000E+00
    20    1.55524E+00  2.60384E-03   .00000E+00      4.80677E-04
 3.53851E-04       1.03023E-04  1.25161E-06   .00000E+00
    21    1.55192E+00  3.12774E-03   .00000E+00      1.08517E-03
 7.98850E-04       2.33081E-04  3.39414E-06   .00000E+00
    22    1.54000E+00  4.45409E-03   .00000E+00      4.78531E-03
 3.52271E-03       1.03578E-03  2.13142E-05   .00000E+00
    23    1.52869E+00  5.81622E-03   .00000E+00      4.51695E-03
 3.32516E-03       9.84931E-04  2.62716E-05   .00000E+00
    24    1.51588E+00  7.36893E-03   .00000E+00      4.14901E-03
 3.05430E-03       9.12344E-04  3.05738E-05   .00000E+00
    25    1.49532E+00  9.79734E-03   .00000E+00      3.06299E-03
 2.25482E-03       6.82796E-04  3.00092E-05   .00000E+00
    26    1.43463E+00  1.63396E-02   .00000E+00      1.53880E-03
 1.13279E-03       3.57536E-04  2.51434E-05   .00000E+00
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 THERMAL 1.52001E+00  6.73262E-03   .00000E+00      2.08958E-02
1.53824E-02      4.58237E-03  1.40683E-04   .00000E+00

  TOTAL  1.47141E+00  2.94970E-03   .00000E+00      8.61821E-02
6.34431E-02      1.95238E-02  2.54211E-04   .00000E+00


0REGION 37     MATERIAL COOLANT     VOLUME  1.717950E+01

                    CROSS-SECTIONS                          FLUXES
                      REACTIONS
 GROUP   DIFFUSION    ABSORPTION   NU*FISSION           RIF          RAF
        TRANSPORT    ABSORPTION   NU*FISSION

    1    2.70642E+00  2.17943E-03   .00000E+00      4.22463E-02
2.45911E-03      5.20321E-03  9.20727E-05   .00000E+00
    2    2.01031E+00  1.01023E-05   .00000E+00      1.43568E-01
8.35695E-03      2.38054E-02  1.45037E-06   .00000E+00
    3    1.31913E+00  3.55934E-06   .00000E+00      1.21319E-01
7.06185E-03      3.06563E-02  4.31816E-07   .00000E+00
    4    7.92566E-01  3.59323E-06   .00000E+00      7.27912E-02
4.23710E-03      3.06142E-02  2.61556E-07   .00000E+00
    5    6.54618E-01  6.63753E-06   .00000E+00      4.72937E-02
2.75292E-03      2.40821E-02  3.13914E-07   .00000E+00
    6    5.56182E-01  1.94203E-05   .00000E+00      6.54015E-02
3.80695E-03      3.91967E-02  1.27012E-06   .00000E+00
    7    5.63543E-01  6.29493E-05   .00000E+00      5.60555E-02
3.26293E-03      3.31566E-02  3.52866E-06   .00000E+00
    8    5.77544E-01  1.71416E-04   .00000E+00      5.37369E-02
3.12796E-03      3.10146E-02  9.21136E-06   .00000E+00
    9    5.80367E-01  3.53087E-04   .00000E+00      2.62592E-02
1.52852E-03      1.50820E-02  9.27179E-06   .00000E+00
   10    5.80777E-01  5.82085E-04   .00000E+00      2.56305E-02
1.49192E-03      1.47105E-02  1.49191E-05   .00000E+00
   11    5.80114E-01  9.59742E-04   .00000E+00      2.49229E-02
1.45074E-03      1.43207E-02  2.39195E-05   .00000E+00
   12    5.78579E-01  1.47308E-03   .00000E+00      1.78226E-02
1.03744E-03      1.02680E-02  2.62541E-05   .00000E+00
   13    5.30274E-01  2.07713E-03   .00000E+00      1.65846E-02
9.65372E-04      1.04252E-02  3.44484E-05   .00000E+00
   14    4.84734E-01  2.83531E-03   .00000E+00      1.59107E-02
9.26146E-04      1.09412E-02  4.51119E-05   .00000E+00
   15    4.56194E-01  3.34008E-03   .00000E+00      1.88491E-03
1.09719E-04      1.37727E-03  6.29575E-06   .00000E+00
   16    4.50003E-01  3.46267E-03   .00000E+00      1.94520E-03
1.13228E-04      1.44088E-03  6.73558E-06   .00000E+00
   17    4.42700E-01  3.60538E-03   .00000E+00      2.44149E-03
1.42117E-04      1.83833E-03  8.80251E-06   .00000E+00
   18    4.19829E-01  4.05921E-03   .00000E+00      1.04712E-02
6.09518E-04      8.31387E-03  4.25048E-05   .00000E+00
   19    3.66366E-01  5.17002E-03   .00000E+00      1.68883E-02
9.83050E-04      1.53656E-02  8.73129E-05   .00000E+00
   20    3.18292E-01  6.29875E-03   .00000E+00      6.62622E-03
3.85705E-04      6.93936E-03  4.17370E-05   .00000E+00
   21    2.76329E-01  7.56452E-03   .00000E+00      1.50157E-02
8.74050E-04      1.81134E-02  1.13587E-04   .00000E+00
   22    2.07862E-01  1.07793E-02   .00000E+00      6.74157E-02
3.92419E-03      1.08110E-01  7.26691E-04   .00000E+00
   23    1.64184E-01  1.40641E-02   .00000E+00      6.56489E-02
3.82135E-03      1.33283E-01  9.23294E-04   .00000E+00
   24    1.30440E-01  1.78189E-02   .00000E+00      6.21909E-02
3.62007E-03      1.58926E-01  1.10817E-03   .00000E+00
   25    1.01321E-01  2.37240E-02   .00000E+00      4.76163E-02
2.77169E-03      1.56651E-01  1.12965E-03   .00000E+00
   26    7.46679E-02  3.97569E-02   .00000E+00      2.52630E-02
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1.47053E-03       1.12779E-01  1.00438E-03   .00000E+00

 THERMAL 1.43940E-01  1.67441E-02   .00000E+00      3.06665E-01
1.78506E-02       7.10168E-01  5.13482E-03   .00000E+00

  TOTAL  3.45247E-01  5.18697E-03   .00000E+00      1.05295E+00
6.12911E-02       1.01661E+00  5.46163E-03   .00000E+00


0REGION 38     MATERIAL FUEL_10     VOLUME  1.332562E+00

                    CROSS-SECTIONS                           FLUXES
                      REACTIONS
 GROUP  DIFFUSION     ABSORPTION   NU*FISSION          RIF          RAF
       TRANSPORT     ABSORPTION   NU*FISSION

    1   2.32885E+00  1.78377E-02  5.28855E-02       4.93015E-03
3.69976E-03       7.05663E-04  8.79427E-05  2.60734E-04
    2   1.91875E+00  1.57905E-02  3.97823E-02       1.66030E-02
1.24594E-02       2.88435E-03  2.62170E-04  6.60505E-04
    3   1.18870E+00  8.57947E-03  1.48435E-02       1.32865E-02
9.97068E-03       3.72579E-03  1.13991E-04  1.97219E-04
    4   8.16399E-01  9.08466E-03  1.44654E-02       7.01210E-03
5.26212E-03       2.86302E-03  6.37026E-05  1.01433E-04
    5   8.11678E-01  1.23757E-02  1.74729E-02       4.01966E-03
3.01649E-03       1.65076E-03  4.97461E-05  7.02352E-05
    6   7.06212E-01  2.23040E-02  2.45211E-02       5.14349E-03
3.85985E-03       2.42774E-03  1.14720E-04  1.26124E-04
    7   6.33971E-01  4.98772E-02  5.54151E-02       4.29196E-03
3.22084E-03       2.25665E-03  2.14071E-04  2.37840E-04
    8   5.71165E-01  1.29766E-01  1.54831E-01       4.03491E-03
3.02793E-03       2.35478E-03  5.23593E-04  6.24730E-04
    9   4.83010E-01  2.42135E-01  2.40387E-01       1.92656E-03
1.44575E-03       1.32955E-03  4.66486E-04  4.63120E-04
   10   3.98856E-01  3.85543E-01  4.87313E-01       1.82223E-03
1.36746E-03       1.52288E-03  7.02548E-04  8.87995E-04
   11   3.78054E-01  4.94895E-01  5.51727E-01       1.73245E-03
1.30009E-03       1.52751E-03  8.57378E-04  9.55837E-04
   12   3.77531E-01  5.09213E-01  1.61345E-01       1.23447E-03
9.26385E-04       1.08995E-03  6.28606E-04  1.99175E-04
   13   6.68706E-01  1.23913E-01  2.00238E-01       1.24766E-03
9.36286E-04       6.21928E-04  1.54601E-04  2.49829E-04
   14   6.13009E-01  1.58972E-01  2.55781E-01       1.18854E-03
8.91922E-04       6.46288E-04  1.88945E-04  3.04006E-04
   15   3.30150E-01  6.22014E-01  1.20522E+00       1.28383E-04
9.63433E-05       1.29621E-04  7.98563E-05  1.54731E-04
   16   4.05488E-01  4.34493E-01  8.86229E-01       1.36974E-04
1.02790E-04       1.12600E-04  5.95145E-05  1.21391E-04
   17   4.60036E-01  3.35121E-01  7.02402E-01       1.75341E-04
1.31582E-04       1.27049E-04  5.87604E-05  1.23160E-04
   18   4.74286E-01  3.11396E-01  6.58316E-01       7.55949E-04
5.67290E-04       5.31289E-04  2.35399E-04  4.97653E-04
   19   3.62015E-01  5.26401E-01  1.08735E+00       1.16620E-03
8.75159E-04       1.07381E-03  6.13890E-04  1.26807E-03
   20   2.31685E-01  1.04238E+00  2.03659E+00       4.12277E-04
3.09387E-04       5.93156E-04  4.29749E-04  8.39640E-04
   21   2.29393E-01  1.05628E+00  2.06603E+00       9.28383E-04
6.96690E-04       1.34904E-03  9.80631E-04  1.91807E-03
   22   1.89976E-01  1.35500E+00  2.75817E+00       3.96345E-03
2.97431E-03       6.95430E-03  5.37049E-03  1.09319E-02
   23   1.45443E-01  1.89028E+00  3.87453E+00       3.51011E-03
2.63411E-03       8.04467E-03  6.63509E-03  1.36001E-02
   24   1.14148E-01  2.51666E+00  5.16504E+00       2.99070E-03
2.24432E-03       8.73336E-03  7.52655E-03  1.54471E-02
   25   8.57929E-02  3.47833E+00  7.14022E+00       1.97112E-03
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1.47919E-03       7.65843E-03  6.85620E-03  1.40742E-02
   26     5.39375E-02  5.75904E+00  1.18196E+01      7.70520E-04
5.78224E-04       4.76180E-03  4.43746E-03  9.10727E-03

 THERMAL 1.33719E-01  2.09066E+00  4.27590E+00      1.57128E-02
1.17914E-02       3.91686E-02  3.28501E-02  6.71863E-02

  TOTAL  4.33355E-01  4.41681E-01  8.59912E-01      8.53831E-02
6.40744E-02       6.56760E-02  3.77121E-02  7.34220E-02


0REGION 39     MATERIAL CLAD        VOLUME  1.358412E+00

                     CROSS-SECTIONS                          FLUXES
                       REACTIONS
 GROUP   DIFFUSION    ABSORPTION   NU*FISSION            RIF          RAF
        TRANSPORT    ABSORPTION   NU*FISSION

    1    3.16321E+00 -5.37083E-05   .00000E+00      4.11570E-03
3.02978E-03       4.33704E-04 -2.21047E-07   .00000E+00
    2    2.49217E+00  3.74237E-04   .00000E+00      1.39859E-02
1.02958E-02       1.87065E-03  5.23404E-06   .00000E+00
    3    1.67902E+00  5.04506E-04   .00000E+00      1.15047E-02
8.46922E-03       2.28401E-03  5.80419E-06   .00000E+00
    4    1.12257E+00  5.15706E-04   .00000E+00      6.43321E-03
4.73583E-03       1.91027E-03  3.31765E-06   .00000E+00
    5    9.21946E-01  6.62955E-04   .00000E+00      3.91292E-03
2.88051E-03       1.41473E-03  2.59409E-06   .00000E+00
    6    9.25362E-01  1.34549E-03   .00000E+00      5.20121E-03
3.82889E-03       1.87358E-03  6.99817E-06   .00000E+00
    7    8.79884E-01  6.78327E-03   .00000E+00      4.39919E-03
3.23848E-03       1.66658E-03  2.98409E-05   .00000E+00
    8    1.34604E+00  1.24927E-02   .00000E+00      4.17677E-03
3.07475E-03       1.03433E-03  5.21790E-05   .00000E+00
    9    1.59365E+00  1.42310E-04   .00000E+00      2.02236E-03
1.48876E-03       4.23002E-04  2.87800E-07   .00000E+00
   10    1.57942E+00  2.24967E-04   .00000E+00      1.94694E-03
1.43325E-03       4.10899E-04  4.37998E-07   .00000E+00
   11    1.57349E+00  3.85370E-04   .00000E+00      1.87481E-03
1.38015E-03       3.97167E-04  7.22497E-07   .00000E+00
   12    1.57020E+00  6.03067E-04   .00000E+00      1.33872E-03
9.85504E-04       2.84194E-04  8.07339E-07   .00000E+00
   13    1.57409E+00  8.53588E-04   .00000E+00      1.29164E-03
9.50843E-04       2.73520E-04  1.10252E-06   .00000E+00
   14    1.56863E+00  1.16803E-03   .00000E+00      1.23519E-03
9.09292E-04       2.62479E-04  1.44274E-06   .00000E+00
   15    1.56887E+00  1.37786E-03   .00000E+00      1.40658E-04
1.03546E-04       2.98853E-05  1.93807E-07   .00000E+00
   16    1.56929E+00  1.42882E-03   .00000E+00      1.47083E-04
1.08276E-04       3.12419E-05  2.10156E-07   .00000E+00
   17    1.56537E+00  1.48846E-03   .00000E+00      1.86217E-04
1.37085E-04       3.96536E-05  2.77177E-07   .00000E+00
   18    1.56347E+00  1.67631E-03   .00000E+00      8.00304E-04
5.89146E-04       1.70626E-04  1.34156E-06   .00000E+00
   19    1.55973E+00  2.13456E-03   .00000E+00      1.26630E-03
9.32190E-04       2.70623E-04  2.70299E-06   .00000E+00
   20    1.55524E+00  2.60384E-03   .00000E+00      4.76582E-04
3.50838E-04       1.02145E-04  1.24094E-06   .00000E+00
   21    1.55192E+00  3.12774E-03   .00000E+00      1.07665E-03
7.92579E-04       2.31251E-04  3.36748E-06   .00000E+00
   22    1.54000E+00  4.45409E-03   .00000E+00      4.75615E-03
3.50125E-03       1.02947E-03  2.11843E-05   .00000E+00
   23    1.52869E+00  5.81622E-03   .00000E+00      4.49180E-03
3.30665E-03       9.79445E-04  2.61253E-05   .00000E+00
   24    1.51588E+00  7.36893E-03   .00000E+00      4.12550E-03
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3.03700E-03       9.07175E-04  3.04006E-05   .00000E+00
   25     1.49532E+00  9.79734E-03    .00000E+00        3.04321E-03
2.24027E-03       6.78388E-04  2.98154E-05   .00000E+00
   26     1.43463E+00  1.63396E-02    .00000E+00        1.52631E-03
1.12360E-03       3.54633E-04  2.49393E-05   .00000E+00

 THERMAL 1.52002E+00  6.73215E-03    .00000E+00        2.07625E-02
1.52844E-02       4.55313E-03  1.39776E-04   .00000E+00

  TOTAL  1.47142E+00  2.95225E-03    .00000E+00        8.54760E-02
6.29235E-02       1.93637E-02  2.52347E-04   .00000E+00


0REGION 40     MATERIAL COOLANT     VOLUME  9.605871E+00

                    CROSS-SECTIONS                                  FLUXES
                      REACTIONS
 GROUP   DIFFUSION    ABSORPTION   NU*FISSION          RIF             RAF
        TRANSPORT    ABSORPTION   NU*FISSION

    1     2.70642E+00  2.17943E-03    .00000E+00        2.32155E-02
2.41681E-03       2.85932E-03  5.05966E-05   .00000E+00
    2     2.01031E+00  1.01023E-05    .00000E+00        7.90873E-02
8.23323E-03       1.31136E-02  7.98966E-07   .00000E+00
    3     1.31913E+00  3.55934E-06    .00000E+00        6.68481E-02
6.95909E-03       1.68920E-02  2.37935E-07   .00000E+00
    4     7.92566E-01  3.59323E-06    .00000E+00        4.01366E-02
4.17834E-03       1.68804E-02  1.44220E-07   .00000E+00
    5     6.54618E-01  6.63753E-06    .00000E+00        2.60833E-02
2.71535E-03       1.32817E-02  1.73128E-07   .00000E+00
    6     5.56182E-01  1.94203E-05    .00000E+00        3.61080E-02
3.75895E-03       2.16404E-02  7.01229E-07   .00000E+00
    7     5.63543E-01  6.29493E-05    .00000E+00        3.09858E-02
3.22571E-03       1.83279E-02  1.95053E-06   .00000E+00
    8     5.77544E-01  1.71416E-04    .00000E+00        2.97279E-02
3.09476E-03       1.71576E-02  5.09583E-06   .00000E+00
    9     5.80367E-01  3.53087E-04    .00000E+00        1.45316E-02
1.51278E-03       8.34622E-03  5.13092E-06   .00000E+00
   10     5.80777E-01  5.82085E-04    .00000E+00        1.41842E-02
1.47662E-03       8.14094E-03  8.25641E-06   .00000E+00
   11     5.80114E-01  9.59742E-04    .00000E+00        1.37931E-02
1.43590E-03       7.92551E-03  1.32378E-05   .00000E+00
   12     5.78579E-01  1.47308E-03    .00000E+00        9.86413E-03
1.02689E-03       5.68296E-03  1.45306E-05   .00000E+00
   13     5.30274E-01  2.07713E-03    .00000E+00        9.18186E-03
9.55859E-04       5.77177E-03  1.90719E-05   .00000E+00
   14     4.84734E-01  2.83531E-03    .00000E+00        8.80741E-03
9.16878E-04       6.05652E-03  2.49718E-05   .00000E+00
   15     4.56194E-01  3.34008E-03    .00000E+00        1.04296E-03
1.08575E-04       7.62075E-04  3.48357E-06   .00000E+00
   16     4.50003E-01  3.46267E-03    .00000E+00        1.07646E-03
1.12062E-04       7.97369E-04  3.72741E-06   .00000E+00
   17     4.42700E-01  3.60538E-03    .00000E+00        1.35122E-03
1.40666E-04       1.01740E-03  4.87165E-06   .00000E+00
   18     4.19829E-01  4.05921E-03    .00000E+00        5.79500E-03
6.03277E-04       4.60108E-03  2.35231E-05   .00000E+00
   19     3.66366E-01  5.17002E-03    .00000E+00        9.34521E-03
9.72865E-04       8.50263E-03  4.83149E-05   .00000E+00
   20     3.18292E-01  6.29875E-03    .00000E+00        3.66650E-03
3.81694E-04       3.83976E-03  2.30944E-05   .00000E+00
   21     2.76329E-01  7.56452E-03    .00000E+00        8.32061E-03
8.66201E-04       1.00371E-02  6.29414E-05   .00000E+00
   22     2.07862E-01  1.07793E-02    .00000E+00        3.75695E-02
3.91110E-03       6.02475E-02  4.04972E-04   .00000E+00
   23     1.64184E-01  1.40641E-02    .00000E+00        3.66998E-02
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3.82056E-03       7.45095E-02  5.16150E-04   .00000E+00
   24    1.30440E-01  1.78189E-02   .00000E+00        3.48549E-02
3.62850E-03       8.90701E-02  6.21075E-04   .00000E+00
   25    1.01321E-01  2.37240E-02   .00000E+00        2.67603E-02
2.78583E-03       8.80378E-02  6.34861E-04   .00000E+00
   26    7.46679E-02  3.97569E-02   .00000E+00        1.42411E-02
1.48254E-03       6.35754E-02  5.66183E-04   .00000E+00

 THERMAL 1.43665E-01  1.67831E-02   .00000E+00        1.71458E-01
1.78493E-02       3.97820E-01  2.87759E-03   .00000E+00

  TOTAL  3.42858E-01  5.24294E-03   .00000E+00        5.83278E-01
6.07210E-02       5.67074E-01  3.05810E-03   .00000E+00


0REGION 41     MATERIAL FUEL_11     VOLUME  6.662644E-01

                    CROSS-SECTIONS                             FLUXES
                      REACTIONS
 GROUP   DIFFUSION    ABSORPTION   NU*FISSION            RIF            RAF
        TRANSPORT    ABSORPTION   NU*FISSION

    1    2.32885E+00  1.78377E-02  5.28855E-02        2.41310E-03
3.62183E-03       3.45391E-04  4.30441E-05  1.27618E-04
    2    1.91875E+00  1.57905E-02  3.97823E-02        8.12581E-03
1.21961E-02       1.41165E-03  1.28311E-04  3.23264E-04
    3    1.18870E+00  8.57947E-03  1.48435E-02        6.48519E-03
9.73366E-03       1.81856E-03  5.56395E-05  9.62629E-05
    4    8.16399E-01  9.08466E-03  1.44654E-02        3.42858E-03
5.14597E-03       1.39988E-03  3.11475E-05  4.95959E-05
    5    8.11678E-01  1.23757E-02  1.74729E-02        1.97107E-03
2.95839E-03       8.09464E-04  2.43934E-05  3.44404E-05
    6    7.06212E-01  2.23040E-02  2.45211E-02        2.53056E-03
3.79813E-03       1.19443E-03  5.64415E-05  6.20521E-05
    7    6.33971E-01  4.98772E-02  5.54151E-02        2.11690E-03
3.17726E-03       1.11304E-03  1.05585E-04  1.17308E-04
    8    5.71165E-01  1.29766E-01  1.54831E-01        1.99370E-03
2.99236E-03       1.16353E-03  2.58714E-04  3.08687E-04
    9    4.83010E-01  2.42135E-01  2.40387E-01        9.53242E-04
1.43073E-03       6.57848E-04  2.30813E-04  2.29147E-04
   10    3.98856E-01  3.85543E-01  4.87313E-01        9.02961E-04
1.35526E-03       7.54626E-04  3.48130E-04  4.40024E-04
   11    3.78054E-01  4.94895E-01  5.51727E-01        8.59347E-04
1.28980E-03       7.57694E-04  4.25286E-04  4.74125E-04
   12    3.77531E-01  5.09213E-01  1.61345E-01        6.12484E-04
9.19281E-04       5.40780E-04  3.11885E-04  9.88212E-05
   13    6.68706E-01  1.23913E-01  2.00238E-01        6.16807E-04
9.25770E-04       3.07463E-04  7.64302E-05  1.23508E-04
   14    6.13009E-01  1.58972E-01  2.55781E-01        5.87533E-04
8.81832E-04       3.19480E-04  9.34013E-05  1.50280E-04
   15    3.30150E-01  6.22014E-01  1.20522E+00        6.36748E-05
9.55699E-05       6.42889E-05  3.96066E-05  7.67424E-05
   16    4.05488E-01  4.34493E-01  8.86229E-01        6.78582E-05
1.01849E-04       5.57831E-05  2.94839E-05  6.01379E-05
   17    4.60036E-01  3.35121E-01  7.02402E-01        8.67953E-05
1.30272E-04       6.28902E-05  2.90869E-05  6.09652E-05
   18    4.74286E-01  3.11396E-01  6.58316E-01        3.74086E-04
5.61468E-04       2.62912E-04  1.16489E-04  2.46267E-04
   19    3.62015E-01  5.26401E-01  1.08735E+00        5.77973E-04
8.67483E-04       5.32181E-04  3.04245E-04  6.28458E-04
   20    2.31685E-01  1.04238E+00  2.03659E+00        2.05073E-04
3.07796E-04       2.95046E-04  2.13764E-04  4.17651E-04
   21    2.29393E-01  1.05628E+00  2.06603E+00        4.62518E-04
6.94196E-04       6.72090E-04  4.88548E-04  9.55577E-04
   22    1.89976E-01  1.35500E+00  2.75817E+00        1.98894E-03
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2.98522E-03        3.48981E-03  2.69502E-03  5.48584E-03
   23    1.45443E-01  1.89028E+00  3.87453E+00       1.76896E-03
2.65504E-03        4.05419E-03  3.34382E-03  6.85388E-03
   24    1.14148E-01  2.51666E+00  5.16504E+00       1.51057E-03
2.26722E-03        4.41113E-03  3.80159E-03  7.80216E-03
   25    8.57929E-02  3.47833E+00  7.14022E+00       9.96844E-04
1.49617E-03        3.87306E-03  3.46736E-03  7.11769E-03
   26    5.39375E-02  5.75904E+00  1.18196E+01       3.89781E-04
5.85024E-04        2.40884E-03  2.24476E-03  4.60707E-03

 THERMAL 1.33437E-01  2.09591E+00  4.28677E+00       7.90066E-03
1.18581E-02        1.97364E-02  1.65591E-02  3.38683E-02

  TOTAL  4.28060E-01  4.50531E-01  8.77815E-01       4.20904E-02
6.31737E-02        3.27761E-02  1.89630E-02  3.69476E-02


0REGION 42     MATERIAL CLAD        VOLUME  6.792168E-01

                   CROSS-SECTIONS                              FLUXES
                     REACTIONS
 GROUP   DIFFUSION   ABSORPTION   NU*FISSION           RIF           RAF
        TRANSPORT   ABSORPTION   NU*FISSION

    1    3.16321E+00 -5.37083E-05  .00000E+00       1.97704E-03
2.91076E-03       2.08337E-04 -1.06183E-07  .00000E+00
    2    2.49217E+00  3.74237E-04  .00000E+00       6.73986E-03
9.92298E-03       9.01472E-04  2.52230E-06  .00000E+00
    3    1.67902E+00  5.04506E-04  .00000E+00       5.55611E-03
8.18017E-03       1.10305E-03  2.80309E-06  .00000E+00
    4    1.12257E+00  5.15706E-04  .00000E+00       3.12875E-03
4.60642E-03       9.29048E-04  1.61352E-06  .00000E+00
    5    9.21946E-01  6.62955E-04  .00000E+00       1.91464E-03
2.81890E-03       6.92247E-04  1.26932E-06  .00000E+00
    6    9.25362E-01  1.34549E-03  .00000E+00       2.55778E-03
3.76578E-03       9.21361E-04  3.44146E-06  .00000E+00
    7    8.79884E-01  6.78327E-03  .00000E+00       2.16978E-03
3.19453E-03       8.21995E-04  1.47182E-05  .00000E+00
    8    1.34604E+00  1.24927E-02  .00000E+00       2.06412E-03
3.03897E-03       5.11157E-04  2.57864E-05  .00000E+00
    9    1.59365E+00  1.42310E-04  .00000E+00       1.00094E-03
1.47367E-03       2.09360E-04  1.42443E-07  .00000E+00
   10    1.57942E+00  2.24967E-04  .00000E+00       9.65156E-04
1.42098E-03       2.03695E-04  2.17129E-07  .00000E+00
   11    1.57349E+00  3.85370E-04  .00000E+00       9.30440E-04
1.36987E-03       1.97107E-04  3.58563E-07  .00000E+00
   12    1.57020E+00  6.03067E-04  .00000E+00       6.64558E-04
9.78417E-04       1.41077E-04  4.00773E-07  .00000E+00
   13    1.57409E+00  8.53588E-04  .00000E+00       6.38740E-04
9.40406E-04       1.35261E-04  5.45220E-07  .00000E+00
   14    1.56863E+00  1.16803E-03  .00000E+00       6.10803E-04
8.99275E-04       1.29796E-04  7.13437E-07  .00000E+00
   15    1.56887E+00  1.37786E-03  .00000E+00       6.98048E-05
1.02772E-04       1.48312E-05  9.61813E-08  .00000E+00
   16    1.56929E+00  1.42882E-03  .00000E+00       7.29030E-05
1.07334E-04       1.54853E-05  1.04165E-07  .00000E+00
   17    1.56537E+00  1.48846E-03  .00000E+00       9.22237E-05
1.35779E-04       1.96384E-05  1.37271E-07  .00000E+00
   18    1.56347E+00  1.67631E-03  .00000E+00       3.96215E-04
5.83340E-04       8.44735E-05  6.64178E-07  .00000E+00
   19    1.55973E+00  2.13456E-03  .00000E+00       6.27920E-04
9.24476E-04       1.34194E-04  1.34033E-06  .00000E+00
   20    1.55524E+00  2.60384E-03  .00000E+00       2.37198E-04
3.49223E-04       5.08385E-05  6.17626E-07  .00000E+00
   21    1.55192E+00  3.12774E-03  .00000E+00       5.36694E-04
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7.90166E-04       1.15275E-04  1.67864E-06   .00000E+00
   22    1.54000E+00  4.45409E-03   .00000E+00       2.38850E-03
3.51655E-03       5.16991E-04  1.06386E-05   .00000E+00
   23    1.52869E+00  5.81622E-03   .00000E+00       2.26556E-03
3.33555E-03       4.94011E-04  1.31770E-05   .00000E+00
   24    1.51588E+00  7.36893E-03   .00000E+00       2.08580E-03
3.07089E-03       4.58656E-04  1.53701E-05   .00000E+00
   25    1.49532E+00  9.79734E-03   .00000E+00       1.54115E-03
2.26901E-03       3.43551E-04  1.50992E-05   .00000E+00
   26    1.43463E+00  1.63396E-02   .00000E+00       7.73919E-04
1.13943E-03       1.79818E-04  1.26455E-05   .00000E+00

 THERMAL 1.51987E+00  6.74847E-03   .00000E+00       1.04567E-02
1.53953E-02       2.29334E-03  7.05671E-05   .00000E+00

  TOTAL  1.46886E+00  2.99940E-03   .00000E+00       4.20066E-02
6.18456E-02       9.53272E-03  1.25994E-04   .00000E+00


0REGION 43     MATERIAL COOLANT     VOLUME  1.165454E+01

                        CROSS-SECTIONS                            FLUXES
                          REACTIONS
 GROUP   DIFFUSION    ABSORPTION   NU*FISSION          RIF             RAF
        TRANSPORT    ABSORPTION   NU*FISSION

    1    2.70642E+00  2.17943E-03   .00000E+00       2.71299E-02
2.32784E-03       3.34142E-03  5.91277E-05   .00000E+00
    2    2.01031E+00  1.01023E-05   .00000E+00       9.27323E-02
7.95675E-03       1.53761E-02  9.36812E-07   .00000E+00
    3    1.31913E+00  3.55934E-06   .00000E+00       7.86210E-02
6.74595E-03       1.98669E-02  2.79839E-07   .00000E+00
    4    7.92566E-01  3.59323E-06   .00000E+00       4.75971E-02
4.08400E-03       2.00181E-02  1.71027E-07   .00000E+00
    5    6.54618E-01  6.63753E-06   .00000E+00       3.11346E-02
2.67145E-03       1.58538E-02  2.06657E-07   .00000E+00
    6    5.56182E-01  1.94203E-05   .00000E+00       4.33145E-02
3.71653E-03       2.59594E-02  8.41182E-07   .00000E+00
    7    5.63543E-01  6.29493E-05   .00000E+00       3.72753E-02
3.19835E-03       2.20481E-02  2.34645E-06   .00000E+00
    8    5.77544E-01  1.71416E-04   .00000E+00       3.58239E-02
3.07381E-03       2.06760E-02  6.14079E-06   .00000E+00
    9    5.80367E-01  3.53087E-04   .00000E+00       1.75328E-02
1.50438E-03       1.00700E-02  6.19061E-06   .00000E+00
   10    5.80777E-01  5.82085E-04   .00000E+00       1.71358E-02
1.47031E-03       9.83501E-03  9.97451E-06   .00000E+00
   11    5.80114E-01  9.59742E-04   .00000E+00       1.66798E-02
1.43119E-03       9.58422E-03  1.60083E-05   .00000E+00
   12    5.78579E-01  1.47308E-03   .00000E+00       1.19326E-02
1.02385E-03       6.87463E-03  1.75776E-05   .00000E+00
   13    5.30274E-01  2.07713E-03   .00000E+00       1.10776E-02
9.50494E-04       6.96342E-03  2.30095E-05   .00000E+00
   14    4.84734E-01  2.83531E-03   .00000E+00       1.06243E-02
9.11599E-04       7.30591E-03  3.01231E-05   .00000E+00
   15    4.56194E-01  3.34008E-03   .00000E+00       1.26135E-03
1.08228E-04       9.21645E-04  4.21299E-06   .00000E+00
   16    4.50003E-01  3.46267E-03   .00000E+00       1.30073E-03
1.11608E-04       9.63501E-04  4.50401E-06   .00000E+00
   17    4.42700E-01  3.60538E-03   .00000E+00       1.63180E-03
1.40014E-04       1.22868E-03  5.88328E-06   .00000E+00
   18    4.19829E-01  4.05921E-03   .00000E+00       6.99711E-03
6.00376E-04       5.55552E-03  2.84027E-05   .00000E+00
   19    3.66366E-01  5.17002E-03   .00000E+00       1.13000E-02
9.69579E-04       1.02812E-02  5.84212E-05   .00000E+00
   20    3.18292E-01  6.29875E-03   .00000E+00       4.44769E-03
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3.81627E-04       4.65787E-03  2.80149E-05   .00000E+00
   21     2.76329E-01  7.56452E-03   .00000E+00       1.01154E-02
8.67940E-04       1.22022E-02  7.65184E-05   .00000E+00
   22     2.07862E-01  1.07793E-02   .00000E+00       4.60700E-02
3.95297E-03       7.38791E-02  4.96601E-04   .00000E+00
   23     1.64184E-01  1.40641E-02   .00000E+00       4.52749E-02
3.88474E-03       9.19189E-02  6.36751E-04   .00000E+00
   24     1.30440E-01  1.78189E-02   .00000E+00       4.31994E-02
3.70666E-03       1.10394E-01  7.69766E-04   .00000E+00
   25     1.01321E-01  2.37240E-02   .00000E+00       3.33110E-02
2.85820E-03       1.09589E-01  7.90270E-04   .00000E+00
   26     7.46679E-02  3.97569E-02   .00000E+00       1.77898E-02
1.52643E-03       7.94175E-02  7.07268E-04   .00000E+00

 THERMAL 1.43199E-01  1.68486E-02   .00000E+00       2.11508E-01
1.81481E-02       4.92339E-01  3.56361E-03   .00000E+00

  TOTAL  3.36466E-01  5.38926E-03   .00000E+00       7.01311E-01
6.01749E-02       6.94782E-01  3.77955E-03   .00000E+00


0REGION 44     MATERIAL FUEL_12     VOLUME  1.332556E+00

                    CROSS-SECTIONS                             FLUXES
                      REACTIONS
 GROUP  DIFFUSION    ABSORPTION   NU*FISSION           RIF            RAF
       TRANSPORT    ABSORPTION   NU*FISSION

    1     2.32885E+00  1.78377E-02  5.28855E-02       4.72248E-03
3.54393E-03       6.75939E-04  8.42383E-05  2.49751E-04
    2     1.91875E+00  1.57905E-02  3.97823E-02       1.59607E-02
1.19775E-02       2.77277E-03  2.52028E-04  6.34955E-04
    3     1.18870E+00  8.57947E-03  1.48435E-02       1.27693E-02
9.58260E-03       3.58076E-03  1.09554E-04  1.89542E-04
    4     8.16399E-01  9.08466E-03  1.44654E-02       6.76723E-03
5.07838E-03       2.76304E-03  6.14780E-05  9.78909E-05
    5     8.11678E-01  1.23757E-02  1.74729E-02       3.89384E-03
2.92209E-03       1.59909E-03  4.81891E-05  6.80368E-05
    6     7.06212E-01  2.23040E-02  2.45211E-02       5.00719E-03
3.75758E-03       2.36340E-03  1.11680E-04  1.22782E-04
    7     6.33971E-01  4.98772E-02  5.54151E-02       4.19489E-03
3.14800E-03       2.20562E-03  2.09229E-04  2.32460E-04
    8     5.71165E-01  1.29766E-01  1.54831E-01       3.95438E-03
2.96752E-03       2.30779E-03  5.13144E-04  6.12263E-04
    9     4.83010E-01  2.42135E-01  2.40387E-01       1.89167E-03
1.41958E-03       1.30547E-03  4.58038E-04  4.54732E-04
   10     3.98856E-01  3.85543E-01  4.87313E-01       1.79199E-03
1.34478E-03       1.49761E-03  6.90889E-04  8.73258E-04
   11     3.78054E-01  4.94895E-01  5.51727E-01       1.70562E-03
1.27996E-03       1.50386E-03  8.44101E-04  9.41035E-04
   12     3.77531E-01  5.09213E-01  1.61345E-01       1.21573E-03
9.12330E-04       1.07340E-03  6.19066E-04  1.96152E-04
   13     6.68706E-01  1.23913E-01  2.00238E-01       1.22418E-03
9.18673E-04       6.10225E-04  1.51692E-04  2.45128E-04
   14     6.13009E-01  1.58972E-01  2.55781E-01       1.16580E-03
8.74861E-04       6.33923E-04  1.85330E-04  2.98190E-04
   15     3.30150E-01  6.22014E-01  1.20522E+00       1.26326E-04
9.47998E-05       1.27544E-04  7.85765E-05  1.52251E-04
   16     4.05488E-01  4.34493E-01  8.86229E-01       1.34643E-04
1.01041E-04       1.10684E-04  5.85013E-05  1.19324E-04
   17     4.60036E-01  3.35121E-01  7.02402E-01       1.72222E-04
1.29242E-04       1.24789E-04  5.77152E-05  1.20969E-04
   18     4.74286E-01  3.11396E-01  6.58316E-01       7.42241E-04
5.57006E-04       5.21656E-04  2.31131E-04  4.88629E-04
   19     3.62015E-01  5.26401E-01  1.08735E+00       1.14674E-03
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8.60559E-04        1.05589E-03  6.03646E-04  1.24691E-03
   20     2.31685E-01  1.04238E+00  2.03659E+00       4.06815E-04
3.05289E-04        5.85298E-04  4.24055E-04  8.28516E-04
   21     2.29393E-01  1.05628E+00  2.06603E+00       9.18869E-04
6.89554E-04        1.33522E-03  9.70582E-04  1.89841E-03
   22     1.89976E-01  1.35500E+00  2.75817E+00       3.97947E-03
2.98634E-03        6.98240E-03  5.39219E-03  1.09760E-02
   23     1.45443E-01  1.89028E+00  3.87453E+00       3.55329E-03
2.66652E-03        8.14363E-03  6.71671E-03  1.37674E-02
   24     1.14148E-01  2.51666E+00  5.16504E+00       3.04367E-03
2.28408E-03        8.88806E-03  7.65987E-03  1.57207E-02
   25     8.57929E-02  3.47833E+00  7.14022E+00       2.01557E-03
1.51256E-03        7.83115E-03  7.01083E-03  1.43916E-02
   26     5.39375E-02  5.75904E+00  1.18196E+01       7.91122E-04
5.93687E-04        4.88912E-03  4.55610E-03  9.35077E-03

 THERMAL 1.33092E-01  2.10235E+00  4.30009E+00       1.58556E-02
1.18986E-02        3.97108E-02  3.33340E-02  6.81803E-02

  TOTAL  4.23974E-01  4.57387E-01  8.91731E-01       8.32961E-02
6.25085E-02        6.54883E-02  3.80986E-02  7.42777E-02


0REGION 45     MATERIAL CLAD        VOLUME  1.358413E+00

                    CROSS-SECTIONS                           FLUXES
                      REACTIONS
 GROUP   DIFFUSION    ABSORPTION   NU*FISSION          RIF            RAF
        TRANSPORT    ABSORPTION   NU*FISSION

    1     3.16321E+00 -5.37083E-05   .00000E+00       3.88429E-03
2.85944E-03        4.09320E-04 -2.08619E-07   .00000E+00
    2     2.49217E+00  3.74237E-04   .00000E+00       1.32718E-02
9.77008E-03        1.77513E-03  4.96680E-06   .00000E+00
    3     1.67902E+00  5.04506E-04   .00000E+00       1.09452E-02
8.05732E-03        2.17293E-03  5.52191E-06   .00000E+00
    4     1.12257E+00  5.15706E-04   .00000E+00       6.17438E-03
4.54529E-03        1.83341E-03  3.18417E-06   .00000E+00
    5     9.21946E-01  6.62955E-04   .00000E+00       3.78165E-03
2.78387E-03        1.36727E-03  2.50706E-06   .00000E+00
    6     9.25362E-01  1.34549E-03   .00000E+00       5.05999E-03
3.72493E-03        1.82271E-03  6.80817E-06   .00000E+00
    7     8.79884E-01  6.78327E-03   .00000E+00       4.29885E-03
3.16461E-03        1.62857E-03  2.91603E-05   .00000E+00
    8     1.34604E+00  1.24927E-02   .00000E+00       4.09310E-03
3.01315E-03        1.01361E-03  5.11337E-05   .00000E+00
    9     1.59365E+00  1.42310E-04   .00000E+00       1.98567E-03
1.46176E-03        4.15328E-04  2.82579E-07   .00000E+00
   10     1.57942E+00  2.24967E-04   .00000E+00       1.91466E-03
1.40948E-03        4.04086E-04  4.30736E-07   .00000E+00
   11     1.57349E+00  3.85370E-04   .00000E+00       1.84588E-03
1.35885E-03        3.91037E-04  7.11346E-07   .00000E+00
   12     1.57020E+00  6.03067E-04   .00000E+00       1.31852E-03
9.70633E-04        2.79906E-04  7.95156E-07   .00000E+00
   13     1.57409E+00  8.53588E-04   .00000E+00       1.26750E-03
9.33075E-04        2.68408E-04  1.08192E-06   .00000E+00
   14     1.56863E+00  1.16803E-03   .00000E+00       1.21171E-03
8.92006E-04        2.57489E-04  1.41532E-06   .00000E+00
   15     1.56887E+00  1.37786E-03   .00000E+00       1.38418E-04
1.01897E-04        2.94094E-05  1.90721E-07   .00000E+00
   16     1.56929E+00  1.42882E-03   .00000E+00       1.44591E-04
1.06441E-04        3.07126E-05  2.06595E-07   .00000E+00
   17     1.56537E+00  1.48846E-03   .00000E+00       1.82923E-04
1.34660E-04        3.89522E-05  2.72274E-07   .00000E+00
   18     1.56347E+00  1.67631E-03   .00000E+00       7.85852E-04
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5.78507E-04       1.67545E-04  1.31733E-06   .00000E+00
   19    1.55973E+00  2.13456E-03   .00000E+00       1.24524E-03
9.16686E-04       2.66122E-04  2.65803E-06   .00000E+00
   20    1.55524E+00  2.60384E-03   .00000E+00       4.70230E-04
3.46161E-04       1.00784E-04  1.22440E-06   .00000E+00
   21    1.55192E+00  3.12774E-03   .00000E+00       1.06560E-03
7.84443E-04       2.28877E-04  3.33292E-06   .00000E+00
   22    1.54000E+00  4.45409E-03   .00000E+00       4.77672E-03
3.51640E-03       1.03392E-03  2.12759E-05   .00000E+00
   23    1.52869E+00  5.81622E-03   .00000E+00       4.54894E-03
3.34872E-03       9.91905E-04  2.64576E-05   .00000E+00
   24    1.51588E+00  7.36893E-03   .00000E+00       4.20100E-03
3.09258E-03       9.23777E-04  3.09569E-05   .00000E+00
   25    1.49532E+00  9.79734E-03   .00000E+00       3.11461E-03
2.29283E-03       6.94302E-04  3.05149E-05   .00000E+00
   26    1.43463E+00  1.63396E-02   .00000E+00       1.56939E-03
1.15532E-03       3.64644E-04  2.56433E-05   .00000E+00

 THERMAL 1.51971E+00  6.76762E-03   .00000E+00       2.09917E-02
1.54531E-02       4.60434E-03  1.42064E-04   .00000E+00

  TOTAL  1.46829E+00  3.02343E-03   .00000E+00       8.32967E-02
6.13191E-02       1.89102E-02  2.51841E-04   .00000E+00


0REGION 46     MATERIAL COOLANT     VOLUME  3.129841E+01

                     CROSS-SECTIONS                           FLUXES
                       REACTIONS
 GROUP   DIFFUSION    ABSORPTION   NU*FISSION            RIF            RAF
        TRANSPORT    ABSORPTION   NU*FISSION

    1    2.70642E+00  2.17943E-03   .00000E+00       6.96976E-02
2.22687E-03       8.58423E-03  1.51901E-04   .00000E+00
    2    2.01031E+00  1.01023E-05   .00000E+00       2.39569E-01
7.65434E-03       3.97234E-02  2.42020E-06   .00000E+00
    3    1.31913E+00  3.55934E-06   .00000E+00       2.03695E-01
6.50816E-03       5.14720E-02  7.25020E-07   .00000E+00
    4    7.92566E-01  3.59323E-06   .00000E+00       1.24214E-01
3.96869E-03       5.22412E-02  4.46329E-07   .00000E+00
    5    6.54618E-01  6.63753E-06   .00000E+00       8.16279E-02
2.60805E-03       4.15651E-02  5.41808E-07   .00000E+00
    6    5.56182E-01  1.94203E-05   .00000E+00       1.14076E-01
3.64477E-03       6.83683E-02  2.21539E-06   .00000E+00
    7    5.63543E-01  6.29493E-05   .00000E+00       9.84924E-02
3.14688E-03       5.82578E-02  6.20003E-06   .00000E+00
    8    5.77544E-01  1.71416E-04   .00000E+00       9.48055E-02
3.02908E-03       5.47176E-02  1.62512E-05   .00000E+00
    9    5.80367E-01  3.53087E-04   .00000E+00       4.64485E-02
1.48405E-03       2.66777E-02  1.64004E-05   .00000E+00
   10    5.80777E-01  5.82085E-04   .00000E+00       4.54162E-02
1.45107E-03       2.60663E-02  2.64360E-05   .00000E+00
   11    5.80114E-01  9.59742E-04   .00000E+00       4.42213E-02
1.41289E-03       2.54096E-02  4.24411E-05   .00000E+00
   12    5.78579E-01  1.47308E-03   .00000E+00       3.16399E-02
1.01091E-03       1.82285E-02  4.66081E-05   .00000E+00
   13    5.30274E-01  2.07713E-03   .00000E+00       2.93335E-02
9.37221E-04       1.84392E-02  6.09295E-05   .00000E+00
   14    4.84734E-01  2.83531E-03   .00000E+00       2.81167E-02
8.98344E-04       1.93348E-02  7.97197E-05   .00000E+00
   15    4.56194E-01  3.34008E-03   .00000E+00       3.34055E-03
1.06732E-04       2.44089E-03  1.11577E-05   .00000E+00
   16    4.50003E-01  3.46267E-03   .00000E+00       3.44379E-03
1.10031E-04       2.55094E-03  1.19247E-05   .00000E+00
   17    4.42700E-01  3.60538E-03   .00000E+00       4.31923E-03
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1.38002E-04       3.25219E-03  1.55725E-05   .00000E+00
  18    4.19829E-01  4.05921E-03   .00000E+00       1.85159E-02
5.91592E-04       1.47011E-02  7.51599E-05   .00000E+00
  19    3.66366E-01  5.17002E-03   .00000E+00       2.99095E-02
9.55623E-04       2.72128E-02  1.54632E-04   .00000E+00
  20    3.18292E-01  6.29875E-03   .00000E+00       1.17821E-02
3.76443E-04       1.23388E-02  7.42122E-05   .00000E+00
  21    2.76329E-01  7.56452E-03   .00000E+00       2.68534E-02
8.57979E-04       3.23930E-02  2.03133E-04   .00000E+00
  22    2.07862E-01  1.07793E-02   .00000E+00       1.23900E-01
3.95866E-03       1.98689E-01  1.33555E-03   .00000E+00
  23    1.64184E-01  1.40641E-02   .00000E+00       1.22639E-01
3.91837E-03       2.48986E-01  1.72480E-03   .00000E+00
  24    1.30440E-01  1.78189E-02   .00000E+00       1.17557E-01
3.75600E-03       3.00411E-01  2.09473E-03   .00000E+00
  25    1.01321E-01  2.37240E-02   .00000E+00       9.10424E-02
2.90885E-03       2.99517E-01  2.15989E-03   .00000E+00
  26    7.46679E-02  3.97569E-02   .00000E+00       4.88056E-02
1.55936E-03       2.17879E-01  1.94036E-03   .00000E+00

 THERMAL 1.42684E-01  1.69214E-02   .00000E+00       5.72488E-01
1.82913E-02       1.33743E+00  9.68731E-03   .00000E+00

  TOTAL  3.30481E-01  5.53255E-03   .00000E+00       1.85346E+00
5.92190E-02       1.86946E+00  1.02544E-02   .00000E+00


0REGION 47     MATERIAL FUEL_13     VOLUME  1.332598E+00

                    CROSS-SECTIONS                             FLUXES
                      REACTIONS
 GROUP   DIFFUSION    ABSORPTION   NU*FISSION          RIF            RAF
        TRANSPORT    ABSORPTION   NU*FISSION

   1    2.32885E+00  1.78377E-02  5.28855E-02       4.60154E-03
3.45306E-03       6.58628E-04  8.20810E-05  2.43355E-04
   2    1.91875E+00  1.57905E-02  3.97823E-02       1.56181E-02
1.17200E-02       2.71325E-03  2.46618E-04  6.21324E-04
   3    1.18870E+00  8.57947E-03  1.48435E-02       1.24703E-02
9.35786E-03       3.49689E-03  1.06988E-04  1.85102E-04
   4    8.16399E-01  9.08466E-03  1.44654E-02       6.61260E-03
4.96219E-03       2.69991E-03  6.00732E-05  9.56541E-05
   5    8.11678E-01  1.23757E-02  1.74729E-02       3.79981E-03
2.85143E-03       1.56048E-03  4.70254E-05  6.63939E-05
   6    7.06212E-01  2.23040E-02  2.45211E-02       4.89179E-03
3.67087E-03       2.30894E-03  1.09106E-04  1.19952E-04
   7    6.33971E-01  4.98772E-02  5.54151E-02       4.10566E-03
3.08094E-03       2.15870E-03  2.04779E-04  2.27516E-04
   8    5.71165E-01  1.29766E-01  1.54831E-01       3.87015E-03
2.90421E-03       2.25863E-03  5.02213E-04  5.99220E-04
   9    4.83010E-01  2.42135E-01  2.40387E-01       1.85147E-03
1.38937E-03       1.27773E-03  4.48306E-04  4.45071E-04
  10    3.98856E-01  3.85543E-01  4.87313E-01       1.75249E-03
1.31509E-03       1.46460E-03  6.75660E-04  8.54010E-04
  11    3.78054E-01  4.94895E-01  5.51727E-01       1.66672E-03
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9.84476E-05       1.07846E-04  5.70016E-05  1.16265E-04
   17    4.60036E-01  3.35121E-01  7.02402E-01         1.67823E-04
1.25937E-04       1.21601E-04  5.62408E-05  1.17879E-04
   18    4.74286E-01  3.11396E-01  6.58316E-01         7.22991E-04
5.42542E-04       5.08126E-04  2.25136E-04  4.75957E-04
   19    3.62015E-01  5.26401E-01  1.08735E+00         1.11569E-03
8.37228E-04       1.02729E-03  5.87299E-04  1.21314E-03
   20    2.31685E-01  1.04238E+00  2.03659E+00         3.95116E-04
2.96501E-04       5.68467E-04  4.11861E-04  8.04691E-04
   21    2.29393E-01  1.05628E+00  2.06603E+00         8.93701E-04
6.70646E-04       1.29865E-03  9.43997E-04  1.84642E-03
   22    1.89976E-01  1.35500E+00  2.75817E+00         3.92175E-03
2.94293E-03       6.88112E-03  5.31397E-03  1.08168E-02
   23    1.45443E-01  1.89028E+00  3.87453E+00         3.52278E-03
2.64354E-03       8.07369E-03  6.65903E-03  1.36491E-02
   24    1.14148E-01  2.51666E+00  5.16504E+00         3.02795E-03
2.27221E-03       8.84215E-03  7.62031E-03  1.56395E-02
   25    8.57929E-02  3.47833E+00  7.14022E+00         2.01232E-03
1.51007E-03       7.81852E-03  6.99953E-03  1.43684E-02
   26    5.39375E-02  5.75904E+00  1.18196E+01         7.92619E-04
5.94792E-04       4.89837E-03  4.56472E-03  9.36847E-03

 THERMAL 1.32645E-01  2.11076E+00  4.31749E+00         1.56819E-02
1.17679E-02       3.94083E-02  3.31007E-02  6.77066E-02

  TOTAL  4.20988E-01  4.62777E-01  9.02879E-01         8.15879E-02
6.12247E-02       6.46003E-02  3.77570E-02  7.36641E-02


0REGION 48     MATERIAL CLAD        VOLUME  1.358422E+00

                      CROSS-SECTIONS                             FLUXES
                        REACTIONS
 GROUP   DIFFUSION    ABSORPTION   NU*FISSION           RIF            RAF
        TRANSPORT    ABSORPTION   NU*FISSION

    1    3.16321E+00 -5.37083E-05   .00000E+00         3.74661E-03
2.75806E-03       3.94811E-04 -2.01224E-07   .00000E+00
    2    2.49217E+00  3.74237E-04   .00000E+00         1.28864E-02
9.48633E-03       1.72359E-03  4.82258E-06   .00000E+00
    3    1.67902E+00  5.04506E-04   .00000E+00         1.06321E-02
7.82677E-03       2.11077E-03  5.36394E-06   .00000E+00
    4    1.12257E+00  5.15706E-04   .00000E+00         6.01539E-03
4.42822E-03       1.78620E-03  3.10217E-06   .00000E+00
    5    9.21946E-01  6.62955E-04   .00000E+00         3.68461E-03
2.71242E-03       1.33218E-03  2.44273E-06   .00000E+00
    6    9.25362E-01  1.34549E-03   .00000E+00         4.94053E-03
3.63696E-03       1.77967E-03  6.64743E-06   .00000E+00
    7    8.79884E-01  6.78327E-03   .00000E+00         4.20675E-03
3.09679E-03       1.59368E-03  2.85355E-05   .00000E+00
    8    1.34604E+00  1.24927E-02   .00000E+00         4.00601E-03
2.94902E-03       9.92046E-04  5.00457E-05   .00000E+00
    9    1.59365E+00  1.42310E-04   .00000E+00         1.94386E-03
1.43097E-03       4.06585E-04  2.76630E-07   .00000E+00
   10    1.57942E+00  2.24967E-04   .00000E+00         1.87293E-03
1.37875E-03       3.95279E-04  4.21348E-07   .00000E+00
   11    1.57349E+00  3.85370E-04   .00000E+00         1.80437E-03
1.32828E-03       3.82243E-04  6.95348E-07   .00000E+00
   12    1.57020E+00  6.03067E-04   .00000E+00         1.28849E-03
9.48522E-04       2.73531E-04  7.77048E-07   .00000E+00
   13    1.57409E+00  8.53588E-04   .00000E+00         1.23817E-03
9.11475E-04       2.62197E-04  1.05688E-06   .00000E+00
   14    1.56863E+00  1.16803E-03   .00000E+00         1.18239E-03
8.70413E-04       2.51257E-04  1.38107E-06   .00000E+00
   15    1.56887E+00  1.37786E-03   .00000E+00         1.34916E-04
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9.93179E-05       2.86652E-05  1.85895E-07   .00000E+00
   16     1.56929E+00  1.42882E-03   .00000E+00       1.40943E-04
1.03755E-04       2.99378E-05  2.01383E-07   .00000E+00
   17     1.56537E+00  1.48846E-03   .00000E+00       1.78322E-04
1.31271E-04       3.79723E-05  2.65425E-07   .00000E+00
   18     1.56347E+00  1.67631E-03   .00000E+00       7.65779E-04
5.63727E-04       1.63265E-04  1.28368E-06   .00000E+00
   19     1.55973E+00  2.13456E-03   .00000E+00       1.21207E-03
8.92262E-04       2.59034E-04  2.58723E-06   .00000E+00
   20     1.55524E+00  2.60384E-03   .00000E+00       4.56927E-04
3.36366E-04       9.79327E-05  1.18977E-06   .00000E+00
   21     1.55192E+00  3.12774E-03   .00000E+00       1.03683E-03
7.63262E-04       2.22698E-04  3.24294E-06   .00000E+00
   22     1.54000E+00  4.45409E-03   .00000E+00       4.71226E-03
3.46893E-03       1.01997E-03  2.09888E-05   .00000E+00
   23     1.52869E+00  5.81622E-03   .00000E+00       4.51748E-03
3.32554E-03       9.85047E-04  2.62747E-05   .00000E+00
   24     1.51588E+00  7.36893E-03   .00000E+00       4.18740E-03
3.08255E-03       9.20787E-04  3.08567E-05   .00000E+00
   25     1.49532E+00  9.79734E-03   .00000E+00       3.11675E-03
2.29439E-03       6.94781E-04  3.05359E-05   .00000E+00
   26     1.43463E+00  1.63396E-02   .00000E+00       1.57685E-03
1.16079E-03       3.66376E-04  2.57651E-05   .00000E+00

 THERMAL 1.51947E+00  6.79464E-03   .00000E+00       2.08166E-02
1.53241E-02       4.56663E-03  1.41441E-04   .00000E+00

  TOTAL  1.46737E+00  3.05264E-03   .00000E+00       8.14851E-02
5.99852E-02       1.85105E-02  2.48745E-04   .00000E+00


0REGION 49     MATERIAL MODERATOR   VOLUME  1.300012E+01

                     CROSS-SECTIONS                            FLUXES
                       REACTIONS
 GROUP   DIFFUSION     ABSORPTION   NU*FISSION            RIF            RAF
        TRANSPORT     ABSORPTION   NU*FISSION

