1 of 3

Characterization of the Sodium Void Reactivity Effect for Advanced Liquid Metal Reactor Fuels

S. F. Kessler

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CHARACTERIZATION OF THE SODIUM VOID REACTIVITY EFFECT FOR
ADVANCED LIQUID METAL REACTOR FUELS

by

Stephen F. Kessler

A thesis submitted in partial fulfillment
of the requirements for the degree of

Master of Science in Nuclear Engineering

University of Washington

1993

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Abstract

CHARACTERIZATION OF THE SODIUM VOID REACTIVITY EFFECT FOR
ADVANCED LIQUID METAL REACTOR FUELS

by Stephen F. Kessler

Chairperson of the Supervisory Committee: Professor Kermit L. Garlid
Department of Chemical Engineering

The problems of a large positive sodium void reactivity effect in liquid metal reactors have received increased attention following the accident at Chernobyl, a light water reactor with a positive coolant void coefficient. While the probability of voiding sodium is small, a large positive sodium void reactivity effect is, in many minds, unacceptable. Analyses were performed on models of an advanced liquid metal reactors to determine the effects fuel type have on the sodium void reactivity effect. Three fuel types were considered; metal, oxide, and nitride. Calculations were performed using three-dimensional, multigroup diffusion theory. Two programs were developed to aid the analyses. One calculated the capture-to-fission ratio and the other calculated reaction rates of selected materials. A one-group equation was derived to determine a theoretical basis for the sodium void reactivity effect. An option was presented for a shortened core having a near-zero sodium-void worth. The effect on the sodium void reactivity effect of using actinides as fuel is also considered.

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List of Acronyms

ALMR	Advanced Liquid Metal Reactor
ANL	Argonne National Laboratory
DOE	U.S. Department of Energy
ENDF/B-V	Evaluated Nuclear Data File/B Version V
FFTF	Fast Flux Test Facility
GE	General Electric Corporation
GEM	Gas Expansion Module
LMR	liquid metal reactors
NUREG	identifier on NRC documents
PRISM	Power Reactor Inherently Safe Module
SDP	slowing down power
SVRE	Sodium void reactivity effect
ULOF	unprotected loss of flow
USA	Ultimate Shutdown Assembly

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I. INTRODUCTION

Following the accidents at light water reactors at Three Mile Island and Chernobyl, reactor designers throughout the world have been working to develop inherently safe reactor designs. These efforts have included designs for breeder reactors using liquid sodium as the coolant. The excellent heat transfer characteristics of sodium make it attractive as a coolant, and, with a boiling point higher than the normal operating temperature of most reactor designs, it does not require high primary system pressure. Accident scenarios, however, may lead to sodium boiling in hotter portions of the core creating voids. In the United States, the Advanced Liquid Metal Reactor (ALMR) program is based on metal as the reference fuel because of its favorable neutronic feedback characteristics for passive safety. In particular, the Power Reactor Inherently Safe Module (PRISM) design combines favorable relationships between core performance and safety to provide a passively safe metal fuel system with a large margin to sodium boiling. This large margin is required because this design has a large positive sodium void reactivity effect (SVRE). A large coolant void worth was a major contributor to the severity of the accident at Chernobyl.

Most efforts to reduce the SVRE have concentrated on changing the physical characteristics of the reactor core while using a specific fuel type. Some studies^{1,2,3} have examined the effect of the fuel type on the SVRE. The fuel types analyzed were ternary metal fuel consisting of an alloy of uranium, plutonium, and zirconium, mixed oxide fuel, and mixed nitride fuel. These studies have shown that fuel type can significantly alter the SVRE. The purpose of this thesis is to determine the effect of fuel type on the sodium void and investigate the phenomena causing the problem. This investigation is limited to these three fuel types in a core configuration similar to the reference core^{4,5} supplied by the lead United States ALMR designer. Alterations to this design are limited such that the rest of the plant design

will not change. The goal is to determine which fuel type can be used to minimize the performance tradeoffs in a near-zero sodium void worth design. Using minor actinides from spent reactor fuel as a source of fuel to minimize waste, one of the promising uses of the ALMR, will also be considered but only with respect to its effect on the SVRE.

Fuel	42
Internal Blanket	25
Radial Blanket	36
Radial Reflector	42
Radial Shield	48
Control/Shutdown	6
Total		199

3

39105057.2

FIGURE 1: Advanced Liquid Metal Reactor Core Without Universal Shutdown Assembly or Gas Expansion Modules

39105057.3

FIGURE 2: Advanced Liquid Metal Reactor Core With Universal Shutdown Assembly and Gas Expansion Modules

II. SUMMARY AND CONCLUSIONS

Designers of liquid metal reactors (LMR) have long been aware of the problems associated with a large positive SVRE. Since the probability of gross sodium voiding in LMRs was thought to be extremely low, severe compromises in LMR designs to reduce the SVRE were generally considered unwarranted⁶. After the accident at Chernobyl, the potential reactivity effects of light water reactor coolant voiding have received increased attention in both the light water and LMR communities, and reducing, if not minimizing the SVRE is receiving increased attention⁷. To achieve this goal, several innovative ALMR designs with low and negative SVRE have been proposed^{8,9,10,11,12} showing that the importance of reducing this effect is receiving international attention. The optimum method for reducing the SVRE will retain other desirable reactor performance characteristics.

Diffusion theory was chosen for this study because it is an acceptable method for calculating the SVRE. Diffusion theory calculations of the SVRE were used in the design of the Fast Flux Test Facility (FFTF) at the U.S. Department of Energy (DOE) Hanford Site and were compared to results of fast test reactor engineering mockup critical assembly sodium void experiments with good agreement¹³. Calculations are typically performed using multigroup diffusion theory. This study used 12 neutron energy groups for the majority of the calculations. One set of one-group calculations was used for comparison to the 12-group calculations. The results are listed in Table 1. The one-group calculation overestimates the SVRE compared to the 12-group calculation for two reasons. First, in one-group, all spectral effects are accounted for by the code used to generate the cross sections. The code used for K_{eff} calculations handles the spectral effects differently and calculates a different result. Second, leakage is conserved when the transport cross section is collapsed by assuming the buckling is constant over all neutron energies. In

TABLE 1: 1-Group to 12-Group SVRE Comparison

Void Worth ($\Delta K/K$)		
Fuel Type	1-Group	12-Group
Oxide	0.0064	0.0046
Nitride	0.0083	0.0070
Metal	0.0243	0.0178

reality, buckling is dependent on energy, and errors in the transport cross sections are produced by collapsing from many fine groups to few coarse groups. These errors are maximized when collapsing to one-group. Many of the references to this paper emphasize the importance of using accurate cross sections to improve SVRE predictions. Other work recommends that multigroup calculations be used for fast reactor neutronics analysis because of the wide range of neutron energies over which most reactions occur¹⁴. Therefore, a one-group SVRE calculation is useful as an educational tool where accuracy is not considered to be of highest importance but where relationships between core designs is desired.

In comparative studies of the three fuel types, metal fuel always has the most positive SVRE and oxide fuel always has the least positive. The SVRE for nitride fuel is slightly more positive than that of oxide fuel but less positive than metal fuel.

To understand the effect of changes in fuel type, the SVRE may be split into three components; leakage, spectral, and capture. As sodium voids, leakage increases. Increasing leakage may have either a positive or negative effect on the SVRE depending on the location of the void in the core. The neutron spectrum hardens as the sodium voids. A harder spectrum results in reduced capture cross sections and a reduced capture rate. A harden spectrum also results in reduced fission cross sections and a reduced fission rate. However, for some isotopes, the capture rate decreases more rapidly than the fission rate causing the SVRE to be positive.

Typical capture-to-fission ratios for an ALMR are shown in Figures 3 and 4. In Figure 3, note how the ratio for ^{239}Pu decreases more rapidly than that for ^{235}U or ^{241}Pu as the neutron energy increases. A core driven by ^{239}Pu should have a more positive SVRE than one driven by either ^{235}U or ^{241}Pu . In Figure 4, the decrease in the ratio for ^{238}U is on a separate

FIGURE 3: Typical Fissile Material Capture-to-Fission Ratios

FIGURE 4: Typical Nonfissile Material Capture-to-Fission Ratios

graph to show how it dominates all other nonfissile isotopes. A fuel design which changes the amount of ^{238}U in the core will affect the SVRE. Removal of ^{238}U from the core will cause the SVRE to be less positive. It should also be noted that the scale of Figure 4 is much larger than the scale of Figure 3. The presence of large quantities of any of these nonfissile isotopes will cause the SVRE to be more positive.

A decreasing capture-to-fission ratio causes K_{eff} to increase unless leakage increases enough to compensate as shown in Section D of this paper. The effect of a decrease in capture-to-fission ratio on the SVRE is most positive in the metal core where the sodium is the most important moderator in the core. In an oxide or nitride core, the oxygen and nitrogen are much better moderators than the sodium and they remain in the core during coolant voiding. Their presence serves to limit spectral hardening in these cores.

The following derivation may serve to help determine the relative importance of the components of the SVRE. By taking the natural log of Equation 1 and differentiating with respect to the number of sodium atoms, the SVRE may be defined as:

$$\alpha_v = \frac{\partial \ln K_{eff}}{\partial N_{Na}} = \frac{1}{K_{eff}} \frac{\partial K_{eff}}{\partial N_{Na}} \quad (1)$$

Carrying out the differentiation of Equation 1 gives the following:

$$\alpha_v = \frac{1}{v} \frac{\partial v}{\partial N_{Na}} + \frac{1}{\Sigma_f} \frac{\partial \Sigma_f}{\partial N_{Na}} - \left(\frac{P_{nl}}{\Sigma_a} + \frac{(P_{nl}-1)}{\Sigma_{tr}} \right) \frac{\partial \Sigma_a}{\partial N_{Na}} - \frac{(P_{nl}-1)}{\Sigma_{tr}} \frac{\partial \Sigma_s}{\partial N_{Na}} \quad (2)$$

Fission
Absorption
Scatter

Where:

- ν = number of neutrons per fission
- P_{nl} = the non-leakage probability
- Σ_a = macroscopic absorption cross section
- Σ_{tr} = macroscopic transport cross section
- Σ_s = macroscopic scatter cross section.

The consequences of sodium voiding may be seen in this formula and are shown in Table 2. Increasing the leakage decreases the nonleakage probability, P_{nl} , contributing to a decrease in the SVRE by the change in the absorption term and through the change in the scattering term. Hardening the spectrum increases the number of neutrons per fission, ν , and reduces both the fission and absorption cross sections of most of the isotopes in the reactors. It has a minimal effect on the scattering cross section. Therefore, the change in ν is a positive contributor to the SVRE, the change in Σ_f is a negative contributor, and the change in Σ_a is a positive contributor. The effect of a reduction in capture will increase the positive contribution of the change in Σ_a on the SVRE. These observations explain why the SVRE can be positive in an ALMR.

It is the effect of the change in moderation that explains the difference in magnitude of the SVRE between metal, oxide, and nitride fuels. In a metal fueled reactor, the scattering cross section of the sodium coolant is a much larger fraction of the total scatter cross section of all the core materials, resulting in a harder spectrum. This causes a large positive contribution to the SVRE. For an oxide or nitride fueled ALMR, the oxygen and nitrogen constitute the largest fraction of the total scatter cross section. Thus, when the sodium voids, the change in moderation is smaller, limiting the spectral hardening, resulting in a smaller positive contribution to the SVRE. The net result is that a metal fueled ALMR will always have the most positive SVRE

TABLE 2. Effects of Hardening Neutron Spectrum on the SVRE

Term	Change due to Spectral Hardening	Effect on the SVRE	Reason
$\alpha_v(\nu)$	Positive	Positive	The number of neutrons per fission increases.
$\alpha_v(\Sigma_f)$	Negative	Negative	The fission cross section decreases.
$\alpha_v(\Sigma_a)$	Negative	Positive	Parasitic absorption decreases releasing more neutrons for fission.
$\alpha_v(\Sigma_s)$	Negative	Negative	Loss of scattering increases the leakage.

when compared to an oxide or nitride fueled ALMR of similar design. Likewise, core design changes to reduce the SVRE will always be more severe and consequently less durable in a metal fueled ALMR.

III. OBJECTIVE

Using the present United States ALMR design as a reference, and the reduction of the positive sodium void worth of the current core designs as an industry directive, this thesis study had three objectives. The first was to determine how the type of fuel affects the SVRE. The second was to investigate the physics of the problem by looking at the spectral effects of sodium voiding and macroscopic reaction rates of selected materials. The third objective was to present a near-zero sodium void worth design and consider the performance tradeoffs.

Another potential use of the ALMR would be to use minor actinides from spent thermal and fast reactor fuels as a fuel source. "Burning" these actinides not only produces energy but also could reduce the amount of high level waste requiring disposal. This was examined in this study only to show the effect minor actinides have on SVRE.

IV. METHODOLOGY

A. Core Models

Two reference core designs, both supplied by the General Electric Corporation (GE), the ALMR lead designer, were used to investigate how fuel types affected the SVRE. The first, shown in Figure 1, consisted of 42 fuel assemblies, 25 internal blanket assemblies, 6 control rods, 42 HT9 radial reflectors, and 48 boron carbide radial shield assemblies. The HT9 alloy is a ferritic stainless steel which has low irradiation swelling characteristics making its use attractive for cores requiring a long fuel lifetime. This core was 128 cm high and was used as the model for the initial sodium void calculations. The second, safety enhanced reference core design shown in Figure 2, had an ultimate shutdown assembly (USA) in place of the central blanket, three gas expansion modules (GEM) in place of three radial blankets, and inconel radial reflectors in place of the HT9 reflectors. The core height was also increased to 134.6 cm. The USA is a sodium channel through the core with boron-carbide balls in the region above the core. These balls would be released into the central core channel to shut the reactor down in the event of an accident. The GEMs are similar to the ones tested at the FFTF. During normal operation, they are sodium filled and act as reflectors. When flow is lost, the GEMs will become sodium voided as a compressed gas bubble in the top of the assembly expands. This void provides a leakage path for neutrons, adding negative reactivity during a loss-of-flow event. This safety enhanced model was used for developing a near-zero worth core by reducing core height. Effects on other reactor performance characteristics (i.e., breeding ratio and linear power, etc.) were also investigated. No attempts were made to optimize the core configuration during this study.

Ternary metal fuel, consisting of an alloy of uranium, plutonium, and zirconium, and binary metal blankets, consisting of an alloy of depleted uranium and zirconium, have been chosen as the reference fuel/blanket system for the United States ALMR. The advantages of this system are its high heat transfer characteristics for accident mitigation and high density for good breeding. While it has not yet been proven to work on an industrial scale, the Pyroprocessing of metal fuel may also be an advantage over other fuel types. Pyroprocessing is a process used to recover actinide elements from spent reactor fuel and recycling them for use as new reactor fuel and is based on electrowinning of metals. The process was developed in this country by Argonne National Laboratory (ANL)¹⁵. This system also has disadvantages which must be considered in the ALMR design. Metal fuel releases virtually all of the fission gas from the fuel matrix and therefore requires a large plenum above the top of the fuel. This large plenum is also required to accommodate the axial expansion of the fuel. A large pellet-to-clad gap is required to allow for the fuel's radial expansion. Because the fuel and cladding may interact to form a eutectic at temperatures slightly higher than normal operating temperature, the fuel has a low linear power limit. Eutectic penetration of the cladding could lead to the release of fission products to the coolant, or, in the worst case, release of fuel into the coolant. Finally, metal fuel has a large positive SVRE.

