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METHOD FOR GENERATING NEUTRONIC DATA BASES
FOR TRANSIENT REACTOR CALCULATIONS*

by

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ABSTRACT

A systematic method has been developed for the construction of neutronic data bases used with the GRASS code in the analyses of postulated transients in Savannah River reactors. In this context, a neutronic data base consists of a set or sets of correlation equations that represent the fewgroup macroscopic cross sections for various cell types in the reactor charge. The cross sections are functions of the reactor state variables (temperatures, densities, material compositions, etc.).

To minimize the size of the data set library required to support the correlations, two steps are taken. First, the state variables are grouped into separate correlations, or "regimes." A state variable is assigned to a particular regime according to the relative magnitude of the reactivity perturbation caused by changing the value of the variable over its allowed range. The separate regime correlation equations are combined linearly within GRASS to produce a single set of fewgroup cross sections for each

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cell type at any time within the transient. The second step involves the use of Box-Behnken or other incomplete factorial data library designs to support each regime correlation. The derivation of the regime correlation equations is performed with the CRASS system of codes.

The above methods are demonstrated by application to the design of a transient data base for a typical Savannah River reactor charge. Transient phenomena included are the temperature feedback of normal reactor operation, coolant flow instability, assembly melting with associated entrainment of melted particles in the moderator, moderator boiling, and gadolinium poison injection.

INTRODUCTION

In recent years, the trend in reactor kinetics has shifted from the classical point reactor model to multi-dimensional analyses of reactor transients. This shift in emphasis is due to several factors, as Ferguson has pointed out in his review of multi-dimensional reactor dynamics.¹ These include:

- The increased size of the nuclear reactors that are being built.
- The increased burden of proof placed on the nuclear industry to certify the safety margins for nuclear reactor operation.
- Increased need for optimization of reactor operation.

- The development of third generation, high-speed computers along with high-speed and high-volume random access devices.

The first three factors have given code developers the incentive to design computer systems capable of multi-dimensional analysis, while the last factor has made such development possible.

Several such code development efforts for thermal reactor transient analysis are currently in progress. These include the development of the MEKIN² code at the Massachusetts Institute of Technology and the HERMITE^{3,4} code at Combustion Engineering Corporation for transient analyses of light water reactors, and the development of the GRASS⁵ modular system at the Savannah River Laboratory (SRL) for transient analyses of the Savannah River Plant (SRP) heavy water reactors. These codes not only allow for increased detail in the geometric description of the reactor systems, but also allow for more accurate modeling of the feedback mechanisms that control the course of the reactor transients. These feedback mechanisms are represented by sets of parametric equations which are correlations of the fewgroup neutronic cross sections and diffusion coefficients as functions of reactor state variables. The derivation of these parametric relationships is extremely important because the accuracy of the predictions of the model depends to a large extent on the adequacy of these relationships.

The MEKIN code makes use of a set of user-supplied, linear correlations in terms of coolant density, coolant temperature, and metal temperature for each material composition in the reactor.

HERMITE utilizes a scheme patterned after the HARMONY⁶ code. The cross sections used in HERMITE may be fixed constants or functions of up to three independent variables. Interpolation tables are used for the cross section "look-up" at each time step. The GRASS code uses list-processed arrays of parametric data in terms of the state variables required to describe all the conditions that may occur during an accident situation in SRP reactors. The list-processed structure reduces the storage requirements and allows for quick access during a transient calculation.

In this paper, a method for the systematic development of neutronic data bases is described for use in the analysis of postulated transients in SRP reactors. In this context, a neutronic data base consists of a set or sets of correlation equations which represent the fewgroup neutronic cross sections (plus diffusion coefficients) in terms of reactor state variables. The techniques used are general in nature and represent a systematic procedure for the reduction of a large volume of data to a readily accessible and compact form. Examples are presented to illustrate this procedure as applied to a transient analysis data base for a specific reactor charge. The accuracy of the resulting correlations is shown.