    1     2.74822E+00  2.21855E-03   .00000E+00       2.75750E-02
2.12114E-03       3.34459E-03  6.11766E-05   .00000E+00
    2     2.00863E+00  9.66120E-06   .00000E+00       9.53263E-02
7.33272E-03       1.58195E-02  9.20966E-07   .00000E+00
    3     1.31914E+00  3.55843E-06   .00000E+00       8.14298E-02
6.26377E-03       2.05765E-02  2.89762E-07   .00000E+00
    4     7.92343E-01  3.59826E-06   .00000E+00       5.01726E-02
3.85939E-03       2.11073E-02  1.80534E-07   .00000E+00
    5     6.53939E-01  6.66158E-06   .00000E+00       3.31629E-02
2.55097E-03       1.69042E-02  2.20917E-07   .00000E+00
    6     5.56006E-01  1.95488E-05   .00000E+00       4.65557E-02
3.58117E-03       2.79107E-02  9.10106E-07   .00000E+00
    7     5.63650E-01  6.32557E-05   .00000E+00       4.03184E-02
3.10138E-03       2.38436E-02  2.55037E-06   .00000E+00
    8     5.77572E-01  1.72185E-04   .00000E+00       3.88354E-02
2.98731E-03       2.24130E-02  6.68687E-06   .00000E+00
    9     5.80369E-01  3.53483E-04   .00000E+00       1.90533E-02
1.46562E-03       1.09432E-02  6.73502E-06   .00000E+00  10     5.80777E-
01  5.82735E-04   .00000E+00        1.86217E-02  1.43242E-03
1.06878E-02  1.08515E-05   .00000E+00
   11     5.80112E-01  9.60720E-04   .00000E+00       1.81259E-02
1.39429E-03       1.04152E-02  1.74139E-05   .00000E+00
   12     5.78577E-01  1.47362E-03   .00000E+00       1.29671E-02
9.97462E-04       7.47071E-03  1.91087E-05   .00000E+00
   13     5.30259E-01  2.07768E-03   .00000E+00       1.19973E-02
9.22860E-04       7.54179E-03  2.49266E-05   .00000E+00
   14     4.84689E-01  2.83608E-03   .00000E+00       1.14908E-02
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8.83897E-04       7.90251E-03  3.25888E-05   .00000E+00
   15    4.56193E-01  3.34009E-03   .00000E+00       1.36675E-03
1.05134E-04       9.98666E-04  4.56508E-06   .00000E+00
   16    4.50003E-01  3.46266E-03   .00000E+00       1.40821E-03
1.08323E-04       1.04311E-03  4.87616E-06   .00000E+00
   17    4.42698E-01  3.60542E-03   .00000E+00       1.76549E-03
1.35805E-04       1.32934E-03  6.36531E-06   .00000E+00
   18    4.19805E-01  4.05969E-03   .00000E+00       7.56454E-03
5.81882E-04       6.00639E-03  3.07097E-05   .00000E+00
   19    3.66218E-01  5.17320E-03   .00000E+00       1.22142E-02
9.39543E-04       1.11174E-02  6.31864E-05   .00000E+00
   20    3.18091E-01  6.30426E-03   .00000E+00       4.80370E-03
3.69512E-04       5.03389E-03  3.02838E-05   .00000E+00
   21    2.74446E-01  7.62670E-03   .00000E+00       1.09283E-02
8.40634E-04       1.32732E-02  8.33472E-05   .00000E+00
   22    2.07048E-01  1.08270E-02   .00000E+00       5.10092E-02
3.92375E-03       8.21212E-02  5.52276E-04   .00000E+00
   23    1.64120E-01  1.40700E-02   .00000E+00       5.07634E-02
3.90484E-03       1.03102E-01  7.14240E-04   .00000E+00
   24    1.30412E-01  1.78227E-02   .00000E+00       4.86809E-02
3.74465E-03       1.24428E-01  8.67627E-04   .00000E+00
   25    1.01309E-01  2.37277E-02   .00000E+00       3.77145E-02
2.90109E-03       1.24091E-01  8.94877E-04   .00000E+00
   26    7.46682E-02  3.97555E-02   .00000E+00       2.02134E-02
1.55486E-03       9.02365E-02  8.03594E-04   .00000E+00

 THERMAL 1.42348E-01  1.69656E-02   .00000E+00       2.36328E-01
1.81789E-02       5.53403E-01  4.00943E-03   .00000E+00

  TOTAL  3.26579E-01  5.62353E-03   .00000E+00       7.54065E-01
5.80044E-02       7.69661E-01  4.24051E-03   .00000E+00


0REGION 50     MATERIAL COOLANT     VOLUME  1.310543E+01

                    CROSS-SECTIONS                              FLUXES
                      REACTIONS
 GROUP  DIFFUSION    ABSORPTION   NU*FISSION          RIF            RAF
       TRANSPORT    ABSORPTION   NU*FISSION

    1    2.70642E+00  2.17943E-03   .00000E+00       2.75618E-02
2.10308E-03       3.39462E-03  6.00689E-05   .00000E+00
    2    2.01031E+00  1.01023E-05   .00000E+00       9.58391E-02
7.31293E-03       1.58913E-02  9.68198E-07   .00000E+00
    3    1.31913E+00  3.55934E-06   .00000E+00       8.18039E-02
6.24198E-03       2.06711E-02  2.91168E-07   .00000E+00
    4    7.92566E-01  3.59323E-06   .00000E+00       5.03775E-02
3.84402E-03       2.11875E-02  1.81018E-07   .00000E+00
    5    6.54618E-01  6.63753E-06   .00000E+00       3.32365E-02
2.53608E-03       1.69241E-02  2.20608E-07   .00000E+00
    6    5.56182E-01  1.94203E-05   .00000E+00       4.66381E-02
3.55869E-03       2.79513E-02  9.05728E-07   .00000E+00
    7    5.63543E-01  6.29493E-05   .00000E+00       4.03394E-02
3.07807E-03       2.38606E-02  2.53934E-06   .00000E+00
    8    5.77544E-01  1.71416E-04   .00000E+00       3.88067E-02
2.96111E-03       2.23975E-02  6.65208E-06   .00000E+00
    9    5.80367E-01  3.53087E-04   .00000E+00       1.90037E-02
1.45006E-03       1.09148E-02  6.70996E-06   .00000E+00
   10    5.80777E-01  5.82085E-04   .00000E+00       1.85584E-02
1.41609E-03       1.06515E-02  1.08026E-05   .00000E+00
   11    5.80114E-01  9.59742E-04   .00000E+00       1.80473E-02
1.37709E-03       1.03700E-02  1.73208E-05   .00000E+00
   12    5.78579E-01  1.47308E-03   .00000E+00       1.29020E-02
9.84474E-04       7.43313E-03  1.90056E-05   .00000E+00
   13    5.30274E-01  2.07713E-03   .00000E+00       1.19490E-02
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9.11762E-04        7.51123E-03  2.48197E-05   .00000E+00
   14    4.84734E-01  2.83531E-03   .00000E+00       1.14398E-02
8.72907E-04        7.86673E-03  3.24355E-05   .00000E+00
   15    4.56194E-01  3.34008E-03   .00000E+00       1.35795E-03
1.03617E-04        9.92232E-04  4.53566E-06   .00000E+00
   16    4.50003E-01  3.46267E-03   .00000E+00       1.39979E-03
1.06810E-04        1.03687E-03  4.84700E-06   .00000E+00
   17    4.42700E-01  3.60538E-03   .00000E+00       1.75548E-03
1.33951E-04        1.32180E-03  6.32918E-06   .00000E+00
   18    4.19829E-01  4.05921E-03   .00000E+00       7.52088E-03
5.73875E-04        5.97138E-03  3.05288E-05   .00000E+00
   19    3.66366E-01  5.17002E-03   .00000E+00       1.21300E-02
9.25571E-04        1.10363E-02  6.27123E-05   .00000E+00
   20    3.18292E-01  6.29875E-03   .00000E+00       4.76941E-03
3.63927E-04        4.99480E-03  3.00414E-05   .00000E+00
   21    2.76329E-01  7.56452E-03   .00000E+00       1.08738E-02
8.29717E-04        1.31170E-02  8.22550E-05   .00000E+00
   22    2.07862E-01  1.07793E-02   .00000E+00       5.05306E-02
3.85570E-03        8.10322E-02  5.44683E-04   .00000E+00
   23    1.64184E-01  1.40641E-02   .00000E+00       5.00762E-02
3.82103E-03        1.01667E-01  7.04278E-04   .00000E+00
   24    1.30440E-01  1.78189E-02   .00000E+00       4.79507E-02
3.65885E-03        1.22536E-01  8.54430E-04   .00000E+00
   25    1.01321E-01  2.37240E-02   .00000E+00       3.70852E-02
2.82976E-03        1.22005E-01  8.79809E-04   .00000E+00
   26    7.46679E-02  3.97569E-02   .00000E+00       1.98590E-02
1.51532E-03        8.86547E-02  7.89531E-04   .00000E+00

 THERMAL 1.42665E-01  1.69231E-02   .00000E+00       2.33275E-01
1.77999E-02        5.45043E-01  3.94774E-03   .00000E+00

  TOTAL  3.29140E-01  5.55578E-03   .00000E+00       7.51812E-01
5.73665E-02        7.61391E-01  4.17690E-03   .00000E+00


0REGION 51     MATERIAL FUEL_14     VOLUME  6.662320E-01

                    CROSS-SECTIONS                          FLUXES
                      REACTIONS
 GROUP   DIFFUSION   ABSORPTION   NU*FISSION        RIF            RAF
        TRANSPORT   ABSORPTION   NU*FISSION

    1    2.32885E+00  1.78377E-02  5.28855E-02       2.28868E-03
3.43526E-03        3.27584E-04  4.08249E-05  1.21038E-04
    2    1.91875E+00  1.57905E-02  3.97823E-02       7.78741E-03
1.16887E-02        1.35286E-03  1.22967E-04  3.09801E-04
    3    1.18870E+00  8.57947E-03  1.48435E-02       6.21847E-03
9.33379E-03        1.74377E-03  5.33512E-05  9.23039E-05
    4    8.16399E-01  9.08466E-03  1.44654E-02       3.29396E-03
4.94416E-03        1.34491E-03  2.99245E-05  4.76485E-05
    5    8.11678E-01  1.23757E-02  1.74729E-02       1.88767E-03
2.83335E-03        7.75213E-04  2.33613E-05  3.29831E-05
    6    7.06212E-01  2.23040E-02  2.45211E-02       2.42715E-03
3.64310E-03        1.14562E-03  5.41350E-05  5.95163E-05
    7    6.33971E-01  4.98772E-02  5.54151E-02       2.03558E-03
3.05536E-03        1.07028E-03  1.01529E-04  1.12802E-04
    8    5.71165E-01  1.29766E-01  1.54831E-01       1.91610E-03
2.87602E-03        1.11824E-03  2.48644E-04  2.96672E-04
    9    4.83010E-01  2.42135E-01  2.40387E-01       9.15423E-04
1.37403E-03        6.31748E-04  2.21656E-04  2.20056E-04
   10    3.98856E-01  3.85543E-01  4.87313E-01       8.64907E-04
1.29821E-03        7.22823E-04  3.33459E-04  4.21480E-04
   11    3.78054E-01  4.94895E-01  5.51727E-01       8.21374E-04
1.23287E-03        7.24213E-04  4.06494E-04  4.53174E-04
   12    3.77531E-01  5.09213E-01  1.61345E-01       5.84951E-04
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8.78000E-04       5.16470E-04  2.97865E-04  9.43789E-05
   13    6.68706E-01  1.23913E-01  2.00238E-01        5.90493E-04
8.86318E-04       2.94346E-04  7.31695E-05  1.18239E-04
   14    6.13009E-01  1.58972E-01  2.55781E-01        5.61363E-04
8.42594E-04       3.05250E-04  8.92410E-05  1.43586E-04
   15    3.30150E-01  6.22014E-01  1.20522E+00        6.05311E-05
9.08558E-05       6.11148E-05  3.76512E-05  7.29534E-05
   16    4.05488E-01  4.34493E-01  8.86229E-01        6.45904E-05
9.69489E-05       5.30968E-05  2.80641E-05  5.72419E-05
   17    4.60036E-01  3.35121E-01  7.02402E-01        8.26819E-05
1.24104E-04       5.99097E-05  2.77084E-05  5.80759E-05
   18    4.74286E-01  3.11396E-01  6.58316E-01        3.56201E-04
5.34650E-04       2.50342E-04  1.10919E-04  2.34493E-04
   19    3.62015E-01  5.26401E-01  1.08735E+00        5.48606E-04
8.23446E-04       5.05141E-04  2.88786E-04  5.96526E-04
   20    2.31685E-01  1.04238E+00  2.03659E+00        1.93615E-04
2.90611E-04       2.78560E-04  2.01820E-04  3.94314E-04
   21    2.29393E-01  1.05628E+00  2.06603E+00        4.38205E-04
6.57736E-04       6.36761E-04  4.62867E-04  9.05346E-04
   22    1.89976E-01  1.35500E+00  2.75817E+00        1.92628E-03
2.89130E-03       3.37986E-03  2.61011E-03  5.31300E-03
   23    1.45443E-01  1.89028E+00  3.87453E+00        1.72867E-03
2.59470E-03       3.96186E-03  3.26767E-03  6.69779E-03
   24    1.14148E-01  2.51666E+00  5.16504E+00        1.48453E-03
2.22825E-03       4.33510E-03  3.73606E-03  7.66768E-03
   25    8.57929E-02  3.47833E+00  7.14022E+00        9.85877E-04
1.47978E-03       3.83045E-03  3.42921E-03  7.03938E-03
   26    5.39375E-02  5.75904E+00  1.18196E+01        3.88244E-04
5.82746E-04       2.39934E-03  2.23592E-03  4.58891E-03

 THERMAL 1.32699E-01  2.10974E+00  4.31542E+00        7.69403E-03
1.15486E-02       1.93271E-02  1.62324E-02  3.32029E-02

  TOTAL  4.23689E-01  4.58163E-01  8.93646E-01        4.04516E-02
6.07169E-02       3.18249E-02  1.85334E-02  3.61494E-02


0REGION 52     MATERIAL CLAD        VOLUME  6.792046E-01

                     CROSS-SECTIONS                               FLUXES
                       REACTIONS
 GROUP   DIFFUSION    ABSORPTION    NU*FISSION          RIF           RAF
       TRANSPORT    ABSORPTION   NU*FISSION

    1    3.16321E+00 -5.37083E-05   .00000E+00        1.87201E-03
2.75618E-03       1.97269E-04 -1.00542E-07   .00000E+00
    2    2.49217E+00  3.74237E-04   .00000E+00        6.45938E-03
9.51021E-03       8.63957E-04  2.41734E-06   .00000E+00
    3    1.67902E+00  5.04506E-04   .00000E+00        5.32160E-03
7.83504E-03       1.05649E-03  2.68478E-06   .00000E+00
    4    1.12257E+00  5.15706E-04   .00000E+00        3.00205E-03
4.41995E-03       8.91425E-04  1.54818E-06   .00000E+00
    5    9.21946E-01  6.62955E-04   .00000E+00        1.83161E-03
2.69670E-03       6.62227E-04  1.21428E-06   .00000E+00
    6    9.25362E-01  1.34549E-03   .00000E+00        2.45171E-03
3.60967E-03       8.83152E-04  3.29874E-06   .00000E+00
    7    8.79884E-01  6.78327E-03   .00000E+00        2.08586E-03
3.07103E-03       7.90203E-04  1.41489E-05   .00000E+00
    8    1.34604E+00  1.24927E-02   .00000E+00        1.98357E-03
2.92043E-03       4.91210E-04  2.47800E-05   .00000E+00
    9    1.59365E+00  1.42310E-04   .00000E+00        9.61206E-04
1.41519E-03       2.01049E-04  1.36789E-07   .00000E+00
   10    1.57942E+00  2.24967E-04   .00000E+00        9.24446E-04
1.36107E-03       1.95103E-04  2.07970E-07   .00000E+00
   11    1.57349E+00  3.85370E-04   .00000E+00        8.89306E-04
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1.30933E-03       1.88394E-04  3.42712E-07    .00000E+00
   12     1.57020E+00  6.03067E-04    .00000E+00        6.34699E-04
9.34473E-04       1.34739E-04  3.82766E-07    .00000E+00
   13     1.57409E+00  8.53588E-04    .00000E+00        6.11656E-04
9.00547E-04       1.29525E-04  5.22102E-07    .00000E+00
   14     1.56863E+00  1.16803E-03    .00000E+00        5.83724E-04
8.59423E-04       1.24041E-04  6.81808E-07    .00000E+00
   15     1.56887E+00  1.37786E-03    .00000E+00        6.63634E-05
9.77075E-05       1.41001E-05  9.14395E-08    .00000E+00
   16     1.56929E+00  1.42882E-03    .00000E+00        6.94032E-05
1.02183E-04       1.47419E-05  9.91649E-08    .00000E+00
   17     1.56537E+00  1.48846E-03    .00000E+00        8.78717E-05
1.29374E-04       1.87116E-05  1.30794E-07    .00000E+00
   18     1.56347E+00  1.67631E-03    .00000E+00        3.77357E-04
5.55587E-04       8.04531E-05  6.32567E-07    .00000E+00
   19     1.55973E+00  2.13456E-03    .00000E+00        5.96134E-04
8.77694E-04       1.27401E-04  1.27248E-06    .00000E+00
   20     1.55524E+00  2.60384E-03    .00000E+00        2.23988E-04
3.29780E-04       4.80071E-05  5.83229E-07    .00000E+00
   21     1.55192E+00  3.12774E-03    .00000E+00        5.08589E-04
7.48800E-04       1.09239E-04  1.59074E-06    .00000E+00
   22     1.54000E+00  4.45409E-03    .00000E+00        2.31539E-03
3.40897E-03       5.01167E-04  1.03129E-05    .00000E+00
   23     1.52869E+00  5.81622E-03    .00000E+00        2.21793E-03
3.26548E-03       4.83623E-04  1.28999E-05    .00000E+00
   24     1.51588E+00  7.36893E-03    .00000E+00        2.05441E-03
3.02473E-03       4.51753E-04  1.51388E-05    .00000E+00
   25     1.49532E+00  9.79734E-03    .00000E+00        1.52833E-03
2.25017E-03       3.40692E-04  1.49735E-05    .00000E+00
   26     1.43463E+00  1.63396E-02    .00000E+00        7.73249E-04
1.13846E-03       1.79662E-04  1.26346E-05    .00000E+00

 THERMAL 1.51949E+00  6.79254E-03    .00000E+00        1.02180E-02
1.50441E-02       2.24154E-03  6.94063E-05    .00000E+00

  TOTAL  1.46838E+00  3.03291E-03    .00000E+00        4.04318E-02
5.95282E-02       9.17833E-03  1.22626E-04    .00000E+00


0REGION 53     MATERIAL COOLANT     VOLUME  2.188514E+01

                    CROSS-SECTIONS                              FLUXES
                      REACTIONS
 GROUP   DIFFUSION    ABSORPTION   NU*FISSION           RIF            RAF
        TRANSPORT    ABSORPTION   NU*FISSION

    1     2.70642E+00  2.17943E-03    .00000E+00        4.47548E-02
2.04499E-03       5.51218E-03  9.75399E-05    .00000E+00
    2     2.01031E+00  1.01023E-05    .00000E+00        1.56967E-01
7.17229E-03       2.60270E-02  1.58573E-06    .00000E+00
    3     1.31913E+00  3.55934E-06    .00000E+00        1.34256E-01
6.13458E-03       3.39254E-02  4.77863E-07    .00000E+00
    4     7.92566E-01  3.59323E-06    .00000E+00        8.31710E-02
3.80034E-03       3.49796E-02  2.98852E-07    .00000E+00
    5     6.54618E-01  6.63753E-06    .00000E+00        5.49831E-02
2.51235E-03       2.79975E-02  3.64952E-07    .00000E+00
    6     5.56182E-01  1.94203E-05    .00000E+00        7.72839E-02
3.53134E-03       4.63181E-02  1.50088E-06    .00000E+00
    7     5.63543E-01  6.29493E-05    .00000E+00        6.67967E-02
3.05215E-03       3.95100E-02  4.20481E-06    .00000E+00
    8     5.77544E-01  1.71416E-04    .00000E+00        6.41472E-02
2.93108E-03       3.70230E-02  1.09958E-05    .00000E+00
    9     5.80367E-01  3.53087E-04    .00000E+00        3.13639E-02
1.43311E-03       1.80138E-02  1.10742E-05    .00000E+00
   10     5.80777E-01  5.82085E-04    .00000E+00        3.05953E-02
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1.39800E-03       1.75600E-02  1.78091E-05   .00000E+00
   11     5.80114E-01  9.59742E-04    .00000E+00        2.97177E-02
1.35789E-03       1.70758E-02  2.85213E-05   .00000E+00
   12     5.78579E-01  1.47308E-03    .00000E+00        2.12261E-02
9.69885E-04       1.22289E-02  3.12676E-05   .00000E+00
   13     5.30274E-01  2.07713E-03    .00000E+00        1.96479E-02
8.97772E-04       1.23508E-02  4.08112E-05   .00000E+00
   14     4.84734E-01  2.83531E-03    .00000E+00        1.87974E-02
8.58912E-04       1.29263E-02  5.32965E-05   .00000E+00
   15     4.56194E-01  3.34008E-03    .00000E+00        2.22916E-03
1.01857E-04       1.62881E-03  7.44557E-06   .00000E+00
   16     4.50003E-01  3.46267E-03    .00000E+00        2.29790E-03
1.04998E-04       1.70213E-03  7.95686E-06   .00000E+00
   17     4.42700E-01  3.60538E-03    .00000E+00        2.88190E-03
1.31683E-04       2.16994E-03  1.03904E-05   .00000E+00
   18     4.19829E-01  4.05921E-03    .00000E+00        1.23420E-02
5.63944E-04       9.79923E-03  5.00987E-05   .00000E+00
   19     3.66366E-01  5.17002E-03    .00000E+00        1.98840E-02
9.08561E-04       1.80912E-02  1.02801E-04   .00000E+00
   20     3.18292E-01  6.29875E-03    .00000E+00        7.81477E-03
3.57081E-04       8.18407E-03  4.92233E-05   .00000E+00
   21     2.76329E-01  7.56452E-03    .00000E+00        1.78343E-02
8.14906E-04       2.15134E-02  1.34908E-04   .00000E+00
   22     2.07862E-01  1.07793E-02    .00000E+00        8.27624E-02
3.78167E-03       1.32720E-01  8.92118E-04   .00000E+00
   23     1.64184E-01  1.40641E-02    .00000E+00        8.17549E-02
3.73563E-03       1.65982E-01  1.14981E-03   .00000E+00
   24     1.30440E-01  1.78189E-02    .00000E+00        7.80808E-02
3.56775E-03       1.99532E-01  1.39131E-03   .00000E+00
   25     1.01321E-01  2.37240E-02    .00000E+00        6.01949E-02
2.75049E-03       1.98033E-01  1.42806E-03   .00000E+00
   26     7.46679E-02  3.97569E-02    .00000E+00        3.21327E-02
1.46824E-03       1.43447E-01  1.27749E-03   .00000E+00

 THERMAL 1.42895E-01  1.68894E-02    .00000E+00        3.80459E-01
1.73843E-02       8.87503E-01  6.42573E-03   .00000E+00

  TOTAL  3.30565E-01  5.51202E-03    .00000E+00        1.23392E+00
5.63815E-02       1.24425E+00  6.80137E-03   .00000E+00


0REGION 54     MATERIAL FUEL_15     VOLUME  1.332588E+00

                      CROSS-SECTIONS                              FLUXES
                        REACTIONS
 GROUP   DIFFUSION    ABSORPTION   NU*FISSION           RIF           RAF
        TRANSPORT    ABSORPTION   NU*FISSION

    1     2.32885E+00  1.78377E-02  5.28855E-02        4.28712E-03
3.21714E-03       6.13625E-04  7.64725E-05  2.26727E-04
    2     1.91875E+00  1.57905E-02  3.97823E-02        1.46963E-02
1.10284E-02       2.55312E-03  2.32063E-04  5.84654E-04
    3     1.18870E+00  8.57947E-03  1.48435E-02        1.17819E-02
8.84140E-03       3.30387E-03  1.01083E-04  1.74885E-04
    4     8.16399E-01  9.08466E-03  1.44654E-02        6.34669E-03
4.76268E-03       2.59134E-03  5.76576E-05  9.18077E-05
    5     8.11678E-01  1.23757E-02  1.74729E-02        3.68737E-03
2.76708E-03       1.51430E-03  4.56338E-05  6.44292E-05
    6     7.06212E-01  2.23040E-02  2.45211E-02        4.78842E-03
3.59332E-03       2.26014E-03  1.06801E-04  1.17417E-04
    7     6.33971E-01  4.98772E-02  5.54151E-02        4.02154E-03
3.01784E-03       2.11447E-03  2.00583E-04  2.22854E-04
    8     5.71165E-01  1.29766E-01  1.54831E-01        3.78210E-03
2.83816E-03       2.20724E-03  4.90788E-04  5.85588E-04
    9     4.83010E-01  2.42135E-01  2.40387E-01        1.80679E-03
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1.35585E-03       1.24690E-03  4.37487E-04  4.34330E-04
   10    3.98856E-01  3.85543E-01  4.87313E-01       1.70831E-03
1.28195E-03       1.42767E-03  6.58626E-04  8.32480E-04
   11    3.78054E-01  4.94895E-01  5.51727E-01       1.62239E-03
1.21748E-03       1.43048E-03  8.02914E-04  8.95118E-04
   12    3.77531E-01  5.09213E-01  1.61345E-01       1.15441E-03
8.66289E-04       1.01926E-03  5.87838E-04  1.86258E-04
   13    6.68706E-01  1.23913E-01  2.00238E-01       1.15671E-03
8.68015E-04       5.76590E-04  1.43331E-04  2.31617E-04
   14    6.13009E-01  1.58972E-01  2.55781E-01       1.09882E-03
8.24572E-04       5.97498E-04  1.74681E-04  2.81056E-04
   15    3.30150E-01  6.22014E-01  1.20522E+00       1.19088E-04
8.93662E-05       1.20237E-04  7.40746E-05  1.43528E-04
   16    4.05488E-01  4.34493E-01  8.86229E-01       1.26809E-04
9.51598E-05       1.04244E-04  5.50976E-05  1.12382E-04
   17    4.60036E-01  3.35121E-01  7.02402E-01       1.62098E-04
1.21642E-04       1.17453E-04  5.43225E-05  1.13858E-04
   18    4.74286E-01  3.11396E-01  6.58316E-01       6.97572E-04
5.23472E-04       4.90261E-04  2.17221E-04  4.59223E-04
   19    3.62015E-01  5.26401E-01  1.08735E+00       1.07553E-03
8.07099E-04       9.90318E-04  5.66160E-04  1.16948E-03
   20    2.31685E-01  1.04238E+00  2.03659E+00       3.81341E-04
2.86165E-04       5.48647E-04  3.97502E-04  7.76635E-04
   21    2.29393E-01  1.05628E+00  2.06603E+00       8.63917E-04
6.48300E-04       1.25537E-03  9.12537E-04  1.78488E-03
   22    1.89976E-01  1.35500E+00  2.75817E+00       3.80879E-03
2.85819E-03       6.68292E-03  5.16092E-03  1.05053E-02
   23    1.45443E-01  1.89028E+00  3.87453E+00       3.41195E-03
2.56040E-03       7.81970E-03  6.44954E-03  1.32197E-02
   24    1.14148E-01  2.51666E+00  5.16504E+00       2.91948E-03
2.19083E-03       8.52540E-03  7.34732E-03  1.50792E-02
   25    8.57929E-02  3.47833E+00  7.14022E+00       1.92798E-03
1.44679E-03       7.49081E-03  6.70614E-03  1.37662E-02
   26    5.39375E-02  5.75904E+00  1.18196E+01       7.54625E-04
5.66285E-04       4.66357E-03  4.34592E-03  8.91940E-03

 THERMAL 1.32920E-01  2.10558E+00  4.30682E+00       1.51436E-02
1.13641E-02       3.79767E-02  3.18860E-02  6.52208E-02

  TOTAL  4.18574E-01  4.65579E-01  9.07798E-01       7.81881E-02
5.86739E-02       6.22654E-02  3.64027E-02  7.09790E-02


0REGION 55     MATERIAL CLAD        VOLUME  1.358407E+00

                    CROSS-SECTIONS                          FLUXES
                      REACTIONS
 GROUP   DIFFUSION    ABSORPTION   NU*FISSION          RIF           RAF
       TRANSPORT    ABSORPTION   NU*FISSION


    1    3.16321E+00 -5.37083E-05   .00000E+00       3.45902E-03
2.54638E-03       3.64505E-04 -1.85778E-07   .00000E+00
    2    2.49217E+00  3.74237E-04   .00000E+00       1.20568E-02
8.87571E-03       1.61263E-03  4.51211E-06   .00000E+00
    3    1.67902E+00  5.04506E-04   .00000E+00       1.00098E-02
7.36881E-03       1.98724E-03  5.05003E-06   .00000E+00
    4    1.12257E+00  5.15706E-04   .00000E+00       5.77164E-03
4.24883E-03       1.71382E-03  2.97647E-06   .00000E+00
    5    9.21946E-01  6.62955E-04   .00000E+00       3.57667E-03
2.63299E-03       1.29316E-03  2.37117E-06   .00000E+00
    6    9.25362E-01  1.34549E-03   .00000E+00       4.83642E-03
3.56036E-03       1.74217E-03  6.50734E-06   .00000E+00
    7    8.79884E-01  6.78327E-03   .00000E+00       4.12049E-03
3.03333E-03       1.56100E-03  2.79504E-05   .00000E+00
    8    1.34604E+00  1.24927E-02   .00000E+00       3.91511E-03
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2.88213E-03       9.69535E-04  4.89101E-05   .00000E+00
    9     1.59365E+00  1.42310E-04   .00000E+00       1.89694E-03
1.39644E-03       3.96769E-04  2.69952E-07   .00000E+00
   10     1.57942E+00  2.24967E-04   .00000E+00       1.82577E-03
1.34405E-03       3.85326E-04  4.10739E-07   .00000E+00
   11     1.57349E+00  3.85370E-04   .00000E+00       1.75641E-03
1.29300E-03       3.72084E-04  6.76869E-07   .00000E+00
   12     1.57020E+00  6.03067E-04   .00000E+00       1.25245E-03
9.22000E-04       2.65880E-04  7.55311E-07   .00000E+00
   13     1.57409E+00  8.53588E-04   .00000E+00       1.19805E-03
8.81953E-04       2.53702E-04  1.02264E-06   .00000E+00
   14     1.56863E+00  1.16803E-03   .00000E+00       1.14253E-03
8.41078E-04       2.42787E-04  1.33451E-06   .00000E+00
   15     1.56887E+00  1.37786E-03   .00000E+00       1.30559E-04
9.61120E-05       2.77396E-05  1.79892E-07   .00000E+00
   16     1.56929E+00  1.42882E-03   .00000E+00       1.36244E-04
1.00297E-04       2.89397E-05  1.94669E-07   .00000E+00
   17     1.56537E+00  1.48846E-03   .00000E+00       1.72251E-04
1.26804E-04       3.66797E-05  2.56389E-07   .00000E+00
   18     1.56347E+00  1.67631E-03   .00000E+00       7.38913E-04
5.43956E-04       1.57537E-04  1.23865E-06   .00000E+00
   19     1.55973E+00  2.13456E-03   .00000E+00       1.16850E-03
8.60196E-04       2.49722E-04  2.49422E-06   .00000E+00
   20     1.55524E+00  2.60384E-03   .00000E+00       4.41033E-04
3.24669E-04       9.45262E-05  1.14838E-06   .00000E+00
   21     1.55192E+00  3.12774E-03   .00000E+00       1.00231E-03
7.37855E-04       2.15283E-04  3.13496E-06   .00000E+00
   22     1.54000E+00  4.45409E-03   .00000E+00       4.57094E-03
3.36493E-03       9.89382E-04  2.03594E-05   .00000E+00
   23     1.52869E+00  5.81622E-03   .00000E+00       4.36713E-03
3.21489E-03       9.52261E-04  2.54002E-05   .00000E+00
   24     1.51588E+00  7.36893E-03   .00000E+00       4.02909E-03
2.96604E-03       8.85975E-04  2.96901E-05   .00000E+00
   25     1.49532E+00  9.79734E-03   .00000E+00       2.97989E-03
2.19366E-03       6.64271E-04  2.91950E-05   .00000E+00
   26     1.43463E+00  1.63396E-02   .00000E+00       1.49810E-03
1.10284E-03       3.48080E-04  2.44784E-05   .00000E+00

 THERMAL 1.51964E+00  6.77572E-03   .00000E+00       2.00570E-02
1.47651E-02       4.39950E-03  1.35901E-04   .00000E+00

  TOTAL  1.46077E+00  3.07908E-03   .00000E+00       7.80531E-02
5.74593E-02       1.78110E-02  2.40332E-04   .00000E+00


0REGION 56     MATERIAL COOLANT     VOLUME  1.290042E+01

                    CROSS-SECTIONS                            FLUXES
                      REACTIONS
 GROUP  DIFFUSION    ABSORPTION   NU*FISSION          RIF           RAF
       TRANSPORT    ABSORPTION   NU*FISSION

    1     2.70642E+00  2.17943E-03   .00000E+00       2.56311E-02
1.98684E-03       3.15683E-03  5.58612E-05   .00000E+00
    2     2.01031E+00  1.01023E-05   .00000E+00       9.07007E-02
7.03084E-03       1.50393E-02  9.16288E-07   .00000E+00
    3     1.31913E+00  3.55934E-06   .00000E+00       7.77819E-02
6.02941E-03       1.96548E-02  2.76852E-07   .00000E+00
    4     7.92566E-01  3.59323E-06   .00000E+00       4.85274E-02
3.76169E-03       2.04094E-02  1.74370E-07   .00000E+00
    5     6.54618E-01  6.63753E-06   .00000E+00       3.21496E-02
2.49214E-03       1.63707E-02  2.13394E-07   .00000E+00
    6     5.56182E-01  1.94203E-05   .00000E+00       4.52627E-02
3.50862E-03       2.71270E-02  8.79017E-07   .00000E+00
    7     5.63543E-01  6.29493E-05   .00000E+00       3.90661E-02
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3.02828E-03       2.31074E-02  2.45918E-06   .00000E+00
    8    5.77544E-01  1.71416E-04   .00000E+00        3.74138E-02
2.90020E-03       2.15936E-02  6.41332E-06   .00000E+00
    9    5.80367E-01  3.53087E-04   .00000E+00        1.82592E-02
1.41539E-03       1.04872E-02  6.44708E-06   .00000E+00
   10    5.80777E-01  5.82085E-04   .00000E+00        1.77793E-02
1.37820E-03       1.02043E-02  1.03491E-05   .00000E+00
   11    5.80114E-01  9.59742E-04   .00000E+00        1.72398E-02
1.33638E-03       9.90600E-03  1.65458E-05   .00000E+00
   12    5.78579E-01  1.47308E-03   .00000E+00        1.23005E-02
9.53500E-04       7.08664E-03  1.81196E-05   .00000E+00
   13    5.30274E-01  2.07713E-03   .00000E+00        1.13805E-02
8.82182E-04       7.15385E-03  2.36388E-05   .00000E+00
   14    4.84734E-01  2.83531E-03   .00000E+00        1.08769E-02
8.43142E-04       7.47962E-03  3.08393E-05   .00000E+00
   15    4.56194E-01  3.34008E-03   .00000E+00        1.28786E-03
9.98311E-05       9.41020E-04  4.30156E-06   .00000E+00
   16    4.50003E-01  3.46267E-03   .00000E+00        1.32776E-03
1.02923E-04       9.83516E-04  4.59758E-06   .00000E+00
   17    4.42700E-01  3.60538E-03   .00000E+00        1.66537E-03
1.29094E-04       1.25395E-03  6.00430E-06   .00000E+00
   18    4.19829E-01  4.05921E-03   .00000E+00        7.12832E-03
5.52565E-04       5.65970E-03  2.89353E-05   .00000E+00
   19    3.66366E-01  5.17002E-03   .00000E+00        1.14646E-02
8.88700E-04       1.04309E-02  5.92722E-05   .00000E+00
   20    3.18292E-01  6.29875E-03   .00000E+00        4.49741E-03
3.48625E-04       4.70995E-03  2.83281E-05   .00000E+00
   21    2.76329E-01  7.56452E-03   .00000E+00        1.02698E-02
7.96084E-04       1.23884E-02  7.76861E-05   .00000E+00
   22    2.07862E-01  1.07793E-02   .00000E+00        4.78013E-02
3.70541E-03       7.66554E-02  5.15262E-04   .00000E+00
   23    1.64184E-01  1.40641E-02   .00000E+00        4.71432E-02
3.65439E-03       9.57121E-02  6.63027E-04   .00000E+00
   24    1.30440E-01  1.78189E-02   .00000E+00        4.49217E-02
3.48219E-03       1.14795E-01  8.00455E-04   .00000E+00
   25    1.01321E-01  2.37240E-02   .00000E+00        3.45378E-02
2.67727E-03       1.13625E-01  8.19375E-04   .00000E+00
   26    7.46679E-02  3.97569E-02   .00000E+00        1.84028E-02
1.42653E-03       8.21539E-02  7.31637E-04   .00000E+00

 THERMAL 1.43030E-01  1.68694E-02   .00000E+00        2.19039E-01
1.69792E-02       5.10471E-01  3.69504E-03   .00000E+00

  TOTAL  3.31816E-01  5.47275E-03   .00000E+00        7.14818E-01
5.54104E-02       7.18086E-01  3.91202E-03   .00000E+00


0REGION 57     MATERIAL FUEL_16     VOLUME  1.332575E+00

                    CROSS-SECTIONS                            FLUXES
                      REACTIONS
 GROUP   DIFFUSION    ABSORPTION   NU*FISSION          RIF            RAF
        TRANSPORT    ABSORPTION   NU*FISSION

    1    2.32885E+00  1.78377E-02  5.28855E-02        4.24579E-03
3.18616E-03       6.07709E-04  7.57353E-05  2.24541E-04
    2    1.91875E+00  1.57905E-02  3.97823E-02        1.46318E-02
1.09801E-02       2.54191E-03  2.31044E-04  5.82087E-04
    3    1.18870E+00  8.57947E-03  1.48435E-02        1.17711E-02
8.83331E-03       3.30082E-03  1.00989E-04  1.74724E-04
    4    8.16399E-01  9.08466E-03  1.44654E-02        6.35219E-03
4.76685E-03       2.59358E-03  5.77075E-05  9.18871E-05
    5    8.11678E-01  1.23757E-02  1.74729E-02        3.68394E-03
2.76453E-03       1.51289E-03  4.55913E-05  6.43692E-05
    6    7.06212E-01  2.23040E-02  2.45211E-02        4.77750E-03
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3.58516E-03       2.25499E-03  1.06557E-04  1.17150E-04
    7    6.33971E-01  4.98772E-02  5.54151E-02        4.00364E-03
3.00444E-03       2.10506E-03  1.99690E-04  2.21862E-04
    8    5.71165E-01  1.29766E-01  1.54831E-01        3.75534E-03
2.81811E-03       2.19163E-03  4.87315E-04  5.81444E-04
    9    4.83010E-01  2.42135E-01  2.40387E-01        1.79039E-03
1.34356E-03       1.23558E-03  4.33515E-04  4.30387E-04
   10    3.98856E-01  3.85543E-01  4.87313E-01        1.68890E-03
1.26740E-03       1.41146E-03  6.51144E-04  8.23023E-04
   11    3.78054E-01  4.94895E-01  5.51727E-01        1.60108E-03
1.20149E-03       1.41168E-03  7.92364E-04  8.83356E-04
   12    3.77531E-01  5.09213E-01  1.61345E-01        1.13848E-03
8.54349E-04       1.00520E-03  5.79731E-04  1.83689E-04
   13    6.68706E-01  1.23913E-01  2.00238E-01        1.14432E-03
8.58726E-04       5.70414E-04  1.41795E-04  2.29136E-04
   14    6.13009E-01  1.58972E-01  2.55781E-01        1.08616E-03
8.15081E-04       5.90614E-04  1.72668E-04  2.77818E-04
   15    3.30150E-01  6.22014E-01  1.20522E+00        1.17195E-04
8.79461E-05       1.18325E-04  7.28968E-05  1.41246E-04
   16    4.05488E-01  4.34493E-01  8.86229E-01        1.24934E-04
9.37538E-05       1.02702E-04  5.42829E-05  1.10720E-04
   17    4.60036E-01  3.35121E-01  7.02402E-01        1.59836E-04
1.19945E-04       1.15814E-04  5.35643E-05  1.12269E-04
   18    4.74286E-01  3.11396E-01  6.58316E-01        6.87784E-04
5.16132E-04       4.83382E-04  2.14173E-04  4.52779E-04
   19    3.62015E-01  5.26401E-01  1.08735E+00        1.05756E-03
7.93624E-04       9.73774E-04  5.56702E-04  1.14994E-03
   20    2.31685E-01  1.04238E+00  2.03659E+00        3.73263E-04
2.80107E-04       5.37026E-04  3.89082E-04  7.60185E-04
   21    2.29393E-01  1.05628E+00  2.06603E+00        8.45910E-04
6.34793E-04       1.22920E-03  8.93516E-04  1.74768E-03
   22    1.89976E-01  1.35500E+00  2.75817E+00        3.73234E-03
2.80085E-03       6.54879E-03  5.05733E-03  1.02944E-02
   23    1.45443E-01  1.89028E+00  3.87453E+00        3.33382E-03
2.50179E-03       7.64064E-03  6.30185E-03  1.29170E-02
   24    1.14148E-01  2.51666E+00  5.16504E+00        2.84444E-03
2.13455E-03       8.30628E-03  7.15848E-03  1.46917E-02
   25    8.57929E-02  3.47833E+00  7.14022E+00        1.87208E-03
1.40486E-03       7.27363E-03  6.51171E-03  1.33670E-02
   26    5.39375E-02  5.75904E+00  1.18196E+01        7.30335E-04
5.48063E-04       4.51346E-03  4.20603E-03  8.63230E-03

 THERMAL 1.33159E-01  2.10110E+00  4.29759E+00        1.47898E-02
1.10986E-02       3.70228E-02  3.10747E-02  6.35602E-02

  TOTAL  4.22548E-01  4.58355E-01  8.93135E-01        7.75501E-02
5.81957E-02       6.11765E-02  3.55455E-02  6.92627E-02


0REGION 58     MATERIAL CLAD        VOLUME  1.358438E+00

                    CROSS-SECTIONS                          FLUXES
                      REACTIONS
 GROUP   DIFFUSION    ABSORPTION   NU*FISSION           RIF          RAF
        TRANSPORT    ABSORPTION   NU*FISSION

    1    3.16321E+00 -5.37083E-05   .00000E+00        3.48469E-03
2.56522E-03       3.67211E-04 -1.87157E-07   .00000E+00
    2    2.49217E+00  3.74237E-04   .00000E+00        1.21871E-02
8.97139E-03       1.63005E-03  4.56086E-06   .00000E+00
    3    1.67902E+00  5.04506E-04   .00000E+00        1.01030E-02
7.43719E-03       2.00573E-03  5.09701E-06   .00000E+00
    4    1.12257E+00  5.15706E-04   .00000E+00        5.80727E-03
4.27496E-03       1.72440E-03  2.99484E-06   .00000E+00
    5    9.21946E-01  6.62955E-04   .00000E+00        3.58076E-03
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2.63594E-03 1.29464E-03 2.37388E-06 .00000E+00
6 9.25362E-01 1.34549E-03 .00000E+00 4.82711E-03
3.55343E-03 1.73882E-03 6.49482E-06 .00000E+00
7 8.79884E-01 6.78327E-03 .00000E+00 4.10208E-03
3.01970E-03 1.55402E-03 2.78255E-05 .00000E+00
8 1.34604E+00 1.24927E-02 .00000E+00 3.88691E-03
2.86131E-03 9.62552E-04 4.85578E-05 .00000E+00
9 1.59365E+00 1.42310E-04 .00000E+00 1.87921E-03
1.38336E-03 3.93061E-04 2.67429E-07 .00000E+00
10 1.57942E+00 2.24967E-04 .00000E+00 1.80430E-03
1.32822E-03 3.80795E-04 4.05909E-07 .00000E+00
11 1.57349E+00 3.85370E-04 .00000E+00 1.73244E-03
1.27531E-03 3.67005E-04 6.67629E-07 .00000E+00
12 1.57020E+00 6.03067E-04 .00000E+00 1.23450E-03
9.08764E-04 2.62069E-04 7.44486E-07 .00000E+00
13 1.57409E+00 8.53588E-04 .00000E+00 1.18507E-03
8.72379E-04 2.50953E-04 1.01156E-06 .00000E+00
14 1.56863E+00 1.16803E-03 .00000E+00 1.12918E-03
8.31233E-04 2.39951E-04 1.31892E-06 .00000E+00
15 1.56887E+00 1.37786E-03 .00000E+00 1.28406E-04
9.45249E-05 2.72821E-05 1.76926E-07 .00000E+00
16 1.56929E+00 1.42882E-03 .00000E+00 1.34173E-04
9.87701E-05 2.84997E-05 1.91709E-07 .00000E+00
17 1.56537E+00 1.48846E-03 .00000E+00 1.69793E-04
1.24991E-04 3.61561E-05 2.52729E-07 .00000E+00
18 1.56347E+00 1.67631E-03 .00000E+00 7.28337E-04
5.36158E-04 1.55282E-04 1.22092E-06 .00000E+00
19 1.55973E+00 2.13456E-03 .00000E+00 1.14850E-03
8.45455E-04 2.45448E-04 2.45154E-06 .00000E+00
20 1.55524E+00 2.60384E-03 .00000E+00 4.31425E-04
3.17589E-04 9.24668E-05 1.12336E-06 .00000E+00
21 1.55192E+00 3.12774E-03 .00000E+00 9.80864E-04
7.22053E-04 2.10677E-04 3.06789E-06 .00000E+00
22 1.54000E+00 4.45409E-03 .00000E+00 4.47641E-03
3.29526E-03 9.68921E-04 1.99383E-05 .00000E+00
23 1.52869E+00 5.81622E-03 .00000E+00 4.26469E-03
3.13941E-03 9.29925E-04 2.48044E-05 .00000E+00
24 1.51588E+00 7.36893E-03 .00000E+00 3.92397E-03
2.88859E-03 8.62858E-04 2.89155E-05 .00000E+00
25 1.49532E+00 9.79734E-03 .00000E+00 2.89328E-03
2.12986E-03 6.44965E-04 2.83465E-05 .00000E+00
26 1.43463E+00 1.63396E-02 .00000E+00 1.45030E-03
1.06763E-03 3.36974E-04 2.36974E-05 .00000E+00

THERMAL 1.51976E+00 6.76283E-03 .00000E+00 1.95694E-02
1.44058E-02 4.29224E-03 1.32345E-04 .00000E+00

TOTAL 1.46190E+00 3.04248E-03 .00000E+00 7.76737E-02
5.71787E-02 1.77107E-02 2.36321E-04 .00000E+00

0REGION 59 MATERIAL COOLANT VOLUME 1.671883E+01

CROSS-SECTIONS FLUXES
REACTIONS
GROUP DIFFUSION ABSORPTION NU*FISSION RIF RAF
TRANSPORT ABSORPTION NU*FISSION

1 2.70642E+00 2.17943E-03 .00000E+00 3.20927E-02
1.91955E-03 3.95266E-03 6.99437E-05 .00000E+00
2 2.01031E+00 1.01023E-05 .00000E+00 1.14626E-01
6.85609E-03 1.90064E-02 1.15799E-06 .00000E+00
3 1.31913E+00 3.55934E-06 .00000E+00 9.86274E-02
5.89918E-03 2.49223E-02 3.51048E-07 .00000E+00
4 7.92566E-01 3.59323E-06 .00000E+00 6.21879E-02

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3.71963E-03       2.61547E-02  2.23455E-07   .00000E+00
    5     6.54618E-01  6.63753E-06    .00000E+00         4.13974E-02
2.47609E-03       2.10797E-02  2.74776E-07   .00000E+00
    6     5.56182E-01  1.94203E-05    .00000E+00         5.84491E-02
3.49600E-03       3.50300E-02  1.13510E-06   .00000E+00
    7     5.63543E-01  6.29493E-05    .00000E+00         5.03822E-02
3.01350E-03       2.98008E-02  3.17153E-06   .00000E+00
    8     5.77544E-01  1.71416E-04    .00000E+00         4.81271E-02
2.87862E-03       2.77769E-02  8.24975E-06   .00000E+00
    9     5.80367E-01  3.53087E-04    .00000E+00         2.34564E-02
1.40299E-03       1.34722E-02  8.28215E-06   .00000E+00
   10     5.80777E-01  5.82085E-04    .00000E+00         2.28121E-02
1.36446E-03       1.30929E-02  1.32786E-05   .00000E+00
   11     5.80114E-01  9.59742E-04    .00000E+00         2.20940E-02
1.32150E-03       1.26952E-02  2.12045E-05   .00000E+00
   12     5.78579E-01  1.47308E-03    .00000E+00         1.57512E-02
9.42122E-04       9.07463E-03  2.32027E-05   .00000E+00
   13     5.30274E-01  2.07713E-03    .00000E+00         1.45447E-02
8.69961E-04       9.14290E-03  3.02113E-05   .00000E+00
   14     4.84734E-01  2.83531E-03    .00000E+00         1.38886E-02
8.30717E-04       9.55067E-03  3.93785E-05   .00000E+00
   15     4.56194E-01  3.34008E-03    .00000E+00         1.64460E-03
9.83684E-05       1.20169E-03  5.49311E-06   .00000E+00
   16     4.50003E-01  3.46267E-03    .00000E+00         1.69490E-03
1.01377E-04       1.25547E-03  5.86888E-06   .00000E+00
   17     4.42700E-01  3.60538E-03    .00000E+00         2.12535E-03
1.27123E-04       1.60030E-03  7.66271E-06   .00000E+00
   18     4.19829E-01  4.05921E-03    .00000E+00         9.09195E-03
5.43815E-04       7.21876E-03  3.69061E-05   .00000E+00
   19     3.66366E-01  5.17002E-03    .00000E+00         1.46096E-02
8.73839E-04       1.32923E-02  7.55317E-05   .00000E+00
   20     3.18292E-01  6.29875E-03    .00000E+00         5.72971E-03
3.42710E-04       6.00048E-03  3.60900E-05   .00000E+00
   21     2.76329E-01  7.56452E-03    .00000E+00         1.30981E-02
7.83433E-04       1.58001E-02  9.90807E-05   .00000E+00
   22     2.07862E-01  1.07793E-02    .00000E+00         6.14250E-02
3.67400E-03       9.85027E-02  6.62116E-04   .00000E+00
   23     1.64184E-01  1.40641E-02    .00000E+00         6.06765E-02
3.62923E-03       1.23188E-01  8.53361E-04   .00000E+00
   24     1.30440E-01  1.78189E-02    .00000E+00         5.78178E-02
3.45824E-03       1.47751E-01  1.03025E-03   .00000E+00
   25     1.01321E-01  2.37240E-02    .00000E+00         4.44134E-02
2.65649E-03       1.46114E-01  1.05366E-03   .00000E+00
   26     7.46679E-02  3.97569E-02    .00000E+00         2.36729E-02
1.41594E-03       1.05681E-01  9.41161E-04   .00000E+00

 THERMAL 1.42938E-01  1.68818E-02    .00000E+00         2.81443E-01
1.68339E-02       6.56329E-01  4.75125E-03   .00000E+00

  TOTAL  3.30471E-01  5.49765E-03    .00000E+00         9.14436E-01
5.46950E-02       9.22357E-01  5.02725E-03   .00000E+00


0REGION 60     MATERIAL FUEL_17     VOLUME  1.776733E+00

                      CROSS-SECTIONS                              FLUXES
                        REACTIONS
 GROUP   DIFFUSION    ABSORPTION   NU*FISSION          RIF            RAF
       TRANSPORT    ABSORPTION   NU*FISSION

    1     2.32885E+00  1.78377E-02  5.28855E-02        5.51591E-03
3.10452E-03       7.89503E-04  9.83912E-05  2.91711E-04
    2     1.91875E+00  1.57905E-02  3.97823E-02        1.91772E-02
1.07935E-02       3.33155E-03  3.02818E-04  7.62914E-04
    3     1.18870E+00  8.57947E-03  1.48435E-02        1.54227E-02
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8.68037E-03       4.32480E-03  1.32319E-04  2.28927E-04
    4    8.16399E-01  9.08466E-03  1.44654E-02       8.39964E-03
4.72757E-03       3.42955E-03  7.63079E-05  1.21504E-04
    5    8.11678E-01  1.23757E-02  1.74729E-02       4.88394E-03
2.74883E-03       2.00570E-03  6.04422E-05  8.53367E-05
    6    7.06212E-01  2.23040E-02  2.45211E-02       6.33935E-03
3.56798E-03       2.99219E-03  1.41393E-04  1.55448E-04
    7    6.33971E-01  4.98772E-02  5.54151E-02       5.27828E-03
2.97078E-03       2.77525E-03  2.63266E-04  2.92496E-04
    8    5.71165E-01  1.29766E-01  1.54831E-01       4.91233E-03
2.76481E-03       2.86685E-03  6.37452E-04  7.60582E-04
    9    4.83010E-01  2.42135E-01  2.40387E-01       2.33211E-03
1.31258E-03       1.60943E-03  5.64685E-04  5.60610E-04
   10    3.98856E-01  3.85543E-01  4.87313E-01       2.18894E-03
1.23200E-03       1.82934E-03  8.43929E-04  1.06670E-03
   11    3.78054E-01  4.94895E-01  5.51727E-01       2.06677E-03
1.16324E-03       1.82229E-03  1.02283E-03  1.14029E-03
   12    3.77531E-01  5.09213E-01  1.61345E-01       1.46704E-03
8.25694E-04       1.29529E-03  7.47035E-04  2.36699E-04
   13    6.68706E-01  1.23913E-01  2.00238E-01       1.47675E-03
8.31158E-04       7.36122E-04  1.82987E-04  2.95701E-04
   14    6.13009E-01  1.58972E-01  2.55781E-01       1.39852E-03
7.87132E-04       7.60469E-04  2.22326E-04  3.57716E-04
   15    3.30150E-01  6.22014E-01  1.20522E+00       1.49966E-04
8.44052E-05       1.51412E-04  9.32807E-05  1.80742E-04
   16    4.05488E-01  4.34493E-01  8.86229E-01       1.60054E-04
9.00834E-05       1.31573E-04  6.95424E-05  1.41845E-04
   17    4.60036E-01  3.35121E-01  7.02402E-01       2.04963E-04
1.15359E-04       1.48512E-04  6.86872E-05  1.43966E-04
   18    4.74286E-01  3.11396E-01  6.58316E-01       8.81167E-04
4.95948E-04       6.19294E-04  2.74392E-04  5.80086E-04
   19    3.62015E-01  5.26401E-01  1.08735E+00       1.34787E-03
7.58625E-04       1.24108E-03  7.09521E-04  1.46561E-03
   20    2.31685E-01  1.04238E+00  2.03659E+00       4.72126E-04
2.65727E-04       6.79263E-04  4.92135E-04  9.61528E-04
   21    2.29393E-01  1.05628E+00  2.06603E+00       1.07216E-03
6.03447E-04       1.55798E-03  1.13251E-03  2.21513E-03
   22    1.89976E-01  1.35500E+00  2.75817E+00       4.81093E-03
2.70774E-03       8.44129E-03  6.51882E-03  1.32693E-02
   23    1.45443E-01  1.89028E+00  3.87453E+00       4.29210E-03
2.41573E-03       9.83687E-03  8.11327E-03  1.66299E-02
   24    1.14148E-01  2.51666E+00  5.16504E+00       3.64430E-03
2.05112E-03       1.06420E-02  9.17144E-03  1.88229E-02
   25    8.57929E-02  3.47833E+00  7.14022E+00       2.37430E-03
1.33633E-03       9.22493E-03  8.25861E-03  1.69530E-02
   26    5.39375E-02  5.75904E+00  1.18196E+01       9.19828E-04
5.17708E-04       5.68453E-03  5.29733E-03  1.08720E-02

 THERMAL 1.33407E-01  2.09646E+00  4.28811E+00       1.89336E-02
1.06564E-02       4.73079E-02  3.96936E-02  8.11895E-02

  TOTAL  4.27353E-01  4.49610E-01  8.75516E-01       1.01189E-01
5.69524E-02       7.89271E-02  4.54957E-02  8.85928E-02


0REGION 61     MATERIAL CLAD        VOLUME  1.811211E+00

                    CROSS-SECTIONS                              FLUXES
                      REACTIONS
 GROUP   DIFFUSION   ABSORPTION   NU*FISSION            RIF          RAF
        TRANSPORT   ABSORPTION  NU*FISSION

    1    3.16321E+00 -5.37083E-05   .00000E+00       4.49872E-03
2.48382E-03       4.74067E-04 -2.41619E-07   .00000E+00
    2    2.49217E+00  3.74237E-04   .00000E+00       1.59457E-02
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8.80391E-03        2.13278E-03  5.96749E-06   .00000E+00
     3   1.67902E+00  5.04506E-04   .00000E+00       1.32497E-02
7.31540E-03        2.63045E-03  6.68458E-06   .00000E+00
     4   1.12257E+00  5.15706E-04   .00000E+00       7.69624E-03
4.24922E-03        2.28531E-03  3.96900E-06   .00000E+00
     5   9.21946E-01  6.62955E-04   .00000E+00       4.75123E-03
2.62324E-03        1.71783E-03  3.14985E-06   .00000E+00
     6   9.25362E-01  1.34549E-03   .00000E+00       6.40582E-03
3.53676E-03        2.30750E-03  8.61897E-06   .00000E+00
     7   8.79884E-01  6.78327E-03   .00000E+00       5.40887E-03
2.98633E-03        2.04909E-03  3.66898E-05   .00000E+00
     8   1.34604E+00  1.24927E-02   .00000E+00       5.08581E-03
2.80796E-03        1.25945E-03  6.35354E-05   .00000E+00
     9   1.59365E+00  1.42310E-04   .00000E+00       2.44858E-03
1.35190E-03        5.12153E-04  3.48456E-07   .00000E+00
    10   1.57942E+00  2.24967E-04   .00000E+00       2.33941E-03
1.29162E-03        4.93728E-04  5.26290E-07   .00000E+00
    11   1.57349E+00  3.85370E-04   .00000E+00       2.23722E-03
1.23521E-03        4.73940E-04  8.62158E-07   .00000E+00
    12   1.57020E+00  6.03067E-04   .00000E+00       1.59136E-03
8.78617E-04        3.37826E-04  9.59697E-07   .00000E+00
    13   1.57409E+00  8.53588E-04   .00000E+00       1.52981E-03
8.44635E-04        3.23956E-04  1.30583E-06   .00000E+00
    14   1.56863E+00  1.16803E-03   .00000E+00       1.45438E-03
8.02990E-04        3.09057E-04  1.69877E-06   .00000E+00
    15   1.56887E+00  1.37786E-03   .00000E+00       1.64381E-04
9.07573E-05        3.49255E-05  2.26493E-07   .00000E+00
    16   1.56929E+00  1.42882E-03   .00000E+00       1.71967E-04
9.49456E-05        3.65275E-05  2.45710E-07   .00000E+00
    17   1.56537E+00  1.48846E-03   .00000E+00       2.17826E-04
1.20266E-04        4.63845E-05  3.24226E-07   .00000E+00
    18   1.56347E+00  1.67631E-03   .00000E+00       9.33521E-04
5.15412E-04        1.99028E-04  1.56487E-06   .00000E+00
    19   1.55973E+00  2.13456E-03   .00000E+00       1.46463E-03
8.08647E-04        3.13009E-04  3.12634E-06   .00000E+00
    20   1.55524E+00  2.60384E-03   .00000E+00       5.46105E-04
3.01514E-04        1.17046E-04  1.42197E-06   .00000E+00
    21   1.55192E+00  3.12774E-03   .00000E+00       1.24415E-03
6.86918E-04        2.67229E-04  3.89139E-06   .00000E+00
    22   1.54000E+00  4.45409E-03   .00000E+00       5.77219E-03
3.18692E-03        1.24939E-03  2.57098E-05   .00000E+00
    23   1.52869E+00  5.81622E-03   .00000E+00       5.49483E-03
3.03379E-03        1.19816E-03  3.19591E-05   .00000E+00
    24   1.51588E+00  7.36893E-03   .00000E+00       5.03449E-03
2.77963E-03        1.10706E-03  3.70988E-05   .00000E+00
    25   1.49532E+00  9.79734E-03   .00000E+00       3.68169E-03
2.03272E-03        8.20715E-04  3.60708E-05   .00000E+00
    26   1.43463E+00  1.63396E-02   .00000E+00       1.83704E-03
1.01426E-03        4.26832E-04  3.00166E-05   .00000E+00