Unlike water, sodium does not need to be pressurized to prevent boiling at the normal operating temperatures in a reactor. It may be that low linear power and large margin to boiling under most accident scenarios would eliminate the problem of sodium voiding. However, as reported in NUREG-1368¹⁶, a bounding event consisting of an unprotected loss of flow (ULOF) with three pump coastdowns could result in sodium boiling within the first minute. A large SVRE would

then cause a power excursion event resulting in both fuel and cladding damage.

Oxide fuel has been chosen as the backup fuel for the United States ALMR because its physics are well known. It releases less fission gas, so plenum size may be reduced and core size may be increased without increasing reactor height. Because of its low density, the breeding ratio is reduced so that the ability to increase core height to improve the breeding becomes an important factor. Oxide fuel does have a higher linear power capability than metal fuel, but its thermal conductivity is much lower. In order to be considered inherently safe, a reactor using oxide fuel requires passive safety devices such as GEMs which act quickly to automatically insert negative reactivity during a loss of flow without scram event. The negative reactivity serves to limit fuel and clad temperatures to acceptable values. These have been incorporated in the design of the backup oxide core along with axial blankets to improve breeding. Another important characteristic of oxide fuel is the SVRE which, while still positive, is much smaller than that of metal fuel.

Nitride fuel is being considered for ALMRs in many countries (e.g., Commonwealth of Independent States [formerly USSR], France, and Japan) as an advanced fuel type that combines the attributes of both metal and oxide. Use of nitride fuel in large sodium cooled reactors is not a new idea. Its thermal conductivity is comparable to that of metal fuel. Fuel density is also comparable to metal fuel giving nitride fuel a similar breeding ratio. Like oxide fuel, nitride fuel has a relatively small positive SVRE. According to Bauer¹⁷, at temperatures below 1400°C, less than 10% of the fission gas is released from the fuel. This property, along with a high linear power capability, allows both a reduction in reactor height and core size as a potential method of reducing the

SVRE. With its high linear power capability, nitride fuel would seem to be the ideal ALMR fuel. However, operating experience with nitride fuels is limited and the buildup of carbon-14 through the (n, p) reaction with nitrogen-14 may be a drawback. Testing is continuing in France. Nitride fuel has been selected for use in the United States space reactor program. Testing of this fuel was performed at FFTF.

B. Factors Affecting Sodium Void

Hummel and Okrent¹⁸ split the SVRE into three components; spectral, capture, and leakage. The change in core reactivity resulting from the change in both the elastic and inelastic scattering cross sections as sodium voids is defined as the spectral component. While the slowing down power (SDP) is a measure of the spectral effect, it does not show in detail the neutron balance and competition among other factors contributing to the SVRE. The effects of capture-to-fission ratio, fuel/clad bond, and heavy metal composition also contribute to the spectral effect of SVRE. Similarly, the change in core reactivity resulting from the change in macroscopic capture cross sections as sodium voids is defined as the capture component. Heavy metal composition of the fuel also contributes to the capture component of the SVRE. The change in core reactivity resulting from the change in the transport cross section as sodium voids is defined as the leakage component. Core size contributes to the leakage component of the SVRE. For both ALMR cores the spectral component is the most important. The leakage component becomes important when shortening the core to obtain a near-zero void worth. The capture component is small in all cases. Hummel and Okrent also point out that the spectral component is much smaller in a ^{235}U driven, core than in a ^{239}Pu driven core. Other analyses performed on the ALMR reference core design¹⁹ showed that this design has a negative sodium void when driven by ^{235}U fuel.

C. Calculation of Reactor Parameters

In order to compare the three fuel types, a methodical approach of limiting the number of core parameters varied per calculation was used. Core size and fuel composition were initially held constant while an individual core parameter,

such as clad thickness, was altered and the sodium void worth calculated. A nitride fueled core containing a generic equilibrium blanket in all core blanket locations using the configuration of the first reference ALMR design was used as the starting point. This procedure continued until both metal and oxide cores of correct density, enrichment, pellet size, and pin dimensions were created. The final step in this process was to create realistic equilibrium cores based on data supplied by the lead designer for the reference metal core.

The ALMR was initially modeled in R-Z geometry and scoping studies performed in two-dimensional, multigroup diffusion theory²⁰. Using the first reference core design, shown in Figure 1, the assemblies were homogenized into concentric cylinders representing different material zones. This model was used only for the initial scoping studies of nitride and metal fuel which showed that use of nitride fuel had a potential for reducing the SVRE. The modeling of the control rods as a thin cylinder of material homogenized over the entire height of the core with the density of boron carbide being smeared over the rod height was considered to be too inaccurate for this analysis. To improve the accuracy and to be consistent with other analyses, three-dimensional, multigroup diffusion theory²¹ was used for all further analyses. The three-dimensional model was a triangular-Z model in which each assembly is modeled as a hexagon consisting of six triangles in the x and y plane. Axially, the assemblies were divided into layers based on the region (i.e., below core, core, above core). The below core region was one layer, the core region was divided into eight layers, and the above core region was divided into three layers. The materials in each assembly were homogenized throughout the hexagonal assembly area. Because this model was not burned and used an assumed equilibrium blanket assembly, material

densities were constant in the axial layers, except for the control rods. The control rods were modeled as HT9 and sodium channels up to the layer containing the boron carbide. This model, and a similar one for the updated ALMR core, was used for the void comparisons and the proposed near-zero rod worth core. A spreadsheet was used to convert the material densities from one fuel type to another and to correct the material densities as the pin parameters were modified. The model for the second reference core design was reduced to a 1/6 core model. Additionally, the GEM and USA were modeled as HT-9 and sodium channels in the core region.

Neutron cross sections were taken from the Evaluated Nuclear Data File/B Version V (ENDF/B-V) cross section library²². Multigroup cross section sets were calculated using the shielding factor method¹⁴ in twelve energy groups with five downscattering terms. The one dimensional diffusion theory code 1DX²³ was used to generate all cross section sets. For the initial studies using the first reference core, a new cross section set was prepared each time the material density was modified. The only exception to this method was for the equilibrium oxide core calculations. Both the equilibrium oxide core developed from the original equilibrium metal core and the equilibrium oxide core supplied by the lead designer used the same cross section set. Only one cross section set was prepared for each fuel type used in the second core model. Fifty-three energy group cross section sets with fifty-two downscattering terms were created for each equilibrium core. All cross section sets included separate sodium-flooded and sodium-voided cross sections.

Void worth was defined in two ways. For the first core, without GEMs or USA, the void worth was defined as the change in core reactivity due to the removal of coolant from the total core. The sodium bond in the pins of the nitride core

remained in the model. For the metal core, it was assumed that the radial expansion of the pellets expelled the sodium out of the bond after a short period of operation. The void worth of the reference core with GEMs and USA was defined as the removal of the coolant only from the fuel and blanket assemblies. This change was made to be consistent with void worths reported in the literature.

A value of less than one dollar for the void worth of the fuel and blankets is attractive for reactor safety in that the reactor will not go prompt critical if the coolant voids. When the affects of the radial reflectors and shields on the void is included, the total core void worth will be near zero.

D. Industry Computer Codes

Industry computer codes were used for the majority of the analyses. Multipurpose, one-dimensional²³ diffusion theory code M1DX was used to prepare resonance shielded cross sections in both 53 and 12 neutron energy groups, using a radial model consisting of concentric rings of material out to the core barrel. The thickness of the rings was determined by the area of the assemblies in each material zone, shown in Figure 5. To prepare the cross sections for the lower shield and upper plenum, plane geometry was used. Basic cross section data was from the ENDF/B-V library²². Cross sections were prepared for both sodium-flooded and sodium-voided cases and for each core parameter change.

Multigroup diffusion code 2DB²⁰ was used for the initial scoping studies on both nitride and metal fuel. R-Z geometry was used with the model consisting of concentric cylinders of material. In this model, the control rods were modeled as a

REGIONS

1. Inner Blanket
2. Inner Fuel
3. Middle Blanket
4. Control Rod
5. Outer Fuel
6. Radial Reflector
7. Radial Reflector
8. Radial Shield
9. Surrounding Sodium
10. Core Barrel

FIGURE 5: 1DX Radial Model

curtain of material. The concentration of material was determined by the amount of control rod insertion and homogenizing with the control rod channel.

Multigroup diffusion code 3DB²¹ was used for void calculations on all three fuel types. Triangular-Z geometry was used to match the hexagonal design of the ALMR core assemblies. Radially, each assembly, was broken down into six triangles. Axially, each assembly was broken down into twelve layers. The lowest layer was the lower shield, the next eight layers were the core region, and the top three layers were the upper shield, load pad, and plenum. This version also contains a burn module which was used to burn cores into equilibrium.

Perturbation theory code 3DP²⁴ was used to calculate the beta effective for the nitride core. For simplicity, this beta was used for all reactivity calculations.

E. New Computer Codes

Two programs were written to support this study. RATIO is a program written to calculate the capture-to-fission ratio. This program uses any cross section file and calculates the capture-to-fission ratios of selected isotopes by neutron energy group. A listing of RATIO is provided in Appendix A.

The second program was titled RATE and was designed to take a modified 3DB input deck, the cross section file, and the output neutron flux from the 3DB calculation and calculate macroscopic reaction rates for selected isotopes. Fluxes are averaged over each zone of the model and the fission, absorption, scatter, and fission reaction rates printed out by neutron energy groups. A listing of this program is provided in Appendix B.

V. RESULTS

A. 2DB Results

A nitride fuel was examined as a possible compromise solution to the sodium void problem of metal fuel and the low breeding ratio of oxide fuel. It has a thermal conductivity and density comparable to metal, a higher linear power capability, and releases very little fission gas. Table 3 lists the results of the scoping studies using 2DB in R-Z geometry. Both helium bonded and sodium bonded fuel and blanket pins were examined. The delayed neutron fraction, beta, was assumed to be the same for all three fuel types. Of interest is that the SVRE is approximately three dollars less than that for metal. Also, sodium bonded fuel has a slightly lower SVRE than helium bonded. The sodium bond also improves the thermal performance of the pin. The models for these cores were developed from data supplied by GE and used two different types of assemblies; one fresh fuel assembly and an estimated equilibrium blanket assembly. Number densities for the nitride cores were developed by substituting nitrogen for zirconium and correcting for density and isotopic fractions. Based on these results, it was decided to study nitride fuel further using sodium bonded pins. To be consistent with GE, all further studies were done in three dimensions, using 3DB, in triangular-Z geometry.

B. 3DB Results

The three dimensional model used for this part of the study consisted of the original reference core configuration using nitride fuel with the pin internal parameters modified, as shown in Figure 6, to take advantage of nitride fuel's swelling characteristics²⁵. This nitride reference core used fresh fuel and an estimated equilibrium blanket assembly

TABLE 3. 2-D Analysis Sodium Void Reactivity
Worths for the Reference Cores

Fuel Type	Full Core Sodium Void Worth (\$)
Metal	5.91
He Bonded Nitride	2.03
Na bonded Nitride	1.76

The delayed neutron fraction used for all fuel types was 0.00374.

Oxide and Nitride Fuel Pin

Metal Fuel Pin

Oxide and Nitride Blanket Pin

Metal Blanket Pin

FIGURE 6: Fuel and Blanket Pin Dimensions.

similar to the previous 2D work. A logical progression of changing an individual parameter and then performing a calculation to determine the effect on the void reactivity worth was used. The equilibrium nitride and oxide cores were developed from data supplied by GE and modified using the ORIGEN computer code²⁶. Oxide fuel and blanket pins had the same dimensions as the nitride pins but contained no sodium bond. The results are listed in Table 4. New cross section sets for both sodium-flooded and sodium-voided cores were prepared for each calculation. Starting with the nitride core as a reference, a metal core was developed by substituting one zirconium atom for each nitrogen atom and calculating the sodium void reactivity worth. Two calculations were performed to determine the effect of changing the fuel and blanket densities to the densities used for the reference metal core. The third calculation was performed with both the fuel and blanket densities corrected. Fuel composition was then corrected to the (Pu,U)Zr alloy used in the reference metal core. The next step was to correct the fuel and blanket pin dimensions to the reference core dimensions. Next, the fuel enrichment was reduced to get the K_{eff} for the flooded core as close as possible to unity. The final calculation was performed on an equilibrium reference metal core. A similar procedure was used for the oxide core. While some of these cores are unrealistic, it is instructive to note how various parameters effect SVRE. Changing the type of fuel has the largest effect followed by changing fuel enrichment and fuel/blanket pin density. Similar to the 2D analyses, the equilibrium nitride core has an SVRE approximately three dollars less than that of the equilibrium metal core. The equilibrium oxide core's void worth is more negative than that for the nitride. These cores are all similar in that they are plutonium driven. The major difference is in the fuel composition. Further study requires breaking down the SVRE effect into three components: leakage, capture, and spectral.

TABLE 4. 3-D Analysis Sodium Void Reactivity Worth Comparison

Fuel	Void Worth	
	($\Delta k/k$)	(\$)
Nitride		
Reference Core	0.0067	1.79
Equilibrium Core	0.0070	1.88
Metal		
One Zirconium Substituted for One Nitrogen	0.0138	3.67
Blanket Density Increased to Metal Density	0.0136	3.64
Fuel Density Increased to Metal Density	0.0130	3.47
Fuel and Blanket Density Increased to Metal Density	0.0127	3.40
Fuel Composition Corrected to (Pu,U)10Zr alloy	0.0124	3.32
Clad Thickness, Gap Thickness, and Pellet Diameter Corrected to Metal Reference Core Dimensions	0.0122	3.26
Enrichment reduced by 4%	0.0134	3.58
Metal Equilibrium Core	0.0178	4.76
Oxide		
One Oxygen Substituted for each Nitrogen	0.0083	2.22
Two Oxygens substituted for each Nitrogen	0.0041	1.10
Blanket Density Decreased to Oxide Density	0.0050	1.34
Fuel Density Decreased to Oxide Density	0.0025	0.67
Fuel and Blanket Density Decreased to Oxide Density	0.0022	0.59
Enrichment Increased by 2%	0.0030	0.80
Equilibrium Oxide Core	0.0047	1.25

Note: The delayed neutron fraction was set equal to 0.00374 for all cores.

C. 3DB Neutron Balance Results

By examining the data from the neutron balance tables in the 3DB output, one can gain further insight into the leakage and capture components of the sodium void. Table 5 contains a listing of these data for all three cores. The change in leakage between the sodium-flooded and -voided cores is similar for the three fuel types. Since the cores are the same size, this is to be expected. This also indicates that the change in leakage due to voiding the sodium is not a major contributor to the difference seen in the SVRE. However, the change in capture as sodium voids leads to a different conclusion. The magnitude of the change in capture for the metal core is approximately twice that of both the oxide and metal cores. Again, since these cores are very similar, the expectation would be that the change in capture should also be similar to the nitride and oxide cores. These data show that the change in capture is a major contributor to the sodium void worth and this change is due to spectral hardening as the sodium voids.