DESCRIPTION OF SRP REACTORS AND POSTULATED TRANSIENTS

The data base described in this paper was constructed to allow transient analysis of an SRP reactor containing a charge of fuel, target, and control assemblies arranged on a 7-inch

center-to-center spacing in a hexagonal pattern. Figure 1 shows a face map of such a lattice. The fuel assemblies are constructed of aluminum-clad, concentric tubes of an alloy or mixture of enriched uranium and aluminum. The target assemblies are constructed of aluminum-clad, concentric tubes of depleted uranium metal. Heavy water coolant flows in the spaces between the tubes and an outer housing. The coolant is injected from a plenum above the assemblies (Figure 2) and is ejected from the assemblies into the moderator space at the bottom of the tank. The heavy water then flows upward and outward to the edge of the tank where it flows downward to the six exit nozzles. The heavy water then flows through an external loop system, which consists of a pump and two parallel heat exchangers, and then flows back to the top plenum.

Transient analysis of an SRP reactor charge entails prediction of the neutronic response to a wide variety of transient initiators under the assumption that the normal safety rod SCRAM system fails to operate. In moderate transients, the response may be limited to small power surges in certain fuel and target assemblies. Such transients can be successfully terminated by the combined effects of moderator, coolant, and metal temperature feedback and adjustments in the control system.

More severe transients may involve certain of the assemblies ejecting super-heated coolant into the moderator space at the coolant outlets. The pressure drop between the coolant channels within

the assemblies and the moderator space in the coolant effluent stream may be sufficient to cause local moderator boiling. This "flashing" has a large negative reactivity effect. If sufficient corrective action is not taken, the transient may continue with subsequent reactivity surges following collapse of the moderator steam voids. These surges may cause sufficient power generation within certain assemblies to cause steam formation directly within the coolant channel spaces. If the coolant steam formation is sufficiently large, the coolant flow may be interrupted (coolant flow instability). The combined effects of coolant voiding and the subsequent metal temperature rise in the fuel assemblies result in a loss of reactivity.

If the transient is not terminated, subsequent power surges may lead to bulk moderator boiling and/or to fuel and target melting in certain of the assemblies. Target melting has, generally, a strong positive reactivity effect. Fuel melting is a strong negative reactivity effect. However, melting of the fuel may lead to particles of molten uranium-aluminum alloy injected into and entrained within the flow of the moderator, which has a strong positive reactivity effect. Further complicating the reactivity effects of melting is the strong negative effect of the bulk moderator boiling.

All of the above phenomena are accounted for within the neutronic base described in this paper. An additional phenomena included within the base is the effect of injection of a solution of heavy water and gadolinium nitrate directly into the moderator space. This supplementary shutdown system is automatically triggered by a sensing system entirely independent of the safety rod SCRAM unit.

DESCRIPTION OF DATA BASE CORRELATION METHODS

The overall data flow structure for the generation of correlated cross sections for the neutronic data base is shown in Figure 3. The first step is the lattice fine structure calculation which considers a single cell or supercell (consisting of fuel, target, and control rod assemblies; see Figure 1). The lattice fine structure is calculated by the GLASS⁷ code which provides a multigroup transport calculation of reactor cells or supercells. GLASS also collapses the multigroup cross sections into selected fewgroups for each cell within a supercell.

Many different lattice calculations, or "states," are required to span adequately the space of all the possible reactor state variables. The Box-Behnken⁸ and other incomplete factorial designs were used to minimize the number of states required to support the data base. The number of states was further reduced by the development of the multi-regime concept which groups the state variables in separate correlations, or "regimes." A state variable is assigned to a particular regime according to

the relative magnitude of the reactivity perturbation caused by changing the variable over its allowed range. Correlation equations developed for each regime are combined linearly within GRASS to produce the fewgroup cross sections at any time during a transient calculation, as discussed in a later section.

The resulting cross sections and lattice state variables are retrieved by lattice state with the CRAB2 code and are converted to a correlation data library which consists of sets of discrete values for each data type (called vectors) and a master list of data types. These data are then analyzed using the CRASS⁵ modular system which provides the following analysis options:

- *Correlation analysis* to determine the relative dependence of each cross section on each lattice variable or cross term of variables. This information is given by the correlation coefficients which are defined as

$$C_{ni} = \frac{\sum_{s=1}^N (Y_{ns}^{\alpha} - \overline{Y_n^{\alpha}}) (X_{is}^{\beta} - \overline{X_i^{\beta}})}{\left\{ \sum_{s=1}^N (Y_{ns}^{\alpha} - \overline{Y_n^{\alpha}})^2 \right\}^{\frac{1}{2}} \left\{ \sum_{s=1}^N (X_{is}^{\beta} - \overline{X_i^{\beta}})^2 \right\}^{\frac{1}{2}}} \quad (1)$$

where Y_n is a particular dependent variable, X_i is a particular reactor state variable, α and β are specified exponents, N is the number of states, and

$$\overline{Y_n^\alpha} = \frac{1}{N} \sum_{s=1}^N Y_{ns}^\alpha$$

$$\overline{X_i^\beta} = \frac{1}{N} \sum_{s=1}^N X_{is}^\beta$$

The correlation coefficient ranges in value from -1 to +1 and indicates the relative dependence of one variable upon another. A value of +1 indicates a strong direct dependence. A value of -1 indicates a strong inverse relation, while a value of zero indicates no dependence. Options are provided for correlation analysis with α and β ranging from -3 to +3 in steps of 1/2. Cross-correlation analysis is also provided for pairs of independent variables versus a particular dependent variable.

- *Mathematical transformation*⁹ of either or both the cross sections and reactor state variables to minimize the curvature of cross section dependence on the state variables. Available transformations of some variable, z , include z^c , $z/(1 + cz)$, $\exp(cz)$, $\ln(1 + cz)$, and $\exp(cz) - 1$, where c is a specified constant.
- *Regression analysis* to fit the discrete cross sections to specified functional forms of the state variables. The correlation equations are derived by least-squares fitting using orthonormal polynomials.¹⁰ The orthonormal polynomials are defined as:

$$P_k(\tilde{X}) = \sum_{j=1}^k a_j T_{kj}(\tilde{X}) \quad (2)$$

where $T_{kj}(\tilde{X})$ are user-specified algebraic terms in the independent variables:

$$\tilde{X} = (X_{1n}, X_{2n}, X_{3n}, \dots) \quad n=1, N \text{ and}$$

N = total number of points.

- *Plots of fitting results* to allow the investigator to evaluate his results and use his experience and intuition to improve the fitting. The plotting is performed by the COMPOSER¹¹ system of codes which provides output in either tabular or graphical form via microfiche or the Savannah River Laboratory interactive graphics system.

The fitting analysis is a trial and error procedure which allows considerable user interaction in the selection of analysis options. The resulting expansion functions are collected (PARMOUT) in a list-processed array (PARMEQ) for quick access during a reactor calculation.

DETAILED DESCRIPTION OF A NEUTRONIC DATA BASE FOR A SAVANNAH RIVER REACTOR

A. Reactor State Variables

Neutronic data bases for many reactor cell types must be included in a GRASS calculation for an actual charge. In addition to the reactor core cells (fuel, target, and control), there are

cells representing top and bottom endfitting regions which normally contain no fissile material (except during melting). Near the outer boundary, lithium-bearing blanket assemblies or only moderator may be present. At six locations within the core, aluminum assemblies (sparjets) are placed whose function is to direct the upward flow in the center of the tank and to inject the gadolinium nitrate solution upon demand. Each cell type must have its own set of macroscopic cross section correlation functions.

In this paper, only the data bases for the fuel and target cells within the core will be discussed. In particular, attention will be confined to the control zone where the control rod, fuel, and target assemblies are present (Figure 1). The data bases for these cells are the most complicated and, hence, encompass all the major problems of data base design.

Table 1 lists the independent variables for the fuel and target cells. At the beginning of the neutron exposure cycle for this charge, all the fuel assemblies had essentially the same material composition. The target initial material composition was also uniform. During the subsequent burnup, the material compositions of the different cells diverged because of the different neutron exposures experienced by each cell. In Table 1, the reactor state variables for the fuel cross section correlations include the average exposure of the neighboring target cells, and vice-versa. Also included in both lists is a variable for the

composition of the neighboring control cell. Inclusion of these neighboring cell material compositions is a reflection of the small effect neighboring cells have on the neutron spectrum at the cell in question; and, hence, on the macroscopic cross sections.

All the other variables in Table 1 reflect transient conditions. All the moderator variables are common to both the fuel and target correlations. Moreover, the coolant density of the fuel cell was found to have a significant but small effect on the cross sections of the neighboring target cell (though not vice-versa). Finally, the fuel and target homogenized density variables were found to have significant effects on the cross sections of neighboring cells; and, hence, each is included in both lists.