 THERMAL 1.51986E+00  6.75150E-03   .00000E+00       2.50751E-02
1.38444E-02        5.49944E-03  1.69295E-04   .00000E+00

  TOTAL  1.45892E+00  3.02088E-03   .00000E+00       1.01206E-01
5.58774E-02        2.31234E-02  3.05731E-04   .00000E+00


0REGION 62     MATERIAL COOLANT     VOLUME  2.910080E+01

                      CROSS-SECTIONS                          FLUXES
                        REACTIONS
 GROUP   DIFFUSION   ABSORPTION   NU*FISSION           RIF           RAF
        TRANSPORT    ABSORPTION  NU*FISSION

    1    2.70642E+00  2.17943E-03   .00000E+00       5.33276E-02
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1.83251E-03       6.56804E-03  1.16224E-04   .00000E+00
    2    2.01031E+00  1.01023E-05   .00000E+00        1.93829E-01
6.66060E-03       3.21392E-02  1.95812E-06   .00000E+00
    3    1.31913E+00  3.55934E-06   .00000E+00        1.67581E-01
5.75863E-03       4.23462E-02  5.96477E-07   .00000E+00
    4    7.92566E-01  3.59323E-06   .00000E+00        1.07487E-01
3.69360E-03       4.52062E-02  3.86224E-07   .00000E+00
    5    6.54618E-01  6.63753E-06   .00000E+00        7.18839E-02
2.47017E-03       3.66035E-02  4.77132E-07   .00000E+00
    6    5.56182E-01  1.94203E-05   .00000E+00        1.01661E-01
3.49341E-03       6.09280E-02  1.97429E-06   .00000E+00
    7    5.63543E-01  6.29493E-05   .00000E+00        8.71229E-02
2.99383E-03       5.15328E-02  5.48433E-06   .00000E+00
    8    5.77544E-01  1.71416E-04   .00000E+00        8.26381E-02
2.83972E-03       4.76951E-02  1.41655E-05   .00000E+00
    9    5.80367E-01  3.53087E-04   .00000E+00        4.01516E-02
1.37974E-03       2.30611E-02  1.41770E-05   .00000E+00
   10    5.80777E-01  5.82085E-04   .00000E+00        3.89108E-02
1.33710E-03       2.23326E-02  2.26494E-05   .00000E+00
   11    5.80114E-01  9.59742E-04   .00000E+00        3.75742E-02
1.29117E-03       2.15901E-02  3.60615E-05   .00000E+00
   12    5.78579E-01  1.47308E-03   .00000E+00        2.67476E-02
9.19137E-04       1.54100E-02  3.94013E-05   .00000E+00
   13    5.30274E-01  2.07713E-03   .00000E+00        2.46528E-02
8.47152E-04       1.54969E-02  5.12071E-05   .00000E+00
   14    4.84734E-01  2.83531E-03   .00000E+00        2.34964E-02
8.07414E-04       1.61576E-02  6.66195E-05   .00000E+00
   15    4.56194E-01  3.34008E-03   .00000E+00        2.77576E-03
9.53844E-05       2.02820E-03  9.27126E-06   .00000E+00
   16    4.50003E-01  3.46267E-03   .00000E+00        2.86067E-03
9.83021E-05       2.11900E-03  9.90555E-06   .00000E+00
   17    4.42700E-01  3.60538E-03   .00000E+00        3.58786E-03
1.23291E-04       2.70150E-03  1.29356E-05   .00000E+00
   18    4.19829E-01  4.05921E-03   .00000E+00        1.53311E-02
5.26828E-04       1.21725E-02  6.22321E-05   .00000E+00
   19    3.66366E-01  5.17002E-03   .00000E+00        2.45553E-02
8.43803E-04       2.23414E-02  1.26952E-04   .00000E+00
   20    3.18292E-01  6.29875E-03   .00000E+00        9.59189E-03
3.29609E-04       1.00452E-02  6.04170E-05   .00000E+00
   21    2.76329E-01  7.56452E-03   .00000E+00        2.19772E-02
7.55211E-04       2.65110E-02  1.66247E-04   .00000E+00
   22    2.07862E-01  1.07793E-02   .00000E+00        1.05401E-01
3.62192E-03       1.69024E-01  1.13614E-03   .00000E+00
   23    1.64184E-01  1.40641E-02   .00000E+00        1.04539E-01
3.59232E-03       2.12240E-01  1.47025E-03   .00000E+00
   24    1.30440E-01  1.78189E-02   .00000E+00        9.94712E-02
3.41816E-03       2.54194E-01  1.77247E-03   .00000E+00
   25    1.01321E-01  2.37240E-02   .00000E+00        7.58031E-02
2.60485E-03       2.49382E-01  1.79835E-03   .00000E+00
   26    7.46679E-02  3.97569E-02   .00000E+00        4.01784E-02
1.38066E-03       1.79365E-01  1.59737E-03   .00000E+00

 THERMAL 1.42913E-01  1.68804E-02   .00000E+00        4.81517E-01
1.65465E-02       1.12310E+00  8.12820E-03   .00000E+00

  TOTAL  3.29945E-01  5.49787E-03   .00000E+00        1.56314E+00
5.37146E-02       1.57919E+00  8.59393E-03   .00000E+00


0REGION 63     MATERIAL FUEL_18     VOLUME  3.997746E+00

                    CROSS-SECTIONS                            FLUXES
                      REACTIONS
 GROUP   DIFFUSION    ABSORPTION   NU*FISSION           RIF            RAF
        TRANSPORT    ABSORPTION   NU*FISSION
```

```
   1    2.32885E+00  1.78377E-02  5.28855E-02        1.14900E-02
2.87413E-03        1.64459E-03  2.04956E-04  6.07656E-04
   2    1.91875E+00  1.57905E-02  3.97823E-02        4.04875E-02
1.01276E-02        7.03367E-03  6.39319E-04  1.61069E-03
   3    1.18870E+00  8.57947E-03  1.48435E-02        3.27480E-02
8.19160E-03        9.18312E-03  2.80960E-04  4.86094E-04
   4    8.16399E-01  9.08466E-03  1.44654E-02        1.83851E-02
4.59887E-03        7.50658E-03  1.67022E-04  2.65948E-04
   5    8.11678E-01  1.23757E-02  1.74729E-02        1.08963E-02
2.72560E-03        4.47479E-03  1.34849E-04  1.90390E-04
   6    7.06212E-01  2.23040E-02  2.45211E-02        1.42866E-02
3.57368E-03        6.74333E-03  3.18649E-04  3.50324E-04
   7    6.33971E-01  4.98774E-02  5.54151E-02        1.17872E-02
2.94846E-03        6.19755E-03  5.87915E-04  6.53190E-04
   8    5.71164E-01  1.29767E-01  1.54832E-01        1.08801E-02
2.72155E-03        6.34964E-03  1.41187E-03  1.68457E-03
   9    4.83008E-01  2.42138E-01  2.40388E-01        5.16198E-03
1.29122E-03        3.56239E-03  1.24991E-03  1.24088E-03
  10    3.98853E-01  3.85549E-01  4.87316E-01        4.83995E-03
1.21067E-03        4.04488E-03  1.86604E-03  2.35858E-03
  11    3.78049E-01  4.94905E-01  5.51734E-01        4.56558E-03
1.14204E-03        4.02555E-03  2.25953E-03  2.51898E-03
  12    3.77521E-01  5.09236E-01  1.61346E-01        3.23809E-03
8.09978E-04        2.85908E-03  1.64895E-03  5.22451E-04
  13    6.68706E-01  1.23913E-01  2.00238E-01        3.21217E-03
8.03495E-04        1.60119E-03  3.98028E-04  6.43198E-04
  14    6.13009E-01  1.58972E-01  2.55781E-01        3.03756E-03
7.59818E-04        1.65172E-03  4.82887E-04  7.76950E-04
  15    3.30150E-01  6.22014E-01  1.20522E+00        3.28434E-04
8.21548E-05        3.31601E-04  2.04290E-04  3.95836E-04
  16    4.05488E-01  4.34493E-01  8.86229E-01        3.49205E-04
8.73504E-05        2.87065E-04  1.51727E-04  3.09475E-04
  17    4.60036E-01  3.35121E-01  7.02402E-01        4.46299E-04
1.11638E-04        3.23380E-04  1.49564E-04  3.13481E-04
  18    4.74286E-01  3.11396E-01  6.58316E-01        1.91408E-03
4.78790E-04        1.34524E-03  5.96037E-04  1.26007E-03
  19    3.62015E-01  5.26401E-01  1.08735E+00        2.93222E-03
7.33468E-04        2.69990E-03  1.54352E-03  3.18834E-03
  20    2.31685E-01  1.04238E+00  2.03659E+00        1.03424E-03
2.58706E-04        1.48800E-03  1.07807E-03  2.10632E-03
  21    2.29393E-01  1.05628E+00  2.06603E+00        2.36368E-03
5.91252E-04        3.43469E-03  2.49670E-03  4.88343E-03
  22    1.89976E-01  1.35500E+00  2.75817E+00        1.11713E-02
2.79441E-03        1.96013E-02  1.51372E-02  3.08124E-02
  23    1.45443E-01  1.89028E+00  3.87453E+00        1.01722E-02
2.54449E-03        2.33133E-02  1.92284E-02  3.94127E-02
  24    1.14148E-01  2.51666E+00  5.16504E+00        8.76003E-03
2.19124E-03        2.55808E-02  2.20460E-02  4.52459E-02
  25    8.57929E-02  3.47833E+00  7.14022E+00        5.65630E-03
1.41487E-03        2.19766E-02  1.96745E-02  4.03873E-02
  26    5.39375E-02  5.75904E+00  1.18196E+01        2.21254E-03
5.53448E-04        1.36735E-02  1.27421E-02  2.61515E-02

 THERMAL 1.32127E-01  2.12056E+00  4.33807E+00        4.43026E-02
1.10819E-02        1.11768E-01  9.39464E-02  1.92188E-01

  TOTAL  4.09647E-01  4.79855E-01  9.37128E-01        2.22357E-01
5.56205E-02        1.80933E-01  1.06699E-01  2.08377E-01


0REGION 64     MATERIAL CLAD        VOLUME  4.075275E+00

                       CROSS-SECTIONS                              FLUXES
                         REACTIONS
 GROUP   DIFFUSION   ABSORPTION   NU*FISSION          RIF          RAF
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          TRANSPORT    ABSORPTION   NU*FISSION

      1    3.16321E+00 -5.37083E-05    .00000E+00       8.98675E-03
  2.20519E-03        9.47007E-04 -4.82663E-07    .00000E+00
      2    2.49217E+00  3.74237E-04    .00000E+00       3.26417E-02
  8.00969E-03        4.36590E-03  1.22157E-05    .00000E+00
      3    1.67902E+00  5.04506E-04    .00000E+00       2.74641E-02
  6.73919E-03        5.45240E-03  1.38558E-05    .00000E+00
      4    1.12257E+00  5.15706E-04    .00000E+00       1.66917E-02
  4.09586E-03        4.95642E-03  8.60804E-06    .00000E+00
      5    9.21946E-01  6.62955E-04    .00000E+00       1.05713E-02
  2.59401E-03        3.82210E-03  7.00830E-06    .00000E+00
      6    9.25362E-01  1.34549E-03    .00000E+00       1.44295E-02
  3.54075E-03        5.19779E-03  1.94148E-05    .00000E+00
      7    8.79884E-01  6.78327E-03    .00000E+00       1.20819E-02
  2.96469E-03        4.57709E-03  8.19550E-05    .00000E+00
      8    1.34604E+00  1.24927E-02    .00000E+00       1.12679E-02
  2.76495E-03        2.79038E-03  1.40766E-04    .00000E+00
      9    1.59365E+00  1.42310E-04    .00000E+00       5.42177E-03
  1.33041E-03        1.13403E-03  7.71570E-07    .00000E+00
     10    1.57942E+00  2.24967E-04    .00000E+00       5.17491E-03
  1.26983E-03        1.09216E-03  1.16418E-06    .00000E+00
     11    1.57349E+00  3.85370E-04    .00000E+00       4.94452E-03
  1.21330E-03        1.04746E-03  1.90547E-06    .00000E+00
     12    1.57020E+00  6.03067E-04    .00000E+00       3.51401E-03
  8.62275E-04        7.45980E-04  2.11918E-06    .00000E+00
     13    1.57409E+00  8.53588E-04    .00000E+00       3.32812E-03
  8.16662E-04        7.04770E-04  2.84084E-06    .00000E+00
     14    1.56863E+00  1.16803E-03    .00000E+00       3.15950E-03
  7.75285E-04        6.71394E-04  3.69040E-06    .00000E+00
     15    1.56887E+00  1.37786E-03    .00000E+00       3.60213E-04
  8.83899E-05        7.65336E-05  4.96323E-07    .00000E+00
     16    1.56929E+00  1.42882E-03    .00000E+00       3.75363E-04
  9.21073E-05        7.97309E-05  5.36327E-07    .00000E+00
     17    1.56537E+00  1.48846E-03    .00000E+00       4.74449E-04
  1.16421E-04        1.01030E-04  7.06199E-07    .00000E+00
     18    1.56347E+00  1.67631E-03    .00000E+00       2.02854E-03
  4.97767E-04        4.32487E-04  3.40046E-06    .00000E+00
     19    1.55973E+00  2.13456E-03    .00000E+00       3.18704E-03
  7.82043E-04        6.81109E-04  6.80292E-06    .00000E+00
     20    1.55524E+00  2.60384E-03    .00000E+00       1.19648E-03
  2.93595E-04        2.56441E-04  3.11545E-06    .00000E+00
     21    1.55192E+00  3.12774E-03    .00000E+00       2.74147E-03
  6.72708E-04        5.88833E-04  8.57461E-06    .00000E+00
     22    1.54000E+00  4.45409E-03    .00000E+00       1.33910E-02
  3.28591E-03        2.89848E-03  5.96446E-05    .00000E+00
     23    1.52869E+00  5.81622E-03    .00000E+00       1.30031E-02
  3.19072E-03        2.83535E-03  7.56287E-05    .00000E+00
     24    1.51588E+00  7.36893E-03    .00000E+00       1.20632E-02
  2.96010E-03        2.65264E-03  8.88932E-05    .00000E+00
     25    1.49532E+00  9.79734E-03    .00000E+00       8.73841E-03
  2.14425E-03        1.94795E-03  8.56133E-05    .00000E+00
     26    1.43463E+00  1.63396E-02    .00000E+00       4.38841E-03
  1.07684E-03        1.01963E-03  7.17049E-05    .00000E+00

   THERMAL 1.51934E+00  6.81287E-03    .00000E+00       5.87091E-02
  1.44062E-02        1.28804E-02  3.99978E-04    .00000E+00

     TOTAL 1.44640E+00  3.16277E-03    .00000E+00       2.21625E-01
  5.43829E-02        5.10751E-02  7.00950E-04    .00000E+00

  0REGION 65     MATERIAL MODERATOR   VOLUME  1.925113E+01

                      CROSS-SECTIONS                          FLUXES
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		REACTIONS			
GROUP	DIFFUSION	ABSORPTION	NU*FISSION	RIF	RAF
	TRANSPORT	ABSORPTION	NU*FISSION		
1	2.74822E+00	2.21855E-03	.00000E+00	3.16832E-02	
1.64578E-03	3.84286E-03	7.02907E-05	.00000E+00		
2	2.00863E+00	9.66120E-06	.00000E+00	1.19145E-01	
6.18899E-03	1.97722E-02	1.15108E-06	.00000E+00		
3	1.31914E+00	3.55843E-06	.00000E+00	1.04499E-01	
5.42822E-03	2.64059E-02	3.71854E-07	.00000E+00		
4	7.92343E-01	3.59826E-06	.00000E+00	7.01321E-02	
3.64301E-03	2.95041E-02	2.52354E-07	.00000E+00		
5	6.53939E-01	6.66158E-06	.00000E+00	4.76398E-02	
2.47465E-03	2.42835E-02	3.17356E-07	.00000E+00		
6	5.56006E-01	1.95488E-05	.00000E+00	6.76717E-02	
3.51521E-03	4.05701E-02	1.32290E-06	.00000E+00		
7	5.63650E-01	6.32557E-05	.00000E+00	5.72867E-02	
2.97576E-03	3.38784E-02	3.62371E-06	.00000E+00		
8	5.77572E-01	1.72185E-04	.00000E+00	5.37329E-02	
2.79116E-03	3.10108E-02	9.25200E-06	.00000E+00		
9	5.80369E-01	3.53483E-04	.00000E+00	2.60509E-02	
1.35321E-03	1.49623E-02	9.20856E-06	.00000E+00		
10	5.80777E-01	5.82735E-04	.00000E+00	2.51366E-02	
1.30572E-03	1.44270E-02	1.46480E-05	.00000E+00		
11	5.80112E-01	9.60720E-04	.00000E+00	2.42017E-02	
1.25716E-03	1.39063E-02	2.32511E-05	.00000E+00		
12	5.78577E-01	1.47362E-03	.00000E+00	1.72140E-02	
8.94181E-04	9.91743E-03	2.53669E-05	.00000E+00		
13	5.30259E-01	2.07768E-03	.00000E+00	1.57466E-02	
8.17959E-04	9.89871E-03	3.27165E-05	.00000E+00		
14	4.84689E-01	2.83608E-03	.00000E+00	1.49769E-02	
7.77976E-04	1.03000E-02	4.24757E-05	.00000E+00		
15	4.56193E-01	3.34009E-03	.00000E+00	1.77054E-03	
9.19705E-05	1.29370E-03	5.91375E-06	.00000E+00		
16	4.50003E-01	3.46266E-03	.00000E+00	1.82214E-03	
9.46511E-05	1.34972E-03	6.30946E-06	.00000E+00		
17	4.42698E-01	3.60542E-03	.00000E+00	2.28470E-03	
1.18679E-04	1.72029E-03	8.23731E-06	.00000E+00		
18	4.19805E-01	4.05969E-03	.00000E+00	9.74430E-03	
5.06168E-04	7.73716E-03	3.95589E-05	.00000E+00		
19	3.66218E-01	5.17320E-03	.00000E+00	1.55678E-02	
8.08672E-04	1.41699E-02	8.05356E-05	.00000E+00		
20	3.18091E-01	6.30426E-03	.00000E+00	6.06802E-03	
3.15203E-04	6.35879E-03	3.82543E-05	.00000E+00		
21	2.74446E-01	7.62670E-03	.00000E+00	1.39775E-02	
7.26062E-04	1.69767E-02	1.06602E-04	.00000E+00		
22	2.07048E-01	1.08270E-02	.00000E+00	7.11390E-02	
3.69532E-03	1.14529E-01	7.70221E-04	.00000E+00		
23	1.64120E-01	1.40700E-02	.00000E+00	7.20306E-02	
3.74163E-03	1.46297E-01	1.01347E-03	.00000E+00		
24	1.30412E-01	1.78227E-02	.00000E+00	6.94558E-02	
3.60788E-03	1.77529E-01	1.23789E-03	.00000E+00		
25	1.01309E-01	2.37277E-02	.00000E+00	5.23801E-02	
2.72089E-03	1.72345E-01	1.24286E-03	.00000E+00		
26	7.46682E-02	3.97555E-02	.00000E+00	2.80586E-02	
1.45750E-03	1.25259E-01	1.11548E-03	.00000E+00		
THERMAL	1.41648E-01	1.70542E-02	.00000E+00	3.28677E-01	
1.70732E-02	7.73463E-01	5.60532E-03	.00000E+00		
TOTAL	3.18097E-01	5.78722E-03	.00000E+00	1.01942E+00	
5.29536E-02	1.06824E+00	5.89958E-03	.00000E+00		

0REGION 66 MATERIAL CLAD VOLUME 7.922059E+00

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                     CROSS-SECTIONS                                  FLUXES
                        REACTIONS
GROUP   DIFFUSION    ABSORPTION   NU*FISSION               RIF              RAF
       TRANSPORT    ABSORPTION   NU*FISSION

   1    3.16321E+00 -5.37083E-05   .00000E+00       2.30906E-02
2.91472E-03       2.43324E-03 -1.24016E-06   .00000E+00
   2    2.49217E+00  3.74237E-04   .00000E+00       7.92421E-02
1.00027E-02       1.05988E-02  2.96553E-05   .00000E+00
   3    1.67902E+00  5.04506E-04   .00000E+00       6.68542E-02
8.43900E-03       1.32725E-02  3.37284E-05   .00000E+00
   4    1.12257E+00  5.15706E-04   .00000E+00       4.02500E-02
5.08075E-03       1.19518E-02  2.07572E-05   .00000E+00
   5    9.21946E-01  6.62955E-04   .00000E+00       2.60944E-02
3.29389E-03       9.43454E-03  1.72994E-05   .00000E+00
   6    9.25362E-01  1.34549E-03   .00000E+00       3.61189E-02
4.55928E-03       1.30107E-02  4.85976E-05   .00000E+00
   7    8.79884E-01  6.78327E-03   .00000E+00       3.09324E-02
3.90459E-03       1.17184E-02  2.09822E-04   .00000E+00
   8    1.34604E+00  1.24927E-02   .00000E+00       2.95605E-02
3.73142E-03       7.32035E-03  3.69289E-04   .00000E+00
   9    1.59365E+00  1.42310E-04   .00000E+00       1.44192E-02
1.82013E-03       3.01595E-03  2.05198E-06   .00000E+00
  10    1.57942E+00  2.24967E-04   .00000E+00       1.40326E-02
1.77134E-03       2.96157E-03  3.15688E-06   .00000E+00
  11    1.57349E+00  3.85370E-04   .00000E+00       1.36158E-02
1.71872E-03       2.88442E-03  5.24713E-06   .00000E+00
  12    1.57020E+00  6.03067E-04   .00000E+00       9.73160E-03
1.22842E-03       2.06590E-03  5.86881E-06   .00000E+00
  13    1.57409E+00  8.53588E-04   .00000E+00       9.07659E-03
1.14574E-03       1.92208E-03  7.74767E-06   .00000E+00
  14    1.56863E+00  1.16803E-03   .00000E+00       8.68960E-03
1.09689E-03       1.84654E-03  1.01497E-05   .00000E+00
  15    1.56887E+00  1.37786E-03   .00000E+00       1.02372E-03
1.29224E-04       2.17508E-04  1.41055E-06   .00000E+00
  16    1.56929E+00  1.42882E-03   .00000E+00       1.05808E-03
1.33561E-04       2.24747E-04  1.51181E-06   .00000E+00
  17    1.56537E+00  1.48846E-03   .00000E+00       1.32931E-03
1.67799E-04       2.83068E-04  1.97863E-06   .00000E+00
  18    1.56347E+00  1.67631E-03   .00000E+00       5.69842E-03
7.19310E-04       1.21491E-03  9.55230E-06   .00000E+00
  19    1.55973E+00  2.13456E-03   .00000E+00       9.15916E-03
1.15616E-03       1.95742E-03  1.95508E-05   .00000E+00
  20    1.55524E+00  2.60384E-03   .00000E+00       3.57431E-03
4.51185E-04       7.66079E-04  9.30694E-06   .00000E+00
  21    1.55192E+00  3.12774E-03   .00000E+00       8.16764E-03
1.03100E-03       1.75431E-03  2.55463E-05   .00000E+00
  22    1.54000E+00  4.45409E-03   .00000E+00       3.81690E-02
4.81806E-03       8.26169E-03  1.70008E-04   .00000E+00
  23    1.52869E+00  5.81622E-03   .00000E+00       3.76400E-02
4.75129E-03       8.20748E-03  2.18922E-04   .00000E+00
  24    1.51588E+00  7.36893E-03   .00000E+00       3.58057E-02
4.51975E-03       7.87347E-03  2.63850E-04   .00000E+00
  25    1.49532E+00  9.79734E-03   .00000E+00       2.73878E-02
3.45716E-03       6.10525E-03  2.68328E-04   .00000E+00
  26    1.43463E+00  1.63396E-02   .00000E+00       1.44702E-02
1.82657E-03       3.36211E-03  2.36437E-04   .00000E+00

 THERMAL 1.51810E+00  6.95029E-03   .00000E+00       1.74374E-01
2.20112E-02       3.82878E-02  1.21195E-03   .00000E+00

  TOTAL  1.44851E+00  3.39809E-03   .00000E+00       5.85192E-01
7.38687E-02       1.34665E-01  1.98853E-03   .00000E+00
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0REGION 67     MATERIAL REFLECTOR   VOLUME  6.474813E+01

                       CROSS-SECTIONS                                    FLUXES
                         REACTIONS
 GROUP   DIFFUSION      ABSORPTION   NU*FISSION              RIF              RAF
       TRANSPORT       ABSORPTION   NU*FISSION

    1    2.30318E+00 -5.97456E-02   .00000E+00       8.21987E-02
1.26952E-03       1.18964E-02 -4.91101E-03   .00000E+00
    2    1.68415E+00 -4.37687E-03   .00000E+00       3.36764E-01
5.20114E-03       6.66536E-02 -1.47397E-03   .00000E+00
    3    1.03682E+00  2.97272E-04   .00000E+00       3.06083E-01
4.72728E-03       9.84039E-02  9.09898E-05   .00000E+00
    4    7.45735E-01  1.25812E-05   .00000E+00       2.31164E-01
3.57021E-03       1.03327E-01  2.90832E-06   .00000E+00
    5    5.77663E-01  1.27952E-05   .00000E+00       1.62570E-01
2.51081E-03       9.38092E-02  2.08012E-06   .00000E+00
    6    5.00725E-01  1.30026E-05   .00000E+00       2.32307E-01
3.58786E-03       1.54647E-01  3.02060E-06   .00000E+00
    7    4.87326E-01  1.35587E-05   .00000E+00       1.89874E-01
2.93250E-03       1.29875E-01  2.57445E-06   .00000E+00
    8    4.86672E-01  3.31188E-05   .00000E+00       1.74546E-01
2.69576E-03       1.19550E-01  5.78074E-06   .00000E+00
    9    4.87141E-01  2.21321E-05   .00000E+00       8.44405E-02
1.30414E-03       5.77797E-02  1.86885E-06   .00000E+00
   10    4.87128E-01  3.45602E-05   .00000E+00       8.11873E-02
1.25389E-03       5.55550E-02  2.80585E-06   .00000E+00
   11    4.87083E-01  5.67886E-05   .00000E+00       7.80552E-02
1.20552E-03       5.34168E-02  4.43265E-06   .00000E+00
   12    4.86989E-01  8.70353E-05   .00000E+00       5.55659E-02
8.58186E-04       3.80336E-02  4.83620E-06   .00000E+00
   13    4.79214E-01  1.22663E-04   .00000E+00       4.97296E-02
7.68047E-04       3.45911E-02  6.09998E-06   .00000E+00
   14    4.79268E-01  1.67422E-04   .00000E+00       4.73905E-02
7.31920E-04       3.29603E-02  7.93420E-06   .00000E+00
   15    4.79884E-01  1.97090E-04   .00000E+00       5.65893E-03
8.73992E-05       3.93077E-03  1.11532E-06   .00000E+00
   16    4.79032E-01  2.04307E-04   .00000E+00       5.79583E-03
8.95135E-05       4.03301E-03  1.18413E-06   .00000E+00
   17    4.78520E-01  2.12804E-04   .00000E+00       7.26337E-03
1.12179E-04       5.05961E-03  1.54568E-06   .00000E+00
   18    4.79656E-01  2.39516E-04   .00000E+00       3.09191E-02
4.77529E-04       2.14870E-02  7.40563E-06   .00000E+00
   19    4.78165E-01  3.05050E-04   .00000E+00       4.96118E-02
7.66227E-04       3.45849E-02  1.51341E-05   .00000E+00
   20    4.77521E-01  3.71704E-04   .00000E+00       1.95925E-02
3.02596E-04       1.36766E-02  7.28261E-06   .00000E+00
   21    4.76578E-01  4.49655E-04   .00000E+00       4.58453E-02
7.08056E-04       3.20656E-02  2.06146E-05   .00000E+00
   22    4.68426E-01  6.38537E-04   .00000E+00       2.50074E-01
3.86226E-03       1.77953E-01  1.59681E-04   .00000E+00
   23    4.60595E-01  8.30284E-04   .00000E+00       2.56607E-01
3.96316E-03       1.85707E-01  2.13057E-04   .00000E+00
   24    4.50824E-01  1.05166E-03   .00000E+00       2.53535E-01
3.91572E-03       1.87461E-01  2.66633E-04   .00000E+00
   25    3.99521E-01  1.39907E-03   .00000E+00       1.85031E-01
2.85770E-03       1.54377E-01  2.58871E-04   .00000E+00
   26    8.36762E-01  2.34584E-03   .00000E+00       1.00671E-01
1.55482E-03       4.01036E-02  2.36159E-04   .00000E+00

 THERMAL 4.68551E-01  1.01418E-03   .00000E+00       1.16097E+00
1.79305E-02       8.25929E-01  1.17743E-03   .00000E+00

  TOTAL  5.79555E-01 -1.52325E-03   .00000E+00       3.32248E+00
5.13139E-02       1.91094E+00 -5.06097E-03   .00000E+00
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0REGION 68     MATERIAL BERYL       VOLUME  6.983765E+01

                     CROSS-SECTIONS                              FLUXES
                       REACTIONS
 GROUP  DIFFUSION    ABSORPTION   NU*FISSION          RIF          RAF
       TRANSPORT    ABSORPTION   NU*FISSION

    1   2.30318E+00 -5.97456E-02   .00000E+00      6.37999E-02
9.13546E-04       9.23359E-03 -3.81176E-03   .00000E+00
    2   1.68415E+00 -4.37687E-03   .00000E+00      2.84310E-01
4.07102E-03       5.62717E-02 -1.24439E-03   .00000E+00
    3   1.03682E+00  2.97272E-04   .00000E+00      2.66552E-01
3.81674E-03       8.56951E-02  7.92386E-05   .00000E+00
    4   7.45735E-01  1.25812E-05   .00000E+00      2.25429E-01
3.22790E-03       1.00764E-01  2.83617E-06   .00000E+00
    5   5.77663E-01  1.27952E-05   .00000E+00      1.66914E-01
2.39002E-03       9.63155E-02  2.13569E-06   .00000E+00
    6   5.00725E-01  1.30026E-05   .00000E+00      2.45676E-01
3.51781E-03       1.63547E-01  3.19442E-06   .00000E+00
    7   4.87326E-01  1.35587E-05   .00000E+00      1.97665E-01
2.83034E-03       1.35204E-01  2.68009E-06   .00000E+00
    8   4.86672E-01  3.31188E-05   .00000E+00      1.78613E-01
2.55754E-03       1.22336E-01  5.91544E-06   .00000E+00
    9   4.87141E-01  2.21321E-05   .00000E+00      8.61646E-02
1.23378E-03       5.89594E-02  1.90701E-06   .00000E+00
   10   4.87128E-01  3.45602E-05   .00000E+00      8.26625E-02
1.18364E-03       5.65646E-02  2.85683E-06   .00000E+00
   11   4.87083E-01  5.67886E-05   .00000E+00      7.93990E-02
1.13691E-03       5.43363E-02  4.50896E-06   .00000E+00
   12   4.86989E-01  8.70353E-05   .00000E+00      5.65385E-02
8.09570E-04       3.86993E-02  4.92085E-06   .00000E+00
   13   4.79215E-01  1.22663E-04   .00000E+00      4.96833E-02
7.11412E-04       3.45589E-02  6.09430E-06   .00000E+00
   14   4.79268E-01  1.67422E-04   .00000E+00      4.72027E-02
6.75892E-04       3.28297E-02  7.90276E-06   .00000E+00
   15   4.79884E-01  1.97090E-04   .00000E+00      5.67710E-03
8.12900E-05       3.94339E-03  1.11890E-06   .00000E+00
   16   4.79032E-01  2.04307E-04   .00000E+00      5.79993E-03
8.30487E-05       4.03586E-03  1.18497E-06   .00000E+00
   17   4.78520E-01  2.12804E-04   .00000E+00      7.26569E-03
1.04037E-04       5.06123E-03  1.54617E-06   .00000E+00
   18   4.79656E-01  2.39516E-04   .00000E+00      3.09245E-02
4.42806E-04       2.14907E-02  7.40693E-06   .00000E+00
   19   4.78165E-01  3.05050E-04   .00000E+00      4.99332E-02
7.14990E-04       3.48089E-02  1.52321E-05   .00000E+00
   20   4.77521E-01  3.71704E-04   .00000E+00      1.99905E-02
2.86243E-04       1.39544E-02  7.43055E-06   .00000E+00
   21   4.76578E-01  4.49655E-04   .00000E+00      4.79392E-02
6.86438E-04       3.35302E-02  2.15561E-05   .00000E+00
   22   4.68425E-01  6.38537E-04   .00000E+00      2.79468E-01
4.00168E-03       1.98871E-01  1.78451E-04   .00000E+00
   23   4.60595E-01  8.30284E-04   .00000E+00      2.90780E-01
4.16365E-03       2.10437E-01  2.41430E-04   .00000E+00
   24   4.50824E-01  1.05166E-03   .00000E+00      2.91104E-01
4.16829E-03       2.15238E-01  3.06142E-04   .00000E+00
   25   3.99521E-01  1.39907E-03   .00000E+00      2.09214E-01
2.99572E-03       1.74554E-01  2.92704E-04   .00000E+00
   26   8.36762E-01  2.34584E-03   .00000E+00      1.12714E-01
1.61394E-03       4.49008E-02  2.64409E-04   .00000E+00

 THERMAL 4.68225E-01  1.02015E-03   .00000E+00      1.30114E+00
1.86310E-02       9.26295E-01  1.32736E-03   .00000E+00

  TOTAL  5.61845E-01 -1.06268E-03   .00000E+00      3.38142E+00
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4.84183E-02      2.00614E+00 -3.59335E-03   .00000E+00

0REGION 69     MATERIAL REFLECTOR   VOLUME  7.492686E+01

                       CROSS-SECTIONS                              FLUXES
                         REACTIONS
 GROUP   DIFFUSION    ABSORPTION    NU*FISSION           RIF            RAF
        TRANSPORT    ABSORPTION   NU*FISSION

    1    2.30318E+00 -5.97456E-02   .00000E+00       5.21078E-02
6.95449E-04       7.54142E-03 -3.11321E-03   .00000E+00
    2    1.68415E+00 -4.37687E-03   .00000E+00       2.45884E-01
3.28165E-03       4.86663E-02 -1.07620E-03   .00000E+00
    3    1.03682E+00  2.97272E-04   .00000E+00       2.34394E-01
3.12830E-03       7.53564E-02  6.96788E-05   .00000E+00
    4    7.45735E-01  1.25812E-05   .00000E+00       2.12978E-01
2.84248E-03       9.51984E-02  2.67952E-06   .00000E+00
    5    5.77663E-01  1.27952E-05   .00000E+00       1.64315E-01
2.19301E-03       9.48162E-02  2.10245E-06   .00000E+00
    6    5.00725E-01  1.30026E-05   .00000E+00       2.49854E-01
3.33464E-03       1.66328E-01  3.24875E-06   .00000E+00
    7    4.87326E-01  1.35587E-05   .00000E+00       2.00963E-01
2.68213E-03       1.37460E-01  2.72481E-06   .00000E+00
    8    4.86672E-01  3.31188E-05   .00000E+00       1.79924E-01
2.40133E-03       1.23234E-01  5.95887E-06   .00000E+00
    9    4.87141E-01  2.21321E-05   .00000E+00       8.65815E-02
1.15555E-03       5.92447E-02  1.91623E-06   .00000E+00
   10    4.87128E-01  3.45602E-05   .00000E+00       8.29207E-02
1.10669E-03       5.67412E-02  2.86575E-06   .00000E+00
   11    4.87083E-01  5.67886E-05   .00000E+00       7.95783E-02
1.06208E-03       5.44590E-02  4.51914E-06   .00000E+00
   12    4.86989E-01  8.70353E-05   .00000E+00       5.66659E-02
7.56283E-04       3.87866E-02  4.93194E-06   .00000E+00
   13    4.79214E-01  1.22663E-04   .00000E+00       4.93520E-02
6.58669E-04       3.43284E-02  6.05366E-06   .00000E+00
   14    4.79268E-01  1.67422E-04   .00000E+00       4.66844E-02
6.23067E-04       3.24693E-02  7.81599E-06   .00000E+00
   15    4.79884E-01  1.97090E-04   .00000E+00       5.62308E-03
7.50476E-05       3.90586E-03  1.10825E-06   .00000E+00
   16    4.79032E-01  2.04307E-04   .00000E+00       5.74070E-03
7.66173E-05       3.99465E-03  1.17287E-06   .00000E+00
   17    4.78520E-01  2.12804E-04   .00000E+00       7.19088E-03
9.59719E-05       5.00911E-03  1.53025E-06   .00000E+00
   18    4.79656E-01  2.39516E-04   .00000E+00       3.06080E-02
4.08504E-04       2.12708E-02  7.33111E-06   .00000E+00
   19    4.78165E-01  3.05050E-04   .00000E+00       4.95974E-02
6.61944E-04       3.45748E-02  1.51297E-05   .00000E+00
   20    4.77521E-01  3.71704E-04   .00000E+00       2.00234E-02
2.67240E-04       1.39774E-02  7.44279E-06   .00000E+00
   21    4.76578E-01  4.49655E-04   .00000E+00       4.92550E-02
6.57375E-04       3.44505E-02  2.21477E-05   .00000E+00
   22    4.68426E-01  6.38537E-04   .00000E+00       3.04766E-01
4.06752E-03       2.16873E-01  1.94605E-04   .00000E+00
   23    4.60595E-01  8.30284E-04   .00000E+00       3.21598E-01
4.29216E-03       2.32741E-01  2.67018E-04   .00000E+00
   24    4.50824E-01  1.05166E-03   .00000E+00       3.24783E-01
4.33467E-03       2.40140E-01  3.41562E-04   .00000E+00
   25    3.99521E-01  1.39907E-03   .00000E+00       2.31996E-01
3.09629E-03       1.93561E-01  3.24578E-04   .00000E+00
   26    8.36762E-01  2.34584E-03   .00000E+00       1.24216E-01
1.65783E-03       4.94827E-02  2.91391E-04   .00000E+00

 THERMAL 4.68017E-01  1.02639E-03   .00000E+00       1.42624E+00
1.90350E-02       1.01580E+00  1.46387E-03   .00000E+00
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  TOTAL  5.49115E-01 -7.60739E-04   .00000E+00      3.41760E+00
4.56125E-02       2.07461E+00 -2.59990E-03   .00000E+00


0REGION 70     MATERIAL BERYL       VOLUME  7.607612E+01

                    CROSS-SECTIONS                              FLUXES
                      REACTIONS
 GROUP   DIFFUSION    ABSORPTION   NU*FISSION          RIF          RAF
        TRANSPORT    ABSORPTION   NU*FISSION

    1    2.30318E+00 -5.97456E-02   .00000E+00      4.15051E-02
5.45574E-04       6.00692E-03 -2.47975E-03   .00000E+00
    2    1.68415E+00 -4.37687E-03   .00000E+00      2.04586E-01
2.68923E-03       4.04924E-02 -8.95446E-04   .00000E+00
    3    1.03682E+00  2.97272E-04   .00000E+00      1.97169E-01
2.59174E-03       6.33888E-02  5.86129E-05   .00000E+00
    4    7.45735E-01  1.25812E-05   .00000E+00      1.88570E-01
2.47871E-03       8.42883E-02  2.37244E-06   .00000E+00
    5    5.77663E-01  1.27952E-05   .00000E+00      1.50259E-01
1.97512E-03       8.67053E-02  1.92259E-06   .00000E+00
    6    5.00725E-01  1.30026E-05   .00000E+00      2.35576E-01
3.09658E-03       1.56823E-01  3.06310E-06   .00000E+00
    7    4.87326E-01  1.35587E-05   .00000E+00      1.90696E-01
2.50665E-03       1.30437E-01  2.58560E-06   .00000E+00
    8    4.86672E-01  3.31188E-05   .00000E+00      1.70116E-01
2.23613E-03       1.16517E-01  5.63404E-06   .00000E+00
    9    4.87141E-01  2.21321E-05   .00000E+00      8.17480E-02
1.07455E-03       5.59373E-02  1.80926E-06   .00000E+00
   10    4.87128E-01  3.45602E-05   .00000E+00      7.82138E-02
1.02810E-03       5.35203E-02  2.70308E-06   .00000E+00
   11    4.87083E-01  5.67886E-05   .00000E+00      7.50266E-02
9.86205E-04       5.13441E-02  4.26066E-06   .00000E+00
   12    4.86989E-01  8.70353E-05   .00000E+00      5.34293E-02
7.02313E-04       3.65711E-02  4.65023E-06   .00000E+00
   13    4.79215E-01  1.22663E-04   .00000E+00      4.63171E-02
6.08826E-04       3.22174E-02  5.68139E-06   .00000E+00
   14    4.79268E-01  1.67422E-04   .00000E+00      4.36667E-02
5.73987E-04       3.03704E-02  7.31076E-06   .00000E+00
   15    4.79884E-01  1.97090E-04   .00000E+00      5.25972E-03
6.91377E-05       3.65347E-03  1.03664E-06   .00000E+00
   16    4.79032E-01  2.04307E-04   .00000E+00      5.36793E-03
7.05600E-05       3.73526E-03  1.09671E-06   .00000E+00
   17    4.78520E-01  2.12804E-04   .00000E+00      6.72326E-03
8.83754E-05       4.68337E-03  1.43074E-06   .00000E+00
   18    4.79656E-01  2.39516E-04   .00000E+00      2.86149E-02
3.76136E-04       1.98857E-02  6.85375E-06   .00000E+00
   19    4.78165E-01  3.05050E-04   .00000E+00      4.64676E-02
6.10804E-04       3.23930E-02  1.41749E-05   .00000E+00
   20    4.77521E-01  3.71704E-04   .00000E+00      1.88737E-02
2.48089E-04       1.31748E-02  7.01541E-06   .00000E+00
   21    4.76578E-01  4.49655E-04   .00000E+00      4.75770E-02
6.25386E-04       3.32768E-02  2.13932E-05   .00000E+00
   22    4.68425E-01  6.38537E-04   .00000E+00      3.10499E-01
4.08142E-03       2.20952E-01  1.98265E-04   .00000E+00
   23    4.60595E-01  8.30284E-04   .00000E+00      3.31935E-01
4.36319E-03       2.40222E-01  2.75600E-04   .00000E+00
   24    4.50824E-01  1.05166E-03   .00000E+00      3.37230E-01
4.43280E-03       2.49343E-01  3.54651E-04   .00000E+00
   25    3.99521E-01  1.39907E-03   .00000E+00      2.40492E-01
3.16121E-03       2.00651E-01  3.36465E-04   .00000E+00
   26    8.36762E-01  2.34584E-03   .00000E+00      1.28406E-01
1.68786E-03       5.11519E-02  3.01220E-04   .00000E+00

 THERMAL 4.67899E-01  1.03237E-03   .00000E+00      1.46148E+00
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1.92108E-02       1.04116E+00  1.50879E-03   .00000E+00

  TOTAL  5.39271E-01 -5.37748E-04   .00000E+00       3.26433E+00
4.29087E-02       2.01774E+00 -1.75538E-03   .00000E+00


0REGION 71     MATERIAL REFLECTOR   VOLUME  4.953617E+01

                         CROSS-SECTIONS                              FLUXES
                           REACTIONS
 GROUP   DIFFUSION      ABSORPTION   NU*FISSION             RIF            RAF
        TRANSPORT     ABSORPTION   NU*FISSION

    1    2.30318E+00 -5.97456E-02   .00000E+00       2.22347E-02
4.48858E-04       3.21797E-03 -1.32843E-03   .00000E+00
    2    1.68415E+00 -4.37687E-03   .00000E+00       1.13321E-01
2.28763E-03       2.24288E-02 -4.95989E-04   .00000E+00
    3    1.03682E+00  2.97272E-04   .00000E+00       1.10004E-01
2.22067E-03       3.53655E-02  3.27010E-05   .00000E+00
    4    7.45735E-01  1.25812E-05   .00000E+00       1.08909E-01
2.19858E-03       4.86809E-02  1.37021E-06   .00000E+00
    5    5.77663E-01  1.27952E-05   .00000E+00       8.86984E-02
1.79058E-03       5.11823E-02  1.13491E-06   .00000E+00
    6    5.00725E-01  1.30026E-05   .00000E+00       1.42487E-01
2.87641E-03       9.48535E-02  1.85270E-06   .00000E+00
    7    4.87326E-01  1.35587E-05   .00000E+00       1.16254E-01
2.34685E-03       7.95183E-02  1.57626E-06   .00000E+00
    8    4.86672E-01  3.31188E-05   .00000E+00       1.03702E-01
2.09346E-03       7.10279E-02  3.43448E-06   .00000E+00
    9    4.87141E-01  2.21321E-05   .00000E+00       4.98179E-02
1.00569E-03       3.40887E-02  1.10258E-06   .00000E+00
   10    4.87128E-01  3.45602E-05   .00000E+00       4.76524E-02
9.61972E-04       3.26077E-02  1.64688E-06   .00000E+00
   11    4.87083E-01  5.67886E-05   .00000E+00       4.57112E-02
9.22785E-04       3.12823E-02  2.59588E-06   .00000E+00
   12    4.86989E-01  8.70353E-05   .00000E+00       3.25606E-02
6.57310E-04       2.22870E-02  2.83392E-06   .00000E+00
   13    4.79214E-01  1.22663E-04   .00000E+00       2.81662E-02
5.68598E-04       1.95919E-02  3.45494E-06   .00000E+00
   14    4.79268E-01  1.67422E-04   .00000E+00       2.65039E-02
5.35041E-04       1.84336E-02  4.43732E-06   .00000E+00
   15    4.79884E-01  1.97090E-04   .00000E+00       3.19173E-03
6.44324E-05       2.21702E-03  6.29060E-07   .00000E+00
   16    4.79032E-01  2.04307E-04   .00000E+00       3.25687E-03
6.57474E-05       2.26629E-03  6.65403E-07   .00000E+00
   17    4.78520E-01  2.12804E-04   .00000E+00       4.07889E-03
8.23416E-05       2.84132E-03  8.68004E-07   .00000E+00
   18    4.79656E-01  2.39516E-04   .00000E+00       1.73590E-02
3.50431E-04       1.20635E-02  4.15777E-06   .00000E+00
   19    4.78165E-01  3.05050E-04   .00000E+00       2.82276E-02
5.69837E-04       1.96777E-02  8.61082E-06   .00000E+00
   20    4.77521E-01  3.71704E-04   .00000E+00       1.15144E-02
2.32445E-04       8.03763E-03  4.27995E-06   .00000E+00
   21    4.76578E-01  4.49655E-04   .00000E+00       2.96153E-02
5.97851E-04       2.07138E-02  1.33166E-05   .00000E+00
   22    4.68426E-01  6.38537E-04   .00000E+00       2.01414E-01
4.06600E-03       1.43327E-01  1.28610E-04   .00000E+00
   23    4.60595E-01  8.30284E-04   .00000E+00       2.17510E-01
4.39093E-03       1.57412E-01  1.80595E-04   .00000E+00
   24    4.50824E-01  1.05166E-03   .00000E+00       2.21818E-01
4.47789E-03       1.64009E-01  2.33277E-04   .00000E+00
   25    3.99521E-01  1.39907E-03   .00000E+00       1.58289E-01
3.19542E-03       1.32065E-01  2.21457E-04   .00000E+00
   26    8.36762E-01  2.34584E-03   .00000E+00       8.44896E-02
1.70562E-03       3.36574E-02  1.98199E-04   .00000E+00
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 THERMAL 4.67853E-01  1.03722E-03   .00000E+00      9.52877E-01
1.92360E-02      6.78900E-01  9.88345E-04   .00000E+00

  TOTAL  5.32335E-01 -3.82593E-04   .00000E+00      2.01678E+00
4.07134E-02      1.26285E+00 -7.71607E-04   .00000E+00


0REGION 72     MATERIAL BERYL       VOLUME  7.899689E+01

                       CROSS-SECTIONS                               FLUXES
                         REACTIONS
 GROUP   DIFFUSION     ABSORPTION   NU*FISSION            RIF             RAF
        TRANSPORT     ABSORPTION   NU*FISSION

    1    2.30318E+00 -5.97456E-02   .00000E+00      2.96366E-02
3.75162E-04      4.28923E-03 -1.77066E-03   .00000E+00
    2    1.68415E+00 -4.37687E-03   .00000E+00      1.54897E-01
1.96080E-03      3.06578E-02 -6.77964E-04   .00000E+00
    3    1.03682E+00  2.97272E-04   .00000E+00      1.50978E-01
1.91118E-03      4.85385E-02  4.48814E-05   .00000E+00
    4    7.45735E-01  1.25812E-05   .00000E+00      1.53229E-01
1.93969E-03      6.84913E-02  1.92781E-06   .00000E+00
    5    5.77663E-01  1.27952E-05   .00000E+00      1.26878E-01
1.60611E-03      7.32133E-02  1.62342E-06   .00000E+00
    6    5.00725E-01  1.30026E-05   .00000E+00      2.08094E-01
2.63421E-03      1.38529E-01  2.70577E-06   .00000E+00
    7    4.87326E-01  1.35587E-05   .00000E+00      1.71306E-01
2.16852E-03      1.17174E-01  2.32270E-06   .00000E+00
    8    4.86672E-01  3.31188E-05   .00000E+00      1.53168E-01
1.93891E-03      1.04908E-01  5.07273E-06   .00000E+00
    9    4.87141E-01  2.21321E-05   .00000E+00      7.36289E-02
9.32048E-04      5.03817E-02  1.62956E-06   .00000E+00
   10    4.87128E-01  3.45602E-05   .00000E+00      7.04637E-02
8.91981E-04      4.82171E-02  2.43524E-06   .00000E+00
   11    4.87083E-01  5.67886E-05   .00000E+00      6.76323E-02
8.56138E-04      4.62838E-02  3.84074E-06   .00000E+00
   12    4.86989E-01  8.70353E-05   .00000E+00      4.82017E-02
6.10172E-04      3.29930E-02  4.19525E-06   .00000E+00
   13    4.79215E-01  1.22663E-04   .00000E+00      4.16641E-02
5.27414E-04      2.89808E-02  5.11064E-06   .00000E+00
   14    4.79268E-01  1.67422E-04   .00000E+00      3.91714E-02
4.95860E-04      2.72439E-02  6.55815E-06   .00000E+00
   15    4.79884E-01  1.97090E-04   .00000E+00      4.71677E-03
5.97084E-05      3.27633E-03  9.29630E-07   .00000E+00
   16    4.79032E-01  2.04307E-04   .00000E+00      4.81272E-03
6.09229E-05      3.34892E-03  9.83273E-07   .00000E+00
   17    4.78520E-01  2.12804E-04   .00000E+00      6.02738E-03
7.62990E-05      4.19863E-03  1.28265E-06   .00000E+00
   18    4.79656E-01  2.39516E-04   .00000E+00      2.56537E-02
3.24744E-04      1.78279E-02  6.14449E-06   .00000E+00
   19    4.78165E-01  3.05050E-04   .00000E+00      4.17717E-02
5.28777E-04      2.91195E-02  1.27425E-05   .00000E+00
   20    4.77521E-01  3.71704E-04   .00000E+00      1.71075E-02
2.16559E-04      1.19419E-02  6.35892E-06   .00000E+00
   21    4.76578E-01  4.49655E-04   .00000E+00      4.49045E-02
5.68433E-04      3.14076E-02  2.01915E-05   .00000E+00
   22    4.68425E-01  6.38537E-04   .00000E+00      3.17771E-01
4.02258E-03      2.26127E-01  2.02909E-04   .00000E+00
   23    4.60595E-01  8.30284E-04   .00000E+00      3.46570E-01
4.38713E-03      2.50813E-01  2.87751E-04   .00000E+00
   24    4.50824E-01  1.05166E-03   .00000E+00      3.54442E-01
4.48679E-03      2.62070E-01  3.72753E-04   .00000E+00
   25    3.99521E-01  1.39907E-03   .00000E+00      2.53575E-01
3.20994E-03      2.11566E-01  3.54769E-04   .00000E+00
   26    8.36762E-01  2.34584E-03   .00000E+00      1.35462E-01
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1.71477E-03       5.39627E-02  3.17772E-04   .00000E+00

 THERMAL 4.67841E-01  1.04210E-03   .00000E+00       1.51160E+00
1.91350E-02       1.07701E+00  1.57525E-03   .00000E+00

  TOTAL  5.26559E-01 -2.56999E-04   .00000E+00       3.04176E+00
3.85048E-02       1.92556E+00 -7.81730E-04   .00000E+00


0REGION 73     MATERIAL GAP         VOLUME  1.570796E+01

                          CROSS-SECTIONS                              FLUXES
                            REACTIONS
 GROUP  DIFFUSION     ABSORPTION   NU*FISSION            RIF            RAF
       TRANSPORT     ABSORPTION   NU*FISSION

    1   1.41755E+05  5.13322E-07   .00000E+00       7.44500E-03
4.73964E-04       1.75067E-08  3.82168E-09   .00000E+00
    2   1.02864E+05  2.81158E-07   .00000E+00       3.60832E-02
2.29713E-03       1.16928E-07  1.01451E-08   .00000E+00
    3   7.59244E+04  6.05228E-08   .00000E+00       3.44133E-02
2.19082E-03       1.51086E-07  2.08279E-09   .00000E+00
    4   4.70167E+04  1.33476E-08   .00000E+00       3.30903E-02
2.10660E-03       2.34600E-07  4.41677E-10   .00000E+00
    5   3.93658E+04  2.66520E-09   .00000E+00       2.68290E-02
1.70799E-03       2.27176E-07  7.15047E-11   .00000E+00
    6   2.80739E+04  3.59990E-09   .00000E+00       4.30012E-02
2.73755E-03       5.10572E-07  1.54800E-10   .00000E+00
    7   2.25878E+04  9.38580E-09   .00000E+00       3.51948E-02
2.24057E-03       5.19377E-07  3.30332E-10   .00000E+00
    8   2.01120E+04  2.55719E-08   .00000E+00       3.15183E-02
2.00652E-03       5.22378E-07  8.05982E-10   .00000E+00
    9   1.89538E+04  5.23663E-08   .00000E+00       1.51620E-02
9.65241E-04       2.66647E-07  7.93976E-10   .00000E+00
   10   1.87168E+04  8.63328E-08   .00000E+00       1.45188E-02
9.24295E-04       2.58570E-07  1.25345E-09   .00000E+00
   11   1.85665E+04  1.42363E-07   .00000E+00       1.39393E-02
8.87406E-04       2.50259E-07  1.98444E-09   .00000E+00
   12   1.84522E+04  2.18329E-07   .00000E+00       9.93424E-03
6.32434E-04       1.79459E-07  2.16894E-09   .00000E+00
   13   1.83667E+04  3.07848E-07   .00000E+00       8.60753E-03
5.47973E-04       1.56217E-07  2.64981E-09   .00000E+00
   14   1.82499E+04  4.20257E-07   .00000E+00       8.10737E-03
5.16131E-04       1.48081E-07  3.40718E-09   .00000E+00
   15   1.81808E+04  4.94670E-07   .00000E+00       9.76436E-04
6.21619E-05       1.79024E-08  4.83014E-10   .00000E+00
   16   1.81627E+04  5.12788E-07   .00000E+00       9.96530E-04
6.34411E-05       1.82889E-08  5.11008E-10   .00000E+00
   17   1.81308E+04  5.34075E-07   .00000E+00       1.24796E-03
7.94479E-05       2.29437E-08  6.66506E-10   .00000E+00
   18   1.80596E+04  6.01494E-07   .00000E+00       5.31129E-03
3.38127E-04       9.80328E-08  3.19471E-09   .00000E+00
   19   1.79037E+04  7.66370E-07   .00000E+00       8.63833E-03
5.49933E-04       1.60830E-07  6.62015E-09   .00000E+00
   20   1.77278E+04  9.34163E-07   .00000E+00       3.52352E-03
2.24315E-04       6.62522E-08  3.29155E-09   .00000E+00
   21   1.75125E+04  1.12938E-06   .00000E+00       9.11370E-03
5.80197E-04       1.73471E-07  1.02929E-08   .00000E+00
   22   1.70165E+04  1.60405E-06   .00000E+00       6.28549E-02
4.00147E-03       1.23125E-06  1.00822E-07   .00000E+00
   23   1.65119E+04  2.08502E-06   .00000E+00       6.81801E-02
4.34048E-03       1.37639E-06  1.42157E-07   .00000E+00
   24   1.59381E+04  2.64136E-06   .00000E+00       6.94476E-02
4.42117E-03       1.45244E-06  1.83436E-07   .00000E+00
   25   1.50606E+04  3.51301E-06   .00000E+00       4.97641E-02
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3.16808E-03       1.10142E-06  1.74822E-07   .00000E+00
   26    1.27683E+04  5.89029E-06   .00000E+00      2.64434E-02
1.68344E-03       6.90339E-07  1.55759E-07   .00000E+00

 THERMAL 1.58854E+04  2.60836E-06   .00000E+00      2.97966E-01
1.89691E-02       6.25239E-06  7.77201E-07   .00000E+00

  TOTAL  2.08773E+04  1.30084E-06   .00000E+00      6.24342E-01
3.97469E-02       9.96842E-06  8.12168E-07   .00000E+00


0REGION 74     MATERIAL AL         VOLUME  5.794665E+00

                     CROSS-SECTIONS                          FLUXES
                       REACTIONS
 GROUP  DIFFUSION   ABSORPTION  NU*FISSION          RIF           RAF
       TRANSPORT   ABSORPTION  NU*FISSION

    1    3.95965E+00  2.27382E-03   .00000E+00      2.67655E-03
4.61899E-04       2.25319E-04  6.08598E-06   .00000E+00
    2    3.10175E+00  5.71386E-05   .00000E+00      1.31444E-02
2.26837E-03       1.41258E-03  7.51056E-07   .00000E+00
    3    2.20964E+00  1.67482E-05   .00000E+00      1.26372E-02
2.18084E-03       1.90638E-03  2.11651E-07   .00000E+00
    4    1.62440E+00  5.00338E-05   .00000E+00      1.22106E-02
2.10722E-03       2.50566E-03  6.10945E-07   .00000E+00
    5    9.26281E-01  1.39174E-04   .00000E+00      9.92034E-03
1.71198E-03       3.56995E-03  1.38065E-06   .00000E+00
    6    3.33061E+00  2.80616E-04   .00000E+00      1.59005E-02
2.74399E-03       1.59135E-03  4.46193E-06   .00000E+00
    7    3.02145E+00  5.19427E-04   .00000E+00      1.30075E-02
2.24474E-03       1.43502E-03  6.75646E-06   .00000E+00
    8    4.08502E+00  2.21816E-04   .00000E+00      1.16449E-02
2.00959E-03       9.50210E-04  2.58302E-06   .00000E+00
    9    4.09110E+00  2.38056E-04   .00000E+00      5.60093E-03
9.66567E-04       4.56351E-04  1.33334E-06   .00000E+00
   10    4.08233E+00  3.84755E-04   .00000E+00      5.36256E-03
9.25431E-04       4.37868E-04  2.06327E-06   .00000E+00
   11    4.06966E+00  6.34997E-04   .00000E+00      5.14789E-03
8.88384E-04       4.21648E-04  3.26889E-06   .00000E+00
   12    4.05320E+00  9.73709E-04   .00000E+00      3.66851E-03
6.33084E-04       3.01697E-04  3.57206E-06   .00000E+00
   13    4.03627E+00  1.37216E-03   .00000E+00      3.17717E-03
5.48293E-04       2.62385E-04  4.35959E-06   .00000E+00
   14    4.01077E+00  1.87212E-03   .00000E+00      2.99181E-03
5.16303E-04       2.48648E-04  5.60103E-06   .00000E+00
   15    4.00087E+00  2.20303E-03   .00000E+00      3.60356E-04
6.21875E-05       3.00231E-05  7.93877E-07   .00000E+00
   16    3.99705E+00  2.28364E-03   .00000E+00      3.67756E-04
6.34646E-05       3.06690E-05  8.39822E-07   .00000E+00
   17    3.98901E+00  2.37847E-03   .00000E+00      4.60503E-04
7.94702E-05       3.84810E-05  1.09529E-06   .00000E+00
   18    3.97288E+00  2.67940E-03   .00000E+00      1.95959E-03
3.38172E-04       1.64414E-04  5.25054E-06   .00000E+00
   19    3.93557E+00  3.41570E-03   .00000E+00      3.18700E-03
5.49989E-04       2.69931E-04  1.08858E-05   .00000E+00
   20    3.89960E+00  4.16532E-03   .00000E+00      1.29970E-03
2.24293E-04       1.11097E-04  5.41369E-06   .00000E+00
   21    3.85636E+00  5.03787E-03   .00000E+00      3.36090E-03
5.79999E-04       2.90507E-04  1.69318E-05   .00000E+00
   22    3.75376E+00  7.15826E-03   .00000E+00      2.31907E-02
4.00208E-03       2.05933E-03  1.66005E-04   .00000E+00
   23    3.65252E+00  9.30090E-03   .00000E+00      2.51546E-02
4.34100E-03       2.29564E-03  2.33961E-04   .00000E+00
   24    3.54298E+00  1.17702E-02   .00000E+00      2.56078E-02
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4.41920E-03        2.40925E-03  3.01409E-04   .00000E+00
   25    3.37501E+00  1.56477E-02   .00000E+00      1.83418E-02
3.16529E-03        1.81153E-03  2.87007E-04   .00000E+00
   26    2.94383E+00  2.62317E-02   .00000E+00      9.73490E-03
1.67998E-03        1.10229E-03  2.55363E-04   .00000E+00