The last data included in the balance table is the fission source. As seen with leakage, only minor differences are seen between the three cores.

A value for the SVRE may also be calculated from these data. As shown in Table 6, these values are similar to the values calculated using the K_{eff} difference method.

TABLE 5. Neutron Balance by Process Component

Process Component	Neutron Change Due to Voiding Sodium*											
	Oxide			Nitride (U,Pu)N			Metal			Nitride (UN)		
	Pu	U	O	Pu	U	N	Pu	U	Zr	Pu	U	N
Fission Source, Fuel	-9.47	2.26	0	-6.50	2.68	0	-8.28	2.31	0	0	-7.73	0
Fission Source, Blanket	8.70	8.98	0	-3.45	8.32	0	-2.32	9.74	0	0.06	10.72	0
Capture Fuel	-10.79	-7.05	0.09	-9.18	-7.83	1.02	-11.16	-13.38	-0.42	0	-9.11	0.13
Capture Blanket	-1.20	-11.41	0.16	-2.16	-19.20	2.02	-2.40	-32.57	-0.72	-1.65	-12.74	0.25
Total HM Matrix Capture		-30.20			-35.33			-60.65			-23.12	

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* Per 1,000 Neutrons Generated for the Sodium-Flooded Case

TABLE 6. Sodium Void Neutron Balance Change

	Oxide		Nitride		Metal	
	Flooded	Voided	Flooded	Voided	Flooded	Voided
Leakage (n/cm ² -sec)	5.594E+17	1.258E+18	4.935E+17	1.023E+18	6.579E+17	1.383E+18
Capture (n/cm ² -sec)	2.556E+19	2.476E+19	2.613E+19	2.536E+19	2.577E+19	2.439E+19
Fission Source (n/cm ² -sec)	4.057E+19	4.062E+19	4.052E+19	4.057E+19	4.047E+19	4.051E+19
Delta Leakage (n/cm ² -sec)	6.626E+17		5.295E+17		7.251E+17	
Delta Capture (n/cm ² -sec)	-8.000E+17		-7.700E+17		-1.380E+18	
Delta Fission (n/cm ² -sec)	4.839E+16		4.474E+16		4.066E+16	
Net (n/cm ² -sec)	1.857E+17		2.852E+17		6.955E+17	
Δk Net	4.579E-03		7.038E-03		1.719E-02	
Rho (\$)	1.224		1.882		4.595	

D. Spectral Effects on the Capture-to-Fission Ratio

The effect of a spectral shift on the capture-to-fission ratio may be illustrated by considering the value of K_{eff} .

$$K_{eff} = \frac{\nu \Sigma_f}{\Sigma_a} \times P_{nl} \quad (3)$$

where:

ν = number of neutrons per fission

Σ_f = macroscopic fission cross section

Σ_a = macroscopic absorption cross section

P_{nl} = non-leakage probability.

Recall that $\Sigma_a = \Sigma_f + \Sigma_c$. As sodium is voided, if Σ_c decreases faster than Σ_f , Σ_a decreases faster than Σ_f , then K_{eff} increases. Conversely, if Σ_f decreases faster than Σ_c , K_{eff} decreases. Figures 7 to 9 show the capture-to-fission ratio of ^{239}Pu in the fuel assemblies for each fuel type. Note that the capture-to-fission ratio decrease as the spectrum hardens. Similarly, Figures 10 to 12 show the results for the ^{238}U in the ALMR fuel. The conclusion is that, for all three fuel types, K_{eff} should increase as sodium voids. Figures 13 through 18 show the same comparison between ^{239}Pu and ^{238}U for the blankets and lead to the same conclusion.

FIGURE 7: Oxide Core Inner Fuel ^{239}Pu Capture-to-Fission Ratio.

FIGURE 8: Nitride Core Inner Fuel ^{239}Pu Capture-to-Fission Ratio.

FIGURE 9: Metal Core Inner Fuel ^{239}Pu Capture-to-Fission Ratio.

FIGURE 10: Oxide Core Inner Fuel ^{238}U Capture-to-Fission Ratio.

FIGURE 11: Nitride Core Inner Fuel ^{238}U Capture-to-Fission Ratio.

FIGURE 12: Metal Core Inner Fuel ^{238}U Capture-to-Fission Ratio.

FIGURE 13: Oxide Core 2-Cycle Radial Blanket ^{239}Pu Capture-to-Fission Ratio.

FIGURE 14: Nitride Core 2-Cycle Radial Blanket ^{239}Pu Capture-to-Fission Ratio.

FIGURE 15: Metal Core 2-Cycle Radial Blanket ^{239}Pu Capture-to-Fission Ratio.

FIGURE 16: Oxide Core 2-Cycle Radial Blanket ^{238}U Capture-to-Fission Ratio.

FIGURE 17: Nitride Core 2-Cycle Radial Blanket ^{238}U Capture-to-Fission Ratio.

FIGURE 18: Metal Core 2-Cycle Radial Blanket ^{238}U Capture-to-Fission Ratio.

The next step in this investigation is to compare the capture-to-fission ratios of all the fissile and nonfissile heavy metal to the difference in flux between the sodium-flooded and -voided cases (i.e., the spectral shift caused by sodium voiding). Figures 19 through 24 show the capture-to-fission ratios of the fissile material for all three fuel types and the flooded-to-void flux difference. In all cases, the capture-to-fission ratios are decreasing as the flux difference peaks with the ^{239}Pu decreasing most rapidly. Similarly, Figures 25 through 30 show the data for the nonfissile material. In these cases, the decrease in the ^{238}U capture-to-fission ratio overwhelms the decreases in the other isotopes. Differences in the ratio between the three fuel types are small. However, the flux difference varies between the three fuel types. While the flux difference for nitride fuel is similar to that of oxide fuel, the metal fuel has a much higher peak flux difference. Thus, K_{eff} for the metal fuel increases more than either nitride or oxide giving it the most positive SVRE. Also of interest is that the ^{235}U capture-to-fission ratio decreases at the slowest rate. This explains why a ^{235}U driven core does not have the similar positive SVRE problem of a ^{239}Pu driven core¹⁹.

E. Slowing Down Power

During normal operation of fast reactors, the moderation of neutrons is not considered to be important. However, when discussing the effects of spectral hardening on the sodium void, it is important to investigate the relative slowing down power (SDP) of the three fuel types. The SDP is defined as

$\xi \Sigma_s$, where ξ is approximately equal to $\frac{2}{A + \frac{2}{3}}$, A is the atomic number, and Σ_s is the macroscopic scattering cross section.

FIGURE 19: Oxide Core Inner Fuel Fissile Material Capture-to-Fission Ratio.

FIGURE 20: Nitride Core Inner Fuel Fissile Material Capture-to-Fission Ratio.

FIGURE 21: Metal Core Inner Fuel Fissile Material Capture-to-Fission Ratio.

FIGURE 22: Oxide Core 2-Cycle Radial Blanket Fissile Material Capture-to-Fission Ratio.

FIGURE 23: Nitride Core 2-Cycle Radial Blanket Fissile Material Capture-to-Fission Ratio.

FIGURE 24: Metal Core 2-Cycle Radial Blanket Fissile Material Capture-to-Fission Ratio.

FIGURE 25: Oxide Core Inner Fuel Nonfissile Material Capture-to-Fission Ratio.

FIGURE 26: Nitride Core Inner Fuel Nonfissile Material Capture-to-Fission Ratio.

FIGURE 27: Metal Core Inner Fuel Nonfissile Material Capture-to-Fission Ratio.

FIGURE 28: Oxide Core 2-Cycle Radial Blanket Nonfissile Material Capture-to-Fission Ratio.

FIGURE 29: Nitride Core 2-Cycle Radial Blanket Nonfissile Material Capture-to-Fission Ratio.

FIGURE 30: Metal Core 2-Cycle Radial Blanket Nonfissile Material Capture-to-Fission Ratio.

Another term used in thermal reactor physics is the moderating ratio which is defined as the SDP divided by the macroscopic absorption cross section, Σ_a . For this investigation, it was desired to separate fission from capture and the SDP was divided by the macroscopic capture cross section, Σ_c . The modified moderating ratio generated by this method was referred to by Russian participants at the 1990 International Fast Reactor Safety meeting held in Snowbird Utah²⁷. In a question and answer session, they claimed that this ratio was the reason for the lower void coefficient in nitride fuel. The results of this investigation support the hypothesis that $\xi\Sigma_g/\Sigma_c$ is a major contributor to the sodium void worth, but also that there are other major contributors. The specific results of this investigation are contained in Tables 7 through 9 for sodium-flooded and sodium-voided cores. Note that for the sodium bonded pins, the bond region is not voided with the coolant. In the metal core, the irradiated fuel and blanket pins are assumed to have expanded into the bond region displacing the sodium. The contribution to the total SDP of the HT-9 steel was considered to be minimal and not included for any core. All macroscopic cross sections were taken from three dimensional analyses of each core and normalized to 1,000 neutrons.

The SDP for the heavy metal in the three fuel types is similar and changes little when sodium is voided. The same is true for capture in the heavy metal. As expected, the SDP for the sodium is greater than that for the heavy metal, but is of the same magnitude for all cores. Capture in sodium for all cores is insignificant. Consider the SDP for the other elements in the fuel matrix. Both the nitrogen and the oxygen have SDP's a factor of ten greater than that for zirconium, a relationship which is constant for sodium-flooded and -voided cores. It is apparent that the major contributor to the SDP in the nitride and oxide cores is the nitrogen and oxygen

respectively while the major contributor in the metal core is the sodium. Also of interest is the slight increase in (n,p) and (n,α) reactions in the nitrogen as sodium is voided. These reactions are not major contributors to the SVRE.

Table 10 is a summary of Tables 7 through 9 using a modified moderating ratio generated by dividing the SDP by Σ_c . The oxide core always has the highest ratio with the metal core having the smallest. Since the sodium is not the major contributor to the SDP in the nitride and oxide cores, voiding of sodium changes the total core SDP approximately 20%. This increases to approximately 60% for the metal core. This analysis points out that the presence of a moderating atom in the fuel matrix acts to limit spectral hardening for both the nitride and oxide cores when sodium is voided. The effect is that the sodium void reactivity worth is lower in nitride and oxide cores than the metal cores of the same general design.

F. Reaction Rates

A comparison of macroscopic reaction rates between sodium-flooded and sodium-voided cores also illustrates the effect of a spectral shift on the sodium void coefficient. The program RATE was written to calculate these reaction rates averaged over an assembly, by using a standard 3DB input deck, cross section file, and the flux file generated by the 3DB analysis. Figures 31 through 36 show the difference in the absorption, capture, and fission reaction rates between the sodium-flooded and sodium-voided cores. Only the ^{239}Pu and the ^{238}U are shown. One conclusion which may be made is that, in the nitride and oxide cores, which have less spectral hardening than the metal core, the reaction rate differences are smaller. A second conclusion is that the ^{238}U is a contributor to the positive sodium void since its fission reaction rate difference is negative at low neutron energies

TABLE 7. Slowing Down Power for the Oxide Core.
(Based on 10^3 Neutrons)

Material	Flooded				Voided			
	ξ	Σ_s	$\xi \Sigma_s$	Σ_c	ξ	Σ_s	$\xi \Sigma_s$	Σ_c
Heavy Metal	0.00834	13578.1	113.241	473.08	0.00834	13763.73	114.79	442.63
Sodium	0.0845	4150.93	350.754	1.70	0.0845	654.79	55.33	0.29
Oxygen	0.12	10189.6	1222.75	2.61	0.12	10472.98	1256.75	2.86
$\frac{\xi \Sigma_s}{\Sigma_c} = \frac{113.241 + 1222.75 + 350.754}{473.08 + 2.61 + 1.70} = \frac{1686.74}{477.39} = 3.53$					$\frac{\xi \Sigma_s}{\Sigma_c} = \frac{114.79 + 1256.75 + 55.33}{442.63 + 2.86 + 0.29} = \frac{1426.87}{445.63} = 3.2$			

TABLE 8. Slowing Down Power for the Nitride Core.
(Based on 10^3 Neutrons)

Material	Flooded				Voided			
	ξ	Σ_s	$\xi\Sigma_s$	Σ_c	ξ	Σ_s	$\xi\Sigma_s$	Σ_c
Heavy Metal	0.00834	13531.05	112.85	419.40	0.00834	13095.18	109.21	380.58
Sodium	0.0845	3413.88	288.45	1.327	0.0845	497.49	42.04	0.192
Nitrogen	0.1364	6080.20	829.34	37.47	0.1364	5744.90	783.60	40.47

$$\frac{\xi\Sigma_s}{\Sigma_c} = \frac{112.85+829.34+288.45}{419.40+37.47+1.3274} = 2.686$$

$$\frac{\xi\Sigma_s}{\Sigma_c} = \frac{109.21+783.60+42.04}{380.58+40.47+0.1922} = 2.219$$

	Flooded				Voided			
Material	ξ	Σ_s	$\xi \Sigma_s$	Σ_c	ξ	Σ_s	$\xi \Sigma_s$	Σ_c
Heavy Metal	0.00834	15024.88	125.31	395.02	0.00834	14306.58	119.32	335.04
Sodium	0.0845	4121.49	348.27	1.1545	0.0845	303.95	25.68	0.08086
Zirconium	0.02206	3831.41	84.52	8.97	0.02206	3743.75	82.59	7.81
	$\frac{\xi \Sigma_s}{\Sigma_c} = \frac{125.31+84.52+348.27}{395.02+8.97+1.1545} = 1.378$				$\frac{\xi \Sigma_s}{\Sigma_c} = \frac{119.32+82.59+25.68}{335.04+7.81+0.08086} = 0.6637$			

TABLE 10. Slowing Down Power Summary.

Core Type	$\xi\Sigma_s/\Sigma_c$		Fractional Contribution for Sodium Flooded Core		
	Sodium Flooded	Sodium Voided	Heavy Metal	N/Zr/O	Sodium
Nitride	2.686	2.219	0.092	0.674 (N)	0.234
Metal	1.378	0.664	0.225	0.151 (Zr)	0.624
Oxide	3.533	3.201	0.067	0.725 (O)	0.208

FIGURE 31: ^{239}Pu Absorption Reaction Rate Difference in an Inner Fuel Assembly.

FIGURE 32: ^{241}Pu Absorption Reaction Rate Difference in an Inner Fuel Assembly.

FIGURE 33: ^{235}U Absorption Reaction Rate Difference in an Inner Fuel Assembly.

FIGURE 34: ^{238}U Absorption Reaction Rate Difference in an Inner Fuel Assembly.

FIGURE 35: ^{240}Pu Absorption Reaction Rate Difference in an Inner Fuel Assembly.