B. Selection of Lattice States

The critical problem in the construction of a data base is the selection of the lattice states required to span adequately the space of the reactor state variables. A GLASS lattice calculation requires three to four minutes of CPU time on an IBM 360/195 computer. Thus, the investment for data generation quickly becomes enormous, unless considerable care is taken in selecting the minimum number of states. For instance, if one approaches the problem and uses a factorial design, which requires all possible combinations of the state variables at each discrete value, then (3^{13}) or 1,594,323 lattice calculations would be required to

support a quadratic-order fit for the target cells. Clearly, drastic reductions must be made to make this operation feasible.

The minimum number of states that would support all terms and cross terms in the correlations would appear to be obtained using the Box-Behnken methodology.⁸ Such an incomplete factorial design for the 13 variables set for target quadratic correlations would require approximately 220 lattice states. However, the effect on the accuracy of the correlation from lumping together such a large number of variables with widely different importances could not be estimated. An alternative approach involving Box-Behnken, other incomplete factorial designs, and a multi-regime correlation structure was utilized.

C. The Multi-Regime Correlation Structure

GRASS has been coded to compute cross sections from four separate correlation functions according to a multi-regime combination formula:

$$\Sigma^j = \Sigma^j_{\text{NORMAL}} + \Delta\Sigma^j_{\text{WEAK}} + \Delta\Sigma^j_{\text{STRONG}} + \Delta\Sigma^j_{\text{SCRAMINK}} \quad (3)$$

where j is the cell type index. The transient state variables of Table 1 are apportioned into four regimes according to the magnitude of the material buckling change they cause when varied over their full range. The material buckling is defined for the supercell of three fuel and three target cells surrounding a central control cell. The NORMAL regime correlations include the feedback effects of moderator, coolant, and metal temperatures;

the range of material buckling changes is $<0.8 \text{ m}^{-2}$. The WEAK regime includes the variation of coolant density for two-phase coolant conditions; the range of material buckling changes can be as large as 2 m^{-2} . The STRONG regime includes the variables of deuterium concentration in the moderator for two-phase conditions, fuel and target densities for melting, and ^{235}U concentration in the moderator space for fuel particle entrainment. These variables can cause material buckling changes on the order of 10 m^{-2} . Finally, the SCRAMINK regime includes the variable for gadolinium concentration in the moderator space; the range of material buckling changes can be as large as 100 m^{-2} . The initial material composition variables were used in all regimes, except that cross-cell, initial material dependencies were dropped for the STRONG and SCRAMINK regimes. Table 2 summarizes the regime variable grouping.

Equation 3 describes, in general terms, the linear additions to the NORMAL cross sections of correction terms due to the effects of the WEAK, STRONG, and SCRAMINK transient phenomena. Each $\Delta\Sigma^j$ is computed by GRASS at each time step by the relation

$$\Delta\Sigma^j = \Sigma^j(f_m, f_t) - \Sigma^j(f_m, f_t^*) \quad (4)$$

where the cross section values have been computed as functions of the sets of state variables f_m , f_t , and f_t^* . The f_m set is the initial material composition variables. The transient variables at time t are given by the set f_t . The f_t^* set represents base values of the transient variables. Whenever the set

$f_t = f_t^*$, the transient has not caused the variables to depart from the base values and $\Delta \Sigma^j$ remains zero. Table 3 lists the definitions of the base values of the transient state variables.

D. Regime Data Base Designs

The number of individual GLASS problems, or states, required for a regime depends on the number of independent variables, the maximum expansion power desired in each variable, and the number of expansion terms to be included in the correlation. Unless some restrictions are placed on the design of a regime data library, the number of required states can still be prohibitively large even with a limited number of variables.

Consider the NORMAL regime correlation for the control-zone fuel assembly. There are six variables: five are quadratic, and one is cubic. If all variables were of equal importance, then all possible terms would have to be supported in the correlation. However, E_f (exposure of the fuel cell) has the dominant effect on the cross sections. The effects of neighboring target and control cell contents are small, and one may eliminate cross terms involving E_t or Li_c with any of the temperature variables with little loss in fitting accuracy. Likewise, experience has shown that cross terms involving the temperature variables alone may be eliminated. With these eliminations, made on the basis of scientific intuition and experience, the number of expansion terms required can be reduced from the 49 possible terms to only 25.