 THERMAL 3.53887E+00  1.16218E-02   .00000E+00      1.09877E-01
1.89618E-02        1.03496E-02  1.27698E-03   .00000E+00

  TOTAL  2.91233E+00  5.77097E-03   .00000E+00      2.30117E-01
3.97118E-02        2.63382E-02  1.32800E-03   .00000E+00


0REGION 75     MATERIAL REFLECTOR   VOLUME  5.696846E+02

                    CROSS-SECTIONS                            FLUXES
                      REACTIONS
 GROUP   DIFFUSION    ABSORPTION   NU*FISSION           RIF           RAF
        TRANSPORT    ABSORPTION   NU*FISSION

    1    2.30318E+00 -5.97456E-02   .00000E+00      1.15839E-01
2.03339E-04        1.67651E-02 -6.92086E-03   .00000E+00
    2    1.68415E+00 -4.37687E-03   .00000E+00      6.17334E-01
1.08364E-03        1.22185E-01 -2.70199E-03   .00000E+00
    3    1.03682E+00  2.97272E-04   .00000E+00      5.90866E-01
1.03718E-03        1.89960E-01  1.75648E-04   .00000E+00
    4    7.45735E-01  1.25812E-05   .00000E+00      6.15128E-01
1.07977E-03        2.74954E-01  7.73905E-06   .00000E+00
    5    5.77663E-01  1.27952E-05   .00000E+00      5.19753E-01
9.12353E-04        2.99917E-01  6.65034E-06   .00000E+00
    6    5.00725E-01  1.30026E-05   .00000E+00      8.87425E-01
1.55775E-03        5.90760E-01  1.15388E-05   .00000E+00
    7    4.87326E-01  1.35587E-05   .00000E+00      7.57291E-01
1.32932E-03        5.17990E-01  1.02679E-05   .00000E+00
    8    4.86672E-01  3.31188E-05   .00000E+00      6.95954E-01
1.22165E-03        4.76675E-01  2.30492E-05   .00000E+00
    9    4.87141E-01  2.21321E-05   .00000E+00      3.38740E-01
5.94611E-04        2.31788E-01  7.49705E-06   .00000E+00
   10    4.87128E-01  3.45602E-05   .00000E+00      3.27855E-01
5.75503E-04        2.24346E-01  1.13307E-05   .00000E+00
   11    4.87083E-01  5.67886E-05   .00000E+00      3.17969E-01
5.58149E-04        2.17600E-01  1.80570E-05   .00000E+00
   12    4.86989E-01  8.70353E-05   .00000E+00      2.28263E-01
4.00683E-04        1.56241E-01  1.98669E-05   .00000E+00
   13    4.79214E-01  1.22663E-04   .00000E+00      1.99200E-01
3.49668E-04        1.38560E-01  2.44345E-05   .00000E+00
   14    4.79268E-01  1.67422E-04   .00000E+00      1.88812E-01
3.31433E-04        1.31320E-01  3.16113E-05   .00000E+00
   15    4.79884E-01  1.97090E-04   .00000E+00      2.28132E-02
4.00453E-05        1.58463E-02  4.49625E-06   .00000E+00
   16    4.79032E-01  2.04307E-04   .00000E+00      2.33021E-02
4.09035E-05        1.62147E-02  4.76079E-06   .00000E+00
   17    4.78520E-01  2.12804E-04   .00000E+00      2.92151E-02
5.12830E-05        2.03511E-02  6.21710E-06   .00000E+00
   18    4.79656E-01  2.39516E-04   .00000E+00      1.24933E-01
2.19302E-04        8.68214E-02  2.99236E-05   .00000E+00
   19    4.78165E-01  3.05050E-04   .00000E+00      2.05817E-01
3.61282E-04        1.43477E-01  6.27844E-05   .00000E+00
   20    4.77521E-01  3.71704E-04   .00000E+00      8.61112E-02
1.51156E-04        6.01099E-02  3.20079E-05   .00000E+00
   21    4.76578E-01  4.49655E-04   .00000E+00      2.49297E-01
4.37605E-04        1.74366E-01  1.12097E-04   .00000E+00
   22    4.68426E-01  6.38537E-04   .00000E+00      2.07116E+00
3.63563E-03        1.47385E+00  1.32252E-03   .00000E+00
   23    4.60595E-01  8.30284E-04   .00000E+00      2.35187E+00
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4.12836E-03        1.70205E+00  1.95272E-03    .00000E+00
   24    4.50824E-01  1.05166E-03    .00000E+00        2.41897E+00
4.24616E-03        1.78856E+00  2.54394E-03    .00000E+00
   25    3.99521E-01  1.39907E-03    .00000E+00        1.78597E+00
3.13502E-03        1.49009E+00  2.49870E-03    .00000E+00
   26    8.36762E-01  2.34584E-03    .00000E+00        9.65381E-01
1.69459E-03        3.84570E-01  2.26463E-03    .00000E+00

 THERMAL 4.68084E-01  1.06461E-03    .00000E+00        1.01346E+01
1.77898E-02        7.21707E+00  1.07894E-02    .00000E+00

  TOTAL  5.09661E-01  9.31940E-05    .00000E+00        1.67353E+01
2.93764E-02        1.09454E+01  1.55963E-03    .00000E+00


0REGION 76     MATERIAL COOLANT     VOLUME  6.016095E+02

                    CROSS-SECTIONS                            FLUXES
                      REACTIONS
 GROUP   DIFFUSION    ABSORPTION   NU*FISSION           RIF            RAF
        TRANSPORT    ABSORPTION   NU*FISSION

    1    2.70642E+00  2.17943E-03    .00000E+00        4.51372E-02
7.50274E-05        5.55927E-03  9.83733E-05    .00000E+00
    2    2.01031E+00  1.01023E-05    .00000E+00        1.89194E-01
3.14479E-04        3.13706E-02  1.91130E-06    .00000E+00
    3    1.31913E+00  3.55934E-06    .00000E+00        1.58355E-01
2.63218E-04        4.00149E-02  5.63638E-07    .00000E+00
    4    7.92566E-01  3.59323E-06    .00000E+00        1.27281E-01
2.11567E-04        5.35311E-02  4.57349E-07    .00000E+00
    5    6.54618E-01  6.63753E-06    .00000E+00        9.79889E-02
1.62878E-04        4.98962E-02  6.50405E-07    .00000E+00
    6    5.56182E-01  1.94203E-05    .00000E+00        1.65055E-01
2.74355E-04        9.89212E-02  3.20542E-06    .00000E+00
    7    5.63543E-01  6.29493E-05    .00000E+00        1.64164E-01
2.72874E-04        9.71020E-02  1.03340E-05    .00000E+00
    8    5.77544E-01  1.71416E-04    .00000E+00        1.76700E-01
2.93712E-04        1.01983E-01  3.02892E-05    .00000E+00
    9    5.80367E-01  3.53087E-04    .00000E+00        9.25671E-02
1.53866E-04        5.31658E-02  3.26842E-05    .00000E+00
   10    5.80777E-01  5.82085E-04    .00000E+00        9.55589E-02
1.58839E-04        5.48454E-02  5.56233E-05    .00000E+00
   11    5.80114E-01  9.59742E-04    .00000E+00        9.81341E-02
1.63119E-04        5.63878E-02  9.41833E-05    .00000E+00
   12    5.78579E-01  1.47308E-03    .00000E+00        7.31778E-02
1.21637E-04        4.21595E-02  1.07797E-04    .00000E+00
   13    5.30274E-01  2.07713E-03    .00000E+00        6.86066E-02
1.14038E-04        4.31265E-02  1.42505E-04    .00000E+00
   14    4.84734E-01  2.83531E-03    .00000E+00        6.72103E-02
1.11718E-04        4.62180E-02  1.90562E-04    .00000E+00
   15    4.56194E-01  3.34008E-03    .00000E+00        8.19062E-03
1.36145E-05        5.98475E-03  2.73573E-05    .00000E+00
   16    4.50003E-01  3.46267E-03    .00000E+00        8.43445E-03
1.40198E-05        6.24770E-03  2.92057E-05    .00000E+00
   17    4.42700E-01  3.60538E-03    .00000E+00        1.05898E-02
1.76025E-05        7.97368E-03  3.81804E-05    .00000E+00
   18    4.19829E-01  4.05921E-03    .00000E+00        4.58799E-02
7.62619E-05        3.64274E-02  1.86236E-04    .00000E+00
   19    3.66366E-01  5.17002E-03    .00000E+00        7.70752E-02
1.28115E-04        7.01259E-02  3.98480E-04    .00000E+00
   20    3.18292E-01  6.29875E-03    .00000E+00        3.41033E-02
5.66868E-05        3.57149E-02  2.14809E-04    .00000E+00
   21    2.76329E-01  7.56452E-03    .00000E+00        1.28486E-01
2.13570E-04        1.54991E-01  9.71932E-04    .00000E+00
   22    2.07862E-01  1.07793E-02    .00000E+00        1.41096E+00
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2.34531E-03       2.26265E+00  1.52091E-02   .00000E+00
   23    1.64184E-01  1.40641E-02   .00000E+00       1.69316E+00
2.81438E-03       3.43752E+00  2.38128E-02   .00000E+00
   24    1.30440E-01  1.78189E-02   .00000E+00       1.74350E+00
2.89807E-03       4.45545E+00  3.10673E-02   .00000E+00
   25    1.01321E-01  2.37240E-02   .00000E+00       1.39873E+00
2.32499E-03       4.60164E+00  3.31835E-02   .00000E+00
   26    7.46679E-02  3.97569E-02   .00000E+00       7.62262E-01
1.26704E-03       3.40290E+00  3.03051E-02   .00000E+00

 THERMAL 1.31160E-01  1.86476E-02   .00000E+00       7.24828E+00
1.20481E-02       1.84210E+01  1.35163E-01   .00000E+00

  TOTAL  1.54799E-01  1.52355E-02   .00000E+00       8.94050E+00
1.48610E-02       1.92519E+01  1.36213E-01   .00000E+00


0REGION 77     MATERIAL GAP         VOLUME  4.071503E+01

                      CROSS-SECTIONS                              FLUXES
                        REACTIONS
 GROUP   DIFFUSION   ABSORPTION   NU*FISSION            RIF            RAF
       TRANSPORT    ABSORPTION   NU*FISSION

    1    1.41755E+05  5.13322E-07   .00000E+00       3.49105E-03
8.57435E-05       8.20912E-09  1.79203E-09   .00000E+00
    2    1.02864E+05  2.81158E-07   .00000E+00       1.48236E-02
3.64082E-04       4.80362E-08  4.16778E-09   .00000E+00
    3    7.59244E+04  6.05228E-08   .00000E+00       1.21729E-02
2.98979E-04       5.34432E-08  7.36739E-10   .00000E+00
    4    4.70167E+04  1.33476E-08   .00000E+00       9.96017E-03
2.44631E-04       7.06145E-08  1.32945E-10   .00000E+00
    5    3.93658E+04  2.66520E-09   .00000E+00       7.60583E-03
1.86806E-04       6.44030E-08  2.02711E-11   .00000E+00
    6    2.80739E+04  3.59990E-09   .00000E+00       1.24671E-02
3.06203E-04       1.48027E-07  4.48801E-11   .00000E+00
    7    2.25878E+04  9.38580E-09   .00000E+00       1.20067E-02
2.94896E-04       1.77185E-07  1.12692E-10   .00000E+00
    8    2.01120E+04  2.55719E-08   .00000E+00       1.25569E-02
3.08408E-04       2.08115E-07  3.21103E-10   .00000E+00
    9    1.89538E+04  5.23663E-08   .00000E+00       6.48855E-03
1.59365E-04       1.14111E-07  3.39782E-10   .00000E+00
   10    1.87168E+04  8.63328E-08   .00000E+00       6.62323E-03
1.62673E-04       1.17955E-07  5.71802E-10   .00000E+00
   11    1.85665E+04  1.42363E-07   .00000E+00       6.73548E-03
1.65430E-04       1.20925E-07  9.58883E-10   .00000E+00
   12    1.84522E+04  2.18329E-07   .00000E+00       4.99028E-03
1.22566E-04       9.01480E-08  1.08952E-09   .00000E+00
   13    1.83667E+04  3.07848E-07   .00000E+00       4.61651E-03
1.13386E-04       8.37843E-08  1.42118E-09   .00000E+00
   14    1.82499E+04  4.20257E-07   .00000E+00       4.49534E-03
1.10410E-04       8.21073E-08  1.88920E-09   .00000E+00
   15    1.81808E+04  4.94670E-07   .00000E+00       5.46692E-04
1.34273E-05       1.00233E-08  2.70432E-10   .00000E+00
   16    1.81627E+04  5.12788E-07   .00000E+00       5.61969E-04
1.38025E-05       1.03136E-08  2.88171E-10   .00000E+00
   17    1.81308E+04  5.34075E-07   .00000E+00       7.05282E-04
1.73224E-05       1.29666E-08  3.76673E-10   .00000E+00
   18    1.80596E+04  6.01494E-07   .00000E+00       3.04792E-03
7.48598E-05       5.62568E-08  1.83331E-09   .00000E+00
   19    1.79037E+04  7.66370E-07   .00000E+00       5.09597E-03
1.25162E-04       9.48775E-08  3.90540E-09   .00000E+00
   20    1.77278E+04  9.34163E-07   .00000E+00       2.23847E-03
5.49789E-05       4.20895E-08  2.09109E-09   .00000E+00
   21    1.75125E+04  1.12938E-06   .00000E+00       8.31822E-03
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2.04303E-04       1.58330E-07  9.39446E-09   .00000E+00
   22     1.70165E+04  1.60405E-06   .00000E+00        9.08327E-02
2.23094E-03       1.77930E-06  1.45700E-07   .00000E+00
   23     1.65119E+04  2.08502E-06   .00000E+00        1.08886E-01
2.67434E-03       2.19814E-06  2.27030E-07   .00000E+00
   24     1.59381E+04  2.64136E-06   .00000E+00        1.11941E-01
2.74938E-03       2.34116E-06  2.95677E-07   .00000E+00
   25     1.50606E+04  3.51301E-06   .00000E+00        8.96114E-02
2.20094E-03       1.98335E-06  3.14806E-07   .00000E+00
   26     1.27683E+04  5.89029E-06   .00000E+00        4.85493E-02
1.19242E-03       1.26744E-06  2.85970E-07   .00000E+00

 THERMAL  1.57286E+04  2.75971E-06   .00000E+00        4.65473E-01
1.14325E-02       9.86469E-06  1.28457E-06   .00000E+00

  TOTAL   1.73222E+04  2.20735E-06   .00000E+00        5.89369E-01
1.44755E-02       1.13413E-05  1.30094E-06   .00000E+00


0REGION 78     MATERIAL AL          VOLUME  1.629733E+01

                     CROSS-SECTIONS                           FLUXES
                       REACTIONS
 GROUP   DIFFUSION     ABSORPTION   NU*FISSION          RIF           RAF
        TRANSPORT     ABSORPTION   NU*FISSION

    1     3.95965E+00  2.27382E-03   .00000E+00        1.32725E-03
8.14399E-05       1.11732E-04  3.01793E-06   .00000E+00
    2     3.10175E+00  5.71386E-05   .00000E+00        5.71909E-03
3.50922E-04       6.14609E-04  3.26781E-07   .00000E+00
    3     2.20964E+00  1.67482E-05   .00000E+00        4.75321E-03
2.91656E-04       7.17042E-04  7.96077E-08   .00000E+00
    4     1.62440E+00  5.00338E-05   .00000E+00        3.86194E-03
2.36968E-04       7.92484E-04  1.93228E-07   .00000E+00
    5     9.26281E-01  1.39174E-04   .00000E+00        2.96259E-03
1.81784E-04       1.06612E-03  4.12315E-07   .00000E+00
    6     3.33061E+00  2.80616E-04   .00000E+00        4.82645E-03
2.96150E-04       4.83039E-04  1.35438E-06   .00000E+00
    7     3.02145E+00  5.19427E-04   .00000E+00        4.68002E-03
2.87165E-04       5.16312E-04  2.43093E-06   .00000E+00
    8     4.08502E+00  2.21816E-04   .00000E+00        4.92249E-03
3.02043E-04       4.01670E-04  1.09189E-06   .00000E+00
    9     4.09110E+00  2.38056E-04   .00000E+00        2.55023E-03
1.56481E-04       2.07787E-04  6.07098E-07   .00000E+00
   10     4.08233E+00  3.84755E-04   .00000E+00        2.60874E-03
1.60072E-04       2.13011E-04  1.00373E-06   .00000E+00
   11     4.06966E+00  6.34997E-04   .00000E+00        2.65763E-03
1.63071E-04       2.17678E-04  1.68759E-06   .00000E+00
   12     4.05320E+00  9.73709E-04   .00000E+00        1.97118E-03
1.20951E-04       1.62109E-04  1.91936E-06   .00000E+00
   13     4.03627E+00  1.37216E-03   .00000E+00        1.82720E-03
1.12116E-04       1.50898E-04  2.50721E-06   .00000E+00
   14     4.01077E+00  1.87212E-03   .00000E+00        1.78021E-03
1.09233E-04       1.47952E-04  3.33277E-06   .00000E+00
   15     4.00087E+00  2.20303E-03   .00000E+00        2.16517E-04
1.32854E-05       1.80392E-05  4.76995E-07   .00000E+00
   16     3.99705E+00  2.28364E-03   .00000E+00        2.22621E-04
1.36600E-05       1.85655E-05  5.08386E-07   .00000E+00
   17     3.98901E+00  2.37847E-03   .00000E+00        2.79375E-04
1.71424E-05       2.33454E-05  6.64487E-07   .00000E+00
   18     3.97288E+00  2.67940E-03   .00000E+00        1.20732E-03
7.40809E-05       1.01297E-04  3.23490E-06   .00000E+00
   19     3.93557E+00  3.41570E-03   .00000E+00        2.01894E-03
1.23882E-04       1.70999E-04  6.89609E-06   .00000E+00
   20     3.89960E+00  4.16532E-03   .00000E+00        8.87572E-04
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5.44612E-05        7.58686E-05  3.69702E-06   .00000E+00
   21     3.85636E+00  5.03787E-03    .00000E+00       3.31123E-03
2.03176E-04        2.86214E-04  1.66815E-05   .00000E+00
   22     3.75376E+00  7.15826E-03    .00000E+00       3.62753E-02
2.22584E-03        3.22124E-03  2.59668E-04   .00000E+00
   23     3.65252E+00  9.30090E-03    .00000E+00       4.34925E-02
2.66869E-03        3.96918E-03  4.04520E-04   .00000E+00
   24     3.54298E+00  1.17702E-02    .00000E+00       4.46810E-02
2.74161E-03        4.20370E-03  5.25906E-04   .00000E+00
   25     3.37501E+00  1.56477E-02    .00000E+00       3.57547E-02
2.19390E-03        3.53131E-03  5.59480E-04   .00000E+00
   26     2.94383E+00  2.62317E-02    .00000E+00       1.93262E-02
1.18585E-03        2.18833E-03  5.06959E-04   .00000E+00

 THERMAL 3.50861E+00  1.22952E-02    .00000E+00       1.85747E-01
1.13974E-02        1.76468E-02  2.28381E-03   .00000E+00

  TOTAL  3.30533E+00  9.86093E-03    .00000E+00       2.34121E-01
1.43656E-02        2.36105E-02  2.30866E-03   .00000E+00


0REGION 79     MATERIAL COOLANT     VOLUME  8.975974E+02

                    CROSS-SECTIONS                           FLUXES
                      REACTIONS
 GROUP   DIFFUSION    ABSORPTION   NU*FISSION          RIF            RAF
        TRANSPORT    ABSORPTION   NU*FISSION

    1     2.70642E+00  2.17943E-03    .00000E+00       2.99083E-02
3.33204E-05        3.68362E-03  6.51830E-05   .00000E+00
    2     2.01031E+00  1.01023E-05    .00000E+00       1.03113E-01
1.14877E-04        1.70974E-02  1.04168E-06   .00000E+00
    3     1.31913E+00  3.55934E-06    .00000E+00       7.80658E-02
8.69719E-05        1.97266E-02  2.77863E-07   .00000E+00
    4     7.92566E-01  3.59323E-06    .00000E+00       5.09675E-02
5.67821E-05        2.14357E-02  1.83138E-07   .00000E+00
    5     6.54618E-01  6.63753E-06    .00000E+00       3.61362E-02
4.02588E-05        1.84006E-02  2.39855E-07   .00000E+00
    6     5.56182E-01  1.94203E-05    .00000E+00       5.57710E-02
6.21336E-05        3.34249E-02  1.08309E-06   .00000E+00
    7     5.63543E-01  6.29493E-05    .00000E+00       5.40448E-02
6.02105E-05        3.19673E-02  3.40209E-06   .00000E+00
    8     5.77544E-01  1.71416E-04    .00000E+00       5.89240E-02
6.56464E-05        3.40084E-02  1.01005E-05   .00000E+00
    9     5.80367E-01  3.53087E-04    .00000E+00       3.13212E-02
3.48944E-05        1.79893E-02  1.10591E-05   .00000E+00
   10     5.80777E-01  5.82085E-04    .00000E+00       3.29012E-02
3.66547E-05        1.88834E-02  1.91513E-05   .00000E+00
   11     5.80114E-01  9.59742E-04    .00000E+00       3.44649E-02
3.83969E-05        1.98035E-02  3.30774E-05   .00000E+00
   12     5.78579E-01  1.47308E-03    .00000E+00       2.61365E-02
2.91183E-05        1.50579E-02  3.85011E-05   .00000E+00
   13     5.30274E-01  2.07713E-03    .00000E+00       2.49135E-02
2.77558E-05        1.56608E-02  5.17487E-05   .00000E+00
   14     4.84734E-01  2.83531E-03    .00000E+00       2.46694E-02
2.74838E-05        1.69642E-02  6.99454E-05   .00000E+00
   15     4.56194E-01  3.34008E-03    .00000E+00       3.01742E-03
3.36166E-06        2.20478E-03  1.00784E-05   .00000E+00
   16     4.50003E-01  3.46267E-03    .00000E+00       3.11256E-03
3.46765E-06        2.30558E-03  1.07778E-05   .00000E+00
   17     4.42700E-01  3.60538E-03    .00000E+00       3.91129E-03
4.35751E-06        2.94502E-03  1.41017E-05   .00000E+00
   18     4.19829E-01  4.05921E-03    .00000E+00       1.69971E-02
1.89362E-05        1.34952E-02  6.89947E-05   .00000E+00
   19     3.66366E-01  5.17002E-03    .00000E+00       2.87697E-02
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3.20519E-05       2.61758E-02  1.48740E-04    .00000E+00
   20     3.18292E-01  6.29875E-03    .00000E+00        1.34922E-02
1.50315E-05       1.41298E-02  8.49841E-05    .00000E+00
   21     2.76329E-01  7.56452E-03    .00000E+00        6.38862E-02
7.11747E-05       7.70654E-02  4.83268E-04    .00000E+00
   22     2.07862E-01  1.07793E-02    .00000E+00        8.30673E-01
9.25441E-04       1.33209E+00  8.95404E-03    .00000E+00
   23     1.64184E-01  1.40641E-02    .00000E+00        1.01468E+00
1.13044E-03       2.06005E+00  1.42706E-02    .00000E+00
   24     1.30440E-01  1.78189E-02    .00000E+00        1.04928E+00
1.16899E-03       2.68139E+00  1.86970E-02    .00000E+00
   25     1.01321E-01  2.37240E-02    .00000E+00        8.44593E-01
9.40948E-04       2.77860E+00  2.00371E-02    .00000E+00
   26     7.46679E-02  3.97569E-02    .00000E+00        4.60267E-01
5.12776E-04       2.05473E+00  1.82988E-02    .00000E+00

 THERMAL 1.30187E-01  1.88066E-02    .00000E+00        4.30564E+00
4.79685E-03       1.10242E+01  8.09745E-02    .00000E+00

  TOTAL  1.46347E-01  1.63617E-02    .00000E+00        4.97402E+00
5.54148E-03       1.13293E+01  8.13835E-02    .00000E+00


0REGION 80     MATERIAL VESSEL      VOLUME  1.854872E+02

                       CROSS-SECTIONS                              FLUXES
                         REACTIONS
 GROUP  DIFFUSION     ABSORPTION    NU*FISSION            RIF            RAF
       TRANSPORT     ABSORPTION   NU*FISSION

    1     3.95965E+00  2.27382E-03    .00000E+00        3.84335E-03
2.07203E-05       3.23544E-04  8.73908E-06    .00000E+00
    2     3.10175E+00  5.71386E-05    .00000E+00        1.23469E-02
6.65647E-05       1.32687E-03  7.05485E-07    .00000E+00
    3     2.20964E+00  1.67482E-05    .00000E+00        9.04159E-03
4.87451E-05       1.36396E-03  1.51430E-07    .00000E+00
    4     1.62440E+00  5.00338E-05    .00000E+00        5.45754E-03
2.94227E-05       1.11991E-03  2.73061E-07    .00000E+00
    5     9.26281E-01  1.39174E-04    .00000E+00        3.76243E-03
2.02841E-05       1.35396E-03  5.23633E-07    .00000E+00
    6     3.33061E+00  2.80616E-04    .00000E+00        5.38804E-03
2.90480E-05       5.39244E-04  1.51197E-06    .00000E+00
    7     3.02144E+00  5.19427E-04    .00000E+00        4.98978E-03
2.69009E-05       5.50485E-04  2.59183E-06    .00000E+00
    8     4.08502E+00  2.21816E-04    .00000E+00        5.26153E-03
2.83660E-05       4.29335E-04  1.16709E-06    .00000E+00
    9     4.09110E+00  2.38056E-04    .00000E+00        2.75634E-03
1.48600E-05       2.24580E-04  6.56164E-07    .00000E+00
   10     4.08233E+00  3.84755E-04    .00000E+00        2.86835E-03
1.54639E-05       2.34209E-04  1.10361E-06    .00000E+00
   11     4.06966E+00  6.34997E-04    .00000E+00        2.98287E-03
1.60813E-05       2.44318E-04  1.89412E-06    .00000E+00
   12     4.05320E+00  9.73709E-04    .00000E+00        2.25197E-03
1.21408E-05       1.85201E-04  2.19276E-06    .00000E+00
   13     4.03627E+00  1.37216E-03    .00000E+00        2.13433E-03
1.15066E-05       1.76263E-04  2.92865E-06    .00000E+00
   14     4.01077E+00  1.87212E-03    .00000E+00        2.10440E-03
1.13453E-05       1.74896E-04  3.93970E-06    .00000E+00
   15     4.00087E+00  2.20303E-03    .00000E+00        2.56933E-04
1.38518E-06       2.14064E-05  5.66033E-07    .00000E+00
   16     3.99705E+00  2.28364E-03    .00000E+00        2.64791E-04
1.42755E-06       2.20822E-05  6.04688E-07    .00000E+00
   17     3.98901E+00  2.37847E-03    .00000E+00        3.32531E-04
1.79274E-06       2.77872E-05  7.90915E-07    .00000E+00
   18     3.97288E+00  2.67940E-03    .00000E+00        1.44093E-03
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7.76833E-06        1.20897E-04  3.86083E-06   .00000E+00
   19     3.93557E+00  3.41570E-03   .00000E+00       2.43045E-03
1.31031E-05        2.05853E-04  8.30168E-06   .00000E+00
   20     3.89960E+00  4.16532E-03   .00000E+00       1.15566E-03
6.23040E-06        9.87845E-05  4.81370E-06   .00000E+00
   21     3.85636E+00  5.03787E-03   .00000E+00       5.82330E-03
3.13946E-05        5.03350E-04  2.93370E-05   .00000E+00
   22     3.75376E+00  7.15826E-03   .00000E+00       7.88672E-02
4.25189E-04        7.00339E-03  5.64552E-04   .00000E+00
   23     3.65252E+00  9.30090E-03   .00000E+00       9.62306E-02
5.18799E-04        8.78212E-03  8.95032E-04   .00000E+00
   24     3.54298E+00  1.17702E-02   .00000E+00       9.87756E-02
5.32520E-04        9.29307E-03  1.16261E-03   .00000E+00
   25     3.37502E+00  1.56477E-02   .00000E+00       7.86930E-02
4.24250E-04        7.77212E-03  1.23137E-03   .00000E+00
   26     2.94383E+00  2.62317E-02   .00000E+00       4.19782E-02
2.26313E-04        4.75324E-03  1.10116E-03   .00000E+00

 THERMAL 3.50546E+00  1.23707E-02   .00000E+00       4.03954E-01
2.17780E-03        3.84119E-02  4.99718E-03   .00000E+00

  TOTAL  3.35418E+00  1.06724E-02   .00000E+00       4.71439E-01
2.54162E-03        4.68509E-02  5.03138E-03   .00000E+00


0REGION 81     MATERIAL COOLANT     VOLUME  5.975334E+01

                      CROSS-SECTIONS                           FLUXES
                        REACTIONS
 GROUP   DIFFUSION      ABSORPTION   NU*FISSION           RIF             RAF
        TRANSPORT     ABSORPTION   NU*FISSION

    1     2.70642E+00  2.17943E-03   .00000E+00       1.12818E-03
1.88807E-05        1.38951E-04  2.45880E-06   .00000E+00
    2     2.01031E+00  1.01023E-05   .00000E+00       3.54090E-03
5.92585E-05        5.87124E-04  3.57713E-08   .00000E+00
    3     1.31913E+00  3.55934E-06   .00000E+00       2.57358E-03
4.30700E-05        6.50322E-04  9.16025E-09   .00000E+00
    4     7.92566E-01  3.59323E-06   .00000E+00       1.57682E-03
2.63888E-05        6.63170E-04  5.66587E-09   .00000E+00
    5     6.54618E-01  6.63753E-06   .00000E+00       1.07150E-03
1.79320E-05        5.45610E-04  7.11211E-09   .00000E+00
    6     5.56182E-01  1.94203E-05   .00000E+00       1.61779E-03
2.70745E-05        9.69583E-04  3.14181E-08   .00000E+00
    7     5.63543E-01  6.29493E-05   .00000E+00       1.50176E-03
2.51327E-05        8.88285E-04  9.45348E-08   .00000E+00
    8     5.77544E-01  1.71416E-04   .00000E+00       1.59045E-03
2.66169E-05        9.17937E-04  2.72628E-07   .00000E+00
    9     5.80367E-01  3.53087E-04   .00000E+00       8.33226E-04
1.39444E-05        4.78563E-04  2.94201E-07   .00000E+00
   10     5.80777E-01  5.82085E-04   .00000E+00       8.66640E-04
1.45036E-05        4.97403E-04  5.04458E-07   .00000E+00
   11     5.80114E-01  9.59742E-04   .00000E+00       9.00831E-04
1.50758E-05        5.17617E-04  8.64565E-07   .00000E+00
   12     5.78579E-01  1.47308E-03   .00000E+00       6.79905E-04
1.13785E-05        3.91710E-04  1.00155E-06   .00000E+00
   13     5.30274E-01  2.07713E-03   .00000E+00       6.45769E-04
1.08072E-05        4.05934E-04  1.34135E-06   .00000E+00
   14     4.84734E-01  2.83531E-03   .00000E+00       6.37763E-04
1.06733E-05        4.38565E-04  1.80826E-06   .00000E+00
   15     4.56194E-01  3.34008E-03   .00000E+00       7.79430E-05
1.30441E-06        5.69517E-05  2.60336E-07   .00000E+00
   16     4.50003E-01  3.46267E-03   .00000E+00       8.03780E-05
1.34516E-06        5.95389E-05  2.78322E-07   .00000E+00
   17     4.42700E-01  3.60538E-03   .00000E+00       1.00957E-04
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1.68956E-06       7.60160E-05  3.63988E-07   .00000E+00
   18    4.19829E-01  4.05921E-03   .00000E+00        4.38032E-04
7.33066E-06       3.47786E-04  1.77806E-06   .00000E+00
   19    3.66366E-01  5.17002E-03   .00000E+00        7.40486E-04
1.23924E-05       6.73722E-04  3.82832E-06   .00000E+00
   20    3.18292E-01  6.29875E-03   .00000E+00        3.52440E-04
5.89824E-06       3.69095E-04  2.21993E-06   .00000E+00
   21    2.76329E-01  7.56452E-03   .00000E+00        1.78856E-03
2.99324E-05       2.15753E-03  1.35296E-05   .00000E+00
   22    2.07862E-01  1.07793E-02   .00000E+00        2.44229E-02
4.08729E-04       3.91653E-02  2.63261E-04   .00000E+00
   23    1.64184E-01  1.40641E-02   .00000E+00        2.99323E-02
5.00931E-04       6.07699E-02  4.20972E-04   .00000E+00
   24    1.30440E-01  1.78189E-02   .00000E+00        3.08602E-02
5.16460E-04       7.88619E-02  5.49895E-04   .00000E+00
   25    1.01321E-01  2.37240E-02   .00000E+00        2.47271E-02
4.13820E-04       8.13489E-02  5.86626E-04   .00000E+00
   26    7.46679E-02  3.97569E-02   .00000E+00        1.33876E-02
2.24048E-04       5.97651E-02  5.32249E-04   .00000E+00

 THERMAL 1.30205E-01  1.87984E-02   .00000E+00        1.26212E-01
2.11221E-03       3.23111E-01  2.37258E-03   .00000E+00

  TOTAL  1.46775E-01  1.63204E-02   .00000E+00        1.46074E-01
2.44462E-03       3.31742E-01  2.38399E-03   .00000E+00


0REGION 82     MATERIAL MODERATOR   VOLUME  6.031861E+01

                    CROSS-SECTIONS                              FLUXES
                      REACTIONS
 GROUP   DIFFUSION    ABSORPTION   NU*FISSION           RIF            RAF
       TRANSPORT    ABSORPTION   NU*FISSION

    1    2.74822E+00  2.21855E-03   .00000E+00        1.08038E-03
1.79112E-05       1.31039E-04  2.39687E-06   .00000E+00
    2    2.00863E+00  9.66120E-06   .00000E+00        3.31763E-03
5.50018E-05       5.50564E-04  3.20523E-08   .00000E+00
    3    1.31914E+00  3.55843E-06   .00000E+00        2.39208E-03
3.96574E-05       6.04455E-04  8.51206E-09   .00000E+00
    4    7.92343E-01  3.59826E-06   .00000E+00        1.46443E-03
2.42782E-05       6.16075E-04  5.26939E-09   .00000E+00
    5    6.53939E-01  6.66158E-06   .00000E+00        9.97092E-04
1.65304E-05       5.08249E-04  6.64221E-09   .00000E+00
    6    5.56006E-01  1.95488E-05   .00000E+00        1.50168E-03
2.48959E-05       9.00280E-04  2.93561E-08   .00000E+00
    7    5.63650E-01  6.32557E-05   .00000E+00        1.39749E-03
2.31685E-05       8.26453E-04  8.83993E-08   .00000E+00
    8    5.77572E-01  1.72185E-04   .00000E+00        1.47800E-03
2.45032E-05       8.52993E-04  2.54489E-07   .00000E+00
    9    5.80369E-01  3.53483E-04   .00000E+00        7.74827E-04
1.28456E-05       4.45020E-04  2.73888E-07   .00000E+00
   10    5.80777E-01  5.82735E-04   .00000E+00        8.05013E-04
1.33460E-05       4.62033E-04  4.69109E-07   .00000E+00
   11    5.80112E-01  9.60720E-04   .00000E+00        8.36100E-04
1.38614E-05       4.80424E-04  8.03258E-07   .00000E+00
   12    5.78577E-01  1.47362E-03   .00000E+00        6.30860E-04
1.04588E-05       3.63455E-04  9.29649E-07   .00000E+00
   13    5.30259E-01  2.07768E-03   .00000E+00        5.99143E-04
9.93296E-06       3.76635E-04  1.24483E-06   .00000E+00
   14    4.84689E-01  2.83608E-03   .00000E+00        5.91493E-04
9.80614E-06       4.06785E-04  1.67752E-06   .00000E+00
   15    4.56193E-01  3.34009E-03   .00000E+00        7.22857E-05
1.19840E-06       5.28180E-05  2.41441E-07   .00000E+00
   16    4.50003E-01  3.46266E-03   .00000E+00        7.45422E-05
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1.23581E-06       5.52161E-05  2.58115E-07   .00000E+00
   17     4.42698E-01  3.60542E-03    .00000E+00       9.36201E-05
1.55209E-06       7.04920E-05  3.37539E-07   .00000E+00
   18     4.19805E-01  4.05969E-03    .00000E+00       4.06232E-04
6.73478E-06       3.22556E-04  1.64918E-06   .00000E+00
   19     3.66218E-01  5.17320E-03    .00000E+00       6.86138E-04
1.13752E-05       6.24526E-04  3.54953E-06   .00000E+00
   20     3.18091E-01  6.30426E-03    .00000E+00       3.24505E-04
5.37984E-06       3.40054E-04  2.04576E-06   .00000E+00
   21     2.74446E-01  7.62670E-03    .00000E+00       1.65090E-03
2.73696E-05       2.00513E-03  1.25909E-05   .00000E+00
   22     2.07048E-01  1.08270E-02    .00000E+00       2.28730E-02
3.79203E-04       3.68239E-02  2.47646E-04   .00000E+00
   23     1.64120E-01  1.40700E-02    .00000E+00       2.81678E-02
4.66984E-04       5.72098E-02  3.96321E-04   .00000E+00
   24     1.30412E-01  1.78227E-02    .00000E+00       2.90872E-02
4.82226E-04       7.43468E-02  5.18414E-04   .00000E+00
   25     1.01309E-01  2.37277E-02    .00000E+00       2.33704E-02
3.87450E-04       7.68949E-02  5.54526E-04   .00000E+00
   26     7.46682E-02  3.97555E-02    .00000E+00       1.27139E-02
2.10779E-04       5.67573E-02  5.05448E-04   .00000E+00

 THERMAL 1.29916E-01  1.88480E-02    .00000E+00       1.18874E-01
1.97077E-03       3.05002E-01  2.24054E-03   .00000E+00

  TOTAL  1.46299E-01  1.63862E-02    .00000E+00       1.37387E-01
2.27769E-03       3.13028E-01  2.25125E-03   .00000E+00


0REGION 83     MATERIAL COOLANT     VOLUME  6.088406E+01

                    CROSS-SECTIONS                              FLUXES
                      REACTIONS
 GROUP   DIFFUSION    ABSORPTION   NU*FISSION          RIF            RAF
       TRANSPORT     ABSORPTION  NU*FISSION

    1     2.70642E+00  2.17943E-03    .00000E+00       1.03626E-03
1.70202E-05       1.27630E-04  2.25845E-06   .00000E+00
    2     2.01031E+00  1.01023E-05    .00000E+00       3.12936E-03
5.13986E-05       5.18885E-04  3.16138E-08   .00000E+00
    3     1.31913E+00  3.55934E-06    .00000E+00       2.24057E-03
3.68005E-05       5.66172E-04  7.97494E-09   .00000E+00
    4     7.92566E-01  3.59323E-06    .00000E+00       1.36833E-03
2.24744E-05       5.75487E-04  4.91674E-09   .00000E+00
    5     6.54618E-01  6.63753E-06    .00000E+00       9.32544E-04
1.53167E-05       4.74854E-04  6.18979E-09   .00000E+00
    6     5.56182E-01  1.94203E-05    .00000E+00       1.39941E-03
2.29848E-05       8.38699E-04  2.71770E-08   .00000E+00
    7     5.63543E-01  6.29493E-05    .00000E+00       1.30201E-03
2.13851E-05       7.70135E-04  8.19609E-08   .00000E+00
    8     5.77544E-01  1.71416E-04    .00000E+00       1.37498E-03
2.25836E-05       7.93579E-04  2.35694E-07   .00000E+00
    9     5.80367E-01  3.53087E-04    .00000E+00       7.20161E-04
1.18284E-05       4.13624E-04  2.54279E-07   .00000E+00
   10     5.80777E-01  5.82085E-04    .00000E+00       7.47982E-04
1.22853E-05       4.29299E-04  4.35389E-07   .00000E+00
   11     5.80114E-01  9.59742E-04    .00000E+00       7.76421E-04
1.27524E-05       4.46131E-04  7.45163E-07   .00000E+00
   12     5.78579E-01  1.47308E-03    .00000E+00       5.85514E-04
9.61686E-06       3.37328E-04  8.62507E-07   .00000E+00
   13     5.30274E-01  2.07713E-03    .00000E+00       5.55893E-04
9.13036E-06       3.49438E-04  1.15466E-06   .00000E+00
   14     4.84734E-01  2.83531E-03    .00000E+00       5.48588E-04
9.01037E-06       3.77243E-04  1.55542E-06   .00000E+00
   15     4.56194E-01  3.34008E-03    .00000E+00       6.70232E-05
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1.10083E-06         4.89728E-05  2.23863E-07    .00000E+00
   16     4.50003E-01  3.46267E-03    .00000E+00        6.91147E-05
1.13518E-06         5.11957E-05  2.39321E-07    .00000E+00
   17     4.42700E-01  3.60538E-03    .00000E+00        8.68018E-05
1.42569E-06         6.53578E-05  3.12954E-07    .00000E+00
   18     4.19829E-01  4.05921E-03    .00000E+00        3.76613E-04
6.18574E-06         2.99021E-04  1.52875E-06    .00000E+00
   19     3.66366E-01  5.17002E-03    .00000E+00        6.35946E-04
1.04452E-05         5.78608E-04  3.28785E-06    .00000E+00
   20     3.18292E-01  6.29875E-03    .00000E+00        3.01442E-04
4.95108E-06         3.15687E-04  1.89871E-06    .00000E+00
   21     2.76329E-01  7.56452E-03    .00000E+00        1.54345E-03
2.53507E-05         1.86185E-03  1.16755E-05    .00000E+00
   22     2.07862E-01  1.07793E-02    .00000E+00        2.14307E-02
3.51992E-04         3.43668E-02  2.31007E-04    .00000E+00
   23     1.64184E-01  1.40641E-02    .00000E+00        2.64033E-02
4.33665E-04         5.36051E-02  3.71339E-04    .00000E+00
   24     1.30440E-01  1.78189E-02    .00000E+00        2.73046E-02
4.48468E-04         6.97756E-02  4.86537E-04    .00000E+00
   25     1.01321E-01  2.37240E-02    .00000E+00        2.19676E-02
3.60810E-04         7.22703E-02  5.21158E-04    .00000E+00
   26     7.46679E-02  3.97569E-02    .00000E+00        1.19664E-02
1.96545E-04         5.34207E-02  4.75748E-04    .00000E+00

 THERMAL 1.29927E-01  1.88488E-02    .00000E+00        1.11553E-01
1.83223E-03         2.86195E-01  2.10265E-03    .00000E+00

  TOTAL  1.46273E-01  1.63933E-02    .00000E+00        1.28871E-01
2.11666E-03         2.93678E-01  2.11262E-03    .00000E+00


0REGION 84      MATERIAL MODERATOR   VOLUME  4.420215E+02

                      CROSS-SECTIONS                                 FLUXES
                         REACTIONS
 GROUP   DIFFUSION    ABSORPTION   NU*FISSION            RIF            RAF
        TRANSPORT    ABSORPTION   NU*FISSION

    1    2.74822E+00  2.21855E-03    .00000E+00        6.21570E-03
1.40620E-05         7.53905E-04  1.37899E-05    .00000E+00
    2    2.00863E+00  9.66120E-06    .00000E+00        1.77443E-02
4.01435E-05         2.94468E-03  1.71431E-07    .00000E+00
    3    1.31914E+00  3.55843E-06    .00000E+00        1.24380E-02
2.81389E-05         3.14296E-03  4.42598E-08    .00000E+00
    4    7.92343E-01  3.59826E-06    .00000E+00        7.52503E-03
1.70241E-05         3.16573E-03  2.70770E-08    .00000E+00
    5    6.53939E-01  6.66158E-06    .00000E+00        5.12725E-03
1.15996E-05         2.61352E-03  3.41556E-08    .00000E+00
    6    5.56006E-01  1.95488E-05    .00000E+00        7.59220E-03
1.71761E-05         4.55163E-03  1.48418E-07    .00000E+00
    7    5.63650E-01  6.32557E-05    .00000E+00        7.03190E-03
1.59085E-05         4.15855E-03  4.44808E-07    .00000E+00
    8    5.77572E-01  1.72185E-04    .00000E+00        7.37479E-03
1.66842E-05         4.25620E-03  1.26983E-06    .00000E+00
    9    5.80369E-01  3.53483E-04    .00000E+00        3.85660E-03
8.72492E-06         2.21503E-03  1.36324E-06    .00000E+00
   10    5.80777E-01  5.82735E-04    .00000E+00        3.99168E-03
9.03052E-06         2.29100E-03  2.32609E-06    .00000E+00
   11    5.80112E-01  9.60720E-04    .00000E+00        4.13060E-03
9.34478E-06         2.37345E-03  3.96835E-06    .00000E+00
   12    5.78577E-01  1.47362E-03    .00000E+00        3.10891E-03
7.03338E-06         1.79112E-03  4.58135E-06    .00000E+00
   13    5.30259E-01  2.07768E-03    .00000E+00        2.94605E-03
6.66496E-06         1.85196E-03  6.12096E-06    .00000E+00
   14    4.84689E-01  2.83608E-03    .00000E+00        2.90212E-03
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6.56556E-06        1.99586E-03  8.23064E-06    .00000E+00
   15    4.56193E-01  3.34009E-03    .00000E+00        3.54404E-04
8.01780E-07        2.58957E-04  1.18374E-06    .00000E+00
   16    4.50003E-01  3.46266E-03    .00000E+00        3.65358E-04
8.26562E-07        2.70634E-04  1.26511E-06    .00000E+00
   17    4.42698E-01  3.60542E-03    .00000E+00        4.58716E-04
1.03777E-06        3.45394E-04  1.65386E-06    .00000E+00
   18    4.19805E-01  4.05969E-03    .00000E+00        1.98872E-03
4.49916E-06        1.57909E-03  8.07362E-06    .00000E+00
   19    3.66218E-01  5.17320E-03    .00000E+00        3.35024E-03
7.57935E-06        3.04940E-03  1.73315E-05    .00000E+00
   20    3.18091E-01  6.30426E-03    .00000E+00        1.57274E-03
3.55805E-06        1.64810E-03  9.91493E-06    .00000E+00
   21    2.74446E-01  7.62670E-03    .00000E+00        8.12069E-03
1.83717E-05        9.86314E-03  6.19340E-05    .00000E+00
   22    2.07048E-01  1.08270E-02    .00000E+00        1.15920E-01
2.62250E-04        1.86624E-01  1.25507E-03    .00000E+00
   23    1.64120E-01  1.40700E-02    .00000E+00        1.43915E-01
3.25585E-04        2.92297E-01  2.02489E-03    .00000E+00
   24    1.30412E-01  1.78227E-02    .00000E+00        1.49034E-01
3.37164E-04        3.80929E-01  2.65619E-03    .00000E+00
   25    1.01309E-01  2.37277E-02    .00000E+00        1.20064E-01
2.71625E-04        3.95043E-01  2.84884E-03    .00000E+00
   26    7.46682E-02  3.97555E-02    .00000E+00        6.54590E-02
1.48090E-04        2.92221E-01  2.60235E-03    .00000E+00

 THERMAL 1.29655E-01  1.88934E-02    .00000E+00        6.07436E-01
1.37422E-03        1.56167E+00  1.14765E-02    .00000E+00

  TOTAL  1.46168E-01  1.64125E-02    .00000E+00        7.02588E-01
1.58949E-03        1.60223E+00  1.15312E-02    .00000E+00


0REGION 85     MATERIAL COOLANT     VOLUME  6.540780E+01

                     CROSS-SECTIONS                               FLUXES
                       REACTIONS
 GROUP   DIFFUSION    ABSORPTION   NU*FISSION          RIF             RAF
        TRANSPORT    ABSORPTION   NU*FISSION

    1    2.70642E+00  2.17943E-03    .00000E+00        7.56977E-04
1.15732E-05        9.32323E-05  1.64978E-06    .00000E+00
    2    2.01031E+00  1.01023E-05    .00000E+00        2.05113E-03
3.13591E-05        3.40102E-04  2.07212E-08    .00000E+00
    3    1.31913E+00  3.55934E-06    .00000E+00        1.40644E-03
2.15026E-05        3.55395E-04  5.00600E-09    .00000E+00
    4    7.92566E-01  3.59323E-06    .00000E+00        8.42038E-04
1.28737E-05        3.54140E-04  3.02564E-09    .00000E+00
    5    6.54618E-01  6.63753E-06    .00000E+00        5.72989E-04
8.76026E-06        2.91768E-04  3.80323E-09    .00000E+00
    6    5.56182E-01  1.94203E-05    .00000E+00        8.39711E-04
1.28381E-05        5.03259E-04  1.63075E-08    .00000E+00
    7    5.63543E-01  6.29493E-05    .00000E+00        7.71343E-04
1.17928E-05        4.56246E-04  4.85555E-08    .00000E+00
    8    5.77544E-01  1.71416E-04    .00000E+00        8.02306E-04
1.22662E-05        4.63056E-04  1.37528E-07    .00000E+00
    9    5.80367E-01  3.53087E-04    .00000E+00        4.17209E-04
6.37859E-06        2.39624E-04  1.47311E-07    .00000E+00
   10    5.80777E-01  5.82085E-04    .00000E+00        4.30567E-04
6.58281E-06        2.47121E-04  2.50626E-07    .00000E+00
   11    5.80114E-01  9.59742E-04    .00000E+00        4.44124E-04
6.79008E-06        2.55193E-04  4.26244E-07    .00000E+00
   12    5.78579E-01  1.47308E-03    .00000E+00        3.33318E-04
5.09600E-06        1.92033E-04  4.91003E-07    .00000E+00
   13    5.30274E-01  2.07713E-03    .00000E+00        3.15209E-04
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4.81913E-06       1.98142E-04  6.54730E-07   .00000E+00
   14     4.84734E-01  2.83531E-03    .00000E+00        3.09968E-04
4.73901E-06       2.13153E-04  8.78856E-07   .00000E+00
   15     4.56194E-01  3.34008E-03    .00000E+00        3.78135E-05
5.78119E-07       2.76297E-05  1.26300E-07   .00000E+00
   16     4.50003E-01  3.46267E-03    .00000E+00        3.89747E-05
5.95873E-07       2.88700E-05  1.34957E-07   .00000E+00
   17     4.42700E-01  3.60538E-03    .00000E+00        4.89248E-05
7.47996E-07       3.68381E-05  1.76392E-07   .00000E+00
   18     4.19829E-01  4.05921E-03    .00000E+00        2.11991E-04
3.24106E-06       1.68315E-04  8.60515E-07   .00000E+00
   19     3.66366E-01  5.17002E-03    .00000E+00        3.56878E-04
5.45620E-06       3.24701E-04  1.84507E-06   .00000E+00
   20     3.18292E-01  6.29875E-03    .00000E+00        1.68949E-04
2.58302E-06       1.76933E-04  1.06417E-06   .00000E+00
   21     2.76329E-01  7.56452E-03    .00000E+00        8.84748E-04
1.35266E-05       1.06726E-03  6.69269E-06   .00000E+00
   22     2.07862E-01  1.07793E-02    .00000E+00        1.25876E-02
1.92449E-04       2.01859E-02  1.35685E-04   .00000E+00
   23     1.64184E-01  1.40641E-02    .00000E+00        1.55816E-02
2.38222E-04       3.16344E-02  2.19141E-04   .00000E+00
   24     1.30440E-01  1.78189E-02    .00000E+00        1.61374E-02
2.46720E-04       4.12385E-02  2.87551E-04   .00000E+00
   25     1.01321E-01  2.37240E-02    .00000E+00        1.29908E-02
1.98613E-04       4.27380E-02  3.08194E-04   .00000E+00
   26     7.46679E-02  3.97569E-02    .00000E+00        7.07540E-03
1.08174E-04       3.15861E-02  2.81296E-04   .00000E+00

 THERMAL 1.29787E-01  1.88721E-02    .00000E+00        6.57834E-02
1.00574E-03       1.68952E-01  1.24147E-03   .00000E+00

  TOTAL  1.46881E-01  1.63255E-02    .00000E+00        7.64145E-02
1.16828E-03       1.73416E-01  1.24750E-03   .00000E+00


0REGION 86     MATERIAL MODERATOR    VOLUME  6.141188E+02

                    CROSS-SECTIONS                              FLUXES
                      REACTIONS
 GROUP   DIFFUSION    ABSORPTION   NU*FISSION           RIF            RAF
        TRANSPORT    ABSORPTION   NU*FISSION

    1     2.74822E+00  2.21855E-03    .00000E+00        5.66937E-03
9.23172E-06       6.87641E-04  1.25778E-05   .00000E+00
    2     2.00863E+00  9.66120E-06    .00000E+00        1.45099E-02
2.36272E-05       2.40794E-03  1.40183E-07   .00000E+00
    3     1.31914E+00  3.55843E-06    .00000E+00        9.75122E-03
1.58784E-05       2.46404E-03  3.46990E-08   .00000E+00
    4     7.92343E-01  3.59826E-06    .00000E+00        5.79989E-03
9.44425E-06       2.43998E-03  2.08695E-08   .00000E+00
    5     6.53939E-01  6.66158E-06    .00000E+00        3.93915E-03
6.41432E-06       2.00791E-03  2.62410E-08   .00000E+00
    6     5.56006E-01  1.95488E-05    .00000E+00        5.73026E-03
9.33086E-06       3.43537E-03  1.12019E-07   .00000E+00
    7     5.63650E-01  6.32557E-05    .00000E+00        5.23111E-03
8.51808E-06       3.09359E-03  3.30898E-07   .00000E+00
    8     5.77572E-01  1.72185E-04    .00000E+00        5.39954E-03
8.79234E-06       3.11623E-03  9.29719E-07   .00000E+00
    9     5.80369E-01  3.53483E-04    .00000E+00        2.79869E-03
4.55724E-06       1.60742E-03  9.89290E-07   .00000E+00
   10     5.80777E-01  5.82735E-04    .00000E+00        2.87578E-03
4.68278E-06       1.65054E-03  1.67582E-06   .00000E+00
   11     5.80112E-01  9.60720E-04    .00000E+00        2.95468E-03
4.81126E-06       1.69777E-03  2.83863E-06   .00000E+00
   12     5.78577E-01  1.47362E-03    .00000E+00        2.21138E-03
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3.60089E-06       1.27403E-03  3.25873E-06   .00000E+00
   13    5.30259E-01  2.07768E-03   .00000E+00       2.08639E-03
3.39737E-06       1.31155E-03  4.33484E-06   .00000E+00
   14    4.84689E-01  2.83608E-03   .00000E+00       2.04768E-03
3.33434E-06       1.40824E-03  5.80738E-06   .00000E+00
   15    4.56193E-01  3.34009E-03   .00000E+00       2.49668E-04
4.06547E-07       1.82428E-04  8.33913E-07   .00000E+00
   16    4.50003E-01  3.46266E-03   .00000E+00       2.57262E-04
4.18912E-07       1.90563E-04  8.90811E-07   .00000E+00
   17    4.42698E-01  3.60542E-03   .00000E+00       3.22847E-04
5.25708E-07       2.43090E-04  1.16400E-06   .00000E+00
   18    4.19805E-01  4.05969E-03   .00000E+00       1.39779E-03
2.27608E-06       1.10987E-03  5.67458E-06   .00000E+00
   19    3.66218E-01  5.17320E-03   .00000E+00       2.34848E-03
3.82415E-06       2.13760E-03  1.21492E-05   .00000E+00
   20    3.18091E-01  6.30426E-03   .00000E+00       1.10123E-03
1.79318E-06       1.15400E-03  6.94242E-06   .00000E+00
   21    2.74446E-01  7.62670E-03   .00000E+00       5.74069E-03
9.34785E-06       6.97247E-03  4.37825E-05   .00000E+00
   22    2.07048E-01  1.08270E-02   .00000E+00       8.27482E-02
1.34743E-04       1.33219E-01  8.95914E-04   .00000E+00
   23    1.64120E-01  1.40700E-02   .00000E+00       1.02892E-01
1.67544E-04       2.08977E-01  1.44768E-03   .00000E+00
   24    1.30412E-01  1.78227E-02   .00000E+00       1.06586E-01
1.73560E-04       2.72434E-01  1.89966E-03   .00000E+00
   25    1.01309E-01  2.37277E-02   .00000E+00       8.58692E-02
1.39825E-04       2.82532E-01  2.03748E-03   .00000E+00
   26    7.46682E-02  3.97555E-02   .00000E+00       4.68100E-02
7.62230E-05       2.08969E-01  1.86095E-03   .00000E+00

 THERMAL 1.29612E-01  1.89003E-02   .00000E+00       4.34096E-01
7.06860E-04       1.11640E+00  8.20456E-03   .00000E+00

  TOTAL  1.47472E-01  1.62542E-02   .00000E+00       5.07329E-01
8.26108E-04       1.14672E+00  8.24620E-03   .00000E+00


0REGION 87     MATERIAL COOLANT     VOLUME  7.106281E+01

                    CROSS-SECTIONS                               FLUXES
                      REACTIONS
 GROUP   DIFFUSION   ABSORPTION   NU*FISSION           RIF           RAF
        TRANSPORT   ABSORPTION   NU*FISSION