FIGURE 36: ^{242}Pu Absorption Reaction Rate Difference in an Inner Fuel Assembly.

but becomes positive as neutron energy increases. The ^{238}U is a significant contributor to the SVRE as is the ^{240}Pu . The contribution of the ^{242}Pu is several orders of magnitude smaller than either of the other isotopes. This seems to support the hypothesis in Reference 4 that replacing the ^{240}Pu with ^{241}Pu may help lower the void coefficient. However, the positive contribution of the ^{238}U is so much higher than that of the ^{240}Pu , other void reduction techniques must be used to design a breeder reactor with a near zero sodium void worth. Appendix C contains the figures from the detailed study of these reaction rates.

The fission reaction rates of the fissile isotopes remain significant as neutron energy increases. Unlike the non-fissile isotopes, the fissile isotopes fission reaction rates are significant over the entire range of neutron energies. Therefore, the reduction in the capture reaction rates for the fissile isotopes is accompanied by a corresponding reduction in the fission reaction rates. For the non-fissile isotopes, the reduction in the capture reaction rates are accompanied by an increase in the fission reaction rates. This also points to the fission to capture ratio as being a major contributor to the sodium void. As discussed previously, the ^{238}U has the largest affect on the void coefficient.

G. Effects of Increased Capture in Nitrogen

Previous work ²⁸ claimed that the reason for a lower SVRE in nitride fuels was due to increased capture in nitrogen as neutron energy increases. Nitrogen does have an increased absorption cross section for both an (n,p) and an (n, α) at neutron energies above 0.4 MeV and 1 MeV respectively. Figure 37 shows the difference macroscopic capture reaction rate between sodium-voided and sodium-flooded cores for the

FIGURE 37: Voided to Flooded Capture Reaction Rate Difference.

nitrogen in the model used for this thesis. This difference is small compared to the difference in total core capture presented in Table 5. While the increase in capture in the voided core aids in lowering the void coefficient, it does not appear to be a major contributor. Figure 37 shows the macroscopic capture reaction rates for oxygen and zirconium. There is no significant high energy absorption in either element.

H. Derivation of the Sodium Void Reactivity Worth

Consider a two region reactor with a core and a reflector.

FIGURE 38: Two Region Reactor with a Core and Reflector

The one-group diffusion equation is:

$$\frac{\partial n(x, t)}{\partial t} = S(x, t) - \Sigma_a(x) \Phi(x, t) + \nabla \cdot D(x) \nabla \Phi(x, t) \quad (1)$$

In steady state and one dimension, assuming constant Σ_a and D , Equation 1 reduces to:

$$0 = S(x) - \Sigma_a \Phi(x) + D \frac{d^2}{dx^2} \Phi(x) \quad (2)$$

The volumetric source in this core is:

$$S(x) = K_{\infty} \Sigma_a \Phi(x) = v \Sigma_f \Phi(x) \quad (3)$$

Rewriting the diffusion equation:

$$0 = K_{\infty} \Sigma_a \Phi(x) - \Sigma_a \Phi(x) + D \frac{d^2}{dx^2} \Phi(x) \quad (4)$$

or

$$D \frac{d^2}{dx^2} \Phi(x) + (K_{\infty} - 1) \Sigma_a \Phi(x) = 0 \quad (5)$$

Let $L^2 = D/\Sigma_a$ and $B^2 = (K_{\infty} - 1)/L^2$. Then the diffusion equation becomes:

$$\frac{d^2}{dx^2} \Phi(x) + B^2 \Phi(x) = 0 \quad (6)$$

For the reflector, the diffusion equation is:

$$\frac{dn(t)}{dt} = S(x) - \Sigma_a \Phi(x) + D \frac{d^2}{dx^2} \Phi(x) \quad (7)$$

In steady state and, since there is no source in the reactor:

$$\begin{aligned} 0 &= -\Sigma_a \Phi(x) + D \frac{d^2}{dx^2} \Phi(x) \\ \frac{d^2}{dx^2} \Phi(x) - \frac{1}{L^2} \Phi(x) &= 0 \end{aligned} \quad (8)$$

The boundary conditions are:

$$\begin{aligned} J_{\text{core}}(0) &= 0 \Rightarrow -D_{\text{core}} \frac{d}{dx} \Phi_{\text{core}}(x) = 0 \Rightarrow \frac{d}{dx} \Phi_{\text{core}}(x) = 0 \\ \Phi_{\text{reflector}}(c) &= 0 \\ \Phi_{\text{core}}(a) &= \Phi_{\text{reflector}}(a) \\ J_{\text{core}}(a) &= J_{\text{reflector}}(a) \Rightarrow -D_{\text{core}} \frac{d}{dx} \Phi_{\text{core}}(a) = -D_{\text{reflector}} \frac{d}{dx} \Phi_{\text{reflector}}(a) \end{aligned} \quad (9)$$

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These differential equations may be solved using Euler's technique. Solving for the flux in the core:

$$\begin{aligned}
 \Phi(x) &= Ce^{\alpha x} \\
 C\alpha^2 e^{\alpha x} + B^2 Ce^{\alpha x} &= 0 \\
 \alpha^2 + B^2 &= 0 \\
 \alpha^2 &= -B^2 \Rightarrow \alpha = \pm iB \\
 \Phi_{core}(x) &= C_1 e^{iBx} + C_2 e^{-iBx} \\
 e^{\pm ix} &= \cos x \pm i \sin x \\
 \Phi_{core}(x) &= C_1 (\cos Bx + i \sin Bx) + C_2 (\cos Bx - i \sin Bx) \\
 &= (C_1 + C_2) \cos Bx + (C_1 - C_2) i \sin Bx \\
 &= A \cos Bx + C \sin Bx
 \end{aligned} \tag{10}$$

Solving for the flux in the reflector:

$$\begin{aligned}
 C\alpha^2 e^{\alpha x} - \frac{1}{L^2} Ce^{\alpha x} &= 0 \\
 \alpha^2 - \frac{1}{L^2} &= 0 \\
 \alpha^2 &= \frac{1}{L^2} \Rightarrow \alpha = \pm \frac{1}{L} \\
 \Phi_{reflector}(x) &= C_1 e^{\frac{x}{L}} + C_2 e^{-\frac{x}{L}} \\
 e^{\pm x} &= \cosh x \pm \sinh x \\
 \Phi_{reflector}(x) &= C_1 (\cosh \frac{x}{L} + \sinh \frac{x}{L}) + C_2 (\cosh \frac{x}{L} - \sinh \frac{x}{L}) \\
 &= (C_1 + C_2) \cosh \frac{x}{L} + (C_1 - C_2) \sinh \frac{x}{L} \\
 &= E \cosh \frac{x}{L} + F \sinh \frac{x}{L}
 \end{aligned} \tag{11}$$

Therefore:

$$\begin{aligned}
 \Phi_{core} &= A \cos Bx + C \sin Bx \\
 \Phi_{reflector} &= E \cosh \frac{x}{L} + F \sinh \frac{x}{L}
 \end{aligned} \tag{12}$$

The boundary conditions are now applied:

$$\begin{aligned}
 & \text{at } x=0, \frac{d}{dx} \Phi_{core}(x) = 0 \\
 & -AB \sin(0) + CB \cos(0) = 0 \Rightarrow C=0 \\
 & \therefore \Phi_{core}(x) = A \cos Bx \\
 & \text{at } x=C, \Phi_{reflector}(x) = 0 \\
 & E \cosh \frac{C}{L} + F \sinh \frac{C}{L} = 0 \\
 & F = -E \frac{\cosh \frac{C}{L}}{\sinh \frac{C}{L}} \\
 & \Phi_{reflector}(x) = E \cosh \frac{x}{L} - E \frac{\cosh \frac{C}{L}}{\sinh \frac{C}{L}} \sinh \frac{x}{L} \quad (13) \\
 & \Phi_{reflector}(x) = \frac{E}{\sinh \frac{C}{L}} \left(\cosh \frac{x}{L} \sinh \frac{C}{L} - \sinh \frac{x}{L} \cosh \frac{C}{L} \right) \\
 & \sinh(x \pm y) = \sinh x \cosh y \pm \sinh y \cosh x \\
 & \therefore \Phi_{reflector}(x) = \frac{E}{\sinh \frac{C}{L}} \sinh \left(\frac{C-x}{L} \right) = G \sinh \left(\frac{C-x}{L} \right) \\
 & \text{where } G = \frac{E}{\sinh \frac{C}{L}}
 \end{aligned}$$

$$\begin{aligned}
 \text{Now } \Phi_{core}(x) &= A \cos Bx \\
 \Phi_{reflector}(x) &= G \sinh \frac{C-x}{L} \quad (14)
 \end{aligned}$$

$$\begin{aligned}
 \text{at } x=a, \Phi_{core}(a) &= \Phi_{reflector}(a) \\
 A \cos Ba &= G \sinh \frac{C-a}{L_{reflector}} = G \sinh \frac{b}{L_{reflector}} \\
 & \text{where } b = C-a \\
 \text{also at } x=a, D_{core} \frac{d}{dx} \Phi_{core}(a) &= D_{reflector} \frac{d}{dx} \Phi_{reflector}(a) \quad (15) \\
 D_{core} A B \sin Ba &= \frac{D_{reflector} G}{L_{reflector}} \cosh \frac{b}{L_{reflector}}
 \end{aligned}$$

This gives two equations in two unknowns:

$$\begin{aligned} \cos Ba \times A - \sinh \frac{b}{L_{\text{reflector}}} \times G &= 0 \\ D_{\text{core}} \sin Ba \times A - \frac{D_{\text{reflector}}}{L_{\text{reflector}}} \cosh \frac{b}{L_{\text{reflector}}} \times G &= 0 \end{aligned} \quad (16)$$

Using Cramer's rule and setting the determinant of the coefficients equal to zero:

$$\begin{vmatrix} \cos Ba & -\sinh \frac{b}{L_{\text{reflector}}} \\ D_{\text{core}} B \sin Ba & -\frac{D_{\text{reflector}}}{L_{\text{reflector}}} \cosh \frac{b}{L_{\text{reflector}}} \end{vmatrix} = 0 \quad (17)$$

$$\begin{aligned} -\frac{D_{\text{reflector}}}{L_{\text{reflector}}} \cos Ba \cosh \frac{b}{L_{\text{reflector}}} + D_{\text{core}} B \sin Ba \sinh \frac{b}{L_{\text{reflector}}} &= 0 \\ \frac{D_{\text{reflector}}}{L_{\text{reflector}}} \cos Ba \cosh \frac{b}{L_{\text{reflector}}} &= D_{\text{core}} B \sin Ba \sinh \frac{b}{L_{\text{reflector}}} \\ \frac{\cosh \frac{b}{L_{\text{reflector}}}}{\sinh \frac{b}{L_{\text{reflector}}}} &= D_{\text{core}} B \frac{\sin Ba}{\cos Ba} \\ \frac{D_{\text{reflector}}}{L_{\text{reflector}}} \tanh \frac{b}{L_{\text{reflector}}} &= D_{\text{core}} B \tan Ba \end{aligned} \quad (18)$$

Recall the diffusion equation:

$$\begin{aligned} \frac{d^2}{dx^2} \Phi(x) + B^2 \Phi(x) &= 0 \\ B^2 &= \frac{K_{\infty} - 1}{L_{\text{core}}^2} \end{aligned} \quad (19)$$

This is a relationship between the dimensions and its composition.

Consider a bare reactor:

$$\begin{aligned}
 &\text{as } b \rightarrow 0, \sinh \frac{b}{L_{\text{reflector}}} \rightarrow 0, \tanh \frac{b}{L_{\text{reflector}}} \rightarrow 0 \\
 &\quad \therefore D_{\text{core}} B \tan Ba \rightarrow \infty \\
 &\text{as } Ba \rightarrow \frac{\pi}{2} \quad \tan Ba \rightarrow \infty \\
 &\quad \text{Let } Ba = \frac{\pi}{2} \\
 &\quad \text{then } B = \frac{\pi}{2a} \\
 &\quad B^2 = \left(\frac{\pi}{2a} \right)^2 = \frac{K_{\infty} - 1}{L_{\text{core}}^2} \tag{20} \\
 &\quad K_{\infty} - 1 = \left(\frac{\pi}{2a} \right)^2 L_{\text{core}}^2 \\
 &\quad K_{\infty} = 1 + \left(\frac{\pi}{2a} \right)^2 L_{\text{core}}^2 \\
 &\quad \frac{K_{\infty}}{1 + \left(\frac{\pi}{2a} \right)^2 L_{\text{core}}^2} = 1
 \end{aligned}$$

The final equation is the criticality condition for a bare slab reactor. Recall that K_{eff} is the effective multiplication constant and, when it equals 1, the reactor is said to be critical. This equation is also for the lowest harmonic of the solution. Higher harmonics are not considered. Therefore, the critical condition for the bare slab reactor is:

$$\begin{aligned}
 K_{\text{eff}} &= \frac{K_{\infty}}{1 + \left(\frac{\pi}{2a} \right)^2 L_{\text{core}}^2} \\
 &= \frac{K_{\infty}}{1 + B^2 L^2} \tag{21}
 \end{aligned}$$

The sodium void reactivity effect, α_{Na} , may be defined as:

$$\alpha_v \equiv \alpha_{Na} = \frac{\partial \rho}{\partial N_{sodium}} \quad (22)$$

Recall that ρ is defined as:

$$\rho = \frac{K_{eff} - 1}{K_{eff}} = 1 - \frac{1}{K_{eff}} = 1 - K_{eff}^{-1} \quad (23)$$

Now α_v may be written as:

$$\alpha_v = \frac{\partial \rho}{\partial N_{sodium}} = \frac{1}{K_{eff}^2} \times \frac{\partial K_{eff}}{\partial N_{sodium}} \quad (24)$$

Since K_{eff} is approximately equal to 1, the equation may be simplified to:

$$\frac{\partial \rho}{\partial N_{sodium}} = \frac{1}{K_{eff}} \times \frac{\partial K_{eff}}{\partial N_{sodium}} \quad (25)$$

In one-group, the critical condition for a bare reactor is:

$$K_{eff} = \frac{K_{\infty}}{1 + L^2 \times B^2} \quad (26)$$

Where the nonleakage probability may be defined as:

$$P_{nl} = \frac{1}{1 + L^2 \times B^2} \quad (27)$$

$$\therefore K_{eff} = K_{\infty} \times P_{nl} \quad (28)$$

$$\text{Recall that } K_{\infty} = \frac{v \times \Sigma_f}{\Sigma_a} \quad (29)$$

$$\text{Then } K_{\text{eff}} = \frac{v \Sigma_f}{\Sigma_a} \times P_{nl} \quad (30)$$

Take the natural log of K_{eff}

$$\begin{aligned} \ln K_{\text{eff}} &= \ln \left(\frac{v \Sigma_f}{\Sigma_a} \times P_{nl} \right) = \ln \left(\frac{v \Sigma_f}{\Sigma_a} \right) + \ln P_{nl} \\ &= \ln v + \ln \Sigma_f - \ln \Sigma_a + \ln P_{nl} \end{aligned} \quad (31)$$