Not having to provide lattice states (as determined by GLASS) to support all the possible cross terms allows one to use an incomplete factorial design where only those states of the full factorial design which directly support the desired terms are included. In the above example, 4×3^5 or 972 states would have been necessary for a full factorial design. It can easily be shown that the incomplete factorial design which supports 25 terms requires only 60 states. It can also be shown that 18 additional states, giving a total of 78, will support not only the fuel correlations but the target correlations as well. In actuality, 103 states were used to support the correlations for the control cell, the fuel cell, and the target cell of the control-zone NORMAL regime.

The GLASS state libraries for the NORMAL, WEAK, and SCRAMINK regimes for all cell types were of the above incomplete factorial design. However, for the STRONG regime, the variables listed in Table 2 were all quite strong in their effect; thus, another type of incomplete factorial design, the Box-Behnken,⁸ was chosen. The Box-Behnken methodology makes no presumptions about the relative importance of the variables or about the significance of cross terms involving pairs of the variables. All terms are supported. However, these designs are applicable only if all variables have a quadratic maximum power. As discussed later, some independent variables had to be transformed to meet this requirement.

In addition to the 103 states of the NORMAL regime, 42 states supported the WEAK regime, 73 states supported the STRONG regime, and 35 states supported the SCRAMINK regime of the control-zone

cell correlations. The number of states used to support the cross section correlations are drastically reduced by using a combination of the multi-regime concept and the special incomplete factorial or Box-Behnken designs within each regime. The apparent advantage of using a multi-regime structure is that the absolute correlation fitting accuracy of each regime is roughly proportional to the strength of the variables in each regime. In the following section, the errors of fitting the adopted design input data and those errors arising from the use of Equation 3 are shown to be acceptable small.

E. Tests of the Neutronic Data Base Correlations

Two types of errors affecting the accuracy of the correlations will be discussed. The first error type involves fitting errors, i.e., those that arise from the approximations of the expansions and from the use of Equation 3 in constructing the combined cross section from the several regime correlations. The second error type involves those errors arising from the assumptions, both explicit and implicit, which were made in constructing the base.

E.1. Fitting Errors

The plotting option of the CRASS system was used to ensure that the correlations accurately fitted the input values. Moreover, the plots were useful in making fitting choices to improve accuracy and in identifying insufficiently supported correlation

terms. Figures 4-8 show selected plots of the core region correlations and input values for the control-zone fuel assemblies. The left sides of the figures show thermal group capture cross sections, and the right sides of the figures show thermal group production cross sections. Figure 4 shows NORMAL regime correlations of a family of exposure values with moderator temperature. Figure 5 shows WEAK regime correlations of a similar family with coolant density. Figures 6 and 7 show STRONG regime correlations of a family of moderator ^{235}U particle concentrations with moderator deuterium concentrations (representing moderator density) or with fuel density. Figure 8 shows SCRAMINK correlations of a family of exposures with gadolinium concentrations in the moderator.

Selected independent and dependent variables were transformed to fit adequately the data from the STRONG and SCRAMINK regimes. The diffusion coefficients in the STRONG regime could not be adequately fitted directly because of the large curvature of the group coefficients versus the moderator density. Satisfactory fits were obtained only by fitting the inverse diffusion coefficients. Also in the STRONG regime, all the cross section correlations used a transformation of the fuel and target density of the form $[\ln (1 + c_j \rho_j)]$ to replace ρ_j . The constants, c_j , were chosen to nearly linearize the fits of Σ_{c_2} and $v\Sigma_{f_2}$ against the transformed densities. The same sort of logarithmic transformation was made for the gadolinium concentration in the SCRAMINK regime.