    1    2.70642E+00  2.17943E-03   .00000E+00       5.20173E-04
7.31990E-06       6.40665E-05  1.13368E-06   .00000E+00
    2    2.01031E+00  1.01023E-05   .00000E+00       1.25507E-03
1.76614E-05       2.08106E-04  1.26791E-08   .00000E+00
    3    1.31913E+00  3.55934E-06   .00000E+00       8.25480E-04
1.16162E-05       2.08592E-04  2.93817E-09   .00000E+00
    4    7.92566E-01  3.59323E-06   .00000E+00       4.87782E-04
6.86410E-06       2.05149E-04  1.75272E-09   .00000E+00
    5    6.54618E-01  6.63753E-06   .00000E+00       3.30754E-04
4.65438E-06       1.68421E-04  2.19539E-09   .00000E+00
    6    5.56182E-01  1.94203E-05   .00000E+00       4.78684E-04
6.73607E-06       2.86887E-04  9.29621E-09   .00000E+00
    7    5.63543E-01  6.29493E-05   .00000E+00       4.33802E-04
6.10448E-06       2.56592E-04  2.73075E-08   .00000E+00
    8    5.77544E-01  1.71416E-04   .00000E+00       4.44406E-04
6.25371E-06       2.56492E-04  7.61783E-08   .00000E+00
    9    5.80367E-01  3.53087E-04   .00000E+00       2.28943E-04
3.22169E-06       1.31493E-04  8.08366E-08   .00000E+00
   10    5.80777E-01  5.82085E-04   .00000E+00       2.34484E-04
3.29967E-06       1.34581E-04  1.36490E-07   .00000E+00
   11    5.80114E-01  9.59742E-04   .00000E+00       2.40041E-04
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3.37787E-06       1.37927E-04  2.30377E-07   .00000E+00
   12    5.78579E-01  1.47308E-03   .00000E+00       1.79048E-04
2.51957E-06       1.03154E-04  2.63751E-07   .00000E+00
   13    5.30274E-01  2.07713E-03   .00000E+00       1.68542E-04
2.37174E-06       1.05947E-04  3.50084E-07   .00000E+00
   14    4.84734E-01  2.83531E-03   .00000E+00       1.65114E-04
2.32349E-06       1.13543E-04  4.68149E-07   .00000E+00
   15    4.56194E-01  3.34008E-03   .00000E+00       2.01108E-05
2.83000E-07       1.46946E-05  6.71715E-08   .00000E+00
   16    4.50003E-01  3.46267E-03   .00000E+00       2.07188E-05
2.91556E-07       1.53471E-05  7.17423E-08   .00000E+00
   17    4.42700E-01  3.60538E-03   .00000E+00       2.59964E-05
3.65822E-07       1.95741E-05  9.37269E-08   .00000E+00
   18    4.19829E-01  4.05921E-03   .00000E+00       1.12489E-04
1.58295E-06       8.93134E-05  4.56617E-07   .00000E+00
   19    3.66366E-01  5.17002E-03   .00000E+00       1.88913E-04
2.65840E-06       1.71880E-04  9.76685E-07   .00000E+00
   20    3.18292E-01  6.29875E-03   .00000E+00       8.91269E-05
1.25420E-06       9.33387E-05  5.61388E-07   .00000E+00
   21    2.76329E-01  7.56452E-03   .00000E+00       4.64010E-04
6.52957E-06       5.59731E-04  3.51001E-06   .00000E+00
   22    2.07862E-01  1.07793E-02   .00000E+00       6.59138E-03
9.27543E-05       1.05701E-02  7.10502E-05   .00000E+00
   23    1.64184E-01  1.40641E-02   .00000E+00       8.16303E-03
1.14871E-04       1.65729E-02  1.14806E-04   .00000E+00
   24    1.30440E-01  1.78189E-02   .00000E+00       8.45960E-03
1.19044E-04       2.16181E-02  1.50741E-04   .00000E+00
   25    1.01321E-01  2.37240E-02   .00000E+00       6.81550E-03
9.59082E-05       2.24221E-02  1.61691E-04   .00000E+00
   26    7.46679E-02  3.97569E-02   .00000E+00       3.71530E-03
5.22819E-05       1.65859E-02  1.47709E-04   .00000E+00

 THERMAL 1.29756E-01  1.88780E-02   .00000E+00       3.44869E-02
4.85301E-04       8.85941E-02  6.51044E-04   .00000E+00

  TOTAL  1.48746E-01  1.60982E-02   .00000E+00       4.06585E-02
5.72149E-04       9.11139E-02  6.54529E-04   .00000E+00


0REGION 88     MATERIAL MODERATOR   VOLUME  5.132737E+02

                    CROSS-SECTIONS                              FLUXES
                      REACTIONS
 GROUP   DIFFUSION    ABSORPTION   NU*FISSION           RIF            RAF
       TRANSPORT    ABSORPTION   NU*FISSION

    1    2.74822E+00  2.21855E-03   .00000E+00       3.16562E-03
6.16750E-06       3.83959E-04  7.02308E-06   .00000E+00
    2    2.00863E+00  9.66120E-06   .00000E+00       7.30664E-03
1.42354E-05       1.21254E-03  7.05909E-08   .00000E+00
    3    1.31914E+00  3.55843E-06   .00000E+00       4.74525E-03
9.24506E-06       1.19908E-03  1.68856E-08   .00000E+00
    4    7.92343E-01  3.59826E-06   .00000E+00       2.79868E-03
5.45260E-06       1.17739E-03  1.00704E-08   .00000E+00
    5    6.53939E-01  6.66158E-06   .00000E+00       1.89541E-03
3.69278E-06       9.66148E-04  1.26264E-08   .00000E+00
    6    5.56006E-01  1.95488E-05   .00000E+00       2.73472E-03
5.32799E-06       1.63950E-03  5.34604E-08   .00000E+00
    7    5.63650E-01  6.32557E-05   .00000E+00       2.47195E-03
4.81605E-06       1.46187E-03  1.56365E-07   .00000E+00
    8    5.77572E-01  1.72185E-04   .00000E+00       2.52265E-03
4.91483E-06       1.45589E-03  4.34363E-07   .00000E+00
    9    5.80369E-01  3.53483E-04   .00000E+00       1.29787E-03
2.52862E-06       7.45431E-04  4.58777E-07   .00000E+00
   10    5.80777E-01  5.82735E-04   .00000E+00       1.32561E-03
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2.58265E-06       7.60825E-04  7.72478E-07   .00000E+00
   11    5.80112E-01  9.60720E-04   .00000E+00       1.35367E-03
2.63732E-06       7.77820E-04  1.30050E-06   .00000E+00
   12    5.78577E-01  1.47362E-03   .00000E+00       1.00800E-03
1.96386E-06       5.80734E-04  1.48541E-06   .00000E+00
   13    5.30259E-01  2.07768E-03   .00000E+00       9.47411E-04
1.84582E-06       5.95565E-04  1.96842E-06   .00000E+00
   14    4.84689E-01  2.83608E-03   .00000E+00       9.26945E-04
1.80595E-06       6.37485E-04  2.62889E-06   .00000E+00
   15    4.56193E-01  3.34009E-03   .00000E+00       1.12870E-04
2.19903E-07       8.24727E-05  3.76997E-07   .00000E+00
   16    4.50003E-01  3.46266E-03   .00000E+00       1.16259E-04
2.26506E-07       8.61175E-05  4.02567E-07   .00000E+00
   17    4.42698E-01  3.60542E-03   .00000E+00       1.45843E-04
2.84142E-07       1.09814E-04  5.25824E-07   .00000E+00
   18    4.19805E-01  4.05969E-03   .00000E+00       6.30724E-04
1.22883E-06       5.00807E-04  2.56055E-06   .00000E+00
   19    3.66218E-01  5.17320E-03   .00000E+00       1.05742E-03
2.06015E-06       9.62470E-04  5.47026E-06   .00000E+00
   20    3.18091E-01  6.30426E-03   .00000E+00       4.93214E-04
9.60918E-07       5.16848E-04  3.10935E-06   .00000E+00
   21    2.74446E-01  7.62670E-03   .00000E+00       2.53010E-03
4.92934E-06       3.07299E-03  1.92963E-05   .00000E+00
   22    2.07048E-01  1.08270E-02   .00000E+00       3.61502E-02
7.04307E-05       5.81993E-02  3.91398E-04   .00000E+00
   23    1.64120E-01  1.40700E-02   .00000E+00       4.49150E-02
8.75069E-05       9.12238E-02  6.31953E-04   .00000E+00
   24    1.30412E-01  1.78227E-02   .00000E+00       4.65256E-02
9.06449E-05       1.18919E-01  8.29214E-04   .00000E+00
   25    1.01309E-01  2.37277E-02   .00000E+00       3.74844E-02
7.30300E-05       1.23334E-01  8.89417E-04   .00000E+00
   26    7.46682E-02  3.97555E-02   .00000E+00       2.04352E-02
3.98135E-05       9.12267E-02  8.12412E-04   .00000E+00

 THERMAL 1.29647E-01  1.88947E-02   .00000E+00       1.89591E-01
3.69376E-04       4.87455E-01  3.58227E-03   .00000E+00

  TOTAL  1.49518E-01  1.60043E-02   .00000E+00       2.25097E-01
4.38552E-04       5.01828E-01  3.60253E-03   .00000E+00


0REGION 89     MATERIAL COOLANT     VOLUME  7.558652E+01

                        CROSS-SECTIONS                            FLUXES
                          REACTIONS
 GROUP   DIFFUSION    ABSORPTION   NU*FISSION           RIF            RAF
        TRANSPORT    ABSORPTION   NU*FISSION

    1    2.70642E+00  2.17943E-03   .00000E+00       3.92272E-04
5.18971E-06       4.83138E-05  8.54929E-07   .00000E+00
    2    2.01031E+00  1.01023E-05   .00000E+00       8.65438E-04
1.14496E-05       1.43500E-04  8.74294E-09   .00000E+00
    3    1.31913E+00  3.55934E-06   .00000E+00       5.54534E-04
7.33641E-06       1.40126E-04  1.97378E-09   .00000E+00
    4    7.92566E-01  3.59323E-06   .00000E+00       3.26287E-04
4.31674E-06       1.37228E-04  1.17242E-09   .00000E+00
    5    6.54618E-01  6.63753E-06   .00000E+00       2.20811E-04
2.92130E-06       1.12438E-04  1.46564E-09   .00000E+00
    6    5.56182E-01  1.94203E-05   .00000E+00       3.18050E-04
4.20777E-06       1.90615E-04  6.17665E-09   .00000E+00
    7    5.63543E-01  6.29493E-05   .00000E+00       2.86420E-04
3.78930E-06       1.69416E-04  1.80300E-08   .00000E+00
    8    5.77544E-01  1.71416E-04   .00000E+00       2.91246E-04
3.85315E-06       1.68094E-04  4.99242E-08   .00000E+00
    9    5.80367E-01  3.53087E-04   .00000E+00       1.49291E-04
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1.97511E-06       8.57454E-05  5.27128E-08   .00000E+00
   10    5.80777E-01  5.82085E-04   .00000E+00        1.52266E-04
2.01446E-06       8.73920E-05  8.86316E-08   .00000E+00
   11    5.80114E-01  9.59742E-04   .00000E+00        1.55201E-04
2.05330E-06       8.91787E-05  1.48953E-07   .00000E+00
   12    5.78579E-01  1.47308E-03   .00000E+00        1.15344E-04
1.52599E-06       6.64526E-05  1.69911E-07   .00000E+00
   13    5.30274E-01  2.07713E-03   .00000E+00        1.08279E-04
1.43252E-06       6.80648E-05  2.24910E-07   .00000E+00
   14    4.84734E-01  2.83531E-03   .00000E+00        1.05840E-04
1.40025E-06       7.27822E-05  3.00090E-07   .00000E+00
   15    4.56194E-01  3.34008E-03   .00000E+00        1.28789E-05
1.70387E-07       9.41042E-06  4.30166E-08   .00000E+00
   16    4.50003E-01  3.46267E-03   .00000E+00        1.32646E-05
1.75489E-07       9.82559E-06  4.59310E-08   .00000E+00
   17    4.42700E-01  3.60538E-03   .00000E+00        1.66389E-05
2.20131E-07       1.25284E-05  5.99897E-08   .00000E+00
   18    4.19829E-01  4.05921E-03   .00000E+00        7.19400E-05
9.51757E-07       5.71184E-05  2.92019E-07   .00000E+00
   19    3.66366E-01  5.17002E-03   .00000E+00        1.20592E-04
1.59541E-06       1.09719E-04  6.23461E-07   .00000E+00
   20    3.18292E-01  6.29875E-03   .00000E+00        5.64631E-05
7.47000E-07       5.91314E-05  3.55647E-07   .00000E+00
   21    2.76329E-01  7.56452E-03   .00000E+00        2.87086E-04
3.79811E-06       3.46309E-04  2.17166E-06   .00000E+00
   22    2.07862E-01  1.07793E-02   .00000E+00        4.02740E-03
5.32820E-05       6.45845E-03  4.34124E-05   .00000E+00
   23    1.64184E-01  1.40641E-02   .00000E+00        4.98241E-03
6.59166E-05       1.01155E-02  7.00732E-05   .00000E+00
   24    1.30440E-01  1.78189E-02   .00000E+00        5.16228E-03
6.82963E-05       1.31920E-02  9.19861E-05   .00000E+00
   25    1.01321E-01  2.37240E-02   .00000E+00        4.15876E-03
5.50199E-05       1.36818E-02  9.86625E-05   .00000E+00
   26    7.46679E-02  3.97569E-02   .00000E+00        2.26706E-03
2.99929E-05       1.01206E-02  9.01313E-05   .00000E+00

 THERMAL 1.29812E-01  1.88688E-02   .00000E+00        2.10621E-02
2.78648E-04       5.40835E-02  3.97416E-04   .00000E+00

  TOTAL  1.50776E-01  1.58531E-02   .00000E+00        2.52181E-02
3.33632E-04       5.57517E-02  3.99785E-04   .00000E+00


0REGION 90     MATERIAL MODERATOR   VOLUME  4.653951E+02

                     CROSS-SECTIONS                           FLUXES
                       REACTIONS
 GROUP   DIFFUSION    ABSORPTION   NU*FISSION         RIF            RAF
       TRANSPORT    ABSORPTION   NU*FISSION

    1    2.74822E+00  2.21855E-03   .00000E+00        2.10358E-03
4.51999E-06       2.55144E-04  4.66690E-06   .00000E+00
    2    2.00863E+00  9.66120E-06   .00000E+00        4.46152E-03
9.58651E-06       7.40393E-04  4.31036E-08   .00000E+00
    3    1.31914E+00  3.55843E-06   .00000E+00        2.83391E-03
6.08925E-06       7.16100E-04  1.00843E-08   .00000E+00
    4    7.92343E-01  3.59826E-06   .00000E+00        1.66708E-03
3.58208E-06       7.01331E-04  5.99860E-09   .00000E+00
    5    6.53939E-01  6.66158E-06   .00000E+00        1.12728E-03
2.42220E-06       5.74609E-04  7.50945E-09   .00000E+00
    6    5.56006E-01  1.95488E-05   .00000E+00        1.62074E-03
3.48250E-06       9.71656E-04  3.16835E-08   .00000E+00
    7    5.63650E-01  6.32557E-05   .00000E+00        1.45791E-03
3.13263E-06       8.62186E-04  9.22213E-08   .00000E+00
    8    5.77572E-01  1.72185E-04   .00000E+00        1.47925E-03
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 3.17849E-06       8.53718E-04  2.54705E-07   .00000E+00
     9     5.80369E-01  3.53483E-04   .00000E+00       7.58046E-04
 1.62882E-06       4.35382E-04  2.67957E-07   .00000E+00
    10     5.80777E-01  5.82735E-04   .00000E+00       7.71689E-04
 1.65814E-06       4.42907E-04  4.49690E-07   .00000E+00
    11     5.80112E-01  9.60720E-04   .00000E+00       7.85309E-04
 1.68740E-06       4.51240E-04  7.54462E-07   .00000E+00
    12     5.78577E-01  1.47362E-03   .00000E+00       5.83055E-04
 1.25282E-06       3.35914E-04  8.59204E-07   .00000E+00
    13     5.30259E-01  2.07768E-03   .00000E+00       5.46805E-04
 1.17493E-06       3.43735E-04  1.13609E-06   .00000E+00
    14     4.84689E-01  2.83608E-03   .00000E+00       5.34035E-04
 1.14749E-06       3.67270E-04  1.51457E-06   .00000E+00
    15     4.56193E-01  3.34009E-03   .00000E+00       6.49769E-05
 1.39617E-07       4.74776E-05  2.17029E-07   .00000E+00
    16     4.50003E-01  3.46266E-03   .00000E+00       6.69128E-05
 1.43776E-07       4.95647E-05  2.31697E-07   .00000E+00
    17     4.42698E-01  3.60542E-03   .00000E+00       8.39213E-05
 1.80323E-07       6.31892E-05  3.02571E-07   .00000E+00
    18     4.19805E-01  4.05969E-03   .00000E+00       3.62700E-04
 7.79338E-07       2.87991E-04  1.47245E-06   .00000E+00
    19     3.66218E-01  5.17320E-03   .00000E+00       6.07101E-04
 1.30449E-06       5.52586E-04  3.14066E-06   .00000E+00
    20     3.18091E-01  6.30426E-03   .00000E+00       2.81017E-04
 6.03826E-07       2.94483E-04  1.77161E-06   .00000E+00
    21     2.74446E-01  7.62670E-03   .00000E+00       1.40429E-03
 3.01740E-06       1.70560E-03  1.07101E-05   .00000E+00
    22     2.07048E-01  1.08270E-02   .00000E+00       1.97717E-02
 4.24837E-05       3.18311E-02  2.14068E-04   .00000E+00
    23     1.64120E-01  1.40700E-02   .00000E+00       2.45324E-02
 5.27130E-05       4.98261E-02  3.45170E-04   .00000E+00
    24     1.30412E-01  1.78227E-02   .00000E+00       2.54053E-02
 5.45886E-05       6.49358E-02  4.52792E-04   .00000E+00
    25     1.01309E-01  2.37277E-02   .00000E+00       2.04668E-02
 4.39772E-05       6.73411E-02  4.85629E-04   .00000E+00
    26     7.46682E-02  3.97555E-02   .00000E+00       1.11579E-02
 2.39750E-05       4.98108E-02  4.43586E-04   .00000E+00

  THERMAL 1.29713E-01  1.88839E-02   .00000E+00       1.03626E-01
 2.22663E-04       2.66298E-01  1.95687E-03   .00000E+00

   TOTAL  1.51548E-01  1.57617E-02   .00000E+00       1.24935E-01
 2.68450E-04       2.74797E-01  1.96919E-03   .00000E+00


 0REGION 91     MATERIAL COOLANT     VOLUME  7.954496E+01

                         CROSS-SECTIONS                              FLUXES
                             REACTIONS
  GROUP   DIFFUSION    ABSORPTION   NU*FISSION          RIF              RAF
        TRANSPORT    ABSORPTION   NU*FISSION

     1     2.70642E+00  2.17943E-03   .00000E+00       3.13644E-04
 3.94298E-06       3.86296E-05  6.83565E-07   .00000E+00
     2     2.01031E+00  1.01023E-05   .00000E+00       6.39679E-04
 8.04173E-06       1.06067E-04  6.46225E-09   .00000E+00
     3     1.31913E+00  3.55934E-06   .00000E+00       4.02576E-04
 5.06099E-06       1.01728E-04  1.43291E-09   .00000E+00
     4     7.92566E-01  3.59323E-06   .00000E+00       2.36643E-04
 2.97496E-06       9.95263E-05  8.50314E-10   .00000E+00
     5     6.54618E-01  6.63753E-06   .00000E+00       1.59963E-04
 2.01098E-06       8.14536E-05  1.06176E-09   .00000E+00
     6     5.56182E-01  1.94203E-05   .00000E+00       2.29830E-04
 2.88932E-06       1.37743E-04  4.46339E-09   .00000E+00
     7     5.63543E-01  6.29493E-05   .00000E+00       2.06254E-04
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2.59293E-06       1.21999E-04  1.29836E-08   .00000E+00
    8     5.77544E-01  1.71416E-04   .00000E+00       2.08855E-04
2.62562E-06       1.20542E-04  3.58010E-08   .00000E+00
    9     5.80367E-01  3.53087E-04   .00000E+00       1.06752E-04
1.34204E-06       6.13132E-05  3.76929E-08   .00000E+00
   10     5.80777E-01  5.82085E-04   .00000E+00       1.08614E-04
1.36544E-06       6.23383E-05  6.32225E-08   .00000E+00
   11     5.80114E-01  9.59742E-04   .00000E+00       1.10422E-04
1.38817E-06       6.34486E-05  1.05977E-07   .00000E+00
   12     5.78579E-01  1.47308E-03   .00000E+00       8.18837E-05
1.02940E-06       4.71752E-05  1.20621E-07   .00000E+00
   13     5.30274E-01  2.07713E-03   .00000E+00       7.67413E-05
9.64754E-07       4.82400E-05  1.59402E-07   .00000E+00
   14     4.84734E-01  2.83531E-03   .00000E+00       7.49112E-05
9.41747E-07       5.15136E-05  2.12397E-07   .00000E+00
   15     4.56194E-01  3.34008E-03   .00000E+00       9.11014E-06
1.14528E-07       6.65663E-06  3.04286E-08   .00000E+00
   16     4.50003E-01  3.46267E-03   .00000E+00       9.38131E-06
1.17937E-07       6.94907E-06  3.24843E-08   .00000E+00
   17     4.42700E-01  3.60538E-03   .00000E+00       1.17659E-05
1.47914E-07       8.85916E-06  4.24204E-08   .00000E+00
   18     4.19829E-01  4.05921E-03   .00000E+00       5.08460E-05
6.39211E-07       4.03704E-05  2.06394E-07   .00000E+00
   19     3.66366E-01  5.17002E-03   .00000E+00       8.51110E-05
1.06997E-06       7.74372E-05  4.40025E-07   .00000E+00
   20     3.18292E-01  6.29875E-03   .00000E+00       3.95221E-05
4.96853E-07       4.13898E-05  2.48940E-07   .00000E+00
   21     2.76329E-01  7.56452E-03   .00000E+00       1.95538E-04
2.45821E-06       2.35876E-04  1.47915E-06   .00000E+00
   22     2.07862E-01  1.07793E-02   .00000E+00       2.70255E-03
3.39751E-05       4.33388E-03  2.91315E-05   .00000E+00
   23     1.64184E-01  1.40641E-02   .00000E+00       3.33918E-03
4.19785E-05       6.77934E-03  4.69626E-05   .00000E+00
   24     1.30440E-01  1.78189E-02   .00000E+00       3.45877E-03
4.34819E-05       8.83873E-03  6.16315E-05   .00000E+00
   25     1.01321E-01  2.37240E-02   .00000E+00       2.78620E-03
3.50267E-05       9.16621E-03  6.60997E-05   .00000E+00
   26     7.46679E-02  3.97569E-02   .00000E+00       1.51884E-03
1.90942E-05       6.78044E-03  6.03845E-05   .00000E+00

 THERMAL 1.29880E-01  1.88577E-02   .00000E+00       1.41257E-02
1.77581E-04       3.62533E-02  2.66378E-04   .00000E+00

  TOTAL  1.52737E-01  1.56224E-02   .00000E+00       1.71636E-02
2.15772E-04       3.74579E-02  2.68136E-04   .00000E+00


0REGION 92     MATERIAL MODERATOR   VOLUME  6.567189E+02

                    CROSS-SECTIONS                             FLUXES
                      REACTIONS
 GROUP   DIFFUSION    ABSORPTION   NU*FISSION          RIF           RAF
        TRANSPORT    ABSORPTION   NU*FISSION

    1     2.74822E+00  2.21855E-03   .00000E+00       2.22659E-03
3.39048E-06       2.70064E-04  4.93980E-06   .00000E+00
    2     2.00863E+00  9.66120E-06   .00000E+00       4.31734E-03
6.57410E-06       7.16466E-04  4.17107E-08   .00000E+00
    3     1.31914E+00  3.55843E-06   .00000E+00       2.69355E-03
4.10152E-06       6.80632E-04  9.58480E-09   .00000E+00
    4     7.92343E-01  3.59826E-06   .00000E+00       1.58481E-03
2.41322E-06       6.66719E-04  5.70256E-09   .00000E+00
    5     6.53939E-01  6.66158E-06   .00000E+00       1.07055E-03
1.63015E-06       5.45693E-04  7.13155E-09   .00000E+00
    6     5.56006E-01  1.95488E-05   .00000E+00       1.53568E-03
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2.33841E-06       9.20661E-04  3.00207E-08   .00000E+00
     7     5.63650E-01  6.32557E-05   .00000E+00       1.37714E-03
2.09700E-06       8.14416E-04  8.71118E-08   .00000E+00
     8     5.77572E-01  1.72185E-04   .00000E+00       1.39200E-03
2.11962E-06       8.03361E-04  2.39681E-07   .00000E+00
     9     5.80369E-01  3.53483E-04   .00000E+00       7.11558E-04
1.08350E-06       4.08681E-04  2.51524E-07   .00000E+00
    10     5.80777E-01  5.82735E-04   .00000E+00       7.22715E-04
1.10049E-06       4.14798E-04  4.21151E-07   .00000E+00
    11     5.80112E-01  9.60720E-04   .00000E+00       7.33693E-04
1.11721E-06       4.21582E-04  7.04874E-07   .00000E+00
    12     5.78577E-01  1.47362E-03   .00000E+00       5.43616E-04
8.27776E-07       3.13191E-04  8.01085E-07   .00000E+00
    13     5.30259E-01  2.07768E-03   .00000E+00       5.09025E-04
7.75103E-07       3.19985E-04  1.05759E-06   .00000E+00
    14     4.84689E-01  2.83608E-03   .00000E+00       4.96502E-04
7.56035E-07       3.41458E-04  1.40812E-06   .00000E+00
    15     4.56193E-01  3.34009E-03   .00000E+00       6.03785E-05
9.19396E-08       4.41176E-05  2.01670E-07   .00000E+00
    16     4.50003E-01  3.46266E-03   .00000E+00       6.21669E-05
9.46628E-08       4.60492E-05  2.15263E-07   .00000E+00
    17     4.42698E-01  3.60542E-03   .00000E+00       7.79566E-05
1.18706E-07       5.86980E-05  2.81066E-07   .00000E+00
    18     4.19805E-01  4.05969E-03   .00000E+00       3.36761E-04
5.12793E-07       2.67395E-04  1.36715E-06   .00000E+00
    19     3.66218E-01  5.17320E-03   .00000E+00       5.62812E-04
8.57006E-07       5.12274E-04  2.91154E-06   .00000E+00
    20     3.18091E-01  6.30426E-03   .00000E+00       2.57918E-04
3.92737E-07       2.70277E-04  1.62598E-06   .00000E+00
    21     2.74446E-01  7.62670E-03   .00000E+00       1.24698E-03
1.89881E-06       1.51455E-03  9.51037E-06   .00000E+00
    22     2.07048E-01  1.08270E-02   .00000E+00       1.72569E-02
2.62775E-05       2.77824E-02  1.86841E-04   .00000E+00
    23     1.64120E-01  1.40700E-02   .00000E+00       2.13860E-02
3.25650E-05       4.34357E-02  3.00901E-04   .00000E+00
    24     1.30412E-01  1.78227E-02   .00000E+00       2.21403E-02
3.37135E-05       5.65904E-02  3.94600E-04   .00000E+00
    25     1.01309E-01  2.37277E-02   .00000E+00       1.78351E-02
2.71580E-05       5.86824E-02  4.23186E-04   .00000E+00
    26     7.46682E-02  3.97555E-02   .00000E+00       9.72334E-03
1.48059E-05       4.34069E-02  3.86556E-04   .00000E+00

 THERMAL 1.29790E-01  1.88712E-02   .00000E+00       9.04094E-02
1.37668E-04       2.32195E-01  1.70613E-03   .00000E+00

  TOTAL  1.53815E-01  1.54986E-02   .00000E+00       1.10861E-01
1.68811E-04       2.40249E-01  1.71820E-03   .00000E+00


0REGION 93     MATERIAL COOLANT     VOLUME  8.463431E+01

                        CROSS-SECTIONS                           FLUXES
                          REACTIONS
 GROUP   DIFFUSION    ABSORPTION   NU*FISSION          RIF            RAF
       TRANSPORT    ABSORPTION   NU*FISSION

     1     2.70642E+00  2.17943E-03   .00000E+00       2.49704E-04
2.95039E-06       3.07545E-05  5.44212E-07   .00000E+00
     2     2.01031E+00  1.01023E-05   .00000E+00       4.60449E-04
5.44045E-06       7.63481E-05  4.65161E-09   .00000E+00
     3     1.31913E+00  3.55934E-06   .00000E+00       2.84627E-04
3.36303E-06       7.19229E-05  1.01309E-09   .00000E+00
     4     7.92566E-01  3.59323E-06   .00000E+00       1.67551E-04
1.97970E-06       7.04677E-05  6.02049E-10   .00000E+00
     5     6.54618E-01  6.63753E-06   .00000E+00       1.13162E-04
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 1.33707E-06       5.76222E-05  7.51115E-10   .00000E+00
    6     5.56182E-01  1.94203E-05   .00000E+00      1.62270E-04
 1.91731E-06       9.72525E-05  3.15135E-09   .00000E+00
    7     5.63543E-01  6.29493E-05   .00000E+00      1.45207E-04
 1.71570E-06       8.58893E-05  9.14068E-09   .00000E+00
    8     5.77544E-01  1.71416E-04   .00000E+00      1.46519E-04
 1.73120E-06       8.45644E-05  2.51157E-08   .00000E+00
    9     5.80367E-01  3.53087E-04   .00000E+00      7.47089E-05
 8.82726E-07       4.29090E-05  2.63787E-08   .00000E+00
   10     5.80777E-01  5.82085E-04   .00000E+00      7.58498E-05
 8.96206E-07       4.35335E-05  4.41510E-08   .00000E+00
   11     5.80114E-01  9.59742E-04   .00000E+00      7.69380E-05
 9.09064E-07       4.42085E-05  7.38406E-08   .00000E+00
   12     5.78579E-01  1.47308E-03   .00000E+00      5.69419E-05
 6.72799E-07       3.28056E-05  8.38797E-08   .00000E+00
   13     5.30274E-01  2.07713E-03   .00000E+00      5.32871E-05
 6.29616E-07       3.34966E-05  1.10684E-07   .00000E+00
   14     4.84734E-01  2.83531E-03   .00000E+00      5.19543E-05
 6.13869E-07       3.57270E-05  1.47307E-07   .00000E+00
   15     4.56194E-01  3.34008E-03   .00000E+00      6.31510E-06
 7.46163E-08       4.61434E-06  2.10929E-08   .00000E+00
   16     4.50003E-01  3.46267E-03   .00000E+00      6.50205E-06
 7.68252E-08       4.81630E-06  2.25144E-08   .00000E+00
   17     4.42700E-01  3.60538E-03   .00000E+00      8.15355E-06
 9.63386E-08       6.13926E-06  2.93967E-08   .00000E+00
   18     4.19829E-01  4.05921E-03   .00000E+00      3.52200E-05
 4.16143E-07       2.79637E-05  1.42965E-07   .00000E+00
   19     3.66366E-01  5.17002E-03   .00000E+00      5.88646E-05
 6.95517E-07       5.35573E-05  3.04331E-07   .00000E+00
   20     3.18292E-01  6.29875E-03   .00000E+00      2.70380E-05
 3.19468E-07       2.83157E-05  1.70306E-07   .00000E+00
   21     2.76329E-01  7.56452E-03   .00000E+00      1.28818E-04
 1.52206E-06       1.55392E-04  9.74448E-07   .00000E+00
   22     2.07862E-01  1.07793E-02   .00000E+00      1.74270E-03
 2.05909E-05       2.79464E-03  1.87850E-05   .00000E+00
   23     1.64184E-01  1.40641E-02   .00000E+00      2.14933E-03
 2.53955E-05       4.36366E-03  3.02284E-05   .00000E+00
   24     1.30440E-01  1.78189E-02   .00000E+00      2.22544E-03
 2.62948E-05       5.68702E-03  3.96549E-05   .00000E+00
   25     1.01321E-01  2.37240E-02   .00000E+00      1.79251E-03
 2.11795E-05       5.89713E-03  4.25256E-05   .00000E+00
   26     7.46679E-02  3.97569E-02   .00000E+00      9.77173E-04
 1.15458E-05       4.36231E-03  3.88493E-05   .00000E+00

 THERMAL 1.29978E-01  1.88414E-02   .00000E+00      9.10188E-03
 1.07544E-04       2.33420E-02  1.71492E-04   .00000E+00

  TOTAL  1.55378E-01  1.53214E-02   .00000E+00      1.12772E-02
 1.33247E-04       2.41931E-02  1.72783E-04   .00000E+00


0REGION 94     MATERIAL MODERATOR   VOLUME  6.082755E+02

                      CROSS-SECTIONS                               FLUXES
                        REACTIONS
 GROUP  DIFFUSION    ABSORPTION   NU*FISSION          RIF            RAF
        TRANSPORT    ABSORPTION  NU*FISSION

    1     2.74822E+00  2.21855E-03   .00000E+00      1.64585E-03
 2.70576E-06       1.99625E-04  3.65139E-06   .00000E+00
    2     2.00863E+00  9.66120E-06   .00000E+00      2.90474E-03
 4.77537E-06       4.82045E-04  2.80633E-08   .00000E+00
    3     1.31914E+00  3.55843E-06   .00000E+00      1.78370E-03
 2.93239E-06       4.50723E-04  6.34717E-09   .00000E+00
    4     7.92343E-01  3.59826E-06   .00000E+00      1.05166E-03
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1.72892E-06       4.42426E-04  3.78414E-09   .00000E+00
    5    6.53939E-01  6.66158E-06   .00000E+00        7.09916E-04
1.16710E-06       3.61867E-04  4.72916E-09   .00000E+00
    6    5.56006E-01  1.95488E-05   .00000E+00        1.01671E-03
1.67147E-06       6.09534E-04  1.98755E-08   .00000E+00
    7    5.63650E-01  6.32557E-05   .00000E+00        9.09406E-04
1.49506E-06       5.37808E-04  5.75251E-08   .00000E+00
    8    5.77572E-01  1.72185E-04   .00000E+00        9.16307E-04
1.50640E-06       5.28827E-04  1.57774E-07   .00000E+00
    9    5.80369E-01  3.53483E-04   .00000E+00        4.67346E-04
7.68313E-07       2.68419E-04  1.65199E-07   .00000E+00
   10    5.80777E-01  5.82735E-04   .00000E+00        4.73776E-04
7.78885E-07       2.71921E-04  2.76086E-07   .00000E+00
   11    5.80112E-01  9.60720E-04   .00000E+00        4.80012E-04
7.89136E-07       2.75816E-04  4.61157E-07   .00000E+00
   12    5.78577E-01  1.47362E-03   .00000E+00        3.55042E-04
5.83686E-07       2.04549E-04  5.23197E-07   .00000E+00
   13    5.30259E-01  2.07768E-03   .00000E+00        3.32023E-04
5.45843E-07       2.08718E-04  6.89837E-07   .00000E+00
   14    4.84689E-01  2.83608E-03   .00000E+00        3.23522E-04
5.31867E-07       2.22494E-04  9.17533E-07   .00000E+00
   15    4.56193E-01  3.34009E-03   .00000E+00        3.93248E-05
6.46497E-08       2.87340E-05  1.31348E-07   .00000E+00
   16    4.50003E-01  3.46266E-03   .00000E+00        4.04841E-05
6.65556E-08       2.99880E-05  1.40183E-07   .00000E+00
   17    4.42698E-01  3.60542E-03   .00000E+00        5.07604E-05
8.34496E-08       3.82204E-05  1.83012E-07   .00000E+00
   18    4.19805E-01  4.05969E-03   .00000E+00        2.19196E-04
3.60356E-07       1.74046E-04  8.89868E-07   .00000E+00
   19    3.66218E-01  5.17320E-03   .00000E+00        3.65857E-04
6.01466E-07       3.33005E-04  1.89265E-06   .00000E+00
   20    3.18091E-01  6.30426E-03   .00000E+00        1.66104E-04
2.73074E-07       1.74064E-04  1.04716E-06   .00000E+00
   21    2.74446E-01  7.62670E-03   .00000E+00        7.75280E-04
1.27455E-06       9.41631E-04  5.91282E-06   .00000E+00
   22    2.07048E-01  1.08270E-02   .00000E+00        1.05029E-02
1.72667E-05       1.69089E-02  1.13715E-04   .00000E+00
   23    1.64120E-01  1.40700E-02   .00000E+00        1.29909E-02
2.13569E-05       2.63849E-02  1.82781E-04   .00000E+00
   24    1.30412E-01  1.78227E-02   .00000E+00        1.34436E-02
2.21011E-05       3.43617E-02  2.39602E-04   .00000E+00
   25    1.01309E-01  2.37277E-02   .00000E+00        1.08283E-02
1.78016E-05       3.56280E-02  2.56930E-04   .00000E+00
   26    7.46682E-02  3.97555E-02   .00000E+00        5.90342E-03
9.70518E-06       2.63540E-02  2.34693E-04   .00000E+00

 THERMAL 1.29888E-01  1.88549E-02   .00000E+00        5.49763E-02
9.03806E-05       1.41086E-01  1.03657E-03   .00000E+00

  TOTAL  1.56388E-01  1.52102E-02   .00000E+00        6.86961E-02
1.12936E-04       1.46422E-01  1.04488E-03   .00000E+00


0REGION 95     MATERIAL COOLANT     VOLUME  8.915880E+01

                      CROSS-SECTIONS                              FLUXES
                        REACTIONS
 GROUP   DIFFUSION    ABSORPTION   NU*FISSION           RIF            RAF
        TRANSPORT    ABSORPTION  NU*FISSION

    1    2.70642E+00  2.17943E-03   .00000E+00        2.26256E-04
2.53767E-06       2.78665E-05  4.93108E-07   .00000E+00
    2    2.01031E+00  1.01023E-05   .00000E+00        3.83991E-04
4.30682E-06       6.36704E-05  3.87920E-09   .00000E+00
    3    1.31913E+00  3.55934E-06   .00000E+00        2.33681E-04
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2.62095E-06       5.90491E-06  8.31749E-10   .00000E+00
    4     7.92566E-01  3.59323E-06   .00000E+00      1.37837E-04
1.54597E-06       5.79707E-05  4.95279E-10   .00000E+00
    5     6.54618E-01  6.63753E-06   .00000E+00      9.30170E-05
1.04327E-06       4.73645E-05  6.17403E-10   .00000E+00
    6     5.56182E-01  1.94203E-05   .00000E+00      1.33150E-04
1.49340E-06       7.98001E-05  2.58582E-09   .00000E+00
    7     5.63543E-01  6.29493E-05   .00000E+00      1.18825E-04
1.33274E-06       7.02847E-05  7.47998E-09   .00000E+00
    8     5.77544E-01  1.71416E-04   .00000E+00      1.19510E-04
1.34041E-06       6.89757E-05  2.04859E-08   .00000E+00
    9     5.80367E-01  3.53087E-04   .00000E+00      6.08033E-05
6.81966E-07       3.49223E-05  2.14688E-08   .00000E+00
   10     5.80777E-01  5.82085E-04   .00000E+00      6.16183E-05
6.91108E-07       3.53655E-05  3.58671E-08   .00000E+00
   11     5.80114E-01  9.59742E-04   .00000E+00      6.23840E-05
6.99696E-07       3.58458E-05  5.98725E-08   .00000E+00
   12     5.78579E-01  1.47308E-03   .00000E+00      4.60967E-05
5.17017E-07       2.65574E-05  6.79039E-08   .00000E+00
   13     5.30274E-01  2.07713E-03   .00000E+00      4.30872E-05
4.83264E-07       2.70849E-05  8.94978E-08   .00000E+00
   14     4.84734E-01  2.83531E-03   .00000E+00      4.19704E-05
4.70738E-07       2.88615E-05  1.18999E-07   .00000E+00
   15     4.56194E-01  3.34008E-03   .00000E+00      5.09946E-06
5.71952E-08       3.72609E-06  1.70326E-08   .00000E+00
   16     4.50003E-01  3.46267E-03   .00000E+00      5.24980E-06
5.88815E-08       3.88871E-06  1.81783E-08   .00000E+00
   17     4.42700E-01  3.60538E-03   .00000E+00      6.58249E-06
7.38288E-08       4.95632E-06  2.37324E-08   .00000E+00
   18     4.19829E-01  4.05921E-03   .00000E+00      2.84240E-05
3.18802E-07       2.25679E-05  1.15379E-07   .00000E+00
   19     3.66366E-01  5.17002E-03   .00000E+00      4.74510E-05
5.32208E-07       4.31728E-05  2.45323E-07   .00000E+00
   20     3.18292E-01  6.29875E-03   .00000E+00      2.16133E-05
2.42413E-07       2.26346E-05  1.36137E-07   .00000E+00
   21     2.76329E-01  7.56452E-03   .00000E+00      9.98536E-05
1.11995E-06       1.20453E-04  7.55344E-07   .00000E+00
   22     2.07862E-01  1.07793E-02   .00000E+00      1.32597E-03
1.48720E-05       2.12636E-03  1.42930E-05   .00000E+00
   23     1.64184E-01  1.40641E-02   .00000E+00      1.63268E-03
1.83120E-05       3.31473E-03  2.29622E-05   .00000E+00
   24     1.30440E-01  1.78189E-02   .00000E+00      1.68989E-03
1.89537E-05       4.31844E-03  3.01120E-05   .00000E+00
   25     1.01321E-01  2.37240E-02   .00000E+00      1.36102E-03
1.52651E-05       4.47757E-03  3.22888E-05   .00000E+00
   26     7.46679E-02  3.97569E-02   .00000E+00      7.41956E-04
8.32173E-06       3.31225E-03  2.94978E-05   .00000E+00

 THERMAL 1.30067E-01  1.88269E-02   .00000E+00      6.92043E-03
7.76192E-05       1.77356E-02  1.30291E-04   .00000E+00

  TOTAL  1.57821E-01  1.50536E-02   .00000E+00      8.72801E-03
9.78929E-05       1.84344E-02  1.31388E-04   .00000E+00


0REGION 96     MATERIAL MODERATOR   VOLUME  7.336240E+02

                      CROSS-SECTIONS                          FLUXES
                        REACTIONS
 GROUP    DIFFUSION    ABSORPTION   NU*FISSION          RIF            RAF
        TRANSPORT     ABSORPTION   NU*FISSION

    1     2.74822E+00  2.21855E-03   .00000E+00      1.86378E-03
2.54051E-06       2.26059E-04  4.13490E-06   .00000E+00
    2     2.00863E+00  9.66120E-06   .00000E+00      3.03854E-03
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4.14182E-06        5.04249E-04  2.93560E-08   .00000E+00
    3    1.31914E+00  3.55843E-06   .00000E+00        1.81236E-03
2.47041E-06        4.57964E-04  6.44914E-09   .00000E+00
    4    7.92343E-01  3.59826E-06   .00000E+00        1.06611E-03
1.45321E-06        4.48506E-04  3.83615E-09   .00000E+00
    5    6.53939E-01  6.66158E-06   .00000E+00        7.17816E-04
9.78452E-07        3.65893E-04  4.78179E-09   .00000E+00
    6    5.56006E-01  1.95488E-05   .00000E+00        1.02399E-03
1.39579E-06        6.13893E-04  2.00177E-08   .00000E+00
    7    5.63650E-01  6.32557E-05   .00000E+00        9.11383E-04
1.24230E-06        5.38977E-04  5.76502E-08   .00000E+00
    8    5.77572E-01  1.72185E-04   .00000E+00        9.13831E-04
1.24564E-06        5.27397E-04  1.57348E-07   .00000E+00
    9    5.80369E-01  3.53483E-04   .00000E+00        4.64696E-04
6.33426E-07        2.66897E-04  1.64262E-07   .00000E+00
   10    5.80777E-01  5.82735E-04   .00000E+00        4.70004E-04
6.40660E-07        2.69756E-04  2.73888E-07   .00000E+00
   11    5.80112E-01  9.60720E-04   .00000E+00        4.75124E-04
6.47639E-07        2.73007E-04  4.56461E-07   .00000E+00
   12    5.78577E-01  1.47362E-03   .00000E+00        3.50794E-04
4.78165E-07        2.02101E-04  5.16938E-07   .00000E+00
   13    5.30259E-01  2.07768E-03   .00000E+00        3.27619E-04
4.46577E-07        2.05950E-04  6.80688E-07   .00000E+00
   14    4.84689E-01  2.83608E-03   .00000E+00        3.18894E-04
4.34683E-07        2.19312E-04  9.04408E-07   .00000E+00
   15    4.56193E-01  3.34009E-03   .00000E+00        3.87447E-05
5.28127E-08        2.83101E-05  1.29411E-07   .00000E+00
   16    4.50003E-01  3.46266E-03   .00000E+00        3.98815E-05
5.43623E-08        2.95416E-05  1.38096E-07   .00000E+00
   17    4.42698E-01  3.60542E-03   .00000E+00        4.99985E-05
6.81528E-08        3.76468E-05  1.80266E-07   .00000E+00
   18    4.19805E-01  4.05969E-03   .00000E+00        2.15825E-04
2.94190E-07        1.71369E-04  8.76184E-07   .00000E+00
   19    3.66218E-01  5.17320E-03   .00000E+00        3.59818E-04
4.90467E-07        3.27508E-04  1.86141E-06   .00000E+00
   20    3.18091E-01  6.30426E-03   .00000E+00        1.62165E-04
2.21047E-07        1.69936E-04  1.02233E-06   .00000E+00
   21    2.74446E-01  7.62670E-03   .00000E+00        7.36209E-04
1.00352E-06        8.94177E-04  5.61484E-06   .00000E+00
   22    2.07048E-01  1.08270E-02   .00000E+00        9.80512E-03
1.33653E-05        1.57856E-02  1.06160E-04   .00000E+00
   23    1.64120E-01  1.40700E-02   .00000E+00        1.21106E-02
1.65079E-05        2.45970E-02  1.70395E-04   .00000E+00
   24    1.30412E-01  1.78227E-02   .00000E+00        1.25286E-02
1.70776E-05        3.20230E-02  2.23293E-04   .00000E+00
   25    1.01309E-01  2.37277E-02   .00000E+00        1.00905E-02
1.37543E-05        3.32004E-02  2.39424E-04   .00000E+00
   26    7.46682E-02  3.97555E-02   .00000E+00        5.50125E-03
7.49874E-06        2.45587E-02  2.18705E-04   .00000E+00

 THERMAL 1.29968E-01  1.88418E-02   .00000E+00        5.12942E-02
6.99189E-05        1.31556E-01  9.66476E-04   .00000E+00

  TOTAL  1.59175E-01  1.49129E-02   .00000E+00        6.53936E-02
8.91377E-05        1.36943E-01  9.75211E-04   .00000E+00


0REGION 97      MATERIAL COOLANT      VOLUME  9.424856E+01

                      CROSS-SECTIONS                              FLUXES
                        REACTIONS
 GROUP   DIFFUSION    ABSORPTION   NU*FISSION          RIF              RAF
        TRANSPORT    ABSORPTION   NU*FISSION

   1     2.70642E+00  2.17943E-03   .00000E+00        2.55472E-04
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```
2.71062E-06       3.14649E-05  5.56782E-07   .00000E+00
    2    2.01031E+00  1.01023E-05   .00000E+00       4.04802E-04
4.29504E-06       6.71211E-05  4.08944E-09   .00000E+00
    3    1.31913E+00  3.55934E-06   .00000E+00       2.31392E-04
2.45513E-06       5.84709E-05  8.23605E-10   .00000E+00
    4    7.92566E-01  3.59323E-06   .00000E+00       1.34399E-04
1.42600E-06       5.65248E-05  4.82926E-10   .00000E+00
    5    6.54618E-01  6.63753E-06   .00000E+00       9.01017E-05
9.56000E-07       4.58800E-05  5.98053E-10   .00000E+00
    6    5.56182E-01  1.94203E-05   .00000E+00       1.27955E-04
1.35763E-06       7.66865E-05  2.48493E-09   .00000E+00
    7    5.63543E-01  6.29493E-05   .00000E+00       1.13262E-04
1.20174E-06       6.69942E-05  7.12979E-09   .00000E+00
    8    5.77544E-01  1.71416E-04   .00000E+00       1.13189E-04
1.20096E-06       6.53277E-05  1.94024E-08   .00000E+00
    9    5.80367E-01  3.53087E-04   .00000E+00       5.73759E-05
6.08772E-07       3.29538E-05  2.02587E-08   .00000E+00
   10    5.80777E-01  5.82085E-04   .00000E+00       5.80041E-05
6.15437E-07       3.32911E-05  3.37633E-08   .00000E+00
   11    5.80114E-01  9.59742E-04   .00000E+00       5.85965E-05
6.21723E-07       3.36695E-05  5.62375E-08   .00000E+00
   12    5.78579E-01  1.47308E-03   .00000E+00       4.32236E-05
4.58613E-07       2.49022E-05  6.36717E-08   .00000E+00
   13    5.30274E-01  2.07713E-03   .00000E+00       4.03466E-05
4.28087E-07       2.53621E-05  8.38051E-08   .00000E+00
   14    4.84734E-01  2.83531E-03   .00000E+00       3.92538E-05
4.16492E-07       2.69933E-05  1.11297E-07   .00000E+00
   15    4.56194E-01  3.34008E-03   .00000E+00       4.76667E-06
5.05755E-08       3.48293E-06  1.59210E-08   .00000E+00
   16    4.50003E-01  3.46267E-03   .00000E+00       4.90638E-06
5.20579E-08       3.63433E-06  1.69892E-08   .00000E+00
   17    4.42700E-01  3.60538E-03   .00000E+00       6.15108E-06
6.52644E-08       4.63149E-06  2.21770E-08   .00000E+00
   18    4.19829E-01  4.05921E-03   .00000E+00       2.65497E-05
2.81699E-07       2.10797E-05  1.07771E-07   .00000E+00
   19    3.66366E-01  5.17002E-03   .00000E+00       4.42752E-05
4.69770E-07       4.02832E-05  2.28903E-07   .00000E+00
   20    3.18292E-01  6.29875E-03   .00000E+00       2.00716E-05
2.12964E-07       2.10201E-05  1.26426E-07   .00000E+00
   21    2.76329E-01  7.56452E-03   .00000E+00       9.08787E-05
9.64245E-07       1.09626E-04  6.87453E-07   .00000E+00
   22    2.07862E-01  1.07793E-02   .00000E+00       1.18933E-03
1.26191E-05       1.90725E-03  1.28201E-05   .00000E+00
   23    1.64184E-01  1.40641E-02   .00000E+00       1.46207E-03
1.55129E-05       2.96835E-03  2.05627E-05   .00000E+00
   24    1.30440E-01  1.78189E-02   .00000E+00       1.51280E-03
1.60511E-05       3.86589E-03  2.69564E-05   .00000E+00
   25    1.01321E-01  2.37240E-02   .00000E+00       1.21828E-03
1.29262E-05       4.00798E-03  2.89025E-05   .00000E+00
   26    7.46679E-02  3.97569E-02   .00000E+00       6.64148E-04
7.04677E-06       2.96490E-03  2.64045E-05   .00000E+00

 THERMAL 1.30138E-01  1.88152E-02   .00000E+00       6.20185E-03
6.58032E-05       1.58853E-02  1.16689E-04   .00000E+00

  TOTAL  1.61227E-01  1.47053E-02   .00000E+00       8.01160E-03
8.50050E-05       1.65638E-02  1.17813E-04   .00000E+00


0REGION 98     MATERIAL MODERATOR   VOLUME  9.481306E+01

                      CROSS-SECTIONS                               FLUXES
                        REACTIONS
 GROUP   DIFFUSION   ABSORPTION   NU*FISSION           RIF             RAF
        TRANSPORT   ABSORPTION   NU*FISSION
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```
    1   2.74822E+00  2.21855E-03   .00000E+00       2.69373E-04
2.84109E-06       3.26723E-05  5.97617E-07   .00000E+00
    2   2.00863E+00  9.66120E-06   .00000E+00       4.22674E-04
4.45797E-06       7.01431E-05  4.08354E-09   .00000E+00
    3   1.31914E+00  3.55843E-06   .00000E+00       2.36000E-04
2.48911E-06       5.96348E-05  8.39789E-10   .00000E+00
    4   7.92343E-01  3.59826E-06   .00000E+00       1.35873E-04
1.43306E-06       5.71609E-05  4.88907E-10   .00000E+00
    5   6.53939E-01  6.66158E-06   .00000E+00       9.09172E-05
9.58910E-07       4.63434E-05  6.05652E-10   .00000E+00
    6   5.56006E-01  1.95488E-05   .00000E+00       1.28883E-04
1.35934E-06       7.72670E-05  2.51950E-09   .00000E+00
    7   5.63650E-01  6.32557E-05   .00000E+00       1.14070E-04
1.20310E-06       6.74592E-05  7.21558E-09   .00000E+00
    8   5.77572E-01  1.72185E-04   .00000E+00       1.13978E-04
1.20213E-06       6.57798E-05  1.96252E-08   .00000E+00
    9   5.80369E-01  3.53483E-04   .00000E+00       5.78177E-05
6.09808E-07       3.32075E-05  2.04376E-08   .00000E+00
   10   5.80777E-01  5.82735E-04   .00000E+00       5.84220E-05
6.16180E-07       3.35309E-05  3.40445E-08   .00000E+00
   11   5.80112E-01  9.60720E-04   .00000E+00       5.90075E-05
6.22356E-07       3.39058E-05  5.66897E-08   .00000E+00
   12   5.78577E-01  1.47362E-03   .00000E+00       4.35310E-05
4.59125E-07       2.50794E-05  6.41483E-08   .00000E+00
   13   5.30259E-01  2.07768E-03   .00000E+00       4.06167E-05
4.28387E-07       2.55326E-05  8.43884E-08   .00000E+00
   14   4.84689E-01  2.83608E-03   .00000E+00       3.94990E-05
4.16598E-07       2.71645E-05  1.12022E-07   .00000E+00
   15   4.56193E-01  3.34009E-03   .00000E+00       4.79613E-06
5.05851E-08       3.50445E-06  1.60195E-08   .00000E+00
   16   4.50003E-01  3.46266E-03   .00000E+00       4.93625E-06
5.20630E-08       3.65645E-06  1.70926E-08   .00000E+00
   17   4.42698E-01  3.60542E-03   .00000E+00       6.18793E-06
6.52645E-08       4.65925E-06  2.23100E-08   .00000E+00
   18   4.19805E-01  4.05969E-03   .00000E+00       2.67021E-05
2.81629E-07       2.12020E-05  1.08402E-07   .00000E+00
   19   3.66218E-01  5.17320E-03   .00000E+00       4.44939E-05
4.69281E-07       4.04986E-05  2.30176E-07   .00000E+00
   20   3.18091E-01  6.30426E-03   .00000E+00       2.00767E-05
2.11750E-07       2.10387E-05  1.26568E-07   .00000E+00
   21   2.74446E-01  7.62670E-03   .00000E+00       9.05630E-05
9.55174E-07       1.09995E-04  6.90696E-07   .00000E+00
   22   2.07048E-01  1.08270E-02   .00000E+00       1.19170E-03
1.25689E-05       1.91855E-03  1.29025E-05   .00000E+00
   23   1.64120E-01  1.40700E-02   .00000E+00       1.46790E-03
1.54820E-05       2.98135E-03  2.06533E-05   .00000E+00
   24   1.30412E-01  1.78227E-02   .00000E+00       1.51806E-03
1.60111E-05       3.88016E-03  2.70560E-05   .00000E+00
   25   1.01309E-01  2.37277E-02   .00000E+00       1.22242E-03
1.28930E-05       4.02209E-03  2.90052E-05   .00000E+00
   26   7.46682E-02  3.97555E-02   .00000E+00       6.66410E-04
7.02867E-06       2.97498E-03  2.64934E-05   .00000E+00

 THERMAL 1.30034E-01  1.88308E-02   .00000E+00       6.22162E-03
6.56199E-05       1.59487E-02  1.17158E-04   .00000E+00

  TOTAL  1.61790E-01  1.46536E-02   .00000E+00       8.07490E-03
8.51666E-05       1.66366E-02  1.18326E-04   .00000E+00


0CELL               VOLUME  7.995988E+03

                      CROSS-SECTIONS                         FLUXES
                         REACTIONS
 GROUP   DIFFUSION    ABSORPTION   NU*FISSION          RIF          RAF
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        TRANSPORT    ABSORPTION   NU*FISSION

   1    2.60661E+00 -1.45265E-02  2.82809E-03       1.45013E+00
1.81357E-04       1.85443E-01 -2.10653E-02  4.10110E-03
   2    1.91714E+00 -7.66892E-04  1.88392E-03       5.55891E+00
6.95212E-04       9.66530E-01 -4.26308E-03  1.04726E-02
   3    1.21542E+00  5.12088E-04  6.41375E-04       4.87635E+00
6.09850E-04       1.33736E+00  2.49712E-03  3.12757E-03
   4    7.90820E-01  3.12255E-04  4.51114E-04       3.61011E+00
4.51490E-04       1.52167E+00  1.12728E-03  1.62857E-03
   5    6.29399E-01  3.40123E-04  4.32890E-04       2.62315E+00
3.28058E-04       1.38923E+00  8.92192E-04  1.13553E-03
   6    5.42694E-01  5.28634E-04  5.16746E-04       3.97034E+00
4.96542E-04       2.43866E+00  2.09886E-03  2.05166E-03
   7    5.35963E-01  1.27949E-03  1.14639E-03       3.36951E+00
4.21399E-04       2.09561E+00  4.31124E-03  3.86275E-03
   8    5.44887E-01  3.16460E-03  3.19772E-03       3.16104E+00
3.95329E-04       1.93376E+00  1.00034E-02  1.01081E-02
   9    5.45938E-01  5.04773E-03  4.84431E-03       1.54538E+00
1.93270E-04       9.43566E-01  7.80068E-03  7.48632E-03
  10    5.44137E-01  7.80873E-03  9.52948E-03       1.50412E+00
1.88109E-04       9.21411E-01  1.17453E-02  1.43335E-02
  11    5.43658E-01  9.89051E-03  1.05255E-02       1.46390E+00
1.83080E-04       8.97563E-01  1.44787E-02  1.54082E-02
  12    5.43328E-01  1.03218E-02  3.05069E-03       1.05155E+00
1.31509E-04       6.45126E-01  1.08539E-02  3.20794E-03
  13    5.27296E-01  3.61183E-03  4.21363E-03       9.49544E-01
1.18753E-04       6.00260E-01  3.42959E-03  4.00103E-03
  14    5.05961E-01  4.71320E-03  5.35932E-03       9.06741E-01
1.13400E-04       5.97373E-01  4.27365E-03  4.85951E-03
  15    4.85318E-01  1.34107E-02  2.28343E-02       1.08475E-01
1.35661E-05       7.45042E-02  1.45472E-03  2.47694E-03
  16    4.84278E-01  1.02325E-02  1.74214E-02       1.11412E-01
1.39335E-05       7.66863E-02  1.14002E-03  1.94096E-03
  17    4.81623E-01  8.47805E-03  1.40781E-02       1.39786E-01
1.74820E-05       9.67465E-02  1.18511E-03  1.96792E-03
  18    4.70067E-01  8.26582E-03  1.32683E-02       5.98529E-01
7.48537E-05       4.24429E-01  4.94733E-03  7.94144E-03
  19    4.35120E-01  1.25785E-02  2.07411E-02       9.74683E-01
1.21897E-04       7.46678E-01  1.22600E-02  2.02160E-02
  20    3.98554E-01  2.03481E-02  3.37046E-02       3.97529E-01
4.97160E-05       3.32476E-01  8.08894E-03  1.33986E-02
  21    3.64541E-01  1.83131E-02  2.82961E-02       1.08920E+00
1.36219E-04       9.95959E-01  1.99466E-02  3.08202E-02
  22    2.97357E-01  1.66709E-02  2.23249E-02       8.17567E+00
1.02247E-03       9.16482E+00  1.36296E-01  1.82521E-01
  23    2.47680E-01  1.98194E-02  2.51481E-02       9.14492E+00
1.14369E-03       1.23074E+01  1.81246E-01  2.29977E-01
  24    2.06923E-01  2.33379E-02  2.83297E-02       9.26688E+00
1.15894E-03       1.49281E+01  2.16270E-01  2.62528E-01
  25    1.61252E-01  2.96005E-02  3.37657E-02       7.06969E+00
8.84155E-04       1.46142E+01  2.09267E-01  2.38713E-01
  26    1.33712E-01  4.18826E-02  4.05504E-02       3.81339E+00
4.76913E-04       9.50645E+00  1.59715E-01  1.54634E-01