Differentiating with respect to N_{Na} , the number of sodium atoms:

$$\alpha_v = \frac{1}{K_{\text{eff}}} \frac{\partial K_{\text{eff}}}{\partial N_{Na}} = \frac{1}{v} \frac{\partial v}{\partial N_{Na}} + \frac{1}{\Sigma_f} \frac{\partial \Sigma_f}{\partial N_{Na}} - \frac{1}{\Sigma_a} \frac{\partial \Sigma_a}{\partial N_{Na}} + \frac{1}{P_{nl}} \frac{\partial P_{nl}}{\partial N_{Na}} \quad (32)$$

Rewriting:

$$\alpha_v = \alpha_v(v) + \alpha_v(\Sigma_f) - \alpha_v(\Sigma_a) + \alpha_v(P_{nl}) \quad (33)$$

The nonleakage probability may be broken down:

$$P_{nl} = \frac{1}{1 + L^2 B^2} \quad (34)$$

Taking the natural log and differentiating:

$$\ln P_{nl} = \ln \frac{1}{1 + L^2 B^2} = -\ln(1 + L^2 B^2) \quad (35)$$

$$\begin{aligned}
\frac{1}{P_{n1}} \frac{\partial P_{n1}}{\partial N_{Na}} &= - \frac{1}{1+L^2 B^2} \frac{\partial (1+L^2 B^2)}{\partial N_{Na}} \\
&= - \frac{1}{1+L^2 B^2} \left(L^2 \frac{\partial B^2}{\partial N_{Na}} + B^2 \frac{\partial L^2}{\partial N_{Na}} \right) \\
&= - \frac{L^2 B^2}{1+L^2 B^2} \left(\frac{1}{B^2} \frac{\partial B^2}{\partial N_{Na}} + \frac{1}{L^2} \frac{\partial L^2}{\partial N_{Na}} \right) \\
&= (P_{n1}-1) (\alpha_v(B^2) + \alpha_v(L^2))
\end{aligned} \tag{36}$$

The equation for the sodium void reactivity worth may now be written as:

$$\alpha_v = \frac{1}{v} \frac{\partial v}{\partial N_{Na}} + \frac{1}{\Sigma_f} \frac{\partial \Sigma_f}{\partial N_{Na}} - \frac{1}{\Sigma_a} \frac{\partial \Sigma_a}{\partial N_{Na}} + (P_{n1}-1) \left(\frac{1}{B^2} \frac{\partial B^2}{\partial N_{Na}} + \frac{1}{L^2} \frac{\partial L^2}{\partial N_{Na}} \right) \tag{37}$$

$$\alpha_v = \alpha_v(v) + \alpha_v(\Sigma_f) - \alpha_v(\Sigma_a) + (P_{n1}-1) (\alpha_v(B^2) + \alpha_v(L^2)) \tag{38}$$

$$\text{Recall that } L^2 = \frac{D}{\Sigma_a} \tag{39}$$

Taking the natural log and differentiating:

$$\ln(L^2) = \ln\left(\frac{D}{\Sigma_a}\right) = \ln(D) - \ln(\Sigma_a) \tag{40}$$

$$\begin{aligned}
&\frac{1}{L^2} \frac{\partial L^2}{\partial N_{Na}} = \frac{1}{D} \frac{\partial D}{\partial N_{Na}} - \frac{1}{\Sigma_a} \frac{\partial \Sigma_a}{\partial N_{Na}} \\
\therefore \alpha_v(L^2) &= \alpha_v(D) - \alpha_v(\Sigma_a)
\end{aligned} \tag{41}$$

Recall that:

$$D \approx \frac{1}{3 \Sigma_{tr}} \quad (42)$$

and that:

$$\Sigma_{tr} = \Sigma_a + (1 - \bar{\mu}) \Sigma_s \quad (43)$$

where $\bar{\mu}$ is the average value of the scattering angle

Assuming the scattering angle is very small, this equation may be rewritten as:

$$\Sigma_{tr} = \Sigma_a + \Sigma_s \quad (44)$$

or:

$$D = \frac{1}{3 (\Sigma_a + \Sigma_s)} \quad (45)$$

$$\ln D = \ln \frac{1}{3 (\Sigma_a + \Sigma_s)} = -\ln 3 - \ln (\Sigma_a + \Sigma_s) \quad (46)$$

$$\frac{1}{D} \frac{\partial D}{\partial N_{Na}} = - \frac{1}{\Sigma_a + \Sigma_s} \frac{\partial (\Sigma_a + \Sigma_s)}{\partial N_{Na}} \quad (47)$$

$$= - \frac{1}{(\Sigma_{tr})} \left(\frac{\partial \Sigma_a}{\partial N_{Na}} + \frac{\partial \Sigma_s}{\partial N_{Na}} \right) = - \frac{\Sigma_a}{\Sigma_{tr}} \frac{1}{\Sigma_a} \frac{\partial \Sigma_a}{\partial N_{Na}} - \frac{\Sigma_s}{\Sigma_{tr}} \frac{1}{\Sigma_s} \frac{\partial \Sigma_s}{\partial N_{Na}} \quad (48)$$

$$\alpha_v(D) = - \frac{\Sigma_a}{\Sigma_{tr}} \alpha_v(\Sigma_a) - \frac{\Sigma_s}{\Sigma_{tr}} \alpha_v(\Sigma_s) \quad (49)$$

This result may be substituted into Equation 41.

$$\therefore \alpha_v(L^2) = -\frac{\Sigma_a}{\Sigma_{tr}} \alpha_v(\Sigma_a) - \frac{\Sigma_s}{\Sigma_{tr}} \alpha_v(\Sigma_s) - \alpha_v(\Sigma_a) \quad (50)$$

Assuming that the buckling is only a function of core geometry and does not contribute to the sodium void reactivity worth, Equation 38 becomes:

$$\alpha_v = \alpha_v(v) + \alpha_v(\Sigma_f) - \alpha_v(\Sigma_a) + (P_{nl}-1) \left(-\frac{\Sigma_a}{\Sigma_{tr}} \alpha_v(\Sigma_a) - \frac{\Sigma_s}{\Sigma_{tr}} \alpha_v(\Sigma_s) - \alpha_v(\Sigma_a) \right) \quad (51)$$

Simplifying Equation 51 further :

$$\begin{aligned} \alpha_v = & \alpha_v(v) + \alpha_v(\Sigma_f) - \alpha_v(\Sigma_a) - (P_{nl}-1) \left(-\frac{\Sigma_a}{\Sigma_{tr}} \alpha_v(\Sigma_a) \right) \\ & - (P_{nl}-1) \left(\frac{\Sigma_s}{\Sigma_{tr}} \alpha_v(\Sigma_s) \right) - P_{nl} \alpha_v(\Sigma_a) + \alpha_v(\Sigma_a) \end{aligned} \quad (52)$$

$$\alpha_v = \alpha_v(v) + \alpha_v(\Sigma_f) - \left(P_{nl} + \frac{(P_{nl}-1)\Sigma_a}{\Sigma_{tr}} \right) \alpha_v(\Sigma_a) - \frac{(P_{nl}-1)\Sigma_s}{\Sigma_{tr}} \alpha_v(\Sigma_s) \quad (53)$$

The effects of the leakage, spectral, and capture components on the SVRE may be seen in Equation 53. Increasing the leakage reduces P_{nl} . A reduction in P_{nl} reduces the coefficient of $\alpha_v(\Sigma_a)$ reducing its contribution to the total SVRE. The coefficient of $\alpha_v(\Sigma_s)$ becomes more negative reducing the SVRE. Therefore, increasing leakage reduces the total SVRE as expected.

Refer to Table 2 for the effects of hardening the spectrum on the SVRE. The net effect is to make the SVRE more positive.

As seen in the third term of Equation 53, a reduction in capture, reducing Σ_a , has a positive effect on the SVRE. Equation 53 also points out an important difference between the three fuel types. In normal operation, moderation of neutrons is not important in a fast reactor. However, when the sodium voids, the presence of a moderator in the fuel is significant. For a metal fueled reactor, the scattering cross section of the sodium is a large fraction of the total scattering cross section of all the core materials. Voiding the sodium removes the moderator causing significant spectral hardening. The harder the spectrum, the more positive the absorption term in Equation 53 becomes resulting in a more positive SVRE. In an oxide or a nitride fueled ALMR, the oxygen or nitrogen in the fuel constitute the largest fraction of the total scatter cross section. Thus, when the sodium voids, their presence serves to limit the spectral hardening and the absorption term has a smaller positive effect on the SVRE than for metal fuel. The net result is that a metal fueled ALMR will always have the most positive SVRE when compared to an oxide or nitride fueled ALMR of similar design. Likewise, core design changes to reduce the SVRE will always be more severe and consequently less durable in a metal fueled ALMR.

I. The Near Zero Sodium Void Worth Core

The safety enhanced nitride core was shortened from 135 cm to 75 cm in order to lower the void coefficient to less than one dollar. The oxide core and metal core were not shortened because this would have required that the reactor vessel diameter be increased to prevent exceeding the linear power capability of each fuel type. Nitride fuel has a linear power capability of 28 kw/ft which allows the core to be shortened without increasing reactor vessel diameter or exceeding the limit. For this part of the study, the model was reduced from full core to a one-sixth core and sodium was voided from only the fuel and blanket assemblies to be consistent with analyses performed by the lead designer. The void coefficient was reduced from \$2.97 and \$0.77 as shown on Figure 39. No attempt was made to optimize this core which retains the same configuration as the safety-enhanced reference core. The disadvantage to shortening the core is that the breeding ratio drops from approximately 1.1 to 0.93. Peak linear power increases from 11 kw/ft to 19 kw/ft for the fuel, and 8 kw/ft to 16 kw/ft for the blankets, within the capability of nitride fuel. An additional advantage to shortening the nitride core is that the design requirement of rail transportability is not compromised.

J. The ALMR as an Actinide Burner

Configuring the ALMR to burn actinides by replacing fuel and blanket assemblies with assemblies containing actinides is a possible solution to the problem of the storage and handling of high level waste. Cross sections for some minor actinides were not available in the ENDF/B-V²² library and were created by combining actinides with similar one-group cross sections. While this increases the inaccuracy of the analysis, it does give an indication of the affect of actinides on the SVRE as

Actinide assemblies were modeled
using fuel assembly pin dimensions

FIGURE 39: Sodium Void Reactivity Effect of a Minor Actinide Burner.

shown on Figure 39. The SVRE for various loadings of actinide target, fuel, and blanket assemblies could not be reduced to less than two dollars.

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APPENDIX A: Ratio Program

```
c    Calculation of capture to fission ratio
c    version 1.0    release 4/11/91
c
c    * * * * * description of subroutines * * * * *
c
c    readi2        reads integer data in generalized format.
c
c    error         prints error messages.
c
c    recs          reads cross section data
c
c    * * * * * internal variables * * * * *
c
c    ninp          input unit
c
c    nout          output unit
c
c    itemp         temporary
c
c    * * * * * input variables * * * * *
c
c    itl           cross section table length
c
c    igm           number of groups
c
c    m1            number of input materials
c
c    * * * * * subscripted variables * * * * *
c
c    atw(m1)       material atomic weight
```

```

c
c   holn(ml)      material name
c
c   alam(ml)      decay constant   (days-1)
c
c   c0(itl,iig)   cross section array for currant material
c
c   cf(iig)       fission cross section for current material
c
c   ca(iig)       absorption cross section for current material
c
c   ncf(iig)      nu sigma f for current material
c
c   cc(iig)       capture cross section for current material
c
c   fcr(iig)      capture to fission ratio for current material
c
c
c   parameter (limc = 800000)
c   common      ninp,      nout,      itl,      igm,      ml
c   common      c(limc)
c   ninp = 5
c   nout = 6
c   read(ninp,5) itl,igm,ml
5   format(3i6)
c   write (nout,5) itl,igm,ml
c   compute dimension pointers
c   latw = 1
c   lholn = latw + ml
c   lalam = lholn + ml*2
c   lc0 = lalam + ml
c   lcf = lc0 + itl*igm*ml
c   lca = lcf + igm*ml
c   lncf = lca + igm*ml
c   lcc = lncf + igm*ml

```

```

    lfcf = lcc + igm*m1
    leta = lfcr + igm*m1
    lidxs = leta + igm*2
    lident = lidxs + m1*100
    last = lident + m1
    if(last .ge. limc) then
    write(nout,6)
6    format('memory limit exceeded')
    stop
    endif
    call recs(c(latw),c(lholn),c(lalam),c(lc0),c(lidxs),
1      c(lident))
    call calc(c(lc0),c(lcf),c(lca),c(lident),c(lholn),
1      c(lncf),c(lcc),c(lfcf),c(leta))
    stop
    end
    subroutine calc(c0,cf,ca,ident,holn,ncf,cc,fcf,eta)
    common ninp, nout, itl, igm, m1
    dimension cc(m1,igm),fcf(m1,igm)
    dimension c0(m1,itl,igm),eta(m1,igm)
    dimension cf(m1,igm), ca(m1,igm)
    character ident(m1)*80
    real ncf(m1,igm)
10    write (nout, 20)
20    format(/'ratios are calculated for the following materials')
    do 100 m=1,m1
    write(nout,55) ident(m)
55    format(/a80)
    write(nout,60)
60    format('group          sigma c      nu sigma f      capture/fission
ratio
1      eta')
    do 75 iig=igm,1,-1

```

```

      i=((igm+1)-iig)
      cf(m,iig)=c0(m,1,iig)
      ca(m,iig)=c0(m,2,iig)
      ncf(m,iig)=c0(m,3,iig)
      cc(m,iig)=ca(m,iig)-cf(m,iig)
      if(ncf(m,iig) .ne. 0) go to 65
      ncf(m,iig)=0.0
      go to 66
65     fcr(m,iig)=cc(m,iig)/ncf(m,iig)
      eta(m,iig)=ncf(m,iig)/ca(m,iig)
66     write(nout,70) i, cc(m,iig), ncf(m,iig), fcr(m,iig), eta(m,iig)
70     format(i6, 6x, 1ple12.5, 2x, 1ple12.5, 5x, 1ple12.5, 5x, 1ple12.5)
75     continue
100    continue
      return
      end
      subroutine reai2(holl,iarray,ncount,jpr)
      common  ninp,      nout,      itl,      igm,      ml
      dimension iarray(1), iv(12), k(12), in(12)
      character holl*(*)
      jflag = 0
      j=1
1     if(jflag) 2,10,2
2     do 4  jj=1,6
      k(jj) = k(jj+6)
      in(jj) = in(jj+6)
4     iv(jj) = iv(jj+6)
      jflag = 0
      go to 21
10    read(ninp,20) (k(i),in(i),iv(i),i=1,6)
20    format(6(i1,i2,i9))
21    do 70  i=1,6
      l=k(i)+1
      go to (30,40,52,80,62,70,22), l
c     fill

```

```

22  jj=j
    do 25 m=jj,ncount
      iarray(j) = iv(i)

```

```

25  j=j+1
    go to 80

```

```

c    no modification

```

```

30  iarray(j)=iv(i)
    j=j+1
    go to 70

```

```

c    repeat

```

```

40  l=in(i)
    do 50 m=1,l
      iarray(j)=iv(i)
      j=j+1

```

```

50  continue
    go to 70

```

```

c    interpolate

```

```

52  if(i-6) 54,53,53

```

```

53  read(ninp,20) (k(m),in(m),iv(m),m=7,12)
    jflag = 1

```

```

54  l = in(i) + 1
    idel = (iv(i+1) - iv(i))/l
    do 56 m=1,l
      iarray(j) = iv(i) + idel*(m-1)

```

```

56  j = j + 1
    go to 70

```

```

c    cycle

```

```

62  l = in(i)
    n = iv(i)
    do 65 ll=1,l
      do 65 nn=1,n
        iarray(j) = iarray(j-n)

```



```

65   j = j + 1
70   continue
    go to 1
c    terminate
80   j=j-1
    if(j .ne. ncount .or. jpr .ne. 0) then
        write (nout,90) holl, j, (iarray(i),i=1,j)
    else
        write (nout,90) holl, j
    endif
90   format(6x,a6,i6/(10i12))
110  return
    end
    subroutine error( hol,jsubr,i)
    common  ninp,    nout,    itl,    igm,    ml
    write(nout,1) hol,jsubr
1    format('*'/ 'error in',a6,'at',i6/ '*' / '*')
    go to (3,4),i
3    stop
4    return
    end
    subroutine recs(atw,holn,alam,c0,idxs,ident)
    common  ninp,    nout,    itl,    igm,    ml
    dimension c0(ml,itl,igm)
    dimension atw(ml),alam(ml),idxs(ml)
    character aa*54, holnx*6, holn(ml)*6
    character ident(ml)*80
    write(nout,8)
8    format('cross section identifiers')
    call rea12(6h idxs ,idxs,ml,1)
    open(15,file='xsinp',form='unformatted',access='sequential',
1    status='old')
10   write(nout,20)
20   format('cross sections are read in for the following materials')
    do 50 n=1,ml

```

m=n

```

35  read(15,end=500) holnx,atwx,alamx,aa
    read(aa,36) itemp
36  format(49x,i5)
    do 45 j=1,m1
        if(itemp .ne. idxs(j)) go to 45
        m=j
        holn(m)=holnx
        atw(m)=atwx
        alam(m)=alamx
        idxs(m)=-idxs(m)
        read(15) ((c0(m,i,iig),i=1,it1),iig=1,igm)
        go to 48
45  continue
    read(15)
    go to 35
48  write(ident(m), 55) m, holn(m), aa
    write(nout,55) m, holn(m), aa
50  continue
55  format(i4, 6x, a6, 6x, a54)
    return
500 write(nout,510)
510 format('requested materials not in library:' /)
    do 520 j=1,m1
        if(idxs(j) .ge. 0) write(nout,530) j,idxs(j)
520 continue
530 format(i6,1h* ,i8)
540 call error('**xs ',540,1)
    end

```

APPENDIX B: Reaction Rate Program

```
program rxn
c   version 1.0   release 5/9/90
c
c   * * * * * description of subroutines * * * * *
c
c   rxn           main routine - controls the operation of the code.
c
c   inp           controls the reading and printing of all input data
c                 and computes the variable dimension pointers.
c
c   erro2         prints error messages.
c
c   recs          reads cross sections from cards and writes cross
c                 sections to drum.
c
c   reag2         reads floating point data in generalized format.
c
c   reai2         reads integer data in generalized format.
c
c   clear         sets an array of a given length equal to a given
c                 constant.
c
c   * * * * * internal variables * * * * *
c
c   ninp          input unit
c   nout          output unit
c   e01           temporary
c   e02           temporary
c   e03           temporary
c   igp           igm + 1
c   iha           position of absorption cross section
c   ihf           position of fission cross section
c   ihs           position of sigma self scatter
c   imjm          im*jm
```

```

c      ip          im + 1
c      itemp       temporary
c      itempl      temporary
c      itemp2      temporary
c      itl         cross section table length
c      itlmt       itl*mt
c      izp         izm + 1
c      jp          jm + 1
c      kp          km + 1
c      ml         absolute value of mcr
c      ngoto       temporary
c      ninit       temporary
c

```

```

c      * * * * * input variables (cards 1-5) * * * * *
c
c      igm         number of groups
c      itl         number of downscattering terms
c      iht         position of sigma transport in cross section table
c      m07         flux guess(0/1/2/3/4/5/6/7=none/phi(x)*phi(y)*phi(z)
c                  /phi(x,y,z)/phi(e,x,y,z) from cards/phi(x,y,z)/
c                  phi(e,x,y,z) from tape/phi(e,x,y) from tape*phi(z)
c                  from cards/rsyn tape)
c      izm         number of material zones
c      mt          total number of materials including mixes
c      m01         number of mixture specifications
c      mcr         number of input cross section materials (neg/
c                  pos=from tape/cards)
c      im          number of intervals in the x direction
c      jm          number of intervals in the y direction
c      km          number of intervals in the z direction
c      nlay        number of material layers
c

```

```

c      * * * * * subscripted variables * * * * *
c
c      atw(m1)      material atomic weight
c      holn(m1)     material name
c      alam(m1)     decay constant   (days-1)
c      c0(m1,it1,igm) cross section array for current group
c      n0(im,jm,km,igm) total flux (old)
c      i0(m01)      mix number
c      i1(m01)      material number for mix
c      i2(m01)      material density
c      m0(im,jm,nlay) zone numbers
c      m2(izm)      material numbers by zone
c      x0(ip)       initial points along the x axis
c      x1(ip)       current points along x axis
c      y0(jp)       initial points along y axis
c      y1(jp)       current points along y axis
c      z0(kp)       initial points along z axis
c      z1(kp)       current points along z axis
c      ndens(m1,izm) average number density by zone
c
      common ninp, nout, iha, ihf, ihs, imjm, ip,
1          jp, kp
      common igm, it1, iht, ige, izm, nlay, mt,
1          m01, m1, im, jm, km, iz, jz,
2          kz, itemp, itempl, m05
      common latw, lholn, lalam, lc0, ln0, lz4, li0,
1          li1, li2, li3, lm0, lm2, lx0, lx1,
2          lv0, ly0, ly1, lvol, lndens, lmatn, lfrrt,
3          lphib, lmstr, lz0, lz1, llyn, last, lmb1,
4          lme1, lidxs, lident, lart, lz5, lphit,
5          lsrt, lfrrt, lx4, lx5, ly4, ly5
      parameter (lima = 3000000)
      common a(lima)
      call inp
      call volume(a(lx0),a(lx1),a(lx4),a(lx5),a(ly0),a(ly1),a(ly4),

```

```

1      a(ly5),a(lz0),a(lz1),a(lz4),a(lz5),
2      a(lv0),im)
  call dens(a(lndens),a(lvol),a(latw),a(lholn),a(lm0),a(lv0),a(li0),
1      a(li1),a(li2),a(lz5),a(llyn),a(lmb1),a(lmel),a(lmatn),
2      a(lm2),a(li3))
  call average(a(lphib),a(lmatn),a(lvol),a(ln0),a(lm0),a(lv0),
1      a(lz5),a(llyn),a(lphit))
  call rate(a(lholn),a(latw),a(lalam),a(lc0),a(lfrt),a(lart),
1      a(lsrt),a(lfnrt),a(lident),a(lndens),a(lphib),
2      a(lmatn))
320  format(10e12.5)
  end
  subroutine inp
  common  ninp,  nout,  iha,  ihf,  ihs,  imjm,  ip,
1      jp,  kp
  common  igm,  itl,  iht,  ige,  izm,  nlay,  mt,
1      m01,  m1,  im,  jm,  km,  iz,  jz,
2      kz,  itemp,  itemp1,  m05
  common  latw, lholn, lalam, lc0, ln0, lz4, li0,
1      li1, li2, li3, lm0, lm2, lx0, lx1,
2      lv0, ly0, ly1, lvol, lndens, lmatn, lfrt,
3      lphib, lmstr, lz0, lz1, llyn, last, lmb1,
4      lmel, lidxs, lident, lart, lz5, lphit,
5      lsrt, lfnrt, lx4, lx5, ly4, ly5
  parameter (lima = 3000000)
  common a(lima)
  character id*66
c  this subroutine controls the reading and printing of initial data
  ninp = 5
  nout = 6
  write(nout, 10)
10  format('1'/20x,'R A T E   Version 1.0   Release 5/ 9/90'/
1      20x,'-----'//)
  read(ninp,20) id, igm, itl,
1      iht, izm,

```

```

2          mt, m01, mcr, im, jm, km,nlay, m05
20  format(a66/12i6)
    write(nout,30) id
30  format(/10x,a66,i6)
    write(nout,60)  igm,  itl, iht
60  format(
1'      igm      number of groups

2                                'i9/
3'      itl      cross section table length

4                                'i9/
5'      iht      position of sigma transport in cross section table
6                                'i9)
    write(nout,80)  izm, mt, m01, mcr
80  format(
1'      izm      number of material zones
2                                'i9/
3'      mt       total number of materials including mixes
4                                'i9/
5'      m01      number of mixture specifications
6                                'i9/
7'      mcr      number of input cross section materials (neg=pos=fr
8om tape/cards)                'i9)
    write(nout,90)  im, jm, km,nlay,m05
90  format(
1'      im       number of intervals in the x direction
2                                'i9/
3'      jm       number of intervals in the y direction
4                                'i9/
5'      km       number of intervals in the z direction
6                                'i9/
7'      nlay     number of material layers
8                                'i9/
9'      m05      number of materials for reaction rate calculations

```

```

1          'i9)
    ihs = iht + 1
    iha = iht - 2
    ihf = iht - 3
    izp = izm + 1
    ip  = im + 1
    jp  = jm + 1
    kp  = km + 1
    m1 = iabs(mcr)
    igp = igm + 1
    imjm=im*jm
    itlm1=itl*igm*m1
    ala = .0
c  compute dimension pointers
    latw = 1
    lholn = latw + m1
    lalam = lholn + m1*(11/4)
    lc0= lalam+m1
    lv0=lc0+itlm1
    lm0=lv0+imjm
    ln0=lm0+imjm*nlay
    lfrt=ln0+imjm
    lart=lfrt+m05*igm*izm
    lsrt=lart+m05*igm*izm
    lfnrt=lsrt+m05*igm*izm
    lx0=lfnrt+m05*igm*izm
    lx1=lx0+ip
    lx4=lx1+ip
    lx5=lx4+im
    ly0=lx5+im
    ly1=ly0+jp
    ly4=ly1+jp
    ly5=ly4+jm
    lz0=ly5+jm
    lz1=lz0+kp

```



```

1z4=1z1+kp
1z5=1z4+km
1lyn=1z5+km
1i0 = 1lyn+km
1i1 = 1i0 + m01
1i2 = 1i1 + m01
1i3=1i2+m01
1vol=1i3+m01
1m2=1vol+izm
1matn=1m2+izm
1mb1=1matn+m05
1me1=1mb1+mt
1ndens=1me1+mt
1phib=1ndens+m1*izp+2*mt
1phit=1phib+igm*izm
1idxs=1phit+izm
1ident=1idxs+m1
last=1ident+m1*((80+3)/4)
write(nout,315) lima,last
if(last .ge. lima) then
write(nout,5)
5  format('memory limit exceeded')
stop
endif
315 format(' lima,last=', 2i8)
c  read cross sections
call recs(a(1atw),a(1holn),a(1alam),a(1c0),a(1idxs),a(1ident))
c  core dimensions for reaction rates
write (nout,330)
330 format (' mesh boundaries (x0/y0/z0=x/y/z) points')
call reag2 (' x0',a(1x0),ip,1)
call reag2 (' y0',a(1y0),jp,1)
call reag2 (' z0',a(1z0),kp,1)
write (nout,335)
335 format (' layer numbers by xy plane')

```

```

call reai2 ('  lyn',a(1lyn),km,1)
itemp=0
do 345 n=1,nlay
write (nout,340) n
340 format(' zone numbers by mesh interval for layer' i3)
call reai2 ('  m0',a(1m0+itemp),imjm,0)
345 itemp=itemp+imjm
write (nout,350)
350 format (' material numbers by zone')
call reai2 ('  m2',a(1m2),izm,1)
write (nout,360)
360 format (' mixture specifications (i0/i1/i2=mix number/mat. number'
1'for mix/material density)')
call reai2 ('  i0',a(1i0),m01,0)
call reai2 ('  i1',a(1i1),m01,0)
call reai2 ('  i2',a(1i2),m01,0)
write(nout,370)
370 format(' materials used for reaction rate calculations')
call reai2 ('  matn',a(1matn),m05,0)
call inpr(a(1matn),a(1ident))
570 return
end
subroutine erro2( hol,jsubr,i)
common ninp, nout, iha, ihf, ihs, imjm, ip,
1 jp, kp
common igm, itl, iht, ige, izm, nlay, mt,
1 m01, m1, im, jm, km, iz, jz,
2 kz, itemp, itemp1, m05
common latw, lholn, lalam, lc0, ln0, lz4, li0,
1 li1, li2, li3, lm0, lm2, lx0, lx1,
2 lv0, ly0, ly1, lv01, lndens, lmatn, lfrt,
3 lphib, lmstr, lz0, lz1, llyn, last, lmb1,
4 lme1, lidxs, lident, lart, lz5, lphit,
5 lsrt, lfprt, lx4, lx5, ly4, ly5
write (nout,1) hol,jsubr

```

```

1   format(' */' error in',a6,' at',i6/' */' *')
   go to (3,4),i
3   stop
4   return
   end
   subroutine reag2(holl,array,ncount,jpr)
   dimension array(1),v(12),k(12),in(12)
   character holl*(*)
   common ninp, nout, iha, ihf, ihs, imjm, ip,
1      jp, kp
   common igm, itl, iht, ige, izm, nlay, mt,
1      m01, m1, im, jm, km, iz, jz,
2      kz, itemp, itempl, m05
   common latw, lholn, lalam, lc0, ln0, lz4, li0,
1      lil, li2, li3, lm0, lm2, lx0, lx1,
2      lv0, ly0, ly1, lvol, lndens, lmatn, lfrrt,
3      lphib, lmstr, lz0, lz1, llyn, last, lmb1,
4      lme1, lidsx, lident, lart, lz5, lphit,
5      lsrt, lfnrt, lx4, lx5, ly4, ly5
   jflag=0
   j=1
10  if(jflag)20,40,20
20  do 30 jj=1,6
     k(jj)=k(jj+6)
     in(jj)=in(jj+6)
30  v(jj)=v(jj+6)
     jflag=0
     go to 60
40  read (ninp,50) (k(i),in(i),v(i),i=1,6)
50  format(6(i1,i2,e9.4))
60  do 140 i=1,6
     l=k(i)+1
     go to (70,80,100,150,132,140,62), l
c   fill
62  jj=j