 THERMAL 2.31472E-01  2.36174E-02  2.83685E-02       3.99320E+01
4.99400E-03       6.25961E+01  9.43089E-01  1.13281E+00

  TOTAL  5.75062E-01  1.29987E-02  1.60263E-02       7.69309E+01
9.62119E-03       7.97420E+01  1.00000E+00  1.23292E+00
1CELL AVERAGE SCATTERING CROSS-SECTIONS


        1            2            3            4            5            6
     7            8            9            10
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  1  3.0306E-02  6.8225E-02  2.2027E-02  1.2443E-02  6.2401E-03
2.9085E-03  2.3670E-04  1.8111E-05  1.5161E-06  7.0287E-07
  2   .0000E+00  7.6033E-02  6.8920E-02  1.8226E-02  7.2324E-03
3.7221E-03  4.4129E-04  5.3862E-05  5.1643E-06  1.8925E-06
  3   .0000E+00   .0000E+00  1.3042E-01  1.0519E-01  2.4159E-02
1.2086E-02  1.6319E-03  2.2078E-04  2.1843E-05  8.0355E-06
  4   .0000E+00   .0000E+00   .0000E+00  2.2025E-01  1.4754E-01
4.6175E-02  6.2452E-03  8.4523E-04  8.3628E-05  3.0764E-05
  5   .0000E+00   .0000E+00   .0000E+00   .0000E+00  2.5365E-01
2.4822E-01  2.3688E-02  3.2057E-03  3.1718E-04  1.1668E-04
  6   .0000E+00   .0000E+00   .0000E+00   .0000E+00   .0000E+00
3.8535E-01  2.0319E-01  2.1747E-02  2.1517E-03  7.9154E-04
  7   .0000E+00   .0000E+00   .0000E+00   .0000E+00   .0000E+00
 .0000E+00  3.5401E-01  2.3665E-01  1.8958E-02  6.9738E-03
  8   .0000E+00   .0000E+00   .0000E+00   .0000E+00   .0000E+00
 .0000E+00   .0000E+00  3.2795E-01  1.9369E-01  5.4962E-02
  9   .0000E+00   .0000E+00   .0000E+00   .0000E+00   .0000E+00
 .0000E+00   .0000E+00   .0000E+00  1.6665E-01  3.0947E-01
 10   .0000E+00   .0000E+00   .0000E+00   .0000E+00   .0000E+00
 .0000E+00   .0000E+00   .0000E+00   .0000E+00  1.6106E-01
 11   .0000E+00   .0000E+00   .0000E+00   .0000E+00   .0000E+00
 .0000E+00   .0000E+00   .0000E+00   .0000E+00   .0000E+00
 12   .0000E+00   .0000E+00   .0000E+00   .0000E+00   .0000E+00
 .0000E+00   .0000E+00   .0000E+00   .0000E+00   .0000E+00
 13   .0000E+00   .0000E+00   .0000E+00   .0000E+00   .0000E+00
 .0000E+00   .0000E+00   .0000E+00   .0000E+00   .0000E+00
 14   .0000E+00   .0000E+00   .0000E+00   .0000E+00   .0000E+00
 .0000E+00   .0000E+00   .0000E+00   .0000E+00   .0000E+00
 15   .0000E+00   .0000E+00   .0000E+00   .0000E+00   .0000E+00
 .0000E+00   .0000E+00   .0000E+00   .0000E+00   .0000E+00
 16   .0000E+00   .0000E+00   .0000E+00   .0000E+00   .0000E+00
 .0000E+00   .0000E+00   .0000E+00   .0000E+00   .0000E+00
 17   .0000E+00   .0000E+00   .0000E+00   .0000E+00   .0000E+00
 .0000E+00   .0000E+00   .0000E+00   .0000E+00   .0000E+00
 18   .0000E+00   .0000E+00   .0000E+00   .0000E+00   .0000E+00
 .0000E+00   .0000E+00   .0000E+00   .0000E+00   .0000E+00
 19   .0000E+00   .0000E+00   .0000E+00   .0000E+00   .0000E+00
 .0000E+00   .0000E+00   .0000E+00   .0000E+00   .0000E+00
 20   .0000E+00   .0000E+00   .0000E+00   .0000E+00   .0000E+00
 .0000E+00   .0000E+00   .0000E+00   .0000E+00   .0000E+00
 21   .0000E+00   .0000E+00   .0000E+00   .0000E+00   .0000E+00
 .0000E+00   .0000E+00   .0000E+00   .0000E+00   .0000E+00
 22   .0000E+00   .0000E+00   .0000E+00   .0000E+00   .0000E+00
 .0000E+00   .0000E+00   .0000E+00   .0000E+00   .0000E+00
 23   .0000E+00   .0000E+00   .0000E+00   .0000E+00   .0000E+00
 .0000E+00   .0000E+00   .0000E+00   .0000E+00   .0000E+00
 24   .0000E+00   .0000E+00   .0000E+00   .0000E+00   .0000E+00
 .0000E+00   .0000E+00   .0000E+00   .0000E+00   .0000E+00
 25   .0000E+00   .0000E+00   .0000E+00   .0000E+00   .0000E+00
 .0000E+00   .0000E+00   .0000E+00   .0000E+00   .0000E+00
 26   .0000E+00   .0000E+00   .0000E+00   .0000E+00   .0000E+00
 .0000E+00   .0000E+00   .0000E+00   .0000E+00   .0000E+00


       11          12          13          14          15          16
      17          18          19          20
  1  1.4526E-07  3.4476E-11  5.0813E-12  2.5332E-12  2.0186E-13
1.9164E-13  2.1992E-13  7.4734E-13  7.1181E-13  1.7853E-13
  2  9.6863E-07  1.1370E-07  4.6187E-14  1.6553E-14  2.5896E-09
2.3370E-09  2.4710E-09  6.5949E-09  4.0420E-09  6.6480E-10
  3  2.9560E-06  1.2140E-06  4.7868E-07   .0000E+00   .0000E+00
 .0000E+00   .0000E+00   .0000E+00   .0000E+00   .0000E+00
  4  1.1317E-05  3.4182E-06  1.6841E-06  1.1952E-06  9.6990E-08
8.7154E-08  6.5951E-08   .0000E+00   .0000E+00   .0000E+00
  5  4.2923E-05  1.2964E-05  5.7080E-06  2.8540E-06  2.3733E-07
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2.2532E-07  3.2129E-07  1.3271E-06  1.2837E-06  5.0144E-09
  6  2.9118E-04  8.7946E-05  3.8722E-05  1.9361E-05  1.6100E-06
1.5285E-06  1.7527E-06  5.8083E-06  5.6045E-06  1.6287E-06
  7  2.5655E-03  7.7484E-04  3.4116E-04  1.7058E-04  1.4185E-05
1.3467E-05  1.5442E-05  5.1174E-05  4.9378E-05  1.2569E-05
  8  2.0215E-02  6.1054E-03  2.6881E-03  1.3441E-03  1.1177E-04
1.0611E-04  1.2167E-04  4.0322E-04  3.8907E-04  9.9037E-05
  9  8.1805E-02  2.4700E-02  1.0875E-02  5.4376E-03  4.5218E-04
4.2929E-04  4.9225E-04  1.6313E-03  1.5740E-03  4.0067E-04
 10  3.1223E-01  6.8246E-02  3.0041E-02  1.5019E-02  1.2490E-03
1.1857E-03  1.3596E-03  4.5057E-03  4.3476E-03  1.1067E-03
 11  1.5546E-01  2.7354E-01  8.2776E-02  4.1383E-02  3.4408E-03
3.2665E-03  3.7454E-03  1.2412E-02  1.1976E-02  3.0485E-03
 12   .0000E+00  7.7250E-02  3.1083E-01  9.7320E-02  8.0929E-03
7.6831E-03  8.8100E-03  2.9194E-02  2.8163E-02  7.1683E-03
 13   .0000E+00  3.0174E-04  8.3845E-02  3.2182E-01  1.6269E-02
1.5203E-02  1.7560E-02  5.7811E-02  5.5949E-02  1.3736E-02
 14   .0000E+00   .0000E+00  8.1030E-04  1.0032E-01  7.2769E-02
6.0630E-02  5.9895E-02  1.4139E-01  1.0616E-01  2.6331E-02
 15   .0000E+00   .0000E+00   .0000E+00  1.7009E-02 -2.3601E-01
1.3494E-01  1.3784E-01  3.0620E-01  1.5736E-01  3.6468E-02
 16   .0000E+00   .0000E+00   .0000E+00  8.4332E-04  1.8053E-02
2.2921E-01  1.6358E-01  3.7782E-01  1.7837E-01  3.9396E-02
 17   .0000E+00   .0000E+00   .0000E+00  4.8886E-05  8.8000E-04
1.6939E-02 -2.0339E-01  4.7696E-01  2.0923E-01  4.2745E-02
 18   .0000E+00   .0000E+00   .0000E+00  6.8507E-07  1.0532E-05
1.7345E-04  5.0326E-03  5.5560E-02  4.0572E-01  5.7490E-02
 19   .0000E+00   .0000E+00   .0000E+00  4.6520E-10  1.4682E-09
9.6587E-09  1.4041E-07  6.5765E-03  2.1666E-01  1.8418E-01
 20   .0000E+00   .0000E+00   .0000E+00  1.7678E-10  1.4755E-10
1.8500E-10  3.8835E-10  9.6142E-07  4.4100E-02  4.9110E-02
 21   .0000E+00   .0000E+00   .0000E+00  6.2659E-11  5.4246E-11
6.7307E-11  1.9617E-10  7.4927E-08  1.4899E-03  2.6286E-02
 22   .0000E+00   .0000E+00   .0000E+00  3.9065E-11  4.7418E-11
6.2243E-11  2.2551E-10  9.8791E-09  2.9666E-05  4.0156E-04
 23   .0000E+00   .0000E+00   .0000E+00  3.0207E-11  5.3540E-11
7.5633E-11  2.6466E-10  5.4504E-09  5.3294E-06  6.6736E-05
 24   .0000E+00   .0000E+00   .0000E+00  9.7954E-12  2.1195E-11
8.8659E-11  3.0462E-10  4.0927E-09  2.9256E-06  3.4183E-05
 25   .0000E+00   .0000E+00   .0000E+00   .0000E+00   .0000E+00
4.9145E-12  1.5442E-10  3.5878E-09  2.2090E-06  2.4947E-05
 26   .0000E+00   .0000E+00   .0000E+00   .0000E+00   .0000E+00
4.8887E-12  1.0810E-10  3.8744E-09  2.2303E-06  2.4893E-05


         21          22          23          24          25           26
  1  2.5569E-13  2.5570E-13  7.6410E-14  5.0139E-14  3.5802E-14
3.5325E-14
  2  6.9345E-10  3.9195E-10  5.8792E-11  2.4120E-11  1.0176E-11
3.0150E-12
  3   .0000E+00   .0000E+00   .0000E+00   .0000E+00   .0000E+00
.0000E+00
  4   .0000E+00   .0000E+00   .0000E+00   .0000E+00   .0000E+00
.0000E+00
  5   .0000E+00   .0000E+00   .0000E+00   .0000E+00   .0000E+00
.0000E+00
  6  2.3435E-06  2.3361E-06  6.8666E-07  6.4232E-08   .0000E+00
.0000E+00
  7  1.7956E-05  1.7956E-05  5.3867E-06  4.0521E-06  3.0842E-06
1.7532E-06
  8  1.4148E-04  1.4148E-04  4.2444E-05  2.8296E-05  2.1222E-05
2.1128E-05
  9  5.7238E-04  5.7238E-04  1.7171E-04  1.1448E-04  8.5857E-05
8.5761E-05
 10  1.5810E-03  1.5810E-03  4.7429E-04  3.1619E-04  2.3714E-04
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2.3705E-04
 11  4.3550E-03  4.3550E-03  1.3065E-03  8.7101E-04  6.5325E-04
6.5316E-04
 12  1.0240E-02  1.0240E-02  3.0721E-03  2.0481E-03  1.5361E-03
1.5360E-03
 13  1.9606E-02  1.7452E-02  4.3398E-03  2.4236E-03  1.4231E-03
8.0759E-04
 14  3.6532E-02  3.2650E-02  8.0562E-03  4.4754E-03  2.6106E-03
1.4718E-03
 15  5.0796E-02  4.5801E-02  1.1215E-02  6.1842E-03  3.5992E-03
2.0261E-03
 16  5.4834E-02  4.9523E-02  1.2134E-02  6.6823E-03  3.8751E-03
2.1751E-03
 17  5.9525E-02  5.3728E-02  1.3181E-02  7.2336E-03  4.1920E-03
2.3504E-03
 18  7.5043E-02  6.8044E-02  1.6603E-02  9.0668E-03  5.2135E-03
2.8935E-03
 19  1.7305E-01  1.1902E-01  2.6935E-02  1.4469E-02  8.1708E-03
4.4217E-03
 20  4.1211E-01  2.2094E-01  4.5952E-02  2.3736E-02  1.3092E-02
6.9650E-03
 21  2.3861E-01  4.7697E-01  7.9707E-02  3.9840E-02  2.1709E-02
1.1467E-02
 22  2.5777E-02  6.0289E-01  2.7859E-01  1.1350E-01  5.5500E-02
2.7625E-02
 23  3.1905E-03  2.2047E-01  6.4975E-01  3.0999E-01  9.8528E-02
4.4004E-02
 24  1.5352E-03  8.5498E-02  2.9292E-01  8.0731E-01  3.3317E-01
6.7091E-02
 25  1.0870E-03  5.5001E-02  1.1759E-01  4.2646E-01  1.0869E+00
3.5047E-01
 26  1.0753E-03  4.9646E-02  9.9327E-02  1.5309E-01  6.3968E-01
1.5082E+00
1
0SLOWING DOWN PAST UPPER ENERGY OF GROUP PER UNIT VOLUME
 REGION       1            2            3            4            5
6            7            8            9           10
 GROUP
     1   .0000E+00   .0000E+00   .0000E+00   .0000E+00   .0000E+00
.0000E+00   .0000E+00   .0000E+00   .0000E+00   .0000E+00
     2  1.9538E-09  2.1385E-04  1.1106E-04  2.4218E-04  2.3525E-04
2.5329E-04  1.7710E-04  2.7631E-04  3.1736E-04  2.3574E-04
     3  4.6588E-09  5.1681E-04  2.2022E-04  1.1444E-03  1.1664E-03
1.1997E-03  3.6814E-04  1.3084E-03  9.4056E-04  4.8601E-04
     4  3.7845E-09  2.7068E-04  1.3682E-04  1.8454E-03  1.8819E-03
1.9295E-03  1.5942E-04  2.0844E-03  5.6641E-04  2.0680E-04
     5  3.9035E-09  7.6308E-05  6.9087E-05  2.1342E-03  2.1682E-03
2.2135E-03  5.1546E-05  2.3507E-03  2.1148E-04  6.3416E-05
     6  3.4313E-09  2.3049E-05  2.3570E-05  2.2472E-03  2.2781E-03
2.3078E-03  1.5823E-05  2.4052E-03  8.7080E-05  1.7788E-05
     7  3.6495E-09  8.8836E-06  1.0901E-05  2.3202E-03  2.3538E-03
2.3529E-03  1.0807E-05  2.3967E-03  4.1716E-05  1.1024E-05
     8  3.9436E-09  5.9277E-06  1.0784E-05  2.3428E-03  2.3700E-03
2.3570E-03  7.4560E-06  2.3729E-03  4.0348E-05  7.4521E-06
     9  4.2362E-09  5.3823E-06  1.0469E-05  2.3338E-03  2.3567E-03
2.3361E-03  7.9140E-06  2.3357E-03  3.8286E-05  7.8203E-06
    10  4.2566E-09  4.8026E-06  1.0220E-05  2.3206E-03  2.3350E-03
2.3200E-03  8.1910E-06  2.3112E-03  3.5919E-05  7.9881E-06
    11  4.2488E-09  5.4616E-06  1.0169E-05  2.3015E-03  2.3091E-03
2.2937E-03  8.2415E-06  2.2756E-03  3.6210E-05  7.9171E-06
    12  4.1981E-09  5.5109E-06  1.0001E-05  2.2731E-03  2.2767E-03
2.2593E-03  8.2468E-06  2.2326E-03  3.3094E-05  7.8336E-06
    13  4.1254E-09  5.3360E-06  9.8304E-06  2.2495E-03  2.2497E-03
2.2321E-03  7.9554E-06  2.2008E-03  3.5066E-05  7.5342E-06
    14  4.3181E-09  7.0530E-06  1.0501E-05  2.2334E-03  2.2323E-03
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2.2169E-03  9.5741E-06  2.1898E-03  3.9894E-05  9.3880E-06
    15  4.4131E-09  7.1105E-06  1.0933E-05  2.2186E-03  2.2169E-03
2.2027E-03  1.0065E-05  2.1777E-03  4.0316E-05  9.8374E-06
    16  4.4105E-09  6.9412E-06  1.0894E-05  2.2163E-03  2.2143E-03
2.2000E-03  9.8096E-06  2.1740E-03  3.7427E-05  9.2462E-06
    17  4.4196E-09  7.0617E-06  1.0919E-05  2.2143E-03  2.2120E-03
2.1976E-03  9.9571E-06  2.1711E-03  3.7487E-05  9.4725E-06
    18  4.4384E-09  7.2814E-06  1.1009E-05  2.2119E-03  2.2094E-03
2.1951E-03  1.0444E-05  2.1682E-03  3.8917E-05  1.0022E-05
    19  4.6386E-09  8.6329E-06  1.1851E-05  2.2035E-03  2.2005E-03
2.1859E-03  1.1947E-05  2.1580E-03  4.2468E-05  1.1493E-05
    20  5.0298E-09  1.0207E-05  1.3395E-05  2.1977E-03  2.1947E-03
2.1767E-03  1.3819E-05  2.1429E-03  4.3107E-05  1.3021E-05
    21  5.2193E-09  1.1388E-05  1.4263E-05  2.2033E-03  2.1992E-03
2.1790E-03  1.5411E-05  2.1387E-03  4.2256E-05  1.3899E-05
    22  7.8639E-09  2.2601E-05  2.4565E-05  2.4244E-03  2.4386E-03
2.3815E-03  2.9261E-05  2.3144E-03  6.7029E-05  2.6020E-05
    23  2.9426E-08  9.8180E-05  9.6372E-05  5.9641E-03  5.9343E-03
5.6515E-03  1.1919E-04  5.2479E-03  2.1783E-04  1.0002E-04
    24  3.8866E-08  1.3022E-04  1.2539E-04  8.0548E-03  7.9225E-03
7.5231E-03  1.5387E-04  6.8446E-03  2.5980E-04  1.2244E-04
    25  3.6881E-08  1.2357E-04  1.1889E-04  7.9064E-03  7.7381E-03
7.3119E-03  1.4368E-04  6.5435E-03  2.1611E-04  1.0886E-04
    26  2.2594E-08  7.5845E-05  7.2626E-05  5.1912E-03  5.0652E-03
4.7638E-03  8.6321E-05  4.1933E-03  1.1050E-04  6.1757E-05
0SLOWING DOWN PAST UPPER ENERGY OF GROUP PER UNIT VOLUME
 REGION      11           12           13           14           15
16          17           18           19           20
 GROUP
     1   .0000E+00   .0000E+00   .0000E+00   .0000E+00   .0000E+00
.0000E+00   .0000E+00   .0000E+00   .0000E+00   .0000E+00
     2  2.7719E-04  3.2535E-04  2.4602E-04  2.6818E-04  2.5628E-04
2.6912E-04  3.1469E-04  2.3513E-04  2.7804E-04  2.9928E-04
     3  1.3101E-03  9.6491E-04  5.0700E-04  1.2662E-03  1.2713E-03
1.2732E-03  9.3260E-04  4.8560E-04  1.3195E-03  8.9096E-04
     4  2.0850E-03  5.8058E-04  2.1526E-04  2.0203E-03  2.0335E-03
2.0322E-03  5.6130E-04  2.0673E-04  2.1006E-03  5.3920E-04
     5  2.3538E-03  2.1569E-04  6.5474E-05  2.2984E-03  2.3111E-03
2.3109E-03  2.0997E-04  6.3448E-05  2.3704E-03  2.0389E-04
     6  2.4106E-03  8.7984E-05  1.8109E-05  2.3745E-03  2.3889E-03
2.3838E-03  8.6791E-05  1.7800E-05  2.4225E-03  8.5529E-05
     7  2.4033E-03  4.1787E-05  1.1047E-05  2.3908E-03  2.4142E-03
2.3938E-03  4.1670E-05  1.1020E-05  2.4045E-03  4.1510E-05
     8  2.3778E-03  4.0314E-05  7.4461E-06  2.3752E-03  2.3961E-03
2.3730E-03  4.0224E-05  7.4308E-06  2.3676E-03  4.0046E-05
     9  2.3381E-03  3.8168E-05  7.7964E-06  2.3395E-03  2.3575E-03
2.3324E-03  3.8011E-05  7.7673E-06  2.3166E-03  3.7777E-05
    10  2.3115E-03  3.5726E-05  7.9455E-06  2.3151E-03  2.3239E-03
2.3053E-03  3.5536E-05  7.9081E-06  2.2835E-03  3.5266E-05
    11  2.2736E-03  3.5902E-05  7.8502E-06  2.2793E-03  2.2834E-03
2.2666E-03  3.5678E-05  7.8076E-06  2.2385E-03  3.5387E-05
    12  2.2282E-03  3.2737E-05  7.7493E-06  2.2355E-03  2.2368E-03
2.2200E-03  3.2489E-05  7.6983E-06  2.1859E-03  3.2191E-05
    13  2.1947E-03  3.4664E-05  7.4483E-06  2.2025E-03  2.2011E-03
2.1851E-03  3.4349E-05  7.3883E-06  2.1477E-03  3.3991E-05
    14  2.1826E-03  3.9695E-05  9.3423E-06  2.1866E-03  2.1834E-03
2.1689E-03  3.9204E-05  9.2309E-06  2.1342E-03  3.8715E-05
    15  2.1695E-03  4.0086E-05  9.7821E-06  2.1714E-03  2.1671E-03
2.1531E-03  3.9571E-05  9.6612E-06  2.1198E-03  3.9086E-05
    16  2.1657E-03  3.7087E-05  9.1358E-06  2.1680E-03  2.1634E-03
2.1494E-03  3.6645E-05  9.0429E-06  2.1155E-03  3.6218E-05
    17  2.1627E-03  3.7122E-05  9.3737E-06  2.1651E-03  2.1603E-03
2.1464E-03  3.6678E-05  9.2704E-06  2.1121E-03  3.6252E-05
    18  2.1596E-03  3.8553E-05  9.9335E-06  2.1621E-03  2.1571E-03
2.1432E-03  3.8077E-05  9.8158E-06  2.1087E-03  3.7625E-05
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    19  2.1489E-03  4.2106E-05  1.1397E-05  2.1513E-03  2.1456E-03
 2.1316E-03  4.1552E-05  1.1255E-05  2.0964E-03  4.1036E-05
    20  2.1325E-03  4.2606E-05  1.2873E-05  2.1368E-03  2.1314E-03
 2.1150E-03  4.2028E-05  1.2710E-05  2.0757E-03  4.1487E-05
    21  2.1270E-03  4.1535E-05  1.3654E-05  2.1335E-03  2.1274E-03
 2.1098E-03  4.0944E-05  1.3474E-05  2.0663E-03  4.0405E-05
    22  2.2916E-03  6.5611E-05  2.5488E-05  2.2997E-03  2.3106E-03
 2.2624E-03  6.3813E-05  2.4814E-05  2.1989E-03  6.2411E-05
    23  5.0715E-03  2.1132E-04  9.7166E-05  5.0962E-03  5.0787E-03
 4.8793E-03  1.9757E-04  9.1001E-05  4.5484E-03  1.8614E-04
    24  6.5596E-03  2.5162E-04  1.1881E-04  6.6273E-03  6.5448E-03
 6.2827E-03  2.3273E-04  1.1005E-04  5.7497E-03  2.1561E-04
    25  6.2394E-03  2.0967E-04  1.0591E-04  6.3496E-03  6.2536E-03
 5.9802E-03  1.9316E-04  9.7795E-05  5.4011E-03  1.7690E-04
    26  3.9833E-03  1.0769E-04  6.0414E-05  4.0947E-03  4.0293E-03
 3.8342E-03  9.9160E-05  5.5815E-05  3.4224E-03  8.9835E-05
0SLOWING DOWN PAST UPPER ENERGY OF GROUP PER UNIT VOLUME
  REGION      21          22          23          24          25
 26           27          28          29          30
  GROUP
     1   .0000E+00   .0000E+00   .0000E+00   .0000E+00   .0000E+00
 .0000E+00   .0000E+00   .0000E+00   .0000E+00   .0000E+00
     2  2.2806E-04  2.7562E-04  2.9699E-04  2.2530E-04  2.6650E-04
 2.9133E-04  2.2123E-04  2.5070E-04  2.8756E-04  2.1771E-04
     3  4.7221E-04  1.3114E-03  8.8431E-04  4.6752E-04  1.2706E-03
 8.6849E-04  4.5987E-04  1.2576E-03  8.5783E-04  4.5299E-04
     4  2.0147E-04  2.0890E-03  5.3467E-04  1.9958E-04  2.0292E-03
 5.2545E-04  1.9640E-04  2.0153E-03  5.1911E-04  1.9354E-04
     5  6.2212E-05  2.3573E-03  2.0195E-04  6.1628E-05  2.3000E-03
 1.9862E-04  6.0670E-05  2.2838E-03  1.9640E-04  5.9870E-05
     6  1.7612E-05  2.4078E-03  8.4594E-05  1.7424E-05  2.3594E-03
 8.3224E-05  1.7149E-05  2.3452E-03  8.2443E-05  1.6967E-05
     7  1.0993E-05  2.3872E-03  4.1011E-05  1.0861E-05  2.3509E-03
 4.0372E-05  1.0691E-05  2.3473E-03  4.0076E-05  1.0612E-05
     8  7.3990E-06  2.3479E-03  3.9555E-05  7.3087E-06  2.3184E-03
 3.8981E-05  7.2021E-06  2.3151E-03  3.8732E-05  7.1562E-06
     9  7.7186E-06  2.2945E-03  3.7276E-05  7.6181E-06  2.2690E-03
 3.6751E-05  7.5101E-06  2.2657E-03  3.6538E-05  7.4669E-06
    10  7.8458E-06  2.2600E-03  3.4761E-05  7.7365E-06  2.2366E-03
 3.4284E-05  7.6300E-06  2.2262E-03  3.4100E-05  7.5897E-06
    11  7.7404E-06  2.2133E-03  3.4824E-05  7.6219E-06  2.1918E-03
 3.4327E-05  7.5121E-06  2.1784E-03  3.4159E-05  7.4767E-06
    12  7.6232E-06  2.1590E-03  3.1635E-05  7.4973E-06  2.1391E-03
 3.1170E-05  7.3858E-06  2.1242E-03  3.1027E-05  7.3535E-06
    13  7.3066E-06  2.1197E-03  3.3383E-05  7.1817E-06  2.1005E-03
 3.2890E-05  7.0743E-06  2.0840E-03  3.2740E-05  7.0436E-06
    14  9.1123E-06  2.1060E-03  3.8126E-05  8.9769E-06  2.0855E-03
 3.7599E-05  8.8527E-06  2.0676E-03  3.7380E-05  8.8015E-06
    15  9.5389E-06  2.0914E-03  3.8471E-05  9.3924E-06  2.0701E-03
 3.7926E-05  9.2592E-06  2.0516E-03  3.7708E-05  9.2067E-06
    16  8.9371E-06  2.0870E-03  3.5592E-05  8.7769E-06  2.0659E-03
 3.5066E-05  8.6413E-06  2.0472E-03  3.4889E-05  8.6049E-06
    17  9.1575E-06  2.0835E-03  3.5611E-05  8.9982E-06  2.0625E-03
 3.5080E-05  8.8615E-06  2.0437E-03  3.4908E-05  8.8213E-06
    18  9.6920E-06  2.0800E-03  3.6965E-05  9.5291E-06  2.0590E-03
 3.6414E-05  9.3869E-06  2.0400E-03  3.6233E-05  9.3408E-06
    19  1.1109E-05  2.0673E-03  4.0324E-05  1.0923E-05  2.0463E-03
 3.9723E-05  1.0759E-05  2.0271E-03  3.9520E-05  1.0705E-05
    20  1.2537E-05  2.0454E-03  4.0694E-05  1.2308E-05  2.0252E-03
 4.0050E-05  1.2111E-05  2.0066E-03  3.9876E-05  1.2061E-05
    21  1.3282E-05  2.0348E-03  3.9503E-05  1.2994E-05  2.0153E-03
 3.8796E-05  1.2751E-05  1.9960E-03  3.8681E-05  1.2721E-05
    22  2.4233E-05  2.1565E-03  6.0675E-05  2.3593E-05  2.1347E-03
 5.9369E-05  2.3072E-05  2.1313E-03  5.9201E-05  2.3011E-05
    23  8.5520E-05  4.3551E-03  1.7805E-04  8.1983E-05  4.3027E-03
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1.7355E-04  7.9922E-05  4.2848E-03  1.7292E-04  7.9621E-05
    24  1.0148E-04  5.4570E-03  2.0526E-04  9.6856E-05  5.4000E-03
2.0005E-04  9.4450E-05  5.3374E-03  1.9944E-04  9.4119E-05
    25  8.9084E-05  5.1001E-03  1.6827E-04  8.5048E-05  5.0595E-03
1.6404E-04  8.2971E-05  4.9875E-03  1.6361E-04  8.2680E-05
    26  5.0255E-05  3.2213E-03  8.5610E-05  4.8127E-05  3.2067E-03
8.3554E-05  4.7023E-05  3.1560E-03  8.3321E-05  4.6819E-05
0SLOWING DOWN PAST UPPER ENERGY OF GROUP PER UNIT VOLUME
 REGION      31          32          33          34          35
36          37          38          39          40
 GROUP
     1  .0000E+00   .0000E+00   .0000E+00   .0000E+00   .0000E+00
.0000E+00   .0000E+00   .0000E+00   .0000E+00   .0000E+00
     2  2.5982E-04  2.8409E-04  2.1641E-04  2.5207E-04  2.7202E-04
2.0877E-04  2.4577E-04  2.7030E-04  2.0653E-04  2.4154E-04
     3  1.2438E-03  8.4958E-04  4.5102E-04  1.2116E-03  8.1763E-04
4.3689E-04  1.1875E-03  8.1318E-04  4.3295E-04  1.1697E-03
     4  1.9891E-03  5.1452E-04  1.9270E-04  1.9398E-03  4.9646E-04
1.8683E-04  1.9029E-03  4.9335E-04  1.8518E-04  1.8751E-03
     5  2.2565E-03  1.9470E-04  5.9544E-05  2.2055E-03  1.8861E-04
5.7849E-05  2.1647E-03  1.8726E-04  5.7337E-05  2.1342E-03
     6  2.3139E-03  8.1646E-05  1.6833E-05  2.2662E-03  7.9410E-05
1.6389E-05  2.2227E-03  7.8737E-05  1.6237E-05  2.1920E-03
     7  2.3051E-03  3.9659E-05  1.0503E-05  2.2638E-03  3.8722E-05
1.0257E-05  2.2191E-03  3.8373E-05  1.0163E-05  2.1908E-03
     8  2.2745E-03  3.8329E-05  7.0814E-06  2.2367E-03  3.7476E-05
6.9230E-06  2.1934E-03  3.7155E-05  6.8636E-06  2.1680E-03
     9  2.2270E-03  3.6154E-05  7.3879E-06  2.1921E-03  3.5383E-05
7.2288E-06  2.1505E-03  3.5093E-05  7.1697E-06  2.1274E-03
    10  2.1955E-03  3.3724E-05  7.5049E-06  2.1614E-03  3.3008E-05
7.3430E-06  2.1204E-03  3.2738E-05  7.2835E-06  2.0982E-03
    11  2.1511E-03  3.3763E-05  7.3889E-06  2.1186E-03  3.3055E-05
7.2303E-06  2.0778E-03  3.2778E-05  7.1708E-06  2.0563E-03
    12  2.0987E-03  3.0653E-05  7.2637E-06  2.0677E-03  3.0011E-05
7.1075E-06  2.0271E-03  2.9755E-05  7.0483E-06  2.0063E-03
    13  2.0604E-03  3.2340E-05  6.9566E-06  2.0302E-03  3.1660E-05
6.8061E-06  1.9900E-03  3.1390E-05  6.7495E-06  1.9697E-03
    14  2.0464E-03  3.6987E-05  8.7084E-06  2.0158E-03  3.6202E-05
8.5217E-06  1.9769E-03  3.5921E-05  8.4558E-06  1.9571E-03
    15  2.0315E-03  3.7306E-05  9.1079E-06  2.0009E-03  3.6514E-05
8.9121E-06  1.9627E-03  3.6225E-05  8.8423E-06  1.9430E-03
    16  2.0272E-03  3.4486E-05  8.4973E-06  1.9967E-03  3.3750E-05
8.3108E-06  1.9584E-03  3.3471E-05  8.2413E-06  1.9387E-03
    17  2.0238E-03  3.4498E-05  8.7146E-06  1.9934E-03  3.3759E-05
8.5235E-06  1.9550E-03  3.3478E-05  8.4533E-06  1.9353E-03
    18  2.0203E-03  3.5812E-05  9.2321E-06  1.9899E-03  3.5043E-05
9.0301E-06  1.9515E-03  3.4752E-05  8.9569E-06  1.9319E-03
    19  2.0076E-03  3.9069E-05  1.0582E-05  1.9775E-03  3.8225E-05
1.0350E-05  1.9390E-03  3.7912E-05  1.0266E-05  1.9194E-03
    20  1.9857E-03  3.9389E-05  1.1912E-05  1.9564E-03  3.8528E-05
1.1646E-05  1.9171E-03  3.8202E-05  1.1550E-05  1.8974E-03
    21  1.9746E-03  3.8165E-05  1.2549E-05  1.9466E-03  3.7336E-05
1.2268E-05  1.9062E-03  3.7009E-05  1.2164E-05  1.8866E-03
    22  2.0867E-03  5.8414E-05  2.2712E-05  2.0598E-03  5.7158E-05
2.2206E-05  2.0145E-03  5.6693E-05  2.2032E-05  1.9950E-03
    23  4.1699E-03  1.7006E-04  7.8317E-05  4.1120E-03  1.6565E-04
7.6204E-05  4.0046E-03  1.6462E-04  7.5740E-05  3.9873E-03
    24  5.2140E-03  1.9583E-04  9.2408E-05  5.1395E-03  1.9012E-04
8.9570E-05  4.9880E-03  1.8905E-04  8.9071E-05  4.9797E-03
    25  4.8693E-03  1.6063E-04  8.1187E-05  4.8040E-03  1.5555E-04
7.8476E-05  4.6433E-03  1.5468E-04  7.8032E-05  4.6463E-03
    26  3.0748E-03  8.1839E-05  4.6001E-05  3.0394E-03  7.9045E-05
4.4339E-05  2.9220E-03  7.8570E-05  4.4053E-05  2.9315E-03
0SLOWING DOWN PAST UPPER ENERGY OF GROUP PER UNIT VOLUME
 REGION      41          42          43          44          45
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46          47          48          49          50
 GROUP
    1  .0000E+00  .0000E+00  .0000E+00  .0000E+00  .0000E+00
.0000E+00  .0000E+00  .0000E+00  .0000E+00  .0000E+00
    2  2.6461E-04  1.9842E-04  2.3265E-04  2.5891E-04  1.9492E-04
2.2256E-04  2.5228E-04  1.8801E-04  2.0283E-04  2.1018E-04
    3  7.9600E-04  4.1703E-04  1.1302E-03  7.8135E-04  4.1043E-04
1.0869E-03  7.6411E-04  3.9803E-04  1.0470E-03  1.0376E-03
    4  4.8219E-04  1.7855E-04  1.8164E-03  4.7405E-04  1.7579E-04
1.7511E-03  4.6316E-04  1.7060E-04  1.6973E-03  1.6778E-03
    5  1.8310E-04  5.5508E-05  2.0802E-03  1.8040E-04  5.4705E-05
2.0165E-03  1.7627E-04  5.3186E-05  1.9667E-03  1.9466E-03
    6  7.7181E-05  1.5826E-05  2.1495E-03  7.6175E-05  1.5616E-05
2.0933E-03  7.4349E-05  1.5199E-05  2.0547E-03  2.0304E-03
    7  3.7758E-05  9.9942E-06  2.1637E-03  3.7353E-05  9.8854E-06
2.1199E-03  3.6491E-05  9.6515E-06  2.1028E-03  2.0680E-03
    8  3.6652E-05  6.7704E-06  2.1484E-03  3.6315E-05  6.7069E-06
2.1127E-03  3.5541E-05  6.5631E-06  2.1005E-03  2.0657E-03
    9  3.4680E-05  7.0862E-06  2.1124E-03  3.4393E-05  7.0259E-06
2.0810E-03  3.3659E-05  6.8764E-06  2.0701E-03  2.0344E-03
   10  3.2398E-05  7.2096E-06  2.0854E-03  3.2145E-05  7.1513E-06
2.0562E-03  3.1461E-05  7.0007E-06  2.0390E-03  2.0095E-03
   11  3.2485E-05  7.1093E-06  2.0461E-03  3.2234E-05  7.0518E-06
2.0187E-03  3.1522E-05  6.8980E-06  1.9990E-03  1.9710E-03
   12  2.9520E-05  6.9957E-06  1.9985E-03  2.9295E-05  6.9394E-06
1.9725E-03  2.8626E-05  6.7833E-06  1.9516E-03  1.9238E-03
   13  3.1149E-05  6.7008E-06  1.9630E-03  3.0914E-05  6.6475E-06
1.9378E-03  3.0198E-05  6.4960E-06  1.9153E-03  1.8885E-03
   14  3.5517E-05  8.3629E-06  1.9483E-03  3.5245E-05  8.2976E-06
1.9223E-03  3.4418E-05  8.1055E-06  1.8970E-03  1.8718E-03
   15  3.5816E-05  8.7448E-06  1.9332E-03  3.5532E-05  8.6741E-06
1.9063E-03  3.4661E-05  8.4641E-06  1.8793E-03  1.8545E-03
   16  3.3146E-05  8.1772E-06  1.9292E-03  3.2882E-05  8.1078E-06
1.9024E-03  3.2055E-05  7.9033E-06  1.8753E-03  1.8504E-03
   17  3.3163E-05  8.3804E-06  1.9259E-03  3.2899E-05  8.3106E-06
1.8991E-03  3.2063E-05  8.1008E-06  1.8719E-03  1.8470E-03
   18  3.4416E-05  8.8718E-06  1.9225E-03  3.4143E-05  8.7986E-06
1.8957E-03  3.3271E-05  8.5772E-06  1.8682E-03  1.8433E-03
   19  3.7523E-05  1.0165E-05  1.9101E-03  3.7225E-05  1.0081E-05
1.8831E-03  3.6258E-05  9.8232E-06  1.8548E-03  1.8296E-03
   20  3.7867E-05  1.1454E-05  1.8896E-03  3.7565E-05  1.1358E-05
1.8627E-03  3.6546E-05  1.1055E-05  1.8348E-03  1.8071E-03
   21  3.6796E-05  1.2107E-05  1.8806E-03  3.6497E-05  1.2001E-05
1.8541E-03  3.5457E-05  1.1662E-05  1.8261E-03  1.7973E-03
   22  5.6488E-05  2.1965E-05  1.9935E-03  5.6109E-05  2.1805E-05
1.9679E-03  5.4569E-05  2.1217E-05  1.9570E-03  1.9056E-03
   23  1.6522E-04  7.6071E-05  4.0227E-03  1.6529E-04  7.6067E-05
4.0195E-03  1.6288E-04  7.5040E-05  4.0289E-03  3.9116E-03
   24  1.9048E-04  8.9849E-05  5.0493E-03  1.9125E-04  9.0203E-05
5.0762E-03  1.8951E-04  8.9578E-05  5.0690E-03  4.9464E-03
   25  1.5621E-04  7.8902E-05  4.7317E-03  1.5730E-04  7.9457E-05
4.7788E-03  1.5640E-04  7.9198E-05  4.7697E-03  4.6553E-03
   26  7.9451E-05  4.4618E-05  2.9981E-03  8.0268E-05  4.5085E-05
3.0413E-03  8.0081E-05  4.5115E-05  3.0353E-03  2.9600E-03
0SLOWING DOWN PAST UPPER ENERGY OF GROUP PER UNIT VOLUME
 REGION      51          52          53          54          55
56          57          58          59          60
 GROUP
    1  .0000E+00  .0000E+00  .0000E+00  .0000E+00  .0000E+00
.0000E+00  .0000E+00  .0000E+00  .0000E+00  .0000E+00
    2  2.5098E-04  1.8788E-04  2.0438E-04  2.3504E-04  1.7358E-04
1.9857E-04  2.3278E-04  1.7486E-04  1.9184E-04  2.2681E-04
    3  7.6181E-04  3.9880E-04  1.0171E-03  7.1805E-04  3.7151E-04
9.9643E-04  7.1439E-04  3.7529E-04  9.7109E-04  7.0142E-04
    4  4.6185E-04  1.7089E-04  1.6480E-03  4.3646E-04  1.5977E-04
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1.6186E-03  4.3520E-04  1.6136E-04  1.5823E-03  4.2741E-04
     5  1.7568E-04  5.3189E-05  1.9206E-03  1.6792E-04  5.0374E-05
1.8966E-03  1.6778E-04  5.0785E-05  1.8689E-03  1.6575E-04
     6  7.3918E-05  1.5142E-05  2.0086E-03  7.1820E-05  1.4615E-05
1.9893E-03  7.1730E-05  1.4665E-05  1.9709E-03  7.1190E-05
     7  3.6216E-05  9.5798E-06  2.0513E-03  3.5712E-05  9.4448E-06
2.0373E-03  3.5631E-05  9.4273E-06  2.0284E-03  3.5457E-05
     8  3.5246E-05  6.5086E-06  2.0484E-03  3.4813E-05  6.4284E-06
2.0327E-03  3.4658E-05  6.3996E-06  2.0229E-03  3.4270E-05
     9  3.3332E-05  6.8097E-06  2.0143E-03  3.2893E-05  6.7203E-06
1.9938E-03  3.2661E-05  6.6718E-06  1.9797E-03  3.2043E-05
    10  3.1114E-05  6.9235E-06  1.9873E-03  3.0702E-05  6.8316E-06
1.9644E-03  3.0424E-05  6.7676E-06  1.9484E-03  2.9722E-05
    11  3.1118E-05  6.8096E-06  1.9471E-03  3.0728E-05  6.7242E-06
1.9215E-03  3.0379E-05  6.6451E-06  1.9036E-03  2.9531E-05
    12  2.8217E-05  6.6866E-06  1.8983E-03  2.7865E-05  6.6029E-06
1.8701E-03  2.7499E-05  6.5127E-06  1.8505E-03  2.6623E-05
    13  2.9750E-05  6.3999E-06  1.8619E-03  2.9354E-05  6.3143E-06
1.8323E-03  2.8949E-05  6.2237E-06  1.8118E-03  2.7978E-05
    14  3.4004E-05  8.0084E-06  1.8443E-03  3.3301E-05  7.8429E-06
1.8137E-03  3.2945E-05  7.7578E-06  1.7911E-03  3.1887E-05
    15  3.4222E-05  8.3573E-06  1.8261E-03  3.3490E-05  8.1787E-06
1.7943E-03  3.3104E-05  8.0831E-06  1.7701E-03  3.1969E-05
    16  3.1592E-05  7.7776E-06  1.8219E-03  3.0994E-05  7.6472E-06
1.7898E-03  3.0570E-05  7.5240E-06  1.7656E-03  2.9432E-05
    17  3.1587E-05  7.9775E-06  1.8183E-03  3.0997E-05  7.8313E-06
1.7860E-03  3.0555E-05  7.7115E-06  1.7616E-03  2.9387E-05
    18  3.2781E-05  8.4531E-06  1.8145E-03  3.2148E-05  8.2854E-06
1.7820E-03  3.1694E-05  8.1668E-06  1.7574E-03  3.0478E-05
    19  3.5731E-05  9.6814E-06  1.8000E-03  3.4984E-05  9.4787E-06
1.7665E-03  3.4494E-05  9.3428E-06  1.7409E-03  3.3145E-05
    20  3.5945E-05  1.0875E-05  1.7760E-03  3.5231E-05  1.0658E-05
1.7402E-03  3.4643E-05  1.0475E-05  1.7133E-03  3.3115E-05
    21  3.4773E-05  1.1433E-05  1.7658E-03  3.4214E-05  1.1256E-05
1.7287E-03  3.3515E-05  1.1011E-05  1.7014E-03  3.1836E-05
    22  5.3518E-05  2.0815E-05  1.8719E-03  5.2749E-05  2.0510E-05
1.8309E-03  5.1650E-05  2.0071E-05  1.8020E-03  4.9098E-05
    23  1.6002E-04  7.3743E-05  3.8375E-03  1.5819E-04  7.2791E-05
3.7592E-03  1.5502E-04  7.1284E-05  3.7232E-03  1.4986E-04
    24  1.8602E-04  8.7960E-05  4.8416E-03  1.8359E-04  8.6598E-05
4.7384E-03  1.7943E-04  8.4565E-05  4.7012E-03  1.7327E-04
    25  1.5339E-04  7.7713E-05  4.5464E-03  1.5089E-04  7.6207E-05
4.4421E-03  1.4708E-04  7.4218E-05  4.4095E-03  1.4142E-04
    26  7.8486E-05  4.4245E-05  2.8826E-03  7.6815E-05  4.3136E-05
2.8101E-03  7.4637E-05  4.1882E-05  2.7884E-03  7.1128E-05
0SLOWING DOWN PAST UPPER ENERGY OF GROUP PER UNIT VOLUME
 REGION     61          62          63          64          65
66          67          68          69          70
 GROUP
     1   .0000E+00   .0000E+00   .0000E+00   .0000E+00   .0000E+00
.0000E+00   .0000E+00   .0000E+00   .0000E+00   .0000E+00
     2  1.6931E-04  1.8314E-04  2.0998E-04  1.5032E-04  1.5737E-04
1.9869E-04  2.0360E-04  1.4651E-04  1.1154E-04  8.7499E-05
     3  3.6741E-04  9.4236E-04  6.5698E-04  3.3283E-04  8.7918E-04
4.1987E-04  4.4424E-04  3.4155E-04  2.7217E-04  2.2117E-04
     4  1.5828E-04  1.5427E-03  4.0190E-04  1.4433E-04  1.4611E-03
1.8138E-04  5.4664E-04  4.3428E-04  3.5264E-04  2.9028E-04
     5  5.0127E-05  1.8457E-03  1.5909E-04  4.6937E-05  1.8062E-03
5.8610E-05  4.2861E-04  3.7774E-04  3.2803E-04  2.8346E-04
     6  1.4539E-05  1.9594E-03  7.0037E-05  1.4086E-05  1.9563E-03
1.7842E-05  3.4394E-04  3.2278E-04  2.9387E-04  2.6334E-04
     7  9.3820E-06  2.0256E-03  3.5501E-05  9.3860E-06  2.0559E-03
1.2084E-05  2.5751E-04  2.5216E-04  2.3885E-04  2.2169E-04
     8  6.3289E-06  2.0111E-03  3.4012E-05  6.2826E-06  2.0206E-03
8.2741E-06  2.2645E-04  2.1855E-04  2.0710E-04  1.9355E-04
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     9  6.5474E-06  1.9550E-03  3.1542E-05  6.4468E-06  1.9419E-03
 8.7000E-06  2.1048E-04  1.9969E-04  1.8749E-04  1.7459E-04
    10  6.6138E-06  1.9191E-03  2.9239E-05  6.5084E-06  1.8938E-03
 8.9037E-06  2.0192E-04  1.9103E-04  1.7892E-04  1.6638E-04
    11  6.4620E-06  1.8690E-03  2.9019E-05  6.3528E-06  1.8350E-03
 8.8613E-06  1.9408E-04  1.8321E-04  1.7130E-04  1.5913E-04
    12  6.3080E-06  1.8114E-03  2.6138E-05  6.1958E-06  1.7716E-03
 8.7764E-06  1.8651E-04  1.7589E-04  1.6432E-04  1.5258E-04
    13  6.0173E-06  1.7705E-03  2.7446E-05  5.9052E-06  1.7276E-03
 8.4121E-06  1.8098E-04  1.7072E-04  1.5949E-04  1.4811E-04
    14  7.5111E-06  1.7474E-03  3.0826E-05  7.2622E-06  1.6970E-03
 1.0188E-05  1.8365E-04  1.7011E-04  1.5750E-04  1.4558E-04
    15  7.8085E-06  1.7240E-03  3.0860E-05  7.5389E-06  1.6689E-03
 1.0666E-05  1.8380E-04  1.6973E-04  1.5647E-04  1.4414E-04
    16  7.2278E-06  1.7190E-03  2.8527E-05  7.0341E-06  1.6636E-03
 1.0256E-05  1.8318E-04  1.6951E-04  1.5633E-04  1.4402E-04
    17  7.4124E-06  1.7147E-03  2.8490E-05  7.1926E-06  1.6588E-03
 1.0435E-05  1.8290E-04  1.6941E-04  1.5626E-04  1.4394E-04
    18  7.8578E-06  1.7100E-03  2.9520E-05  7.6071E-06  1.6535E-03
 1.0966E-05  1.8268E-04  1.6931E-04  1.5618E-04  1.4385E-04
    19  8.9813E-06  1.6916E-03  3.1999E-05  8.6738E-06  1.6326E-03
 1.2534E-05  1.8289E-04  1.6959E-04  1.5646E-04  1.4406E-04
    20  1.0019E-05  1.6599E-03  3.2017E-05  9.6895E-06  1.5982E-03
 1.4325E-05  1.8416E-04  1.7182E-04  1.5905E-04  1.4675E-04
    21  1.0454E-05  1.6456E-03  3.0958E-05  1.0179E-05  1.5832E-03
 1.5641E-05  1.8251E-04  1.7175E-04  1.5983E-04  1.4804E-04
    22  1.9094E-05  1.7404E-03  4.8099E-05  1.8699E-05  1.6944E-03
 2.8659E-05  2.1459E-04  2.0728E-04  1.9781E-04  1.8757E-04
    23  6.8940E-05  3.6593E-03  1.5465E-04  7.1081E-05  3.7485E-03
 1.0423E-04  4.9540E-04  5.1236E-04  5.2001E-04  5.2112E-04
    24  8.1720E-05  4.6415E-03  1.8221E-04  8.5945E-05  4.8062E-03
 1.2798E-04  4.6186E-04  4.8178E-04  4.9314E-04  4.9806E-04
    25  7.1420E-05  4.3552E-03  1.5077E-04  7.6052E-05  4.5627E-03
 1.1611E-04  2.5589E-04  2.6865E-04  2.7653E-04  2.8061E-04
    26  3.9974E-05  2.7407E-03  7.5402E-05  4.2169E-05  2.8626E-03
 6.7967E-05  9.9477E-05  1.0461E-04  1.0787E-04  1.0965E-04
0SLOWING DOWN PAST UPPER ENERGY OF GROUP PER UNIT VOLUME
 REGION       71           72           73           74           75
76            77           78           79           80
 GROUP
     1   .0000E+00   .0000E+00   .0000E+00   .0000E+00   .0000E+00
 .0000E+00   .0000E+00   .0000E+00   .0000E+00   .0000E+00
     2  7.1988E-05  6.0168E-05  3.7875E-10  2.1087E-05  3.2611E-05
 7.4983E-06  6.8519E-11  3.7179E-06  3.3301E-06  9.4593E-07
     3  1.8697E-04  1.5952E-04  1.2595E-09  5.8889E-05  8.7851E-05
 4.4122E-05  2.0014E-10  9.2309E-06  1.6308E-05  1.8146E-06
     4  2.4760E-04  2.1244E-04  1.1766E-09  4.1419E-05  1.1552E-04
 7.0868E-05  1.6082E-10  5.7478E-06  2.4016E-05  1.0019E-06
     5  2.4994E-04  2.1963E-04  1.9372E-09  3.2390E-05  1.2208E-04
 9.9509E-05  2.2502E-10  3.7268E-06  2.8489E-05  4.8496E-07
     6  2.3798E-04  2.1301E-04  2.0631E-09  1.3770E-05  1.2085E-04
 1.2120E-04  2.2566E-10  1.4784E-06  3.1366E-05  1.6931E-07
     7  2.0587E-04  1.8849E-04  2.4936E-09  7.4530E-06  1.1145E-04
 1.5471E-04  2.7891E-10  8.0474E-07  3.5562E-05  7.9054E-08
     8  1.8120E-04  1.6743E-04  2.5264E-09  6.9251E-06  1.0264E-04
 1.7929E-04  3.3252E-10  8.8592E-07  3.9753E-05  8.2995E-08
     9  1.6345E-04  1.5138E-04  2.5045E-09  6.1951E-06  9.5383E-05
 1.9836E-04  3.8496E-10  9.3113E-07  4.4292E-05  8.7446E-08
    10  1.5571E-04  1.4431E-04  2.4568E-09  5.9152E-06  9.2065E-05
 2.0684E-04  4.0562E-10  9.5764E-07  4.6657E-05  9.0941E-08
    11  1.4890E-04  1.3806E-04  2.3636E-09  5.6612E-06  8.9079E-05
 2.1431E-04  4.1598E-10  9.7922E-07  4.9063E-05  9.4599E-08
    12  1.4276E-04  1.3245E-04  2.2783E-09  5.4305E-06  8.6352E-05
 2.2087E-04  4.2472E-10  9.9682E-07  5.1487E-05  9.8301E-08
    13  1.3862E-04  1.2868E-04  2.2136E-09  5.2764E-06  8.4497E-05
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2.2529E-04  4.2900E-10  1.0081E-06  5.3270E-05  1.0119E-07
    14  1.3596E-04  1.2611E-04  2.1757E-09  5.2899E-06  8.3611E-05
2.2840E-04  4.5020E-10  1.0817E-06  5.4775E-05  1.1102E-07
    15  1.3436E-04  1.2452E-04  2.1797E-09  5.3957E-06  8.3229E-05
2.3130E-04  4.6627E-10  1.1416E-06  5.6136E-05  1.1857E-07
    16  1.3424E-04  1.2441E-04  2.1805E-09  5.3840E-06  8.3235E-05
2.3167E-04  4.6832E-10  1.1459E-06  5.6296E-05  1.1930E-07
    17  1.3415E-04  1.2432E-04  2.1791E-09  5.3736E-06  8.3268E-05
2.3205E-04  4.7073E-10  1.1521E-06  5.6462E-05  1.2024E-07
    18  1.3405E-04  1.2422E-04  2.1832E-09  5.4051E-06  8.3317E-05
2.3253E-04  4.7412E-10  1.1644E-06  5.6670E-05  1.2171E-07
    19  1.3422E-04  1.2438E-04  2.2592E-09  5.7646E-06  8.3945E-05
2.3477E-04  5.0003E-10  1.2628E-06  5.7571E-05  1.3242E-07
    20  1.3691E-04  1.2704E-04  2.4378E-09  6.4881E-06  8.6766E-05
2.4044E-04  5.5479E-10  1.4614E-06  5.9537E-05  1.5457E-07
    21  1.3848E-04  1.2882E-04  2.5609E-09  7.0010E-06  8.9201E-05
2.4888E-04  6.1816E-10  1.6901E-06  6.2810E-05  1.9227E-07
    22  1.7885E-04  1.6962E-04  4.1644E-09  1.2991E-05  1.2907E-04
3.7038E-04  1.4536E-09  4.5461E-06  1.1216E-04  7.0204E-07
    23  5.1867E-04  5.1268E-04  1.9552E-08  6.3865E-05  4.6189E-04
2.1297E-03  1.0895E-08  3.5519E-05  8.2968E-04  6.7850E-06
    24  4.9873E-04  4.9589E-04  2.6303E-08  8.4676E-05  4.5766E-04
3.3037E-03  1.6044E-08  5.1851E-05  1.3153E-03  1.0074E-05
    25  2.8196E-04  2.8126E-04  2.5485E-08  8.2080E-05  2.6266E-04
3.4347E-03  1.5808E-08  5.0880E-05  1.3782E-03  9.8832E-06
    26  1.1034E-04  1.1022E-04  1.4444E-08  4.5758E-05  1.0364E-04
2.3153E-03  9.7979E-09  3.1371E-05  9.3354E-04  6.0690E-06
0SLOWING DOWN PAST UPPER ENERGY OF GROUP PER UNIT VOLUME
 REGION       81          82          83          84          85
86            87          88          89          90
 GROUP
     1  .0000E+00   .0000E+00   .0000E+00   .0000E+00   .0000E+00
.0000E+00   .0000E+00   .0000E+00   .0000E+00   .0000E+00
     2  1.8870E-06  1.7127E-06  1.7010E-06  1.3446E-06  1.1566E-06
8.8275E-07  7.3156E-07  5.8975E-07  5.1866E-07  4.3221E-07
     3  8.4672E-06  7.9159E-06  7.3649E-06  5.8084E-06  4.5318E-06
3.4485E-06  2.5782E-06  2.0966E-06  1.6857E-06  1.4234E-06
     4  1.2029E-05  1.1192E-05  1.0324E-05  8.0077E-06  6.1118E-06
4.5782E-06  3.3472E-06  2.6967E-06  2.1341E-06  1.7911E-06
     5  1.3550E-05  1.2560E-05  1.1569E-05  8.8648E-06  6.6983E-06
4.9668E-06  3.6034E-06  2.8869E-06  2.2767E-06  1.9037E-06
     6  1.4282E-05  1.3250E-05  1.2197E-05  9.3167E-06  7.0061E-06
5.1756E-06  3.7386E-06  2.9897E-06  2.3520E-06  1.9647E-06
     7  1.5578E-05  1.4481E-05  1.3235E-05  1.0014E-05  7.4181E-06
5.4553E-06  3.9006E-06  3.1196E-06  2.4388E-06  2.0404E-06
     8  1.6692E-05  1.5527E-05  1.4200E-05  1.0671E-05  7.8467E-06
5.7264E-06  4.0699E-06  3.2424E-06  2.5287E-06  2.1105E-06
     9  1.8035E-05  1.6751E-05  1.5308E-05  1.1419E-05  8.3322E-06
6.0316E-06  4.2573E-06  3.3771E-06  2.6259E-06  2.1859E-06
    10  1.8767E-05  1.7364E-05  1.5923E-05  1.1809E-05  8.6146E-06
6.1926E-06  4.3689E-06  3.4471E-06  2.6842E-06  2.2243E-06
    11  1.9525E-05  1.8028E-05  1.6548E-05  1.2221E-05  8.8972E-06
6.3622E-06  4.4783E-06  3.5205E-06  2.7402E-06  2.2643E-06
    12  2.0311E-05  1.8729E-05  1.7193E-05  1.2650E-05  9.1856E-06
6.5389E-06  4.5887E-06  3.5964E-06  2.7959E-06  2.3054E-06
    13  2.0908E-05  1.9254E-05  1.7683E-05  1.2975E-05  9.4064E-06
6.6724E-06  4.6732E-06  3.6534E-06  2.8384E-06  2.3360E-06
    14  2.1415E-05  1.9707E-05  1.8102E-05  1.3252E-05  9.5926E-06
6.7857E-06  4.7439E-06  3.7015E-06  2.8735E-06  2.3616E-06
    15  2.1877E-05  2.0123E-05  1.8482E-05  1.3505E-05  9.7594E-06
6.8884E-06  4.8070E-06  3.7451E-06  2.9049E-06  2.3849E-06
    16  2.1933E-05  2.0172E-05  1.8527E-05  1.3534E-05  9.7796E-06
6.9006E-06  4.8148E-06  3.7503E-06  2.9088E-06  2.3877E-06
    17  2.1991E-05  2.0223E-05  1.8575E-05  1.3565E-05  9.8004E-06
6.9133E-06  4.8228E-06  3.7558E-06  2.9128E-06  2.3906E-06
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18 2.2062E-05 2.0287E-05 1.8634E-05 1.3604E-05 9.8263E-06
6.9292E-06 4.8328E-06 3.7627E-06 2.9179E-06 2.3943E-06
19 2.2371E-05 2.0567E-05 1.8888E-05 1.3774E-05 9.9392E-06
7.0000E-06 4.8770E-06 3.7939E-06 2.9405E-06 2.4115E-06
20 2.3079E-05 2.1221E-05 1.9469E-05 1.4177E-05 1.0208E-05
7.1795E-06 4.9921E-06 3.8798E-06 3.0032E-06 2.4616E-06
21 2.4436E-05 2.2436E-05 2.0585E-05 1.4946E-05 1.0779E-05
7.5592E-06 5.2608E-06 4.0756E-06 3.1559E-06 2.5788E-06
22 4.6068E-05 4.2950E-05 3.8957E-05 2.8769E-05 2.0674E-05
1.4614E-05 1.0013E-05 7.7567E-06 5.8781E-06 4.7952E-06
23 3.6535E-04 3.4333E-04 3.1442E-04 2.3716E-04 1.7172E-04
1.2180E-04 8.2778E-05 6.3704E-05 4.7591E-05 3.8463E-05
24 5.8182E-04 5.4380E-04 5.0290E-04 3.7821E-04 2.7582E-04
1.9452E-04 1.3299E-04 1.0164E-04 7.6357E-05 6.1265E-05
25 6.0899E-04 5.6919E-04 5.2794E-04 3.9719E-04 2.9011E-04
2.0439E-04 1.3995E-04 1.0677E-04 8.0314E-05 6.4321E-05
26 4.1116E-04 3.8493E-04 3.5775E-04 2.6937E-04 1.9678E-04
1.3864E-04 9.4991E-05 7.2418E-05 5.4503E-05 4.3618E-05
0SLOWING DOWN PAST UPPER ENERGY OF GROUP PER UNIT VOLUME
REGION 91 92 93 94 95
96 97 98
1 .0000E+00 .0000E+00 .0000E+00 .0000E+00 .0000E+00
.0000E+00 .0000E+00 .0000E+00
2 3.9407E-07 3.2420E-07 2.9486E-07 2.5873E-07 2.5362E-07
2.4293E-07 2.7090E-07 2.7167E-07
3 1.1936E-06 9.8518E-07 8.1662E-07 7.2322E-07 6.5298E-07
6.3332E-07 6.5700E-07 6.8499E-07
4 1.4830E-06 1.2157E-06 9.9378E-07 8.7586E-07 7.8035E-07
7.4632E-07 7.4680E-07 7.6826E-07
5 1.5735E-06 1.2858E-06 1.0497E-06 9.2301E-07 8.2128E-07
7.7952E-07 7.6735E-07 7.8039E-07
6 1.6214E-06 1.3239E-06 1.0793E-06 9.4875E-07 8.4297E-07
7.9748E-07 7.7784E-07 7.8755E-07
7 1.6755E-06 1.3707E-06 1.1123E-06 9.8011E-07 8.6666E-07
8.1916E-07 7.8941E-07 7.9923E-07
8 1.7312E-06 1.4134E-06 1.1460E-06 1.0081E-06 8.9057E-07
8.3830E-07 8.0410E-07 8.1265E-07
9 1.7904E-06 1.4585E-06 1.1811E-06 1.0370E-06 9.1495E-07
8.5813E-07 8.2059E-07 8.2865E-07
10 1.8261E-06 1.4812E-06 1.2024E-06 1.0514E-06 9.2982E-07
8.6794E-07 8.3143E-07 8.3649E-07
11 1.8598E-06 1.5046E-06 1.2221E-06 1.0660E-06 9.4337E-07
8.7799E-07 8.4132E-07 8.4507E-07
12 1.8929E-06 1.5283E-06 1.2411E-06 1.0807E-06 9.5624E-07
8.8807E-07 8.5081E-07 8.5390E-07
13 1.9180E-06 1.5459E-06 1.2554E-06 1.0915E-06 9.6595E-07
8.9548E-07 8.5806E-07 8.6039E-07
14 1.9386E-06 1.5605E-06 1.2671E-06 1.1004E-06 9.7378E-07
9.0158E-07 8.6385E-07 8.6558E-07
15 1.9569E-06 1.5737E-06 1.2775E-06 1.1085E-06 9.8077E-07
9.0720E-07 8.6899E-07 8.7027E-07
16 1.9592E-06 1.5754E-06 1.2788E-06 1.1095E-06 9.8167E-07
9.0790E-07 8.6966E-07 8.7085E-07
17 1.9616E-06 1.5771E-06 1.2802E-06 1.1106E-06 9.8262E-07
9.0865E-07 8.7037E-07 8.7146E-07
18 1.9646E-06 1.5792E-06 1.2819E-06 1.1119E-06 9.8380E-07
9.0959E-07 8.7126E-07 8.7225E-07
19 1.9782E-06 1.5893E-06 1.2898E-06 1.1183E-06 9.8930E-07
9.1420E-07 8.7548E-07 8.7618E-07
20 2.0173E-06 1.6198E-06 1.3134E-06 1.1384E-06 1.0062E-06
9.2948E-07 8.8937E-07 8.9010E-07
21 2.1141E-06 1.6915E-06 1.3716E-06 1.1852E-06 1.0480E-06
9.6543E-07 9.2478E-07 9.2464E-07
22 3.8445E-06 3.0560E-06 2.4155E-06 2.0792E-06 1.7991E-06
1.6551E-06 1.5615E-06 1.5786E-06

```
    23  3.0377E-05  2.3819E-05  1.8437E-05  1.5673E-05  1.3334E-05
1.2146E-05  1.1325E-05  1.1429E-05
    24  4.8659E-05  3.7874E-05  2.9464E-05  2.4863E-05  2.1264E-05
1.9232E-05  1.8027E-05  1.8053E-05
    25  5.1152E-05  3.9740E-05  3.0950E-05  2.6066E-05  2.2320E-05
2.0150E-05  1.8910E-05  1.8901E-05
    26  3.4705E-05  2.6942E-05  2.0991E-05  1.7666E-05  1.5134E-05
1.3653E-05  1.2818E-05  1.2802E-05
1                                MODIFIED
          GROUP      REMOVALS       SPECTRUM           P            ETA-F
   Q-DOWN              Q-UP
                1     1.625609E-01  1.398540E-01  1.162361E+00 -1.946850E-01
 .000000E+00    .000000E+00
                2     6.117572E-01  4.436645E-01  1.009125E+00 -2.456568E+00
1.625609E-01    .000000E+00
                3     8.955963E-01  2.887429E-01  9.945543E-01  1.252472E+00
6.117572E-01    .000000E+00
                4     9.887041E-01  9.561380E-02  9.974718E-01  1.444697E+00
8.955964E-01    .000000E+00
                5     1.011980E+00  2.463331E-02  9.986602E-01  1.272746E+00
9.887042E-01    .000000E+00
                6     1.016903E+00  7.110230E-03  9.978542E-01  9.775113E-01
1.011980E+00    .000000E+00
                7     1.013187E+00  3.622916E-04  9.959906E-01  8.959728E-01
1.016903E+00    .000000E+00
                8     1.003680E+00  1.807576E-05  9.905990E-01  1.010467E+00
1.013187E+00    .000000E+00
                9     9.959365E-01  7.330379E-07  9.922844E-01  9.597006E-01
1.003680E+00    .000000E+00
               10     9.842483E-01  1.644464E-07  9.882640E-01  1.220361E+00
9.959368E-01    .000000E+00
               11     9.698272E-01  3.669195E-08  9.853480E-01  1.064197E+00
9.842486E-01    .000000E+00
               12     9.590092E-01  7.022710E-09  9.888454E-01  2.955571E-01
9.698274E-01  2.865148E-04
               13     9.556502E-01   .000000E+00  9.964974E-01  1.166619E+00
9.592961E-01  7.347281E-04
               14     9.514346E-01   .000000E+00  9.955888E-01  1.137088E+00
9.563851E-01  1.946231E-03
               15     9.499811E-01   .000000E+00  9.984723E-01  1.702690E+00
9.533809E-01  2.241896E-03
               16     9.488475E-01   .000000E+00  9.988068E-01  1.702559E+00
9.522231E-01  2.608208E-03
               17     9.476689E-01   .000000E+00  9.987578E-01  1.660531E+00
9.514558E-01  3.122845E-03
               18     9.427658E-01   .000000E+00  9.948262E-01  1.605196E+00
9.507920E-01  6.410807E-03
               19     9.305620E-01   .000000E+00  9.870552E-01  1.648939E+00
9.491767E-01  1.949682E-02
               20     9.225011E-01   .000000E+00  9.913377E-01  1.656404E+00
9.500589E-01  3.507790E-02
               21     9.026630E-01   .000000E+00  9.784952E-01  1.545132E+00
9.575791E-01  2.707590E-01
               22     7.663805E-01   .000000E+00  8.490218E-01  1.339154E+00
1.173422E+00  3.443086E+00
               23     5.850540E-01   .000000E+00  7.633989E-01  1.268866E+00
4.209466E+00  5.321652E+00
               24     3.687429E-01   .000000E+00  6.302715E-01  1.213891E+00
5.906706E+00  5.399039E+00
               25     1.595788E-01   .000000E+00  4.327644E-01  1.140713E+00
5.767782E+00  3.595445E+00

             FAST  9.427680E-01  1.000000E+00  9.430710E-01  1.759102E+00

          THERMAL  0.            0.            0.            1.201168E+00
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1                                        *.

 1998-08-31 06:58:06  Starting 14 Leakage        WIMS-AECL Developmental
97-06-05  CPU Time223533.016 Secs

 SLOWPOKE-2 WIMS-AECL CASE


 * ENDCAP CORRECTED *
0LEAKAGE EDIT - RADIAL BUCKLING    1.41680E-03          AXIAL BUCKLING
3.03570E-03
0B(1) FLUX SOLUTION
 BENOIST DIFFUSION COEFFICIENTS

 26 GROUPS.....  K-INFINITY   1.23251   K-EFFECTIVE   1.00071

                PARTIAL            FISSION    EFFECTIVE
 GROUP       DIFFUSION   ABSORPTION   REMOVAL     NU-FISSION   FLUX-EFF
FLUX-INF       SLOWING DOWN        SPECTRUM   DIFFUSION
         RADIAL   AXIAL
             RADIAL    AXIAL                  COEFFICIENT
   1       2.588   2.599 -1.4527E-02  1.1210E-01  2.8281E-03  1.5909E+00
1.4333E+00     3.569       3.584  1.3985E-01  2.5082E+00
   2       1.909   1.918 -7.6689E-04  9.8604E-02  1.8839E-03  6.1904E+00
5.5342E+00    11.118      11.169  4.4366E-01  1.8719E+00
   3       1.220   1.230  5.1209E-04  1.4332E-01  6.4138E-04  5.4801E+00
4.8786E+00     7.587       7.644  2.8874E-01  1.2460E+00
   4        .790    .811  3.1226E-04  2.0094E-01  4.5111E-04  4.0294E+00
3.6149E+00     4.353       4.470  9.5614E-02  9.0589E-01
   5        .635    .641  3.4012E-04  2.7562E-01  4.3289E-04  2.9080E+00
2.6265E+00     2.665       2.691  2.4633E-02  7.3473E-01
   6        .549    .556  5.2863E-04  2.2834E-01  5.1675E-04  4.3638E+00
3.9749E+00     2.668       2.704  7.1102E-03  6.0394E-01
   7        .544    .551  1.2795E-03  2.6664E-01  1.1464E-03  3.6727E+00
3.3725E+00     2.113       2.142  3.6229E-04  5.5809E-01
   8        .554    .566  3.1646E-03  2.8064E-01  3.1977E-03  3.4172E+00
3.1622E+00     1.988       2.031  1.8076E-05  5.5487E-01
   9        .552    .569  5.0477E-03  4.3887E-01  4.8443E-03  1.6628E+00
1.5459E+00     1.264       1.302  7.3304E-07  5.5470E-01
  10        .553    .561  7.8087E-03  4.4372E-01  9.5295E-03  1.6122E+00
1.5046E+00     1.244       1.263  1.6445E-07  5.5289E-01
  11        .552    .560  9.8905E-03  4.4778E-01  1.0525E-02  1.5633E+00
1.4642E+00     1.226       1.244  3.6692E-08  5.5273E-01
  12        .549    .575  1.0322E-02  5.2593E-01  3.0507E-03  1.1195E+00
1.0517E+00     1.047       1.097  7.0227E-09  5.5599E-01
  13        .540    .515  3.6118E-03  5.4470E-01  4.2136E-03  1.0083E+00
9.4962E-01     1.022        .975   .0000E+00  5.4739E-01
  14        .512    .516  4.7132E-03  5.5378E-01  5.3593E-03  9.6044E-01
9.0676E-01      .995       1.004   .0000E+00  5.4929E-01
  15        .490    .497  1.3411E-02  9.0944E-01  2.2834E-02  1.1478E-01
1.0848E-01      .590        .600   .0000E+00  5.3992E-01 16        .486
.502  1.0232E-02  9.0729E-01  1.7421E-02  1.1785E-01  1.1141E-01
.590       .609    .0000E+00  5.3904E-01
  17        .486    .495  8.4780E-03  8.8701E-01  1.4078E-02  1.4782E-01
1.3979E-01      .603        .614   .0000E+00  5.3504E-01
  18        .475    .484  8.2658E-03  6.4529E-01  1.3268E-02  6.3219E-01
5.9850E-01      .816        .830   .0000E+00  5.2710E-01
  19        .445    .454  1.2578E-02  5.3683E-01  2.0741E-02  1.0273E+00
9.7459E-01      .927        .947   .0000E+00  4.9822E-01
  20        .416    .426  2.0348E-02  7.6690E-01  3.3705E-02  4.1739E-01
3.9746E-01      .610        .624   .0000E+00  4.5980E-01
  21        .393    .405  1.8313E-02  6.5747E-01  2.8296E-02  1.1231E+00
1.0889E+00      .653        .673   .0000E+00  4.0934E-01
  22        .350    .365  1.6671E-02  5.0143E-01  2.2325E-02  8.1771E+00
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8.1734E+00       .754       .788    .0000E+00  3.3227E-01
  23       .319    .335  1.9819E-02  6.7625E-01  2.5148E-02  9.0969E+00
9.1423E+00       .532       .558    .0000E+00  2.8754E-01
  24       .295    .311  2.3338E-02  7.8026E-01  2.8330E-02  9.2044E+00
9.2641E+00       .446       .471    .0000E+00  2.5147E-01
  25       .249    .265  2.9601E-02  9.5063E-01  3.3766E-02  7.0178E+00
7.0674E+00       .320       .341    .0000E+00  2.0313E-01
  26       .394    .421  4.1883E-02  9.4285E-01  4.0550E-02  3.7832E+00
3.8120E+00       .646       .690    .0000E+00  2.1592E-01

  2 GROUPS.....  K-INFINITY   1.23562   K-EFFECTIVE   1.01055

                PARTIAL
 GROUP      DIFFUSION    ABSORPTION   REMOVAL     NU-FISSION   FLUX-EFF
FLUX-INF      SLOWING DOWN
         RADIAL   AXIAL
            RADIAL    AXIAL
   1       .971    .981  1.4882E-03  2.4769E-02  2.6600E-03  4.0218E+01
3.8343E+01    36.982    37.361
   2       .268    .282  2.3580E-02  1.6956E-04  2.8348E-02  3.9871E+01
3.9989E+01    11.280    11.867

                     BUCKLING SEARCH
 GIVEN      RADIAL          AXIAL          TOTAL          NUMBER OF
ITERATIONS
 *** Buckling search failed, keff =      1.00000
 RATIO   1.421822E-03  3.046451E-03  4.468272E-03                   5
1

 1998-08-31 07:00:25  Starting 15 Reactions      WIMS-AECL Developmental
97-06-05  CPU Time223671.344 Secs

 SLOWPOKE-2 WIMS-AECL CASE



1

 1998-08-31 07:00:25  Starting 12 Power          WIMS-AECL Developmental
97-06-05  CPU Time223671.344 Secs

 SLOWPOKE-2 WIMS-AECL CASE



                    MACROSCOPIC FISSION CROSS-SECTIONS
                       CELL       MATERIALS
 GROUP   CELL FLUX   CROSS-SECT      ROD         FUEL_1       FUEL_2
FUEL_3      FUEL_4      FUEL_5      FUEL_6      FUEL_7
                                   FUEL_8      FUEL_9     FUEL_10
FUEL_11     FUEL_12     FUEL_13     FUEL_14     FUEL_15
                                  FUEL_16     FUEL_17     FUEL_18
   1    6.0080E-03  9.0529E-04   .0000E+00  1.6929E-02  1.6929E-02
1.6929E-02  1.6929E-02  1.6929E-02  1.6929E-02  1.6929E-02
                                1.6929E-02  1.6929E-02  1.6929E-02
1.6929E-02  1.6929E-02  1.6929E-02  1.6929E-02  1.6929E-02
                                1.6929E-02  1.6929E-02  1.6929E-02
   2    2.3031E-02  7.0098E-04   .0000E+00  1.4802E-02  1.4802E-02
1.4802E-02  1.4802E-02  1.4802E-02  1.4802E-02  1.4802E-02
                                1.4802E-02  1.4802E-02  1.4802E-02
1.4802E-02  1.4802E-02  1.4802E-02  1.4802E-02  1.4802E-02
                                1.4802E-02  1.4802E-02  1.4802E-02
   3    2.0203E-02  2.5414E-04   .0000E+00  5.8815E-03  5.8815E-03
5.8815E-03  5.8815E-03  5.8815E-03  5.8815E-03  5.8815E-03
                                5.8815E-03  5.8815E-03  5.8815E-03
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5.8815E-03  5.8815E-03  5.8815E-03  5.8815E-03  5.8815E-03
                                    5.8815E-03  5.8815E-03  5.8815E-03
   4    1.4957E-02  1.8291E-04   .0000E+00  5.8651E-03  5.8651E-03
5.8651E-03  5.8651E-03  5.8651E-03  5.8651E-03  5.8651E-03
                                    5.8651E-03  5.8651E-03  5.8651E-03
5.8651E-03  5.8651E-03  5.8651E-03  5.8651E-03  5.8651E-03
                                    5.8651E-03  5.8651E-03  5.8651E-03
   5    1.0868E-02  1.7698E-04   .0000E+00  7.1435E-03  7.1435E-03
7.1435E-03  7.1435E-03  7.1435E-03  7.1435E-03  7.1435E-03
                                    7.1435E-03  7.1435E-03  7.1435E-03
7.1435E-03  7.1435E-03  7.1435E-03  7.1435E-03  7.1435E-03
                                    7.1435E-03  7.1435E-03  7.1435E-03
   6    1.6450E-02  2.1199E-04   .0000E+00  1.0060E-02  1.0060E-02
1.0060E-02  1.0060E-02  1.0060E-02  1.0060E-02  1.0060E-02
                                    1.0060E-02  1.0060E-02  1.0060E-02
1.0060E-02  1.0060E-02  1.0060E-02  1.0060E-02  1.0060E-02
                                    1.0060E-02  1.0060E-02  1.0060E-02
   7    1.3960E-02  4.7047E-04   .0000E+00  2.2742E-02  2.2742E-02
2.2742E-02  2.2742E-02  2.2742E-02  2.2742E-02  2.2742E-02
                                    2.2742E-02  2.2742E-02  2.2742E-02
2.2742E-02  2.2742E-02  2.2742E-02  2.2742E-02  2.2742E-02
                                    2.2742E-02  2.2742E-02  2.2742E-02
   8    1.3097E-02  1.3123E-03   .0000E+00  6.3541E-02  6.3541E-02
6.3541E-02  6.3541E-02  6.3541E-02  6.3541E-02  6.3541E-02
                                    6.3541E-02  6.3541E-02  6.3541E-02
6.3541E-02  6.3541E-02  6.3541E-02  6.3541E-02  6.3541E-02
                                    6.3541E-02  6.3541E-02  6.3542E-02
   9    6.4027E-03  1.9881E-03   .0000E+00  9.8653E-02  9.8653E-02
9.8653E-02  9.8653E-02  9.8653E-02  9.8653E-02  9.8653E-02
                                    9.8653E-02  9.8653E-02  9.8653E-02
9.8653E-02  9.8653E-02  9.8653E-02  9.8653E-02  9.8653E-02
                                    9.8653E-02  9.8653E-02  9.8653E-02
  10    6.2317E-03  3.9108E-03   .0000E+00  1.9999E-01  1.9999E-01
1.9999E-01  1.9999E-01  1.9999E-01  1.9999E-01  1.9999E-01
                                    1.9999E-01  1.9999E-01  1.9999E-01
1.9999E-01  1.9999E-01  1.9999E-01  1.9999E-01  1.9999E-01
                                    1.9999E-01  1.9999E-01  1.9999E-01
  11    6.0651E-03  4.3196E-03   .0000E+00  2.2642E-01  2.2642E-01
2.2642E-01  2.2642E-01  2.2642E-01  2.2642E-01  2.2642E-01
                                    2.2642E-01  2.2642E-01  2.2642E-01
2.2642E-01  2.2642E-01  2.2642E-01  2.2642E-01  2.2642E-01
                                    2.2642E-01  2.2642E-01  2.2643E-01
  12    4.3567E-03  1.2520E-03   .0000E+00  6.6215E-02  6.6215E-02
6.6215E-02  6.6215E-02  6.6215E-02  6.6215E-02  6.6215E-02
                                    6.6215E-02  6.6215E-02  6.6215E-02
6.6215E-02  6.6215E-02  6.6215E-02  6.6215E-02  6.6215E-02
                                    6.6215E-02  6.6215E-02  6.6215E-02
  13    3.9341E-03  1.7292E-03   .0000E+00  8.2176E-02  8.2176E-02
8.2176E-02  8.2176E-02  8.2176E-02  8.2176E-02  8.2176E-02
                                    8.2176E-02  8.2176E-02  8.2176E-02
8.2176E-02  8.2176E-02  8.2176E-02  8.2176E-02  8.2176E-02
                                    8.2176E-02  8.2176E-02  8.2176E-02
  14    3.7567E-03  2.1994E-03   .0000E+00  1.0497E-01  1.0497E-01
1.0497E-01  1.0497E-01  1.0497E-01  1.0497E-01  1.0497E-01
                                    1.0497E-01  1.0497E-01  1.0497E-01
1.0497E-01  1.0497E-01  1.0497E-01  1.0497E-01  1.0497E-01
                                    1.0497E-01  1.0497E-01  1.0497E-01
  15    4.4942E-04  9.3710E-03   .0000E+00  4.9461E-01  4.9461E-01
4.9461E-01  4.9461E-01  4.9461E-01  4.9461E-01  4.9461E-01
                                    4.9461E-01  4.9461E-01  4.9461E-01
4.9461E-01  4.9461E-01  4.9461E-01  4.9461E-01  4.9461E-01
                                    4.9461E-01  4.9461E-01  4.9461E-01
  16    4.6159E-04  7.1496E-03   .0000E+00  3.6370E-01  3.6370E-01
3.6370E-01  3.6370E-01  3.6370E-01  3.6370E-01  3.6370E-01
                                    3.6370E-01  3.6370E-01  3.6370E-01
```

```
3.6370E-01  3.6370E-01  3.6370E-01  3.6370E-01  3.6370E-01
                                    3.6370E-01  3.6370E-01  3.6370E-01
  17    5.7915E-04  5.7775E-03   .0000E+00  2.8826E-01  2.8826E-01
2.8826E-01  2.8826E-01  2.8826E-01  2.8826E-01  2.8826E-01
                                    2.8826E-01  2.8826E-01  2.8826E-01
2.8826E-01  2.8826E-01  2.8826E-01  2.8826E-01  2.8826E-01
                                    2.8826E-01  2.8826E-01  2.8826E-01
  18    2.4798E-03  5.4452E-03   .0000E+00  2.7017E-01  2.7017E-01
2.7017E-01  2.7017E-01  2.7017E-01  2.7017E-01  2.7017E-01
                                    2.7017E-01  2.7017E-01  2.7017E-01
2.7017E-01  2.7017E-01  2.7017E-01  2.7017E-01  2.7017E-01
                                    2.7017E-01  2.7017E-01  2.7017E-01
  19    4.0382E-03  8.5120E-03   .0000E+00  4.4624E-01  4.4624E-01
4.4624E-01  4.4624E-01  4.4624E-01  4.4624E-01  4.4624E-01
                                    4.4624E-01  4.4624E-01  4.4624E-01
4.4624E-01  4.4624E-01  4.4624E-01  4.4624E-01  4.4624E-01
                                    4.4624E-01  4.4624E-01  4.4624E-01
  20    1.6470E-03  1.3832E-02   .0000E+00  8.3580E-01  8.3580E-01
8.3580E-01  8.3580E-01  8.3580E-01  8.3580E-01  8.3580E-01
                                    8.3580E-01  8.3580E-01  8.3580E-01
8.3580E-01  8.3580E-01  8.3580E-01  8.3580E-01  8.3580E-01
                                    8.3580E-01  8.3580E-01  8.3580E-01
  21    4.5127E-03  1.1612E-02   .0000E+00  8.4788E-01  8.4788E-01
8.4788E-01  8.4788E-01  8.4788E-01  8.4788E-01  8.4788E-01
                                    8.4788E-01  8.4788E-01  8.4788E-01
8.4788E-01  8.4788E-01  8.4788E-01  8.4788E-01  8.4788E-01
                                    8.4788E-01  8.4788E-01  8.4788E-01
  22    3.3873E-02  9.1619E-03   .0000E+00  1.1319E+00  1.1319E+00
1.1319E+00  1.1319E+00  1.1319E+00  1.1319E+00  1.1319E+00
                                    1.1319E+00  1.1319E+00  1.1319E+00
1.1319E+00  1.1319E+00  1.1319E+00  1.1319E+00  1.1319E+00
                                    1.1319E+00  1.1319E+00  1.1319E+00
  23    3.7888E-02  1.0321E-02   .0000E+00  1.5901E+00  1.5901E+00
1.5901E+00  1.5901E+00  1.5901E+00  1.5901E+00  1.5901E+00
                                    1.5901E+00  1.5901E+00  1.5901E+00
1.5901E+00  1.5901E+00  1.5901E+00  1.5901E+00  1.5901E+00
                                    1.5901E+00  1.5901E+00  1.5901E+00
  24    3.8394E-02  1.1626E-02   .0000E+00  2.1197E+00  2.1197E+00
2.1197E+00  2.1197E+00  2.1197E+00  2.1197E+00  2.1197E+00
                                    2.1197E+00  2.1197E+00  2.1197E+00
2.1197E+00  2.1197E+00  2.1197E+00  2.1197E+00  2.1197E+00
                                    2.1197E+00  2.1197E+00  2.1197E+00
  25    2.9290E-02  1.3857E-02   .0000E+00  2.9303E+00  2.9303E+00
2.9303E+00  2.9303E+00  2.9303E+00  2.9303E+00  2.9303E+00
                                    2.9303E+00  2.9303E+00  2.9303E+00
2.9303E+00  2.9303E+00  2.9303E+00  2.9303E+00  2.9303E+00
                                    2.9303E+00  2.9303E+00  2.9303E+00
  26    1.5799E-02  1.6641E-02   .0000E+00  4.8507E+00  4.8507E+00
4.8507E+00  4.8507E+00  4.8507E+00  4.8507E+00  4.8507E+00
                                    4.8507E+00  4.8507E+00  4.8507E+00
4.8507E+00  4.8507E+00  4.8507E+00  4.8507E+00  4.8507E+00
                                    4.8507E+00  4.8507E+00  4.8507E+00

 RELATIVE POWER DENSITIES                .0000       1.3263       1.2933
1.2042       1.1133       1.0658       1.0409       1.0373
                                        1.0199        .9888        .9825
.9889        .9940        .9858        .9676        .9499
                                         .9270        .8892        .9297

 RELATIVE POWER PER UNIT MASS HE         .0000       1.3263       1.2933
1.2042       1.1133       1.0658       1.0409       1.0373
                                        1.0199        .9888        .9825
.9889        .9940        .9858        .9676        .9499
                                         .9270        .8892        .9297
```

```
 1998-08-31 07:00:25  Starting  1 Main Data      WIMS-AECL Developmental
97-06-05  CPU Time223671.563 Secs
 >

 END OF WIMS INPUT
 THIS WIMS CASE USED    1307102 OF    1500000 WORDS OF DYNAMIC STORAGE
AVAILABLE

 WIMS COMPLETED WITH     0 SEVERE ERRORS AND     1 WARNINGS
```

ANNEX B

MICROSHIELD Version 5 SLOWPOKE-2 Case File

This Annex contains a sample MS 5 case file containing source volume dimensions and geometry, dose point position, the shielding dimensions and compositions, the source term inputs, the reference buildup material, the integration parameters, and finally, the case output results by energy fluence rate and exposure rate.

MicroShield v5.03 (5.03-00022)
Royal Military College of Canada

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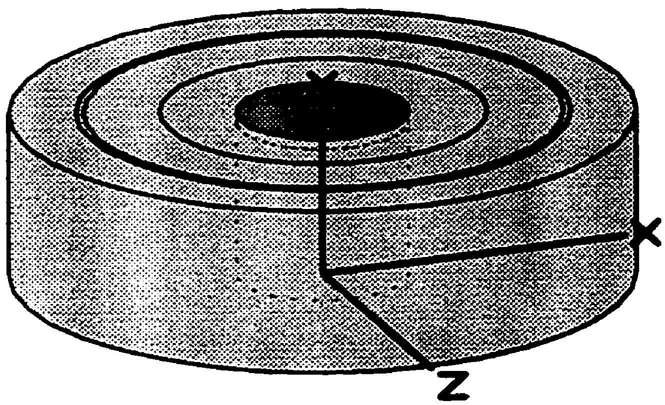
File Ref: ________
Date: ________
By: ________
Checked: ________

Case Title: SL2GAM Case
Description: Gamma Dose Rate Calculations with SL2 Reactor
Geometry: 7 - Cylinder Volume - Side Shields



Source Dimensions

Height	22.0 cm	8.7 in
Radius	11.0 cm	4.3 in

Dose Points

	X	Y	Z
# 1	40 cm	11 cm	0 cm
	1 ft 3.7 in	4.3 in	0.0 in

Shields

Shield Name	Dimension	Material	Density
Source	8362.92 cm³	Mixed ->	1.88013
		Uranium	0.5418
		Water	0.8807
		Zirconium	0.3848
		Oxygen	0.07283
Shield 1	10.0 cm	Mixed ->	1.84671
		Carbon	0.0028
		Iron	0.00241
		Beryllium	1.8415
Shield 2	9.6 cm	Water	0.9959
Shield 3	.95 cm	Mixed ->	2.69996
		Aluminum	2.6438
		Chromium	0.0054
		Copper	0.00756
		Silicon	0.0162
		Sodium	0.027
Shield 4	8.45 cm	Water	0.9959
Transition		Air	0.00122
Air Gap		Air	0.00122
Top Clad	1.5 cm	Mixed ->	1.84671
		Carbon	0.0028
		Iron	0.00241
		Beryllium	1.8415

Source Input
Grouping Method : User Defined Energies

Group #	Energy (MeV)	Activity Photons/sec	Volume Source Photons/sec/cm³	% Energy Activity
1	0.25	4.2500e+014	5.0820e+010	2.324
2	0.5	2.8200e+013	3.3720e+009	.308
3	0.65	1.9100e+015	2.2839e+011	27.160
4	1.0	3.2500e+012	3.8862e+008	.071
5	1.125	2.4500e+014	2.9296e+010	6.030
6	1.5	6.7100e+014	8.0235e+010	22.019
7	2.0	8.3800e+013	1.0020e+010	3.667

Group #	Energy (MeV)	Activity Photons/sec	Volume Source Photons/sec/cm³	% Energy Activity
8	2.23	1.6200e+014	1.9371e+010	7.903
9	2.5	2.2900e+014	2.7383e+010	12.525
10	2.7	1.8600e+013	2.2241e+009	1.099
11	2.9	1.5500e+013	1.8534e+009	.983
12	3.5	9.6100e+013	1.1491e+010	7.358
13	4.0	9.8100e+012	1.1730e+009	.858
14	4.5	3.1000e+013	3.7068e+009	3.052
15	5.5	6.2000e+012	7.4137e+008	.746
16	6.0	6.7700e+012	8.0953e+008	.889
17	6.13	1.0700e+013	1.2795e+009	1.435
18	6.5	2.1700e+012	2.5948e+008	.309
19	7.1	5.5100e+011	6.5886e+007	.086
20	8.0	6.7000e+012	8.0116e+008	1.173
21	8.5	2.3000e+010	2.7502e+006	.004
22	9.35	2.6300e+008	3.1448e+004	.000
23	9.5	7.4400e+008	8.8964e+004	.000

Buildup
The material reference is : Shield 1

Integration Parameters

Radial	39
Circumferential	39
Y Direction (axial)	39

Results

Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.25	4.250e+14	5.441e+06	1.104e+09	1.004e+04	2.037e+06
0.5	2.820e+13	5.154e+06	2.004e+08	1.012e+04	3.935e+05
0.65	1.910e+15	8.101e+08	1.790e+10	1.573e+06	3.475e+07
1.0	3.250e+12	4.808e+06	4.826e+07	8.863e+03	8.895e+04
1.125	2.450e+14	4.986e+08	4.149e+09	8.988e+05	7.481e+06
1.5	6.710e+14	2.850e+09	1.584e+10	4.795e+06	2.665e+07
2.0	8.380e+13	6.978e+08	2.799e+09	1.079e+06	4.328e+06
2.23	1.620e+14	1.715e+09	6.181e+09	2.561e+06	9.233e+06
2.5	2.290e+14	3.093e+09	1.005e+10	4.451e+06	1.446e+07
2.7	1.860e+13	2.946e+08	8.964e+08	4.135e+05	1.258e+06
2.9	1.550e+13	2.836e+08	8.149e+08	3.890e+05	1.118e+06
3.5	9.610e+13	2.533e+09	6.346e+09	3.273e+06	8.201e+06
4.0	9.810e+12	3.306e+08	7.584e+08	4.090e+05	9.382e+05
4.5	3.100e+13	1.286e+09	2.741e+09	1.528e+06	3.256e+06
5.5	6.200e+12	3.596e+08	6.899e+08	4.004e+05	7.681e+05
6.0	6.770e+12	4.508e+08	8.316e+08	4.896e+05	9.030e+05
6.13	1.070e+13	7.367e+08	1.346e+09	7.952e+05	1.453e+06
6.5	2.170e+12	1.635e+08	2.907e+08	1.736e+05	3.087e+05
7.1	5.510e+11	4.737e+07	8.094e+07	4.912e+04	8.391e+04
8.0	6.700e+12	6.843e+08	1.115e+09	6.878e+05	1.121e+06
8.5	2.300e+10	2.557e+06	4.086e+06	2.532e+03	4.046e+03

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Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
9.35	2.630e+08	3.329e+04	5.188e+04	3.221e+01	5.021e+01
9.5	7.440e+08	9.618e+04	1.493e+05	9.273e+01	1.440e+02
TOTALS:	3.961e+15	1.685e+10	7.418e+10	2.400e+07	1.188e+08

ANNEX C

FORTRAN Program - Radiative Capture in Shielding Materials Calculation

This Annex contains the FORTRAN code used to numerically approximate the total radiative capture gamma contribution at a dose point in the beryllium reflector. Only the beryllium reflector and water annulus between the reflector and the container wall were considered as (n,γ) target materials for this calculation. The results of the numerical approximation were given as a 7-energy group gamma flux distribution.

```
C     PROGRAM GAMMA COUNT  .
C
C     The purpose of this program is to calculate the gamma
C     contribution at a dose point due to radiative capture
C     of thermal neutrons in the shielding around the SL-2
C     reactor core.
C
C     Each shield (ie Be reflector, H2O annulus surrounding the
C     reflector) has been divided into 16 equivolume segments at
C     22.5 deg angular quadrants around the core. Reflectional
C     symmetry is assumed (ie identical gamma contributions assumed
C     for segments from 0 to 180 deg and 180 to 360 deg).  This is
C     done to reduce the computational time and is a reasonable
C     assumption.
C
C     In the axial direction, each shield has been divided into
C     11 planes in the positive or negative z-direction of 1cm depth
C     each.  Reflectional symmetry is assumed (ie identical gamma
C     contributions from identical segments above and below the core
C     midplane).  This is done in order to reduce the computational
C     time and is a reasonable assumption.
C
C     List of Variables
C
C     S(8,11)   Array of gamma source terms resulting from radiative
C               capture in Be reflector (8 gamma energies from radiative
C               capture reactions in Be,Fe and 11 axial planes)
C     SH(11)    Vector of gamma source terms resulting from radiative
C               capture in H2O annulus (1 gamma energy from radiative
C               capture reaction in H, and 11 axial planes)
C     FACT      Scaling factor applied to gamma source terms S due to
C               decrease in thermal flux value in z-direction
C     P11(4,11)  Array of gamma contributions at dose point from
C                radiative capture in Be reflector (capture in Be,
C                4 angular segments,at gamma energy 4 MeV and
C                11 axial planes).
C                These gamma contributions pass through only the Be
C                reflector on their way to the dose point.
C     P12(4,11)  Array of gamma contributions at dose point from
C                radiative capture in Be reflector (capture in Be,
C                4 angular segments,at gamma energy 6 MeV and
C                11 axial planes).
C                These gamma contributions pass through only the Be
C                reflector on their way to the dose point.
C     P13(4,11)  Array of gamma contributions at dose point from
C                radiative capture in Be reflector (capture in Fe,
C                4 angular segments,at gamma energy 0.5 MeV and
C                11 axial planes).
C                These gamma contributions pass through only the Be
C                reflector on their way to the dose point.
C     P14(4,11)  Array of gamma contributions at dose point from
C                radiative capture in Be reflector (capture in Fe,
C                4 angular segments,at gamma energy 1.5 MeV and
C                11 axial planes).
C                These gamma contributions pass through only the Be
C                reflector on their way to the dose point.
C     P15(4,11)  Array of gamma contributions at dose point from
C                radiative capture in Be reflector (capture in Fe,
C                4 angular segments,at gamma energy 2 MeV and
C                11 axial planes).
C                These gamma contributions pass through only the Be
C                reflector on their way to the dose point.
C     P16(4,11)  Array of gamma contributions at dose point from
C                radiative capture in Be reflector (capture in Fe,
C                4 angular segments,at gamma energy 4 MeV and
```

```
C              11 axial planes).
C              These gamma contributions pass through only the Be
C              reflector on their way to the dose point.
C    P17(4,11) Array of gamma contributions at dose point from
C              radiative capture in Be reflector (capture in Fe,
C              4 angular segments,at gamma energy 6 MeV and
C              11 axial planes).
C              These gamma contributions pass through only the Be
C              reflector on their way to the dose point.
C    P18(4,11) Array of gamma contributions at dose point from
C              radiative capture in Be reflector (capture in Fe,
C              4 angular segments,at gamma energy 8.5 MeV and
C              11 axial planes).
C              These gamma contributions pass through only the Be
C              reflector on their way to the dose point.
C    P21(4,11) Array of gamma contributions at dose point from
C              radiative capture in Be reflector (capture in Be,
C              4 angular segments,at gamma energy 4 MeV and
C              11 axial planes).
C              These gamma contributions pass through both the Be
C              reflector and the core on their way to the dose point
C    P22(4,11) Array of gamma contributions at dose point from
C              radiative capture in Be reflector (capture in Be,
C              4 angular segments,at gamma energy 6 MeV and
C              11 axial planes).
C              These gamma contributions pass through both the Be
C              reflector and the core on their way to the dose point
C    P23(4,11) Array of gamma contributions at dose point from
C              radiative capture in Be reflector (capture in Fe,
C              4 angular segments,at gamma energy 0.5 MeV and
C              11 axial planes).
C              These gamma contributions pass through both the Be
C              reflector and the core on their way to the dose point
C    P24(4,11) Array of gamma contributions at dose point from
C              radiative capture in Be reflector (capture in Fe,
C              4 angular segments,at gamma energy 1.5 MeV and
C              11 axial planes).
C              These gamma contributions pass through both the Be
C              reflector and the core on their way to the dose point
C    P25(4,11) Array of gamma contributions at dose point from
C              radiative capture in Be reflector (capture in Fe,
C              4 angular segments,at gamma energy 2 MeV and
C              11 axial planes).
C              These gamma contributions pass through both the Be
C              reflector and the core on their way to the dose point
C    P26(4,11) Array of gamma contributions at dose point from
C              radiative capture in Be reflector (capture in Fe,
C              4 angular segments,at gamma energy 4 MeV and
C              11 axial planes).
C              These gamma contributions pass through both the Be
C              reflector and the core on their way to the dose point
C    P27(4,11) Array of gamma contributions at dose point from
C              radiative capture in Be reflector (capture in Fe,
C              4 angular segments,at gamma energy 6 MeV and
C              11 axial planes).
C              These gamma contributions pass through both the Be
C              reflector and the core on their way to the dose point
C    P28(4,11) Array of gamma contributions at dose point from
C              radiative capture in Be reflector (capture in Fe,
C              4 angular segments,at gamma energy 8.5 MeV and
C              11 axial planes).
C              These gamma contributions pass through both the Be
C              reflector and the core on their way to the dose point
C    PW1(5,11) Array of gamma contributions at dose point from
C              radiative capture in H2O annulus where path between
```

```
C                  centre of segment and dose point passes through Be
C                  reflector and H2O annulus only (capture in H, 5 angular
C                  segments and 11 axial planes, at gamma energy 2.23 MeV)
C       PW2(3,11)  Array of gamma contributions at dose point from
C                  radiative capture in H2O annulus where path between
C                  centre of segment and dose point passes through Be
C                  reflector, H2O annulus, and core (capture in H, 3
C                  angular segments and 11 axial planes, at gamma energy
C                  2.23 MeV)
C       UR(6)      Linear attenuation coefficients in Be reflector (6 gamma
C                  energies)
C       UC(6)      Linear attenuation coefficients in core (6 gamma
C                  energies)
C       UW         Linear attenuation coefficient in H2O annulus (at gamma
C                  energy 2.23 MeV)
C       D(8,11)    Distance between centre of Be segment and dose point (8
C                  angular segments and 11 axial planes)
C       DW(8,11)   Distance between centre of H2O segment and dose point
C                  (8 angular segments and 11 axial planes)
C       DR(4,11)   Distance between centre of segment and dose point that
C                  lies within the Be reflector for capture in Be reflector
C                  (for gammas passing through both reflector and core
C                  only, 4 angular segments and 11 axial planes)
C       DWR(8,11)  Distance between centre of H2O segment and dose point
C                  that lies with the Be reflector for capture in H2O
C                  annulus (8 segments and 11 axial planes)
C       DC(4,11)   Distance between centre of segment and dose point that
C                  lies within the core (for gammas passing through both
C                  reflector and core only, 4 angular segments and 11 axial
C                  planes)
C       DWC(3,11)  Distance between centre of H2O segment and dose point
C                  that lies within the core for capture in H2O segment
C                  (3 segments and 11 axial planes)
C       DWW(8,11)  Distance between centre of H2O segment and dose point
C                  that lies within the H2O annulus for capture in H2O
C                  (8 segments and 11 axial planes)
C       B1(4,11)   Buildup factors for segments where path between centre
C                  of segment and dose point passes through Be reflector
C                  only (4 segments for gamma energy 0.5MeV and 11
C                  axial planes)
C       B2(4,11)   Buildup factors for segments where path between centre
C                  of segment and dose point passes through Be reflector
C                  only (4 segments for gamma energy 1.5MeV and 11
C                  axial planes)
C       B3(4,11)   Buildup factors for segments where path between centre
C                  of segment and dose point passes through Be reflector
C                  only (4 segments for gamma energy 2 MeV and 11 axial
C                  planes)
C       B4(4,11)   Buildup factors for segments where path between centre
C                  of segment and dose point passes through Be reflector
C                  only (4 segments for gamma energy 4 MeV and 11 axial
C                  planes)
C       B5(4,11)   Buildup factors for segments where path between centre
C                  of segment and dose point passes through Be reflector
C                  only (4 segments for gamma energy 6 MeV and 11 axial
C                  planes)
C       B6(4,11)   Buildup factors for segments where path between centre
C                  of segment and dose point passes through Be reflector
C                  only (4 segments for gamma energy 8.5 MeV and 11 axial
C                  planes)
C       BR1(4,11)  Buildup factors in Be reflector for segments where path
C                  between centre of segment and dose point passes through
C                  both Be reflector and core (4 segments for gamma
C                  energy 0.5 MeV and 11 axial planes)
C       BR2(4,11)  Buildup factors in Be reflector for segments where path
```

```
C                between centre of segment and dose point passes through
C                both Be reflector and core (4 segments for gamma
C                energy 1.5 MeV and 11 axial planes)
C     BR3(4,11)  Buildup factors in Be reflector for segments where path
C                between centre of segment and dose point passes through
C                both Be reflector and core (4 segments for gamma
C                energy 2.0 MeV and 11 axial planes)
C     BR4(4,11)  Buildup factors in Be reflector for segments where path
C                between centre of segment and dose point passes through
C                both Be reflector and core (4 segments for gamma
C                energy 4.0 MeV and 11 axial planes)
C     BR5(4,11)  Buildup factors in Be reflector for segments where path
C                between centre of segment and dose point passes through
C                both Be reflector and core (4 segments for gamma
C                energy 6.0 MeV and 11 axial planes)
C     BR6(4,11)  Buildup factors in Be reflector for segments where path
C                between centre of segment and dose point passes through
C                both Be reflector and core (4 segments for gamma
C                energy 8.5 MeV and 11 axial planes)
C     BC1        Buildup factor in core for segments where path between
C                centre of segment and dose point passes through both
C                Be reflector and core (for gamma energy 0.5 MeV)
C     BC2        Buildup factor in core for segments where path between
C                centre of segment and dose point passes through both
C                Be reflector and core (for gamma energy 1.5 MeV)
C     BC3        Buildup factor in core for segments where path between
C                centre of segment and dose point passes through both
C                Be reflector and core (for gamma energy 2.0 MeV)
C     BC4        Buildup factor in core for segments where path between
C                centre of segment and dose point passes through both
C                Be reflector and core (for gamma energy 4.0 MeV)
C     BC5        Buildup factor in core for segments where path between
C                centre of segment and dose point passes through both
C                Be reflector and core (for gamma energy 6.0 MeV)
C     BC6        Buildup factor in core for segments where path between
C                centre of segment and dose point passes through both
C                Be reflector and core (for gamma energy 8.5 MeV)
C     BWW(8,11)    Array of buildup factors in H2O for capture reactions
C                  in H2O annulus (for gamma energy 2.23 MeV, 8 segments
C                  and 11 axial planes)
C     BWR(8,11)    Array of buildup factors in Be reflector for capture
C                  reactions in H2O annulus (for gamma energy 2.23 MeV,
C                  8 segments and 11 axial planes)
C     PT1          Summation of gamma contribution at dose point due
C                  to capture reactions in Be and H2O shielding at gamma
C                  energy 0.5 MeV
C     PT2          Summation of gamma contributions at dose point due
C                  to capture reactions in Be and H2O shielding at gamma
C                  energy 1.5 MeV
C     PT3          Summation of gamma contributions at dose point due
C                  to capture reactions in Be and H2O shielding at gamma
C                  energy 2 MeV
C     PT4          Summation of gamma contributions at dose point due
C                  to capture reactions in Be and H2O shielding at gamma
C                  energy 4 MeV
C     PT5          Summation of gamma contributions at dose point due
C                  to capture reactions in Be and H2O shielding at gamma
C                  energy 6 MeV
C     PT6          Summation of gamma contributions at dose point due
C                  to capture reactions in Be and H2O shielding at gamma
C                  energy 8.5 MeV
C     PT7          Summation of gamma contributions at dose point due
C                  to capture reactions in Be and H2O shielding at gamma
C                  energy 2.23 MeV
C
```

```
      REAL S(8,11),FACT,P11(4,11),P12(4,11),P13(4,11),P14(4,11)
      REAL P15(4,11),P16(4,11),P17(4,11),P18(4,11)
      REAL P21(4,11),P22(4,11),P23(4,11),P24(4,11),P25(4,11)
      REAL P26(4,11),P27(4,11),P28(4,11),UR(6),UC(6),D(8,11)
      REAL DR(4,11),DC(4,11)
      REAL B1(4,11),B2(4,11),B3(4,11),B4(4,11),B5(4,11),B6(4,11)
      REAL BR1(4,11),BR2(4,11),BR3(4,11),BR4(4,11),BR5(4,11),BR6(4,11)
      REAL BC1,BC2,BC3,BC4,BC5,BC6
      REAL PW1(5,11),PW2(5,11),SH(11),UW,DW(8,11),DWR(8,11),DWC(3,11)
      REAL DWW(8,11),BWW(8,11),BWR(8,11)
      REAL PT1,PT2,PT3,PT4,PT5,PT6,PT7
      INTEGER I,J
C
C     Open the output file for values of gamma contributions at the
C     dose point
C
      OPEN(91,FILE='GAMMA',STATUS='OLD')
C
C     Initialization of linear attenuation coefficients in core,
C     reflector at the 6 main gamma energies
C
      UR(1)=0.1428
      UR(2)=0.0848
      UR(3)=0.0726
      UR(4)=0.0491
      UR(5)=0.0391
      UR(6)=0.0335
      UC(1)=0.224
      UC(2)=0.102
      UC(3)=0.089
      UC(4)=0.069
      UC(5)=0.0639
      UC(6)=0.0626
C
C     Initialization of gamma source terms as a result of capture in
C     Be reflector
C
      DO 10 I=1,11
      IF (I.LE.5) THEN
      FACT=1.0
      ENDIF
      IF (I.GT.5) THEN
      FACT=0.97**(I-5)
      ENDIF
      S(1,I)=1.64E10*FACT
      S(2,I)=2.46E10*FACT
      S(3,I)=1.40E9*FACT
      S(4,I)=1.118E9*FACT
      S(5,I)=1.864E8*FACT
      S(6,I)=4.47E8*FACT
      S(7,I)=4.1E8*FACT
      S(8,I)=9.321E8*FACT
10    CONTINUE
C
C     Initialization of distances
      D(1,1)=3.16
      D(2,1)=9.35
      D(3,1)=15.33
      D(4,1)=20.57
      D(5,1)=24.92
      D(6,1)=28.37
      D(7,1)=30.77
      D(8,1)=31.82
      D(1,2)=3.49
      D(2,2)=9.57
```