```

```

        do 65 m=jj,ncount
        array(j) = v(i)
65      j=j+1
        go to 150
c      no modification
70      array(j)=v(i)
        j=j+1
        go to 140
c      repeat
80      l=in(i)
        do 90 m=1,l
        array(j)=v(i)
        j=j+1
90      continue
        go to 140
c      interpolate
100     if(i-6) 120,110,110
110     read (ninp,50) (k(jj),in(jj),v(jj),jj=7,12)
        jflag=1
120     l=in(i)+1
        del=(v(i+1)-v(i))/float (l)
        do 130 m=1,l
        array(j)=v(i)+del*float (m-1)
        j=j+1
130     continue
        go to 140
c      cycle
132     l=in(i)
        n=int(.00001+v(i))
        do 135 ll=1,l
        do 135 nn=1,n
        array(j) = array(j-n)
135     j=j+1
140     continue
        go to 10

```

```

c      terminate
150    j=j-1
      if(j .ne. ncount .or. jpr .ne. 0) then
        write (nout,160) holl, j, (array(i),i=1,j)
      else
        write (nout,160) holl, j
      endif
160    format(6x,a6,i6/(10e12.5))
170    if(j .ne. ncount) call erro2( '**reag',170,1)
180    return
      end
      subroutine reai2(holl,iarray,ncount,jpr)
      common ninp, nout, iha, ihf, ihs, imjm, ip,
1         jp, kp
      common igm, itl, iht, ige, izm, nlay, mt,
1         m01, m1, im, jm, km, iz, jz,
2         kz, itemp, itempl, m05
      common latw, lholn, lalam, lc0, ln0, lz4, li0,
1         li1, li2, li3, lm0, lm2, lx0, lx1,
2         lv0, ly0, ly1, lvol, lndens, lmatn, lftr,
3         lphib, lmstr, lz0, lz1, llyn, last, lmb1,
4         lme1, lidxs, lident, lart, lz5, lphit,
5         lsrt, lftrt, lx4, lx5, ly4, ly5
      dimension iarray(1), iv(12), k(12), in(12)
      character holl*(*)
      jflag = 0
      j=1
1     if(jflag) 2,10,2
2     do 4 jj=1,6
      k(jj) = k(jj+6)
      in(jj) = in(jj+6)
4     iv(jj) = iv(jj+6)
      jflag = 0
      go to 21
10    read(ninp,20) (k(i),in(i),iv(i),i=1,6)

```

```

20  format(6(i1,i2,i9))
21  do 70 i=1,6
    l=k(i)+1
    go to (30,40,52,80,62,70,22), l
c   fill
22  jj=j
    do 25 m=jj,ncount
    iarray(j) = iv(i)
25  j=j+1
    go to 80
c   no modification
30  iarray(j)=iv(i)
    j=j+1
    go to 70
c   repeat
40  l=in(i)
    do 50 m=1,l
    iarray(j)=iv(i)
    j=j+1
50  continue
    go to 70
c   interpolate
52  if(i-6) 54,53,53
53  read(ninp,20) (k(m),in(m),iv(m),m=7,12)
    jflag = 1
54  l = in(i) + 1
    idel = (iv(i+1) - iv(i))/l
    do 56 m=1,l
    iarray(j) = iv(i) + idel*(m-1)
56  j = j + 1
    go to 70
c   cycle
62  l = in(i)
    n = iv(i)
    do 65 ll=1,l

```

```

do 65 nn=1,n
  iarray(j) = iarray(j-n)
65  j = j + 1
70  continue
   go to 1
c   terminate
80  j=j-1
   if(j .ne. ncount .or. jpr .ne. 0) then
     write (nout,90) holl, j, (iarray(i),i=1,j)
   else
     write (nout,90) holl, j
   endif
90  format(6x,a6,i6/(10i12))
110 return
   end
   subroutine clear (x,y,n)
   dimension y(n)
   do 1 i=1,n
1    y(i)=x
   return
   end
   subroutine rate(holn,atw,alam,c0,frt,art,srt,fnrt,ident,ndens,
1 phib,matn)
c   this subroutine calculates reaction rates by zone
   common ninp, nout, iha, ihf, ihs, imjm, ip,
1      jp, kp
   common igm, itl, iht, ige, izm, nlay, mt,
1      m0l, ml, im, jm, km, iz, jz,
2      kz, itemp, itempl, m05
   common latw, lholn, lalam, lc0, ln0, lz4, li0,
1      li1, li2, li3, lm0, lm2, lx0, lx1,
2      lv0, ly0, ly1, lv0l, lndens, lmatn, lfrt,
3      lphib, lmstr, lz0, lz1, llyn, last, lmb1,
4      lmel, lidxs, lident, lart, lz5, lphit,
5      lsrt, lfnrt, lx4, lx5, ly4, ly5

```

```

dimension c0(m1,it1,igm),frt(m05,izm,igm),art(m05,izm,igm)
dimension srt(m05,izm,igm),fnrt(m05,izm,igm),phib(izm,igm)
dimension atw(m1),alam(m1),matn(m05)
real ndens(m1,izm)
character ident(m1)*80
open(16,file='rxout',form='formatted',access='sequential',
1   status='unknown')
do 250 i=1,m05
  l=matn(i)
  write(16,55) ident(l)
  write(16,60)
  do 200 kz=1,izm
    if (ndens(l,kz) .eq. 0) go to 200
    write(16,65)
    do 200 iig=igm,1,-1
      iq=((igm+1)-iig)
      frt(l,kz,iig)=ndens(l,kz)*phib(kz,iig)*c0(l,1,iig)*1000
      art(l,kz,iig)=ndens(l,kz)*phib(kz,iig)*c0(l,2,iig)*1000
      fnrt(l,kz,iig)=ndens(l,kz)*phib(kz,iig)*c0(l,3,iig)*1000
      srt(l,kz,iig)=(ndens(l,kz)*phib(kz,iig)*c0(l,iht,iig)*1000)
      srt(l,kz,iig)=srt(l,kz,iig)-art(l,kz,iig)
55   format(/a80)
60   format(' group   zone   fission reaction rate   absorption reaction

         lrate   scatter reaction rate   nu fission reaction rate')
65   format(/)
      write(16,100) iq,kz,frt(l,kz,iig),art(l,kz,iig),srt(l,kz,iig),
1   fnrt(l,kz,iig)
100  format(i6,i6,6x,1ple12.5,12x,1ple12.5,12x,1ple12.5,12x,1ple12.5)
200  continue
250  continue
      return
      end
      subroutine dens(ndens,vol,atw,holn,m0,v0,i0,i1,i2,z5,lyn,
1          mbl,mel,matn,m2,i3)

```



```

common  ninp,  nout,  iha,  ihf,  ihs,  imjm,  ip,
1      jp,  kp
common  igm,  itl,  iht,  ige,  izm,  nlay,  mt,
1      m01,  m1,  im,  jm,  km,  iz,  jz,
2      kz,  itemp,  itempl,  m05
common  latw,  lholn,  lalam,  lc0,  ln0,  lz4,  li0,
1      li1,  li2,  li3,  lm0,  lm2,  lx0,  lx1,
2      lv0,  ly0,  ly1,  lvol,  lndens,  lmatn,  lfirt,
3      lphib,  lmstr,  lz0,  lz1,  llyn,  last,  lmb1,
4      lmel,  lidxs,  lident,  lart,  lz5,  lphit,
5      lsrt,  lfirt,  lx4,  lx5,  ly4,  ly5
dimension vol(izm), atw(m1), holn(m1), m2(m1), i0(m01),
1      i1(m01),z5(izm), lyn(nlay), mbl(m01),mel(m01),
2      matn(m05),m0(im,jm,km), v0(im,jm)
real      i2(m01), i3(m01), ndens(m1,izm)
character holn(m1)*6, zone*6, total*6
data  total/' total'/
data  zone/' zone'/
c  this subroutine calculates the isotopic number densities by zone
call clear (0,mbl,2*mt)
do 15 i=1,m01
n=i0(i)
if (mbl(n) .le. 0) mbl(n)=i
mel(n)=i
15 continue
call clear (0.0,vol,izm)
call clear (0.0,ndens,m1*izp)
do 30 kk=1,km
l=lyn(kk)
do 30 i=1,im
do 30 j=1,jm
k=m0(i,j,l)
vol(k)=vol(k)+(v0(i,j)*.001*z5(kk))
30 continue
do 39 m=1,m01

```

```

      i3(m)=i2(m)
      if (i0(m)-i1(m)) 39,35,39
35    if (i2(m)) 39,36,39
36    itemp=i0(m)
      itempl=mb1(itemp)
      do 38 mm=templ,m
      if (itemp-i0(mm)) 38,37,38
37    i3(mm)=i2(mm)
38    continue
39    continue
      do 190 n=1,izm
      nn=m2(n)
      itemp=mb1(nn)
      itempl=mel(nn)
      do 190 m=itemp,templ
      if (i0(m)-nn) 190,40,190
40    l=i1(m)
      if (l-m1) 170,170,50
50    nnaa=1
      if (l-i0(m)) 130,190,130
130   maab=mel(nnaa)
      maae=mel(nnaa)
      do 160 maa=maab,maae
      if (i0(maa)-nnaa) 160,140,160
140   l=i1(maa)
      if (l) 160,160,150
150   e01=i3(maa)*i3(m)
      ndens(l,n)=(e01*vol(n))+ndens(l,n)
160   continue
      go to 190
170   if (l) 190,190,180
180   e01=i3(m)
      ndens(l,n)=(e01*vol(n))+ndens(l,n)
190   continue
      do 260 l=1,izm, 10

```

```

      ll=1+9
      if (ll-izm) 210,210,200
200  ll=izm
210  write(nout,220) (zone,k,k=1,ll)
220  format(///' material   atwt   ',2x,10(a6,i4,1x))
      write(nout,230) (vol(k),k=1,ll)
230  format(16x,' vol='1p10e11.3)
      do 255 k=1,m1
      do 235 i=1,ll
      if(ndens(k,i) .gt. 0.0) go to 240
235  continue
      go to 255
240  write(nout,250) k,holn(k),atw(k),(ndens(k,i),i=1,ll)
250  format(i4,1x,a6,f9.3,1x,1p10e11.3)
255  continue
260  continue
400  return
      end
      subroutine volume(x0,x1,x4,x5,y0,y1,y4,y5,z0,z1,
1          z4,z5,v0,jm)
      common ninp, nout, iha, ihf, ihs, imjm, ip,
1          jp, kp
      common igm, itl, iht, ige, izm, nlay, mt,
1          m01, m1, im, jm, km, iz, jz,
2          kz, itemp, itempl, m05
      common latw, lholn, lalam, lc0, ln0, lz4, li0,
1          li1, li2, li3, lm0, lm2, lx0, lx1,
2          lv0, ly0, ly1, lvol, lndens, lmatn, lftrt,
3          lphib, lmstr, lz0, lz1, llyn, last, lmb1,
4          lme1, lidxs, lident, lart, lz5, lphit,
5          lsrt, lftrt, lx4, lx5, ly4, ly5
      dimension x0(im), x1(im), x4(im), x5(im), y0(jm),
1          y1(jm), y4(jm), y5(jm), z0(km), z1(km),
2          z4(km), z5(km)
      dimension v0(im,jm)

```

```
255  do 260  i=1,ip
      x1(i)=x0(i)
260  continue
      do 265  j=1,jp
265  y1(j)=y0(j)
      do 268  k=1,kp
268  z1(k) = z0(k)
305  continue
c    calculate areas and volumes.
315  continue
      do 345  i=1,im
      x4(i)=(x1(i+1)+x1(i))*0.5
      x5(i)=x1(i+1)-x1(i)
      if( x5(i) ) 320, 320, 325
320  call erro2 (6h*x5(i),320,1)
325  continue
345  continue
      do 370  j=1,jm
      y4(j)=(y1(j+1)+y1(j))*0.5
      y5(j)=y1(j+1)-y1(j)
      if(y5(j)) 350, 350, 355
350  call erro2 (6h*y5(j),350,1)
355  continue
      do 370  i=1,im
360  v0(i,j)=x5(i)*y5(j)
370  continue
      do 373  k=1,km
      z4(k) = (z1(k+1) + z1(k))*0.5
      z5(k) = z1(k+1) - z1(k)
      if(z5(k)) 372, 372, 373
372  call erro2(6h*z5(k),372,1)
373  continue
375  continue
380  return
      end
```

```

subroutine average(phib,matn,vol,n0,m0,v0,z5,lyn,phit)
common ninp, nout, iha, ihf, ihs, imjm, ip,
1      jp, kp
common igm, itl, iht, ige, izm, nlay, mt,
1      m0l, ml, im, jm, km, iz, jz,
2      kz, itemp, itempl, m05
common latw, lholn, lalam, lc0, ln0, lz4, li0,
1      li1, li2, li3, lm0, lm2, lx0, lx1,
2      lv0, ly0, ly1, lvol, lndens, lmatn, lfirt,
3      lphib, lmstr, lz0, lz1, llyn, last, lmb1,
4      lme1, lidxs, lident, lart, lz5, lphit,
5      lsrt, lfnrt, lx4, lx5, ly4, ly5
dimension phib(izm,igm),matn(m05),vol(izm),z5(km),lyn(km)
dimension phit(izm),v0(imjm)
dimension m0(imjm,izm)
real      n0(imjm)
call clear (0.0,phib,izm*igm)
call clear (0.0,phit,izm)
open(3,file='flxinp',form='unformatted',access='sequential',
1 status='old')
do 105 iig=1,igm
do 105 kk=1,km
l=lyn(kk)
40 read(3) (n0(i),i=1,imjm)
do 50 ii=1,imjm
kz=m0(ii,l)
phib(kz,iig)=phib(kz,iig)+n0(ii)*v0(ii)*z5(kk)
50 continue
105 continue
do 250 kz=1,izm
do 250 iig=1,igm
volkz=vol(kz)*1000
phib(kz,iig)=phib(kz,iig)/volkz
phit(kz)=phit(kz)+phib(kz,iig)
250 continue

```

```

do 300 kz=1,izm
do 300 iig=1,igm
phib(kz,iig)=phib(kz,iig)/phit(kz)
300 continue
return
end
subroutine inpr(matn,ident)
c reads materials to be used for reaction rate calculations
dimension matn(1)
character ident(m1)*80
common ninp, nout, iha, ihf, ihs, imjm, ip,
1      jp, kp
common igm, itl, iht, ige, izm, nlay, mt,
1      m01, m1, im, jm, km, iz, jz,
2      kz, itemp, itempl, m05
common latw, lholn, lalam, lc0, ln0, lz4, li0,
1      li1, li2, li3, lm0, lm2, lx0, lx1,
2      lv0, ly0, ly1, lvol, lndens, lmatn, lfrt,
3      lphib, lmstr, lz0, lz1, llyn, last, lmb1,
4      lme1, lidxs, lident, lart, lz5, lphit,
5      lsrt, lfnrt, lx4, lx5, ly4, ly5
write (nout,60)
60 format (' Reaction Rate Data'//)
write (nout,70)
70 format (' material      name')
do 90 n=1,m05
80 format (a80)
m=matn(n)
write (nout,80) ident(m)
90 continue
return
end
subroutine recs(atw,holn,alam,c0,idxs,ident)
common ninp, nout, iha, ihf, ihs, imjm, ip,
1      jp, kp