D(3,2)=15.46
D(4,2)=20.60
D(5,2)=24.95
D(6,2)=28.39
D(7,2)=30.79
D(8,2)=31.84
D(1,3)=3.91
D(2,3)=9.78
D(3,3)=15.65
D(4,3)=20.70
D(5,3)=25.03
D(6,3)=28.46
D(7,3)=30.85
D(8,3)=32.05
D(1,4)=4.61
D(2,4)=10.08
D(3,4)=15.84
D(4,4)=20.85
D(5,4)=25.14
D(6,4)=28.57
D(7,4)=30.95
D(8,4)=32.14
D(1,5)=5.41
D(2,5)=10.47
D(3,5)=16.09
D(4,5)=21.04
D(5,5)=25.30
D(6,5)=28.70
D(7,5)=31.08
D(8,5)=32.27
D(1,6)=6.26
D(2,6)=10.93
D(3,6)=16.40
D(4,6)=21.27
D(5,6)=25.50
D(6,6)=28.88
D(7,6)=31.24
D(8,6)=32.42
D(1,7)=7.16
D(2,7)=11.47
D(3,7)=16.76
D(4,7)=21.55
D(5,7)=25.73
D(6,7)=29.09
D(7,7)=31.43
D(8,7)=32.60
D(1,8)=8.08
D(2,8)=12.06
D(3,8)=17.17
D(4,8)=21.88
D(5,8)=26.00
D(6,8)=29.33
D(7,8)=31.65
D(8,8)=32.82
D(1,9)=9.01
D(2,9)=12.71
D(3,9)=17.63
D(4,9)=22.24
D(5,9)=26.31
D(6,9)=29.60
D(7,9)=31.90
D(8,9)=33.06
D(1,10)=9.96
D(2,10)=13.40
D(3,10)=18.14

```
      D(4,10)=22.64
      D(5,10)=26.65
      D(6,10)=29.90
      D(7,10)=32.18
      D(8,10)=33.33
      D(1,11)=10.92
      D(2,11)=14.13
      D(3,11)=18.68
      D(4,11)=23.08
      D(5,11)=27.02
      D(6,11)=30.23
      D(7,11)=32.49
      D(8,11)=33.63
C
      DR(1,1)=15.27
      DR(2,1)=11.86
      DR(3,1)=10.51
      DR(4,1)=9.76
      DR(1,2)=15.48
      DR(2,2)=11.87
      DR(3,2)=10.52
      DR(4,2)=9.77
      DR(1,3)=15.53
      DR(2,3)=11.89
      DR(3,3)=10.54
      DR(4,3)=10.09
      DR(1,4)=15.60
      DR(2,4)=11.95
      DR(3,4)=10.57
      DR(4,4)=10.11
      DR(1,5)=15.69
      DR(2,5)=12.00
      DR(3,5)=10.62
      DR(4,5)=10.15
      DR(1,6)=15.82
      DR(2,6)=12.07
      DR(3,6)=10.67
      DR(4,6)=10.19
      DR(1,7)=15.97
      DR(2,7)=12.17
      DR(3,7)=10.74
      DR(4,7)=10.25
      DR(1,8)=16.13
      DR(2,8)=12.27
      DR(3,8)=10.80
      DR(4,8)=10.32
      DR(1,9)=16.32
      DR(2,9)=12.37
      DR(3,9)=10.89
      DR(4,9)=10.40
      DR(1,10)=16.54
      DR(2,10)=12.50
      DR(3,10)=10.98
      DR(4,10)=10.48
      DR(1,11)=16.76
      DR(2,11)=12.64
      DR(3,11)=11.09
      DR(4,11)=10.58
C
      DC(1,1)=9.65
      DC(2,1)=16.51
      DC(3,1)=20.26
      DC(4,1)=22.06
      DC(1,2)=9.47
      DC(2,2)=16.52
```

```
      DC(3,2)=20.27
      DC(4,2)=22.07
      DC(1,3)=9.50
      DC(2,3)=16.57
      DC(3,3)=20.31
      DC(4,3)=21.96
      DC(1,4)=9.54
      DC(2,4)=16.62
      DC(3,4)=20.38
      DC(4,4)=22.03
      DC(1,5)=9.61
      DC(2,5)=16.70
      DC(3,5)=20.46
      DC(4,5)=22.12
      DC(1,6)=9.68
      DC(2,6)=16.81
      DC(3,6)=20.57
      DC(4,6)=22.23
      DC(1,7)=9.76
      DC(2,7)=16.92
      DC(3,7)=20.69
      DC(4,7)=22.35
      DC(1,8)=9.87
      DC(2,8)=17.06
      DC(3,8)=20.85
      DC(4,8)=22.50
      DC(1,9)=9.99
      DC(2,9)=17.23
      DC(3,9)=21.01
      DC(4,9)=22.66
      DC(1,10)=10.11
      DC(2,10)=17.40
      DC(3,10)=21.20
      DC(4,10)=22.85
      DC(1,11)=10.26
      DC(2,11)=17.59
      DC(3,11)=21.40
      DC(4,11)=23.05
C
C     Calculation of buildup factors
C
      DO 20 J=1,11
      DO 30 I=1,4
      B1(I,J)=(177.872*EXP(0.1237*UR(1)*D(I,J)))-
     + (176.872*EXP(0.0801*UR(1)*D(I,J)))
      B2(I,J)=(49.051*EXP(0.0441*UR(2)*D(I,J)))-
     + (48.051*EXP(0.0058*UR(2)*D(I,J)))
      B3(I,J)=(40.343*EXP(0.0309*UR(3)*D(I,J)))-
     + (39.343*EXP(-0.0027*UR(3)*D(I,J)))
      B4(I,J)=(13.104*EXP(0.022*UR(4)*D(I,J)))-
     + (12.104*EXP(-0.037*UR(4)*D(I,J)))
      B5(I,J)=(10.841*EXP(0.0146*UR(5)*D(I,J)))-
     + (9.841*EXP(-0.0372*UR(5)*D(I,J)))
      B6(I,J)=(6.306*EXP(0.0179*UR(6)*D(I,J)))-
     + (5.306*EXP(-0.0593*UR(6)*D(I,J)))
30    CONTINUE
20    CONTINUE
C
      BC1=3.08
      BC2=1.65
      BC3=1.632
      BC4=1.551
      BC5=1.531
      BC6=1.539
C
```

```
      DO 40 J=1,11
      DO 50 I=1,4
      BR1(I,J)=(177.872*EXP(0.1237*UR(1)*DR(I,J)))-
     + (176.872*EXP(0.0801*UR(1)*DR(I,J)))
      BR2(I,J)=(49.051*EXP(0.0441*UR(2)*DR(I,J)))-
     + (48.051*EXP(0.0058*UR(2)*DR(I,J)))
      BR3(I,J)=(40.343*EXP(0.0309*UR(3)*DR(I,J)))-
     + (39.343*EXP(-0.0027*UR(3)*DR(I,J)))
      BR4(I,J)=(13.104*EXP(0.022*UR(4)*DR(I,J)))-
     + (12.104*EXP(-0.037*UR(4)*DR(I,J)))
      BR5(I,J)=(10.841*EXP(0.0146*UR(5)*DR(I,J)))-
     + (9.841*EXP(-0.0372*UR(5)*DR(I,J)))
      BR6(I,J)=(6.306*EXP(0.0179*UR(6)*DR(I,J)))-
     + (5.306*EXP(-0.0593*UR(6)*DR(I,J)))
50    CONTINUE
40    CONTINUE
C
C     Calculation of gamma contributions at dose point from gammas
C     passing through Be reflector only
C
      DO 60 J=1,11
      DO 70 I=1,4
      P11(I,J)=(S(1,J)/(4.0*3.14159*D(I,J)**2.0))*
     +  EXP(-UR(4)*D(I,J))*B4(I,J)
      P12(I,J)=(S(2,J)/(4.0*3.14159*D(I,J)**2.0))*
     +  EXP(-UR(5)*D(I,J))*B5(I,J)
      P13(I,J)=(S(3,J)/(4.0*3.14159*D(I,J)**2.0))*
     +  EXP(-UR(1)*D(I,J))*B1(I,J)
      P14(I,J)=(S(4,J)/(4.0*3.14159*D(I,J)**2.0))*
     +  EXP(-UR(2)*D(I,J))*B2(I,J)
      P15(I,J)=(S(5,J)/(4.0*3.14159*D(I,J)**2.0))*
     +  EXP(-UR(3)*D(I,J))*B3(I,J)
      P16(I,J)=(S(6,J)/(4.0*3.14159*D(I,J)**2.0))*
     +  EXP(-UR(4)*D(I,J))*B4(I,J)
      P17(I,J)=(S(7,J)/(4.0*3.14159*D(I,J)**2.0))*
     +  EXP(-UR(5)*D(I,J))*B5(I,J)
      P18(I,J)=(S(8,J)/(4.0*3.14159*D(I,J)**2.0))*
     +  EXP(-UR(6)*D(I,J))*B6(I,J)
70    CONTINUE
60    CONTINUE
C
C     Calculation of gamma contributions at dose point from gammas
C     passing through Be reflector and core
C
      DO 80 J=1,11
      DO 90 I=1,4
      P21(I,J)=(S(1,J)/(4.0*3.14159*D(I+4,J)**2.0))*EXP(-UR(4)*DR(I,J))*
     +  EXP(-UC(4)*DC(I,J))*BC4*BR4(I,J)
      P22(I,J)=(S(2,J)/(4.0*3.14159*D(I+4,J)**2.0))*EXP(-UR(5)*DR(I,J))*
     +  EXP(-UC(5)*DC(I,J))*BC5*BR5(I,J)
      P23(I,J)=(S(3,J)/(4.0*3.14159*D(I+4,J)**2.0))*EXP(-UR(1)*DR(I,J))*
     +  EXP(-UC(1)*DC(I,J))*BC1*BR1(I,J)
      P24(I,J)=(S(4,J)/(4.0*3.14159*D(I+4,J)**2.0))*EXP(-UR(2)*DR(I,J))*
     +  EXP(-UC(2)*DC(I,J))*BC2*BR2(I,J)
      P25(I,J)=(S(5,J)/(4.0*3.14159*D(I+4,J)**2.0))*EXP(-UR(3)*DR(I,J))*
     +  EXP(-UC(3)*DC(I,J))*BC3*BR3(I,J)
      P26(I,J)=(S(6,J)/(4.0*3.14159*D(I+4,J)**2.0))*EXP(-UR(4)*DR(I,J))*
     +  EXP(-UC(4)*DC(I,J))*BC4*BR4(I,J)
      P27(I,J)=(S(7,J)/(4.0*3.14159*D(I+4,J)**2.0))*EXP(-UR(5)*DR(I,J))*
     +  EXP(-UC(5)*DC(I,J))*BC5*BR5(I,J)
      P28(I,J)=(S(8,J)/(4.0*3.14159*D(I+4,J)**2.0))*EXP(-UR(6)*DR(I,J))*
     +  EXP(-UC(6)*DC(I,J))*BC6*BR6(I,J)
90    CONTINUE
80    CONTINUE
C
```

```
C     Next, initialize gamma source terms as a result of capture in H2O
C     annulus
C
      DO 100 I=1,11
      IF (I.LE.5) THEN
      FACT=1.0
      ENDIF
      IF (I.GT.5) THEN
      FACT=0.97**(I-5)
      ENDIF
      SH(I)=6.451E11*FACT
100   CONTINUE
C
C     Initialization of distance terms required in gamma contribution
C     calculations as a result of capture reactions in H2O annulus
C
      DW(1,1)=10.66
      DW(2,1)=15.61
      DW(3,1)=21.97
      DW(4,1)=27.75
      DW(5,1)=33.24
      DW(6,1)=37.50
      DW(7,1)=40.35
      DW(8,1)=41.85
      DW(1,2)=10.76
      DW(2,2)=15.67
      DW(3,2)=22.01
      DW(4,2)=27.79
      DW(5,2)=33.27
      DW(6,2)=37.53
      DW(7,2)=40.38
      DW(8,2)=41.88
      DW(1,3)=10.94
      DW(2,3)=15.80
      DW(3,3)=22.10
      DW(4,3)=27.86
      DW(5,3)=33.33
      DW(6,3)=37.58
      DW(7,3)=40.43
      DW(8,3)=41.92
      DW(1,4)=11.21
      DW(2,4)=15.99
      DW(3,4)=22.24
      DW(4,4)=27.97
      DW(5,4)=33.42
      DW(6,4)=37.66
      DW(7,4)=40.50
      DW(8,4)=42.00
      DW(1,5)=11.56
      DW(2,5)=16.24
      DW(3,5)=22.42
      DW(4,5)=28.11
      DW(5,5)=33.54
      DW(6,5)=37.77
      DW(7,5)=40.60
      DW(8,5)=42.09
      DW(1,6)=11.99
      DW(2,6)=16.54
      DW(3,6)=22.64
      DW(4,6)=28.29
      DW(5,6)=33.69
      DW(6,6)=37.90
      DW(7,6)=40.72
      DW(8,6)=42.21
      DW(1,7)=12.48
```

```
      DW(2,7)=16.90
      DW(3,7)=22.90
      DW(4,7)=28.50
      DW(5,7)=33.87
      DW(6,7)=38.06
      DW(7,7)=40.87
      DW(8,7)=42.35
      DW(1,8)=13.03
      DW(2,8)=17.31
      DW(3,8)=23.21
      DW(4,8)=28.75
      DW(5,8)=34.08
      DW(6,8)=38.24
      DW(7,8)=41.04
      DW(8,8)=42.52
      DW(1,9)=13.63
      DW(2,9)=17.77
      DW(3,9)=23.55
      DW(4,9)=29.02
      DW(5,9)=34.31
      DW(6,9)=38.45
      DW(7,9)=41.24
      DW(8,9)=42.70
      DW(1,10)=14.27
      DW(2,10)=18.26
      DW(3,10)=23.93
      DW(4,10)=29.33
      DW(5,10)=34.57
      DW(6,10)=38.68
      DW(7,10)=41.45
      DW(8,10)=42.91
      DW(1,11)=14.96
      DW(2,11)=18.80
      DW(3,11)=24.34
      DW(4,11)=29.67
      DW(5,11)=34.86
      DW(6,11)=38.94
      DW(7,11)=41.69
      DW(8,11)=43.15
C
      DWW(1,1)=4.95
      DWW(2,1)=6.37
      DWW(3,1)=6.82
      DWW(4,1)=6.00
      DWW(5,1)=5.55
      DWW(6,1)=5.34
      DWW(7,1)=4.95
      DWW(8,1)=4.65
      DWW(1,2)=5.00
      DWW(2,2)=6.39
      DWW(3,2)=6.82
      DWW(4,2)=6.01
      DWW(5,2)=5.55
      DWW(6,2)=5.34
      DWW(7,2)=4.96
      DWW(8,2)=4.66
      DWW(1,3)=5.09
      DWW(2,3)=6.44
      DWW(3,3)=6.85
      DWW(4,3)=6.02
      DWW(5,3)=5.56
      DWW(6,3)=5.35
      DWW(7,3)=4.96
      DWW(8,3)=4.65
      DWW(1,4)=5.21
```

```
      DWW(2,4)=6.52
      DWW(3,4)=6.90
      DWW(4,4)=6.05
      DWW(5,4)=5.58
      DWW(6,4)=5.36
      DWW(7,4)=4.97
      DWW(8,4)=4.67
      DWW(1,5)=5.37
      DWW(2,5)=6.62
      DWW(3,5)=6.96
      DWW(4,5)=6.08
      DWW(5,5)=5.60
      DWW(6,5)=5.38
      DWW(7,5)=4.98
      DWW(8,5)=4.68
      DWW(1,6)=5.58
      DWW(2,6)=6.74
      DWW(3,6)=7.02
      DWW(4,6)=6.12
      DWW(5,6)=5.62
      DWW(6,6)=5.40
      DWW(7,6)=4.99
      DWW(8,6)=4.69
      DWW(1,7)=5.80
      DWW(2,7)=6.89
      DWW(3,7)=7.10
      DWW(4,7)=6.16
      DWW(5,7)=5.66
      DWW(6,7)=5.42
      DWW(7,7)=5.01
      DWW(8,7)=4.70
      DWW(1,8)=6.06
      DWW(2,8)=7.06
      DWW(3,8)=7.20
      DWW(4,8)=6.22
      DWW(5,8)=5.69
      DWW(6,8)=5.44
      DWW(7,8)=5.03
      DWW(8,8)=4.73
      DWW(1,9)=6.34
      DWW(2,9)=7.25
      DWW(3,9)=7.30
      DWW(4,9)=6.27
      DWW(5,9)=5.73
      DWW(6,9)=5.47
      DWW(7,9)=5.06
      DWW(8,9)=4.74
      DWW(1,10)=6.63
      DWW(2,10)=7.44
      DWW(3,10)=7.42
      DWW(4,10)=6.34
      DWW(5,10)=5.77
      DWW(6,10)=5.50
      DWW(7,10)=5.08
      DWW(8,10)=4.76
      DWW(1,11)=6.96
      DWW(2,11)=7.66
      DWW(3,11)=7.55
      DWW(4,11)=6.41
      DWW(5,11)=5.82
      DWW(6,11)=5.54
      DWW(7,11)=5.11
      DWW(8,11)=4.80
C
      DWR(1,1)=5.71
```

```
DWR(2,1)=9.24
DWR(3,1)=15.15
DWR(4,1)=21.75
DWR(5,1)=27.69
DWR(6,1)=19.41
DWR(7,1)=16.20
DWR(8,1)=15.30
DWR(1,2)=5.76
DWR(2,2)=9.28
DWR(3,2)=15.19
DWR(4,2)=21.78
DWR(5,2)=27.72
DWR(6,2)=19.43
DWR(7,2)=16.20
DWR(8,2)=15.30
DWR(1,3)=5.85
DWR(2,3)=9.36
DWR(3,3)=15.25
DWR(4,3)=21.84
DWR(5,3)=27.77
DWR(6,3)=19.45
DWR(7,3)=16.23
DWR(8,3)=15.33
DWR(1,4)=6.00
DWR(2,4)=9.47
DWR(3,4)=15.34
DWR(4,4)=21.92
DWR(5,4)=27.84
DWR(6,4)=19.50
DWR(7,4)=16.26
DWR(8,4)=15.36
DWR(1,5)=6.19
DWR(2,5)=9.62
DWR(3,5)=15.46
DWR(4,5)=22.03
DWR(5,5)=27.94
DWR(6,5)=19.55
DWR(7,5)=16.30
DWR(8,5)=15.38
DWR(1,6)=6.41
DWR(2,6)=9.80
DWR(3,6)=15.62
DWR(4,6)=22.17
DWR(5,6)=28.07
DWR(6,6)=19.61
DWR(7,6)=16.35
DWR(8,6)=15.43
DWR(1,7)=6.68
DWR(2,7)=10.01
DWR(3,7)=15.80
DWR(4,7)=22.34
DWR(5,7)=28.21
DWR(6,7)=19.70
DWR(7,7)=16.41
DWR(8,7)=15.49
DWR(1,8)=6.97
DWR(2,8)=10.25
DWR(3,8)=16.01
DWR(4,8)=22.53
DWR(5,8)=28.39
DWR(6,8)=19.80
DWR(7,8)=16.48
DWR(8,8)=15.54
DWR(1,9)=7.29
DWR(2,9)=10.52
```

```
      DWR(3,9)=16.25
      DWR(4,9)=22.75
      DWR(5,9)=28.58
      DWR(6,9)=19.91
      DWR(7,9)=16.56
      DWR(8,9)=15.61
      DWR(1,10)=7.64
      DWR(2,10)=10.82
      DWR(3,10)=16.51
      DWR(4,10)=22.99
      DWR(5,10)=28.80
      DWR(6,10)=20.03
      DWR(7,10)=16.65
      DWR(8,10)=15.69
      DWR(1,11)=8.00
      DWR(2,11)=11.14
      DWR(3,11)=16.79
      DWR(4,11)=23.26
      DWR(5,11)=29.04
      DWR(6,11)=20.16
      DWR(7,11)=16.74
      DWR(8,11)=15.77
C
      DWC(1,1)=12.75
      DWC(2,1)=19.20
      DWC(3,1)=21.90
      DWC(1,2)=12.76
      DWC(2,2)=19.22
      DWC(3,2)=21.92
      DWC(1,3)=12.78
      DWC(2,3)=19.24
      DWC(3,3)=21.94
      DWC(1,4)=12.80
      DWC(2,4)=19.27
      DWC(3,4)=21.97
      DWC(1,5)=12.84
      DWC(2,5)=19.32
      DWC(3,5)=22.03
      DWC(1,6)=12.89
      DWC(2,6)=19.38
      DWC(3,6)=22.09
      DWC(1,7)=12.94
      DWC(2,7)=19.45
      DWC(3,7)=22.16
      DWC(1,8)=13.00
      DWC(2,8)=19.53
      DWC(3,8)=22.25
      DWC(1,9)=13.07
      DWC(2,9)=19.62
      DWC(3,9)=22.35
      DWC(1,10)=13.15
      DWC(2,10)=19.72
      DWC(3,10)=22.46
      DWC(1,11)=13.24
      DWC(2,11)=19.84
      DWC(3,11)=22.58
C
C     Next, calculate the buildup factors in the core, Be reflector,
C     and H2O annulus following capture reaction in H2O annulus
C
      DO 110 J=1,11
      DO 120 I=1,8
      BWR(I,J)=(40.343*EXP(0.0309*UR(3)*DWR(I,J)))-
     + (39.343*EXP(-0.0027*UR(3)*DWR(I,J)))
120   CONTINUE
```

```
110   CONTINUE
C
C     Initialize value of UW
      UW=0.04932
C
      DO 130 J=1,11
      DO 140 I=1,8
      BWW(I,J)=(32.003*EXP(0.0274*UW*DWW(I,J)))-
     + (31.003*EXP(-0.0041*UW*DWW(I,J)))
140   CONTINUE
130   CONTINUE
C
C     Calculation of gamma contributions at dose point due to radiative
C     capture in H2O annulus, passing through H2O annulus and Be
C     reflector only
C
      DO 150 J=1,11
      DO 160 I=1,5
      PW1(I,J)=(SH(J)/(4.0*3.14159*DW(I,J)**2.0))*EXP(-UW*DWW(I,J))*
     +  EXP(-UR(3)*DWR(I,J))*BWW(I,J)*BWR(I,J)
160   CONTINUE
150   CONTINUE
C
C     Calculation of gamma contributions at dose point due to radiative
C     capture in H2O annulus, passing through H2O annulus, Be reflector
C     and core
C
      DO 170 J=1,11
      DO 180 I=1,3
      PW2(I,J)=(SH(J)/(4.0*3.14159*DW(I+5,J)**2.0))*EXP(-UW*DWW(I+5,J))*
     +  EXP(-UR(3)*DWR(I+5,J))*EXP(-UC(3)*DWC(I,J))*BC3*BWR(I+5,J)*
     +  BWW(I+5,J)
180   CONTINUE
170   CONTINUE
C
C     Summation of gamma contribution terms at the dose point due to
C     radiative capture reactions in both the Be reflector and the H2O
C     annulus.  Gamma contributions at gamma energies 0.5 MeV,1.5MeV,
C     2 MeV,2.23 MeV,4 MeV,6 MeV,8.5 MeV
C
      PT1=0.0
      PT2=0.0
      PT3=0.0
      PT4=0.0
      PT5=0.0
      PT6=0.0
      PT7=0.0
      DO 190 J=1,11
      DO 200 I=1,4
      PT1=PT1+P13(I,J)+P23(I,J)
      PT2=PT2+P14(I,J)+P24(I,J)
      PT3=PT3+P15(I,J)+P25(I,J)
      PT4=PT4+P11(I,J)+P21(I,J)+P16(I,J)+P26(I,J)
      PT5=PT5+P12(I,J)+P22(I,J)+P17(I,J)+P27(I,J)
      PT6=PT6+P18(I,J)+P28(I,J)
200   CONTINUE
190   CONTINUE
C
      DO 210 J=1,11
      DO 220 I=1,5
      PT7=PT7+PW1(I,J)
220   CONTINUE
210   CONTINUE
      DO 230 J=1,11
      DO 240 I=1,3
```

```
      PT7=PT7+PW2(I,J)
240   CONTINUE
230   CONTINUE
C
C     Gamma contribution terms must be multiplied by 4 to account for
C     symmetry in the azimuthal direction as well as in axial direction
C
      PT1=PT1*4.0
      PT2=PT2*4.0
      PT3=PT3*4.0
      PT4=PT4*4.0
      PT5=PT5*4.0
      PT6=PT6*4.0
      PT7=PT7*4.0
C
C     Print out gamma contribution values to the destination file
C
      WRITE(91,55)
55    FORMAT('I',2X,'J',4X,'P11',9X,'P12',9X,'P13',9X,'P14')
56    FORMAT(I1,1X,I2,1X,E11.5E2,1X,E11.5E2,1X,E11.5E2,1X,E11.5E2)
      DO 250 J=1,11
      DO 260 I=1,4
      WRITE(91,56)I,J,P11(I,J),P12(I,J),P13(I,J),P14(I,J)
260   CONTINUE
250   CONTINUE
      WRITE(91,57)
57    FORMAT('I',2X,'J',4X,'P15',9X,'P16',9X,'P17',9X,'P18')
      DO 270 J=1,11
      DO 280 I=1,4
      WRITE(91,56)I,J,P15(I,J),P16(I,J),P17(I,J),P18(I,J)
280   CONTINUE
270   CONTINUE
      WRITE(91,58)
58    FORMAT('I',2X,'J',4X,'P21',9X,'P22',9X,'P23',9X,'P24')
      DO 290 J=1,11
      DO 310 I=1,4
      WRITE(91,56)I,J,P21(I,J),P22(I,J),P23(I,J),P24(I,J)
310   CONTINUE
290   CONTINUE
      WRITE(91,59)
59    FORMAT('I',2X,'J',4X,'P25',9X,'P26',9X,'P27',9X,'P28')
      DO 320 J=1,11
      DO 330 I=1,4
      WRITE(91,56)I,J,P25(I,J),P26(I,J),P27(I,J),P28(I,J)
330   CONTINUE
320   CONTINUE
      WRITE(91,61)
61    FORMAT('I',2X,'J',4X,'PW1')
76    FORMAT(I1,1X,I2,1X,E11.5E2)
      DO 340 J=1,11
      DO 350 I=1,5
      WRITE(91,76)I,J,PW1(I,J)
350   CONTINUE
340   CONTINUE
      WRITE(91,62)
62    FORMAT('I',2X,'J',4X,'PW2')
      DO 360 J=1,11
      DO 370 I=1,3
      WRITE(91,76)I,J,PW2(I,J)
370   CONTINUE
360   CONTINUE
C
C     Finally, print out the gamma contribution results to the screen
      PRINT*,'GAMMA CONTRIBUTIONS DUE TO RADIATIVE CAPTURE IN SHIELDING'
      PRINT*,'0.5 MeV',PT1
```

```
PRINT*,'1.5 MeV',PT2 ,
PRINT*,'2 MeV',PT3
PRINT*,'2.23 MeV',PT7
PRINT*,'4 MeV',PT4
PRINT*,'6 MeV',PT5
PRINT*,'8.5 MeV',PT6
STOP
END
```

ANNEX D

FORTRAN Program - Gamma Backscatter Calculation

This Annex contains the FORTRAN code used to numerically approximate the gamma backscatter contribution at a dose point around the SLOWPOKE-2 reactor core. Volume integral approximations in the hemispherical volume forward of the dose point were calculated. The sample code contained within is for a dose point in the pool around the reactor container. MS 5-generated gamma source terms were used in the construction of the volume integral approximation.

```
C     PROGRAM BACKSCATTER .
C
C     The purpose of this program is to approximate numerically the
C     volume integral representing the backscatter contribution to
C     the dose point originating from the hemisphere in the positive
C     r-direction from the dose point.
C
C     Microshield 5 was used to calculate the gamma source term at
C     various mesh points along 5 planes (0 deg,22.5 deg,45 deg,
C     67.5 deg, and 90 deg in the azimuthal direction) through the
C     hemisphere volume.  Using this program those points will be used
C     to approximate the backscatter contribution.  The volume to be
C     integrated lies between theta equals 0 deg and theta equals 90
C     deg.  Rotational symmetry is assumed (ie equal backscatter
C     contributions from upper and lower regions of the hemisphere).
C
      REAL JS,V(7),PHIX(7,7),PHIZ(7,7),TH,MU1,MU2,MU3,MU4,MU5,MU6,MU7
      REAL PHI45(7,7),PHI225(7,7),PHI675(7,7)
      REAL MUSC,PHI1
      REAL JX1,JX2,JX3,JX4,JX5,JX6,JX7
      REAL JZ1,JZ2,JZ3,JZ4,JZ5,JZ6,JZ7
      REAL JXZ1,JXZ2,JXZ3,JXZ4,JXZ5,JXZ6,JXZ7
      REAL J2251,J2252,J2253,J2254,J2255,J2256,J2257
      REAL J6751,J6752,J6753,J6754,J6755,J6756,J6757
      REAL SX1,SX2,SX3,SX4,SX5,SX6,SX7
      REAL SZ1,SZ2,SZ3,SZ4,SZ5,SZ6,SZ7
      REAL SXZ1,SXZ2,SXZ3,SXZ4,SXZ5,SXZ6,SXZ7
      REAL S2251,S2252,S2253,S2254,S2255,S2256,S2257
      REAL S6751,S6752,S6753,S6754,S6755,S6756,S6757
      INTEGER J
C
C     ..............................................................
C     Initialization of Variables
C     ..............................................................
C
      V(1)=1.028
      V(2)=7.196
      V(3)=19.531
      V(4)=38.034
      V(5)=62.705
      V(6)=93.544
      V(7)=130.550
C
      PHI1=1.663E11
C
      PHIX(1,1)=5.827E10
      PHIX(1,2)=5.972E10
      PHIX(1,3)=6.429E10
      PHIX(1,4)=7.262E10
      PHIX(1,5)=8.596E10
      PHIX(1,6)=1.064E11
      PHIX(1,7)=1.372E11
      PHIX(2,1)=2.287E10
      PHIX(2,2)=2.367E10
      PHIX(2,3)=2.625E10
      PHIX(2,4)=3.148E10
      PHIX(2,5)=4.067E10
      PHIX(2,6)=5.655E10
      PHIX(2,7)=8.500E10
      PHIX(3,1)=9.728E9
      PHIX(3,2)=1.011E10
      PHIX(3,3)=1.143E10
      PHIX(3,4)=1.410E10
      PHIX(3,5)=1.885E10
      PHIX(3,6)=2.740E10
```

```
      PHIX(3,7)=4.340E10
      PHIX(4,1)=4.416E9
      PHIX(4,2)=4.593E9
      PHIX(4,3)=5.253E9
      PHIX(4,4)=6.533E9
      PHIX(4,5)=8.817E9
      PHIX(4,6)=1.289E10
      PHIX(4,7)=2.025E10
      PHIX(5,1)=2.118E9
      PHIX(5,2)=2.208E9
      PHIX(5,3)=2.530E9
      PHIX(5,4)=3.145E9
      PHIX(5,5)=4.233E9
      PHIX(5,6)=6.106E9
      PHIX(5,7)=9.201E9
      PHIX(6,1)=1.067E9
      PHIX(6,2)=1.114E9
      PHIX(6,3)=1.274E9
      PHIX(6,4)=1.577E9
      PHIX(6,5)=2.101E9
      PHIX(6,6)=2.964E9
      PHIX(6,7)=4.211E9
      PHIX(7,1)=5.602E8
      PHIX(7,2)=5.856E8
      PHIX(7,3)=6.675E8
      PHIX(7,4)=8.209E8
      PHIX(7,5)=1.081E9
      PHIX(7,6)=1.487E9
      PHIX(7,7)=1.976E9
C
      PHIZ(1,1)=1.400E11
      PHIZ(2,1)=8.893E10
      PHIZ(3,1)=4.801E10
      PHIZ(4,1)=2.413E10
      PHIZ(5,1)=1.190E10
      PHIZ(6,1)=5.902E9
      PHIZ(7,1)=2.988E9
      PHIZ(1,2)=1.398E11
      PHIZ(2,2)=8.626E10
      PHIZ(3,2)=4.761E10
      PHIZ(4,2)=2.385E10
      PHIZ(5,2)=1.176E10
      PHIZ(6,2)=5.849E9
      PHIZ(7,2)=2.967E9
      PHIZ(1,3)=1.394E11
      PHIZ(2,3)=8.739E10
      PHIZ(3,3)=4.697E10
      PHIZ(4,3)=2.357E10
      PHIZ(5,3)=1.159E10
      PHIZ(6,3)=5.743E9
      PHIZ(7,3)=2.906E9
      PHIZ(1,4)=1.388E11
      PHIZ(2,4)=8.671E10
      PHIZ(3,4)=4.622E10
      PHIZ(4,4)=2.290E10
      PHIZ(5,4)=1.114E10
      PHIZ(6,4)=5.478E9
      PHIZ(7,4)=2.757E9
      PHIZ(1,5)=1.381E11
      PHIZ(2,5)=8.595E10
      PHIZ(3,5)=4.507E10
      PHIZ(4,5)=2.185E10
      PHIZ(5,5)=1.043E10
      PHIZ(6,5)=5.042E9
      PHIZ(7,5)=2.507E9
```

```
      PHIZ(1,6)=1.374E11
      PHIZ(2,6)=8.530E10
      PHIZ(3,6)=4.390E10
      PHIZ(4,6)=2.076E10
      PHIZ(5,6)=9.607E9
      PHIZ(6,6)=4.502E9
      PHIZ(7,6)=2.174E9
      PHIZ(1,7)=1.372E11
      PHIZ(2,7)=8.500E10
      PHIZ(3,7)=4.340E10
      PHIZ(4,7)=2.025E10
      PHIZ(5,7)=9.201E9
      PHIZ(6,7)=4.211E9
      PHIZ(7,7)=1.976E9
C
      PHI45(1,1)=7.559E10
      PHI45(1,2)=7.685E10
      PHI45(1,3)=8.083E10
      PHI45(1,4)=8.785E10
      PHI45(1,5)=9.870E10
      PHI45(1,6)=1.147E11
      PHI45(1,7)=1.372E11
      PHI45(2,1)=3.567E10
      PHI45(2,2)=3.641E10
      PHI45(2,3)=3.875E10
      PHI45(2,4)=4.351E10
      PHI45(2,5)=5.141E10
      PHI45(2,6)=6.408E10
      PHI45(2,7)=8.500E10
      PHI45(3,1)=1.766E10
      PHI45(3,2)=1.801E10
      PHI45(3,3)=1.929E10
      PHI45(3,4)=2.172E10
      PHI45(3,5)=2.564E10
      PHI45(3,6)=3.205E10
      PHI45(3,7)=4.340E10
      PHI45(4,1)=9.150E9
      PHI45(4,2)=9.307E9
      PHI45(4,3)=9.971E9
      PHI45(4,4)=1.109E10
      PHI45(4,5)=1.281E10
      PHI45(4,6)=1.546E10
      PHI45(4,7)=2.025E10
      PHI45(5,1)=4.934E9
      PHI45(5,2)=5.021E9
      PHI45(5,3)=5.344E9
      PHI45(5,4)=5.853E9
      PHI45(5,5)=6.558E9
      PHI45(5,6)=7.507E9
      PHI45(5,7)=9.201E9
      PHI45(6,1)=2.760E9
      PHI45(6,2)=2.809E9
      PHI45(6,3)=2.968E9
      PHI45(6,4)=3.198E9
      PHI45(6,5)=3.471E9
      PHI45(6,6)=3.743E9
      PHI45(6,7)=4.211E9
      PHI45(7,1)=1.595E9
      PHI45(7,2)=1.623E9
      PHI45(7,3)=1.702E9
      PHI45(7,4)=1.806E9
      PHI45(7,5)=1.902E9
      PHI45(7,6)=1.933E9
      PHI45(7,7)=1.976E9
C
```

```
      PHI225(1,1)=1.024E11
      PHI225(1,2)=1.031E11
      PHI225(1,3)=1.055E11
      PHI225(1,4)=1.095E11
      PHI225(1,5)=1.158E11
      PHI225(1,6)=1.248E11
      PHI225(1,7)=1.372E11
      PHI225(2,1)=6.017E10
      PHI225(2,2)=6.039E10
      PHI225(2,3)=6.113E10
      PHI225(2,4)=6.336E10
      PHI225(2,5)=6.722E10
      PHI225(2,6)=7.376E10
      PHI225(2,7)=8.500E10
      PHI225(3,1)=3.552E10
      PHI225(3,2)=3.542E10
      PHI225(3,3)=3.548E10
      PHI225(3,4)=3.579E10
      PHI225(3,5)=3.638E10
      PHI225(3,6)=3.813E10
      PHI225(3,7)=4.340E10
      PHI225(4,1)=2.151E10
      PHI225(4,2)=2.129E10
      PHI225(4,3)=2.107E10
      PHI225(4,4)=2.049E10
      PHI225(4,5)=1.956E10
      PHI225(4,6)=1.884E10
      PHI225(4,7)=2.025E10
      PHI225(5,1)=1.342E10
      PHI225(5,2)=1.325E10
      PHI225(5,3)=1.289E10
      PHI225(5,4)=1.208E10
      PHI225(5,5)=1.080E10
      PHI225(5,6)=9.373E09
      PHI225(5,7)=9.201E09
      PHI225(6,1)=8.603E09
      PHI225(6,2)=8.478E09
      PHI225(6,3)=8.135E09
      PHI225(6,4)=7.380E09
      PHI225(6,5)=6.197E09
      PHI225(6,6)=4.808E09
      PHI225(6,7)=4.211E09
      PHI225(7,1)=5.654E09
      PHI225(7,2)=5.563E09
      PHI225(7,3)=5.273E09
      PHI225(7,4)=4.660E09
      PHI225(7,5)=3.700E09
      PHI225(7,6)=2.570E09
      PHI225(7,7)=1.976E09
C
      PHI675(1,1)=6.227E10
      PHI675(1,2)=6.372E10
      PHI675(1,3)=6.816E10
      PHI675(1,4)=7.626E10
      PHI675(1,5)=8.907E10
      PHI675(1,6)=1.085E11
      PHI675(1,7)=1.372E11
      PHI675(2,1)=2.561E10
      PHI675(2,2)=2.641E10
      PHI675(2,3)=2.900E10
      PHI675(2,4)=3.419E10
      PHI675(2,5)=4.318E10
      PHI675(2,6)=5.843E10
      PHI675(2,7)=8.500E10
      PHI675(3,1)=1.132E10
```

```
      PHI675(3,2)=1.172E10
      PHI675(3,3)=1.307E10
      PHI675(3,4)=1.574E10
      PHI675(3,5)=2.040E10
      PHI675(3,6)=2.854E10
      PHI675(3,7)=4.340E10
      PHI675(4,1)=5.312E09
      PHI675(4,2)=5.497E09
      PHI675(4,3)=6.186E09
      PHI675(4,4)=7.480E09
      PHI675(4,5)=9.707E09
      PHI675(4,6)=1.352E10
      PHI675(4,7)=2.025E10
      PHI675(5,1)=2.625E09
      PHI675(5,2)=2.720E09
      PHI675(5,3)=3.058E09
      PHI675(5,4)=3.686E09
      PHI675(5,5)=4.737E09
      PHI675(5,6)=6.444E09
      PHI675(5,7)=9.201E09
      PHI675(6,1)=1.356E09
      PHI675(6,2)=1.407E09
      PHI675(6,3)=1.577E09
      PHI675(6,4)=1.886E09
      PHI675(6,5)=2.389E09
      PHI675(6,6)=3.150E09
      PHI675(6,7)=4.211E09
      PHI675(7,1)=7.289E08
      PHI675(7,2)=7.564E08
      PHI675(7,3)=8.447E08
      PHI675(7,4)=1.002E09
      PHI675(7,5)=1.248E09
      PHI675(7,6)=1.592E09
      PHI675(7,7)=1.976E09
C
      MU1=0.101
      MU2=0.0985
      MU3=0.096
      MU4=0.093
      MU5=0.09
      MU6=0.0877
      MU7=0.0853
      MUSC=0.0294
C
C     Calculate the volume integral by dividing the volume into 7
C     equi-volume segments
C
C     First, calculate the volume segments represented by the points
C     along the plane z=0
C
      JX1=0.0
      JX2=0.0
      JX3=0.0
      JX4=0.0
      JX5=0.0
      JX6=0.0
      JX7=0.0
      DO 10 J=1,6
      IF(J.EQ.1)THEN
      TH=1.4399
      ENDIF
      IF(J.EQ.2)THEN
      TH=1.1781
      ENDIF
      IF(J.EQ.3)THEN
```

```
      TH=0.9163
      ENDIF
      IF(J.EQ.4)THEN
      TH=0.6545
      ENDIF
      IF(J.EQ.5)THEN
      TH=0.3927
      ENDIF
      IF(J.EQ.6)THEN
      TH=0.1309
      ENDIF
      SX1=((((PHI1+PHIX(1,J))/2.0)+PHIX(1,J+1))/2.0)*0.5*V(1)*SIN(TH)*
     +  COS(TH)*EXP(-MU1*5.0)
      JX1=JX1+SX1
10    CONTINUE
      DO 20 J=1,6
      IF(J.EQ.1)THEN
      TH=1.4399
      ENDIF
      IF(J.EQ.2)THEN
      TH=1.1781
      ENDIF
      IF(J.EQ.3)THEN
      TH=0.9163
      ENDIF
      IF(J.EQ.4)THEN
      TH=0.6545
      ENDIF
      IF(J.EQ.5)THEN
      TH=0.3927
      ENDIF
      IF(J.EQ.6)THEN
      TH=0.1309
      ENDIF
      SX2=(((PHIX(1,J)+PHIX(2,J+1))/2.0)*0.5*V(2))*SIN(TH)*
     +  COS(TH)*EXP(-MU2*15.0)
      JX2=JX2+SX2
20    CONTINUE
      DO 30 J=1,6
      IF(J.EQ.1)THEN
      TH=1.4399
      ENDIF
      IF(J.EQ.2)THEN
      TH=1.1781
      ENDIF
      IF(J.EQ.3)THEN
      TH=0.9163
      ENDIF
      IF(J.EQ.4)THEN
      TH=0.6545
      ENDIF
      IF(J.EQ.5)THEN
      TH=0.3927
      ENDIF
      IF(J.EQ.6)THEN
      TH=0.1309
      ENDIF
      SX3=(((PHIX(2,J)+PHIX(3,J+1))/2.0)*0.5*V(3))*SIN(TH)*
     +  COS(TH)*EXP(-MU3*25.0)
      JX3=JX3+SX3
30    CONTINUE
      DO 40 J=1,6
      IF(J.EQ.1)THEN
      TH=1.4399
      ENDIF
```

```
      IF(J.EQ.2)THEN
      TH=1.1781
      ENDIF
      IF(J.EQ.3)THEN
      TH=0.9163
      ENDIF
      IF(J.EQ.4)THEN
      TH=0.6545
      ENDIF
      IF(J.EQ.5)THEN
      TH=0.3927
      ENDIF
      IF(J.EQ.6)THEN
      TH=0.1309
      ENDIF
      SX4=(((PHIX(3,J)+PHIX(4,J+1))/2.0)*0.5*V(4))*SIN(TH)*
     +  COS(TH)*EXP(-MU4*35.0)
      JX4=JX4+SX4
40    CONTINUE
      DO 50 J=1,6
      IF(J.EQ.1)THEN
      TH=1.4399
      ENDIF
      IF(J.EQ.2)THEN
      TH=1.1781
      ENDIF
      IF(J.EQ.3)THEN
      TH=0.9163
      ENDIF
      IF(J.EQ.4)THEN
      TH=0.6545
      ENDIF
      IF(J.EQ.5)THEN
      TH=0.3927
      ENDIF
      IF(J.EQ.6)THEN
      TH=0.1309
      ENDIF
      SX5=(((PHIX(4,J)+PHIX(5,J+1))/2.0)*0.5*V(5))*SIN(TH)*
     +  COS(TH)*EXP(-MU5*45.0)
      JX5=JX5+SX5
50    CONTINUE
      DO 60 J=1,6
      IF(J.EQ.1)THEN
      TH=1.4399
      ENDIF
      IF(J.EQ.2)THEN
      TH=1.1781
      ENDIF
      IF(J.EQ.3)THEN
      TH=0.9163
      ENDIF
      IF(J.EQ.4)THEN
      TH=0.6545
      ENDIF
      IF(J.EQ.5)THEN
      TH=0.3927
      ENDIF
      IF(J.EQ.6)THEN
      TH=0.1309
      ENDIF
      SX6=(((PHIX(5,J)+PHIX(6,J+1))/2.0)*0.5*V(6))*SIN(TH)*
     +  COS(TH)*EXP(-MU6*55.0)
      JX6=JX6+SX6
60    CONTINUE
```

```
      DO 70 J=1,6
      IF(J.EQ.1)THEN
      TH=1.4399
      ENDIF
      IF(J.EQ.2)THEN
      TH=1.1781
      ENDIF
      IF(J.EQ.3)THEN
      TH=0.9163
      ENDIF
      IF(J.EQ.4)THEN
      TH=0.6545
      ENDIF
      IF(J.EQ.5)THEN
      TH=0.3927
      ENDIF
      IF(J.EQ.6)THEN
      TH=0.1309
      ENDIF
      SX7=(((PHIX(6,J)+PHIX(7,J+1))/2.0)*0.5*V(7))*SIN(TH)*
     +  COS(TH)*EXP(-MU7*65.0)
      JX7=JX7+SX7
70    CONTINUE
C
C     Next, calculate the volume segments represented by the points
C     along the plane x=0
C
      JZ1=0.0
      JZ2=0.0
      JZ3=0.0
      JZ4=0.0
      JZ5=0.0
      JZ6=0.0
      JZ7=0.0
      DO 80 J=1,6
      IF(J.EQ.1)THEN
      TH=1.4399
      ENDIF
      IF(J.EQ.2)THEN
      TH=1.1781
      ENDIF
      IF(J.EQ.3)THEN
      TH=0.9163
      ENDIF
      IF(J.EQ.4)THEN
      TH=0.6545
      ENDIF
      IF(J.EQ.5)THEN
      TH=0.3927
      ENDIF
      IF(J.EQ.6)THEN
      TH=0.1309
      ENDIF
      SZ1=((((PHI1+PHIZ(1,J))/2.0)+PHIZ(1,J+1))/2.0)*0.5*V(1)*SIN(TH)*
     +  COS(TH)*EXP(-MU1*5.0)
      JZ1=JZ1+SZ1
80    CONTINUE
      DO 90 J=1,6
      IF(J.EQ.1)THEN
      TH=1.4399
      ENDIF
      IF(J.EQ.2)THEN
      TH=1.1781
      ENDIF
      IF(J.EQ.3)THEN
```

```
      TH=0.9163
      ENDIF
      IF(J.EQ.4)THEN
      TH=0.6545
      ENDIF
      IF(J.EQ.5)THEN
      TH=0.3927
      ENDIF
      IF(J.EQ.6)THEN
      TH=0.1309
      ENDIF
      SZ2=(((PHIZ(1,J)+PHIZ(2,J+1))/2.0)*0.5*V(2))*SIN(TH)*
     +  COS(TH)*EXP(-MU2*15.0)
      JZ2=JZ2+SZ2
90    CONTINUE
      DO 100 J=1,6
      IF(J.EQ.1)THEN
      TH=1.4399
      ENDIF
      IF(J.EQ.2)THEN
      TH=1.1781
      ENDIF
      IF(J.EQ.3)THEN
      TH=0.9163
      ENDIF
      IF(J.EQ.4)THEN
      TH=0.6545
      ENDIF
      IF(J.EQ.5)THEN
      TH=0.3927
      ENDIF
      IF(J.EQ.6)THEN
      TH=0.1309
      ENDIF
      SZ3=(((PHIZ(2,J)+PHIZ(3,J+1))/2.0)*0.5*V(3))*SIN(TH)*
     +  COS(TH)*EXP(-MU3*25.0)
      JZ3=JZ3+SZ3
100   CONTINUE
      DO 110 J=1,6
      IF(J.EQ.1)THEN
      TH=1.4399
      ENDIF
      IF(J.EQ.2)THEN
      TH=1.1781
      ENDIF
      IF(J.EQ.3)THEN
      TH=0.9163
      ENDIF
      IF(J.EQ.4)THEN
      TH=0.6545
      ENDIF
      IF(J.EQ.5)THEN
      TH=0.3927
      ENDIF
      IF(J.EQ.6)THEN
      TH=0.1309
      ENDIF
      SZ4=(((PHIZ(3,J)+PHIZ(4,J+1))/2.0)*0.5*V(4))*SIN(TH)*
     +  COS(TH)*EXP(-MU4*35.0)
      JZ4=JZ4+SZ4
110   CONTINUE
      DO 120 J=1,6
      IF(J.EQ.1)THEN
      TH=1.4399
      ENDIF
```

```
      IF(J.EQ.2)THEN
      TH=1.1781
      ENDIF
      IF(J.EQ.3)THEN
      TH=0.9163
      ENDIF
      IF(J.EQ.4)THEN
      TH=0.6545
      ENDIF
      IF(J.EQ.5)THEN
      TH=0.3927
      ENDIF
      IF(J.EQ.6)THEN
      TH=0.1309
      ENDIF
      SZ5=(((PHIZ(4,J)+PHIZ(5,J+1))/2.0)*0.5*V(5))*SIN(TH)*
     +  COS(TH)*EXP(-MU5*45.0)
      JZ5=JZ5+SZ5
120   CONTINUE
      DO 130 J=1,6
      IF(J.EQ.1)THEN
      TH=1.4399
      ENDIF
      IF(J.EQ.2)THEN
      TH=1.1781
      ENDIF
      IF(J.EQ.3)THEN
      TH=0.9163
      ENDIF
      IF(J.EQ.4)THEN
      TH=0.6545
      ENDIF
      IF(J.EQ.5)THEN
      TH=0.3927
      ENDIF
      IF(J.EQ.6)THEN
      TH=0.1309
      ENDIF
      SZ6=(((PHIZ(5,J)+PHIZ(6,J+1))/2.0)*0.5*V(6))*SIN(TH)*
     +  COS(TH)*EXP(-MU6*55.0)
      JZ6=JZ6+SZ6
130   CONTINUE
      DO 140 J=1,6
      IF(J.EQ.1)THEN
      TH=1.4399
      ENDIF
      IF(J.EQ.2)THEN
      TH=1.1781
      ENDIF
      IF(J.EQ.3)THEN
      TH=0.9163
      ENDIF
      IF(J.EQ.4)THEN
      TH=0.6545
      ENDIF
      IF(J.EQ.5)THEN
      TH=0.3927
      ENDIF
      IF(J.EQ.6)THEN
      TH=0.1309
      ENDIF
      SZ7=(((PHIZ(6,J)+PHIZ(7,J+1))/2.0)*0.5*V(7))*SIN(TH)*
     +  COS(TH)*EXP(-MU7*65.0)
      JZ7=JZ7+SZ7
140    CONTINUE
```

```
C
C     Next, calculate the volume segments represented by the points
C     along the 45 deg plane from the dose point
C
      JXZ1=0.0
      JXZ2=0.0
      JXZ3=0.0
      JXZ4=0.0
      JXZ5=0.0
      JXZ6=0.0
      JXZ7=0.0
      DO 150 J=1,6
      IF(J.EQ.1)THEN
      TH=1.4399
      ENDIF
      IF(J.EQ.2)THEN
      TH=1.1781
      ENDIF
      IF(J.EQ.3)THEN
      TH=0.9163
      ENDIF
      IF(J.EQ.4)THEN
      TH=0.6545
      ENDIF
      IF(J.EQ.5)THEN
      TH=0.3927
      ENDIF
      IF(J.EQ.6)THEN
      TH=0.1309
      ENDIF
      SXZ1=((((PHI1+PHI45(1,J))/2.0)+PHI45(1,J+1))/2.0)*V(1)*
     +  SIN(TH)*COS(TH)*EXP(-MU1*5.0)
      JXZ1=JXZ1+SXZ1
150   CONTINUE
      DO 160 J=1,6
      IF(J.EQ.1)THEN
      TH=1.4399
      ENDIF
      IF(J.EQ.2)THEN
      TH=1.1781
      ENDIF
      IF(J.EQ.3)THEN
      TH=0.9163
      ENDIF
      IF(J.EQ.4)THEN
      TH=0.6545
      ENDIF
      IF(J.EQ.5)THEN
      TH=0.3927
      ENDIF
      IF(J.EQ.6)THEN
      TH=0.1309
      ENDIF
      SXZ2=(((PHI45(1,J)+PHI45(2,J+1))/2.0)*V(2))*SIN(TH)*
     +  COS(TH)*EXP(-MU2*15.0)
      JXZ2=JXZ2+SXZ2
160   CONTINUE
      DO 170 J=1,6
      IF(J.EQ.1)THEN
      TH=1.4399
      ENDIF
      IF(J.EQ.2)THEN
      TH=1.1781
      ENDIF
      IF(J.EQ.3)THEN
```

```
      TH=0.9163
      ENDIF
      IF(J.EQ.4)THEN
      TH=0.6545
      ENDIF
      IF(J.EQ.5)THEN
      TH=0.3927
      ENDIF
      IF(J.EQ.6)THEN
      TH=0.1309
      ENDIF
      SXZ3=(((PHI45(2,J)+PHI45(3,J+1))/2.0)*V(3))*SIN(TH)*
     +  COS(TH)*EXP(-MU3*25.0)
      JXZ3=JXZ3+SXZ3
170   CONTINUE
      DO 180 J=1,6
      IF(J.EQ.1)THEN
      TH=1.4399
      ENDIF
      IF(J.EQ.2)THEN
      TH=1.1781
      ENDIF
      IF(J.EQ.3)THEN
      TH=0.9163
      ENDIF
      IF(J.EQ.4)THEN
      TH=0.6545
      ENDIF
      IF(J.EQ.5)THEN
      TH=0.3927
      ENDIF
      IF(J.EQ.6)THEN
      TH=0.1309
      ENDIF
      SXZ4=(((PHI45(3,J)+PHI45(4,J+1))/2.0)*V(4))*SIN(TH)*
     +  COS(TH)*EXP(-MU4*35.0)
      JXZ4=JXZ4+SXZ4
180   CONTINUE
      DO 190 J=1,6
      IF(J.EQ.1)THEN
      TH=1.4399
      ENDIF
      IF(J.EQ.2)THEN
      TH=1.1781
      ENDIF
      IF(J.EQ.3)THEN
      TH=0.9163
      ENDIF
      IF(J.EQ.4)THEN
      TH=0.6545
      ENDIF
      IF(J.EQ.5)THEN
      TH=0.3927
      ENDIF
      IF(J.EQ.6)THEN
      TH=0.1309
      ENDIF
      SXZ5=(((PHI45(4,J)+PHI45(5,J+1))/2.0)*V(5))*SIN(TH)*
     +  COS(TH)*EXP(-MU5*45.0)
      JXZ5=JXZ5+SXZ5
190   CONTINUE
      DO 200 J=1,6
      IF(J.EQ.1)THEN
      TH=1.4399
      ENDIF
```

```
      IF(J.EQ.2)THEN
      TH=1.1781
      ENDIF
      IF(J.EQ.3)THEN
      TH=0.9163
      ENDIF
      IF(J.EQ.4)THEN
      TH=0.6545
      ENDIF
      IF(J.EQ.5)THEN
      TH=0.3927
      ENDIF
      IF(J.EQ.6)THEN
      TH=0.1309
      ENDIF
      SXZ6=(((PHI45(5,J)+PHI45(6,J+1))/2.0)*V(6))*SIN(TH)*
     +  COS(TH)*EXP(-MU6*55.0)
      JXZ6=JXZ6+SXZ6
200   CONTINUE
      DO 210 J=1,6
      IF(J.EQ.1)THEN
      TH=1.4399
      ENDIF
      IF(J.EQ.2)THEN
      TH=1.1781
      ENDIF
      IF(J.EQ.3)THEN
      TH=0.9163
      ENDIF
      IF(J.EQ.4)THEN
      TH=0.6545
      ENDIF
      IF(J.EQ.5)THEN
      TH=0.3927
      ENDIF
      IF(J.EQ.6)THEN
      TH=0.1309
      ENDIF
      SXZ7=(((PHI45(6,J)+PHI45(7,J+1))/2.0)*V(7))*SIN(TH)*
     +  COS(TH)*EXP(-MU7*65.0)
      JXZ7=JXZ7+SXZ7
210   CONTINUE
C
C     Next, calculate the volume segments represented by the points
C     along the 22.5 deg plane from the dose point
C
      J2251=0.0
      J2252=0.0
      J2253=0.0
      J2254=0.0
      J2255=0.0
      J2256=0.0
      J2257=0.0
      DO 220 J=1,6
      IF(J.EQ.1)THEN
      TH=1.4399
      ENDIF
      IF(J.EQ.2)THEN
      TH=1.1781
      ENDIF
      IF(J.EQ.3)THEN
      TH=0.9163
      ENDIF
      IF(J.EQ.4)THEN
      TH=0.6545
```

```
      ENDIF
      IF(J.EQ.5)THEN
      TH=0.3927
      ENDIF
      IF(J.EQ.6)THEN
      TH=0.1309
      ENDIF
      S2251=((((PHI1+PHI225(1,J))/2.0)+PHI225(1,J+1))/2.0)*V(1)*
     +  SIN(TH)*COS(TH)*EXP(-MU1*5.0)
      J2251=J2251+S2251
220   CONTINUE
      DO 230 J=1,6
      IF(J.EQ.1)THEN
      TH=1.4399
      ENDIF
      IF(J.EQ.2)THEN
      TH=1.1781
      ENDIF
      IF(J.EQ.3)THEN
      TH=0.9163
      ENDIF
      IF(J.EQ.4)THEN
      TH=0.6545
      ENDIF
      IF(J.EQ.5)THEN
      TH=0.3927
      ENDIF
      IF(J.EQ.6)THEN
      TH=0.1309
      ENDIF
      S2252=(((PHI225(1,J)+PHI225(2,J+1))/2.0)*V(2))*SIN(TH)*
     +  COS(TH)*EXP(-MU2*15.0)
      J2252=J2252+S2252
230   CONTINUE
      DO 240 J=1,6
      IF(J.EQ.1)THEN
      TH=1.4399
      ENDIF
      IF(J.EQ.2)THEN
      TH=1.1781
      ENDIF
      IF(J.EQ.3)THEN
      TH=0.9163
      ENDIF
      IF(J.EQ.4)THEN
      TH=0.6545
      ENDIF
      IF(J.EQ.5)THEN
      TH=0.3927
      ENDIF
      IF(J.EQ.6)THEN
      TH=0.1309
      ENDIF
      S2253=(((PHI225(2,J)+PHI225(3,J+1))/2.0)*V(3))*SIN(TH)*
     +  COS(TH)*EXP(-MU3*25.0)
      J2253=J2253+S2253
240   CONTINUE
      DO 250 J=1,6
      IF(J.EQ.1)THEN
      TH=1.4399
      ENDIF
      IF(J.EQ.2)THEN
      TH=1.1781
      ENDIF
      IF(J.EQ.3)THEN
      TH=1.1781
```

```
      TH=0.9163
      ENDIF
      IF(J.EQ.4)THEN
      TH=0.6545
      ENDIF
      IF(J.EQ.5)THEN
      TH=0.3927
      ENDIF
      IF(J.EQ.6)THEN
      TH=0.1309
      ENDIF
      S2254=(((PHI225(3,J)+PHI225(4,J+1))/2.0)*V(4))*SIN(TH)*
     +  COS(TH)*EXP(-MU4*35.0)
      J2254=J2254+S2254
250   CONTINUE
      DO 260 J=1,6
      IF(J.EQ.1)THEN
      TH=1.4399
      ENDIF
      IF(J.EQ.2)THEN
      TH=1.1781
      ENDIF
      IF(J.EQ.3)THEN
      TH=0.9163
      ENDIF
      IF(J.EQ.4)THEN
      TH=0.6545
      ENDIF
      IF(J.EQ.5)THEN
      TH=0.3927
      ENDIF
      IF(J.EQ.6)THEN
      TH=0.1309
      ENDIF
      S2255=(((PHI225(4,J)+PHI225(5,J+1))/2.0)*V(5))*SIN(TH)*
     +  COS(TH)*EXP(-MU5*45.0)
      J2255=J2255+S2255
260   CONTINUE
      DO 270 J=1,6
      IF(J.EQ.1)THEN
      TH=1.4399
      ENDIF
      IF(J.EQ.2)THEN
      TH=1.1781
      ENDIF
      IF(J.EQ.3)THEN
      TH=0.9163
      ENDIF
      IF(J.EQ.4)THEN
      TH=0.6545
      ENDIF
      IF(J.EQ.5)THEN
      TH=0.3927
      ENDIF
      IF(J.EQ.6)THEN
      TH=0.1309
      ENDIF
      S2256=(((PHI225(5,J)+PHI225(6,J+1))/2.0)*V(6))*SIN(TH)*
     +  COS(TH)*EXP(-MU6*55.0)
      J2256=J2256+S2256
270   CONTINUE
      DO 280 J=1,6
      IF(J.EQ.1)THEN
      TH=1.4399
      ENDIF
```

```
      IF(J.EQ.2)THEN
      TH=1.1781
      ENDIF
      IF(J.EQ.3)THEN
      TH=0.9163
      ENDIF
      IF(J.EQ.4)THEN
      TH=0.6545
      ENDIF
      IF(J.EQ.5)THEN
      TH=0.3927
      ENDIF
      IF(J.EQ.6)THEN
      TH=0.1309
      ENDIF
      S2257=(((PHI225(6,J)+PHI225(7,J+1))/2.0)*V(7))*SIN(TH)*
     +  COS(TH)*EXP(-MU7*65.0)
      J2257=J2257+S2257
280   CONTINUE
C
C     Next, calculate the volume segments represented by the points
C     along the 67.5 deg plane from the dose point
C
      J6751=0.0
      J6752=0.0
      J6753=0.0
      J6754=0.0
      J6755=0.0
      J6756=0.0
      J6757=0.0
      DO 290 J=1,6
      IF(J.EQ.1)THEN
      TH=1.4399
      ENDIF
      IF(J.EQ.2)THEN
      TH=1.1781
      ENDIF
      IF(J.EQ.3)THEN
      TH=0.9163
      ENDIF
      IF(J.EQ.4)THEN
      TH=0.6545
      ENDIF
      IF(J.EQ.5)THEN
      TH=0.3927
      ENDIF
      IF(J.EQ.6)THEN
      TH=0.1309
      ENDIF
      S6751=((((PHI1+PHI675(1,J))/2.0)+PHI675(1,J+1))/2.0)*V(1)*
     +  SIN(TH)*COS(TH)*EXP(-MU1*5.0)
      J6751=J6751+S6751
290   CONTINUE
      DO 300 J=1,6
      IF(J.EQ.1)THEN
      TH=1.4399
      ENDIF
      IF(J.EQ.2)THEN
      TH=1.1781
      ENDIF
      IF(J.EQ.3)THEN
      TH=0.9163
      ENDIF
      IF(J.EQ.4)THEN
      TH=0.6545
```

```
      ENDIF
      IF(J.EQ.5)THEN
      TH=0.3927
      ENDIF
      IF(J.EQ.6)THEN
      TH=0.1309
      ENDIF
      S6752=(((PHI675(1,J)+PHI675(2,J+1))/2.0)*V(2))*SIN(TH)*
     +  COS(TH)*EXP(-MU2*15.0)
      J6752=J6752+S6752
300   CONTINUE
      DO 310 J=1,6
      IF(J.EQ.1)THEN
      TH=1.4399
      ENDIF
      IF(J.EQ.2)THEN
      TH=1.1781
      ENDIF
      IF(J.EQ.3)THEN
      TH=0.9163
      ENDIF
      IF(J.EQ.4)THEN
      TH=0.6545
      ENDIF
      IF(J.EQ.5)THEN
      TH=0.3927
      ENDIF
      IF(J.EQ.6)THEN
      TH=0.1309
      ENDIF
      S6753=(((PHI675(2,J)+PHI675(3,J+1))/2..0)*V(3))*SIN(TH)*
     +  COS(TH)*EXP(-MU3*25.0)
      J6753=J6753+S6753
310   CONTINUE
      DO 320 J=1,6
      IF(J.EQ.1)THEN
      TH=1.4399
      ENDIF
      IF(J.EQ.2)THEN
      TH=1.1781
      ENDIF
      IF(J.EQ.3)THEN
      TH=0.9163
      ENDIF
      IF(J.EQ.4)THEN
      TH=0.6545
      ENDIF
      IF(J.EQ.5)THEN
      TH=0.3927
      ENDIF
      IF(J.EQ.6)THEN
      TH=0.1309
      ENDIF
      S6754=(((PHI675(3,J)+PHI675(4,J+1))/2.0)*V(4))*SIN(TH)*
     +  COS(TH)*EXP(-MU4*35.0)
      J6754=J6754+S6754
320   CONTINUE
      DO 330 J=1,6
      IF(J.EQ.1)THEN
      TH=1.4399
      ENDIF
      IF(J.EQ.2)THEN
      TH=1.1781
      ENDIF
      IF(J.EQ.3)THEN
```

```
      TH=0.9163
      ENDIF
      IF(J.EQ.4)THEN
      TH=0.6545
      ENDIF
      IF(J.EQ.5)THEN
      TH=0.3927
      ENDIF
      IF(J.EQ.6)THEN
      TH=0.1309
      ENDIF
      S6755=(((PHI675(4,J)+PHI675(5,J+1))/2.0)*V(5))*SIN(TH)*
     +  COS(TH)*EXP(-MU5*45.0)
      J6755=J6755+S6755
330   CONTINUE
      DO 340 J=1,6
      IF(J.EQ.1)THEN
      TH=1.4399
      ENDIF
      IF(J.EQ.2)THEN
      TH=1.1781
      ENDIF
      IF(J.EQ.3)THEN
      TH=0.9163
      ENDIF
      IF(J.EQ.4)THEN
      TH=0.6545
      ENDIF
      IF(J.EQ.5)THEN
      TH=0.3927
      ENDIF
      IF(J.EQ.6)THEN
      TH=0.1309
      ENDIF
      S6756=(((PHI675(5,J)+PHI675(6,J+1))/2.0)*V(6))*SIN(TH)*
     +  COS(TH)*EXP(-MU6*55.0)
      J6756=J6756+S6756
340   CONTINUE
      DO 350 J=1,6
      IF(J.EQ.1)THEN
      TH=1.4399
      ENDIF
      IF(J.EQ.2)THEN
      TH=1.1781
      ENDIF
      IF(J.EQ.3)THEN
      TH=0.9163
      ENDIF
      IF(J.EQ.4)THEN
      TH=0.6545
      ENDIF
      IF(J.EQ.5)THEN
      TH=0.3927
      ENDIF
      IF(J.EQ.6)THEN
      TH=0.1309
      ENDIF
      S6757=(((PHI675(6,J)+PHI675(7,J+1))/2.0)*V(7))*SIN(TH)*
     +  COS(TH)*EXP(-MU7*65.0)
      J6757=J6757+S6757
350   CONTINUE
C
C     Now, sum the segments, and multiply by 4 to account for the
C     backscatter contribution within the entire spherical volume
C     in the positive x-direction
```

```
C
      JS=((MUSC/3.14159)*(JX1+JX2+JX3+JX4+JX5+JX6+JX7+JZ1+JZ2+JZ3+
     + JZ4+JZ5+JZ6+JZ7+JXZ1+JXZ2+JXZ3+JXZ4+JXZ5+JXZ6+JXZ7+J2251+
     + J2252+J2253+J2254+J2255+J2256+J2257+J6751+J6752+J6753+J6754+
     + J6755+J6756+J6757))
C
C     Print out the result to the screen
C
      PRINT*,'BACKSCATTER JS IN FWD HEMISPHERE'
      PRINT*,JS
      STOP
      END
```