```

```

common   igm,   itl,   iht,   ige,   izm,   nlay,   mt,
1         m01,   m1,    im,    jm,    km,    iz,    jz,
2         kz,   itemp, itempl, m05
common   latw, lholn, lalam, lc0,  ln0,   lz4,  li0,
1         li1, li2,   li3,   lm0,  lm2,   lx0,  lx1,
2         lv0, ly0,   ly1,  lvol, lndens, lmatn, lfrt,
3         lphib, lmstr, lz0,  lz1,  llyn, last, lmb1,
4         lme1, lidxs, lident, lart,  lz5,  lphit,
5         lsrt, lfnrt, lx4,  lx5,  ly4,  ly5
dimension c0(m1,itl,igm)
dimension atw(m1),alam(m1),idxs(m1)
character aa*54, holnx*6, holn(m1)*6
character ident(m1)*80
write(nout,8)
8  format(' cross section identifiers')
   call reai2(6h idxs ,idxs,m1,1)
   open(15,file='xsinp',form='unformatted',access='sequential',
1  status='old')
10  write(nout,20)
20  format(' cross sections are read in for the following materials')
   do 50 n=1,m1
      m=n
35  read(15,end=500) holnx,atwx,alamx,aa
      read(aa,36) itemp
36  format(49x,i5)
      do 45 j=1,m1
         if(itemp .ne. idxs(j)) go to 45
         m=j
         holn(m)=holnx
         atw(m)=atwx
         alam(m)=alamx
         idxs(m)=-idxs(m)
         read(15) ((c0(m,i,iig),i=1,itl),iig=1,igm)
         go to 48
45  continue

```

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```
    read(15)
    go to 35
48  write(ident(m), 55)
```


APPENDIX C: Reaction Rates

Figures C-1 through C-108 show the absorption, fission, and capture macroscopic reaction rates for the three fuel types. These figures present a detailed study of each major heavy metal isotope in the reactor to show how the reaction rates change as the neutron spectrum hardens. In the oxide and nitride cores, the oxygen and nitrogen serve to limit the spectral hardening as the sodium voids. Thus, the change in the reaction rates is small compared to the metal core. This can be seen in the figures. Also, it can be seen that all the nonfissile isotopes, i.e., ^{239}U , ^{240}Pu , and ^{242}Pu , contribute to the positive sodium void effect. This is due to their fission reaction rates which are not significant at low neutron energies but become significant as the neutron spectrum hardens. Therefore, the more the spectrum hardens as the sodium voids, the more positive the sodium void effect.

FIGURE C-1: Fission Reaction Rate for ²³⁹Pu in an Oxide Inner Fuel Assembly.

FIGURE C-2: Fission Reaction Rate for ^{239}Pu in an Oxide 2 Cycle Radial Blanket.

FIGURE C-3: Fission Reaction Rate for ^{239}Pu in a Nitride Inner Fuel Assembly.

FIGURE C-4: Fission Reaction Rate for ^{239}Pu in a Nitride 2 Cycle Radial Blanket.

FIGURE C-5: Fission Reaction Rate for ^{239}Pu in a Metal Inner Fuel Assembly.

FIGURE C-6: Fission Reaction Rate for ^{239}Pu in a Metal 2 Cycle Radial Blanket.

FIGURE C-7: Absorption Reaction Rate for ^{239}Pu in an Oxide Inner Fuel Assembly.

FIGURE C-8: Absorption Reaction Rate for ^{239}Pu in an Oxide 2 Cycle Radial Blanket.

FIGURE C-9: Absorption Reaction Rate for ^{239}Pu in a Nitride Inner Fuel Assembly.

FIGURE C-10: Absorption Reaction Rate for ^{239}Pu in a Nitride 2 Cycle Radial Blanket.

FIGURE C-11: Absorption Reaction Rate for ^{239}Pu in a Metal Inner Fuel Assembly.

FIGURE C-12: Absorption Reaction Rate for ^{239}Pu in a Metal 2 Cycle Radial Blanket.

FIGURE C-13: Capture Reaction Rate for ^{239}Pu in an Oxide Inner Fuel Assembly.

FIGURE C-14: Capture Reaction Rate for ^{239}Pu in an Oxide 2 Cycle Radial Blanket.

FIGURE C-15: Capture Reaction Rate for ^{239}Pu in a Nitride Inner Fuel Assembly.

FIGURE C-16: Capture Reaction Rate for ^{239}Pu in a Nitride 2 Cycle Radial Blanket.

FIGURE C-17: Capture Reaction Rate for ^{239}Pu in a Metal Inner Fuel Assembly.

FIGURE C-18: Capture Reaction Rate for ^{239}Pu in a Metal 2 Cycle Radial Blanket.

FIGURE C-19: Fission Reaction Rate for ^{241}Pu in an Oxide Inner Fuel Assembly.

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FIGURE C-20: Fission Reaction Rate for ^{241}Pu in an Oxide 2 Cycle Radial Blanket.

FIGURE C-21: Fission Reaction Rate for ^{241}Pu in a Nitride Inner Fuel Assembly.

FIGURE C-22: Fission Reaction Rate for ^{241}Pu in a Nitride 2 Cycle Radial Blanket.

FIGURE C-23: Fission Reaction Rate for ^{241}Pu in a Metal Inner Fuel Assembly.

FIGURE C-24: Fission Reaction Rate for ^{241}Pu in a Metal 2 Cycle Radial Blanket.

FIGURE C-25: Absorption Reaction Rate for ^{241}Pu in an Oxide Inner Fuel Assembly.

FIGURE C-26: Absorption Reaction Rate for ^{241}Pu in an Oxide 2 Cycle Radial Blanket.

FIGURE C-27: Absorption Reaction Rate for ^{241}Pu in a Nitride Inner Fuel Assembly.

FIGURE C-28: Absorption Reaction Rate for ^{241}Pu in a Nitride 2 Cycle Radial Blanket.

FIGURE C-29: Absorption Reaction Rate for ^{241}Pu in a Metal Inner Fuel Assembly.

FIGURE C-30: Absorption Reaction Rate for ^{241}Pu in a Metal 2 Cycle Radial Blanket.

FIGURE C-31: Capture Reaction Rate for ^{241}Pu in an Oxide Inner Fuel Assembly.

FIGURE C-32: Capture Reaction Rate for ^{241}Pu in an Oxide 2 Cycle Radial Blanket.

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FIGURE C-33: Capture Reaction Rate for ^{241}Pu in a Nitride Inner Fuel Assembly.

FIGURE C-34: Capture Reaction Rate for ^{241}Pu in a Nitride 2 Cycle Radial Blanket.

FIGURE C-35: Capture Reaction Rate for ^{241}Pu in a Metal Inner Fuel Assembly.

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FIGURE C-36: Capture Reaction Rate for ^{241}Pu in a Metal 2 Cycle Radial Blanket.

FIGURE C-37: Fission Reaction Rate for ^{235}U in an Oxide Inner Fuel Assembly.

FIGURE C-38: Fission Reaction Rate for ^{235}U in an Oxide 2 Cycle Radial Blanket.

FIGURE C-39: Fission Reaction Rate for ^{235}U in a Nitride Inner Fuel Assembly.

FIGURE C-40: Fission Reaction Rate for ^{235}U in a Nitride 2 Cycle Radial Blanket.

FIGURE C-41: Fission Reaction Rate for ^{235}U in a Metal Inner Fuel Assembly.

FIGURE C-42: Fission Reaction Rate for ^{235}U in a Metal 2 Cycle Radial Blanket.

FIGURE C-43: Absorption Reaction Rate for ^{235}U in an Oxide Inner Fuel Assembly.

FIGURE C-44: Absorption Reaction Rate for ^{235}U in an Oxide 2 Cycle Radial Blanket.

FIGURE C-45: Absorption Reaction Rate for ^{235}U in a Nitride Inner Fuel Assembly.

FIGURE C-46: Absorption Reaction Rate for ^{235}U in a Nitride 2 Cycle Radial Blanket.

FIGURE C-47: Absorption Reaction Rate for ^{235}U in a Metal Inner Fuel Assembly.

FIGURE C-48: Absorption Reaction Rate for ^{235}U in a Metal 2 Cycle Radial Blanket.

FIGURE C-49: Capture Reaction Rate for ^{235}U in an Oxide Inner Fuel Assembly.

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FIGURE C-50: Capture Reaction Rate for ^{235}U in an Oxide 2 Cycle Radial Blanket.

FIGURE C-51: Capture Reaction Rate for ^{235}U in a Nitride Inner Fuel Assembly.

FIGURE C-52: Capture Reaction Rate for ^{235}U in a Nitride 2 Cycle Radial Blanket.

FIGURE C-53: Capture Reaction Rate for ^{235}U in a Metal Inner Fuel Assembly.

FIGURE C-54: Capture Reaction Rate for ^{235}U in a Metal 2 Cycle Radial Blanket.

FIGURE C-55: Fission Reaction Rate for ^{238}U in an Oxide Inner Fuel Assembly.

FIGURE C-56: Fission Reaction Rate for ^{238}U in an Oxide 2 Cycle Radial Blanket.

FIGURE C-57: Fission Reaction Rate for ^{238}U in a Nitride Inner Fuel Assembly.

FIGURE C-58: Fission Reaction Rate for ^{238}U in a Nitride 2 Cycle Radial Blanket.

FIGURE C-59: Fission Reaction Rate for ^{238}U in a Metal Inner Fuel Assembly.

FIGURE C-60: Fission Reaction Rate for ^{238}U in a Metal 2 Cycle Radial Blanket.

FIGURE C-61: Absorption Reaction Rate for ^{238}U in an Oxide Inner Fuel Assembly.

FIGURE C-62: Absorption Reaction Rate for ^{238}U in an Oxide 2 Cycle Radial Blanket.

FIGURE C-63: Absorption Reaction Rate for ^{238}U in a Nitride Inner Fuel Assembly.

FIGURE C-64: Absorption Reaction Rate for ^{238}U in a Nitride 2 Cycle Radial Blanket.

FIGURE C-65: Absorption Reaction Rate for ^{238}U in a Metal Inner Fuel Assembly.

FIGURE C-66: Absorption Reaction Rate for ^{238}U in a Metal 2 Cycle Radial Blanket.

FIGURE C-67: Capture Reaction Rate for ^{238}U in an Oxide Inner Fuel Assembly.

FIGURE C-68: Capture Reaction Rate for ^{238}U in an Oxide 2 Cycle Radial Blanket.

FIGURE C-69: Capture Reaction Rate for ^{238}U in a Nitride Inner Fuel Assembly.

FIGURE C-70: Capture Reaction Rate for ^{238}U in a Nitride 2 Cycle Radial Blanket.

FIGURE C-71: Capture Reaction Rate for ^{238}U in a Metal Inner Fuel Assembly.

FIGURE C-72: Capture Reaction Rate for ^{238}U in a Metal 2 Cycle Radial Blanket.

FIGURE C-73: Fission Reaction Rate for ^{240}Pu in an Oxide Inner Fuel Assembly.

FIGURE C-74: Fission Reaction Rate for ^{240}Pu in an Oxide 2 Cycle Radial Blanket.

FIGURE C-75: Fission Reaction Rate for ^{240}Pu in a Nitride Inner Fuel Assembly.

FIGURE C-76: Fission Reaction Rate for ^{240}Pu in a Nitride 2 Cycle Radial Blanket.

FIGURE C-77: Fission Reaction Rate for ^{240}Pu in a Metal Inner Fuel Assembly.

FIGURE C-78: Fission Reaction Rate for ^{240}Pu in a Metal 2 Cycle Radial Blanket.

FIGURE C-79: Absorption Reaction Rate for ^{240}Pu in an Oxide Inner Fuel Assembly.

FIGURE C-80: Absorption Reaction Rate for ^{240}Pu in an Oxide 2 Cycle Radial Blanket.

FIGURE C-81: Absorption Reaction Rate for ^{240}Pu in a Nitride Inner Fuel Assembly.

FIGURE C-82: Absorption Reaction Rate for ^{240}Pu in a Nitride 2 Cycle Radial Blanket.

FIGURE C-83: Absorption Reaction Rate for ^{240}Pu in a Metal Inner Fuel Assembly.

FIGURE C-84: Absorption Reaction Rate for ^{240}Pu in a Metal 2 Cycle Radial Blanket.

FIGURE C-85: Capture Reaction Rate for ^{240}Pu in an Oxide Inner Fuel Assembly.

FIGURE C-86: Capture Reaction Rate for ^{240}Pu in an Oxide 2 Cycle Radial Blanket.

FIGURE C-87: Capture Reaction Rate for ^{240}Pu in a Nitride Inner Fuel Assembly.

FIGURE C-88: Capture Reaction Rate for ^{240}Pu in a Nitride 2 Cycle Radial Blanket.

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FIGURE C-89: Capture Reaction Rate for ^{240}Pu in a Metal Inner Fuel Assembly.

FIGURE C-90: Capture Reaction Rate for ^{240}Pu in a Metal 2 Cycle Radial Blanket.

FIGURE C-91: Fission Reaction Rate for ^{242}Pu in an Oxide Inner Fuel Assembly.

FIGURE C-92: Fission Reaction Rate for ^{242}Pu in an Oxide 2 Cycle Radial Blanket.

FIGURE C-93: Fission Reaction Rate for ^{242}Pu in a Nitride Inner Fuel Assembly.

FIGURE C-94: Fission Reaction Rate for ^{242}Pu in a Nitride 2 Cycle Radial Blanket.

FIGURE C-95: Fission Reaction Rate for ^{242}Pu in a Metal Inner Fuel Assembly.

FIGURE C-96: Fission Reaction Rate for ^{242}Pu in a Metal 2 Cycle Radial Blanket.

FIGURE C-97: Absorption Reaction Rate for ^{242}Pu in an Oxide Inner Fuel Assembly.

FIGURE C-98: Absorption Reaction Rate for ^{242}Pu in an Oxide 2 Cycle Radial Blanket.

FIGURE C-99: Absorption Reaction Rate for ^{242}Pu in a Nitride Inner Fuel Assembly.

FIGURE C-100: Absorption Reaction Rate for ^{242}Pu in a Nitride 2 Cycle Radial Blanket.

FIGURE C-101: Absorption Reaction Rate for ^{242}Pu in a Metal Inner Fuel Assembly.

FIGURE C-102: Absorption Reaction Rate for ^{242}Pu in a Metal 2 Cycle Radial Blanket.

FIGURE C-103: Capture Reaction Rate for ^{242}Pu in an Oxide Inner Fuel Assembly.

FIGURE C-104: Capture Reaction Rate for ^{242}Pu in an Oxide 2 Cycle Radial Blanket.

FIGURE C-105: Capture Reaction Rate for ^{242}Pu in a Nitride Inner Fuel Assembly.

FIGURE C-106: Capture Reaction Rate for ^{242}Pu in a Nitride 2 Cycle Radial Blanket.

FIGURE C-107: Capture Reaction Rate for ^{242}Pu in a Metal Inner Fuel Assembly.

FIGURE C-108: Capture Reaction Rate for ^{242}Pu in a Metal 2 Cycle Radial Blanket.

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