A more stringent test for fitting errors involved comparisons of the 7-cell material buckling (or k_{∞}) computed with diffusion theory using either the GLASS fewgroup cross sections or the correlated cross sections from Equation 3. Reactivity errors were directly demonstrated in this manner. In terms of the infinite lattice reactivity, the error may be defined as

$$\frac{[\Delta k_{\infty} (\text{transport, diffusion}) - \Delta k_{\infty} (\text{correlation, diffusion})]}{\Delta k_{\infty} (\text{transport, diffusion})}$$

where the change in lattice reactivity is occasioned by a change in a reactor state variable. The maximum error of the NORMAL regime correlations was about 10% and occurred with the variation of the target metal temperature over its full range. The maximum error for WEAK regime correlations was about 5% and occurred with the variation of the fuel coolant density over its full range. The STRONG regime correlations, not surprisingly, showed the largest errors. The maximum reactivity error of about 13% occurred when the moderator density was varied over its full range.

An alternative method of displaying reactivity errors is shown in Figures 9 and 10. Figure 9 shows 7-cell material bucklings computed with infinite lattice diffusion theory using the GLASS cross sections and the correlation cross sections in the WEAK regime. Also shown are the transport (GLASS) material bucklings. The diffusion theory bucklings agree quite closely, but both differ considerably from the transport bucklings. The reasons for this difference are discussed in the next section.

Figure 10 extends the buckling comparisons to the STRONG regime. The diffusion theory bucklings agree closely, the largest observed absolute error being approximately 0.83 m^{-2} when the moderator was completely voided. No transport 7-cell material bucklings were available because the STRONG transport problems were solved in 19-cell geometry which did not permit a 7-cell edit.

E.2 Errors from Assumptions

Figure 9 demonstrates a basic error arising from the adoption of diffusion theory on a cell-by-cell basis in GRASS. The cell flux-ratios in the infinite lattice, diffusion theory calculation deviated slightly from the original GLASS transport theory results. In all plots of the NORMAL and WEAK regime material bucklings, this deviation caused a fairly constant difference of about 0.50 m^{-2} between the GLASS and diffusion theory results, except for the variation with the fuel coolant density. For this latter case, the absolute error tended toward zero at low coolant densities. Presumably, the cross sections of the data base could be arbitrarily altered to force the diffusion theory bucklings into better agreement with the transport bucklings. However, as long as the error between the transport and the diffusion results versus an updated state variable remains reasonably constant, no particular harm will result because the error will be compensated for in the initial control rod configuration. A possible important exception arises with two-phase coolant flow in the assemblies. The error between

transport and diffusion theory calculations is not constant. This causes a somewhat larger power reduction in the fuel assembly undergoing flow decay when calculated by diffusion theory rather than by transport theory. If the transient proceeds into fuel or target melting, the errors arising from the approximations made in the development of the models for GRASS melting and material transport will make the errors of two-phase coolant flow conditions less significant. For the moment, the data base errors have been judged acceptable, but normalization of the correlations may be attempted in the future.

CONCLUSIONS

A method has been devised for the generation of neutronic data bases for use in reactor transient calculations using diffusion theory. The method makes use of scientific knowledge and experience in the trial and error approach to the correlation process. The data requirements for the support of the data base are made practical by the use of the multi-regime concept and the application of the Box-Behnken or other incomplete factorial design concepts. The method has been thoroughly tested and has been found to produce adequate representations of the fewgroup cross sections for detailed transient analysis studies with multi-dimensional reactor code systems.

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TABLE 1

Independent Variables

<i>Mnemonic^a</i>		<i>Definition</i>
<i>Fuel</i>	<i>Target</i>	
<u>Initial Material Composition Variables:</u>		
E_f^2	E_f^2	exposure of fuel cell, mwd/ft
E_t^2	E_t^2	exposure of target cell, mwd/ft
Li_c^3	Li_c^3	concentration of 6Li in control cell, g/ft
<u>Transient Variables:</u>		
T_M^2	T_M^2	moderator temperature, °C
T_{cf}^2	T_{ct}^2	coolant temperature, °C
T_{mf}^2	T_{mt}^2	metal temperature (fissile), °C

ρ_{cf}^2	ρ_{cf}^2	coolant density in fuel coolant channels, g/cm ³
	ρ_{ct}^2	coolant density in target coolant channels, g/cm ³

$C(D)^2$	$C(D)^2$	concentration of deuterium atoms in moderator (reflecting moderator density), atom/barn-cm
ρ_f^2	ρ_f^2	density of homogenized fuel fissile material plus cladding, g/cm ³
ρ_t^2	ρ_t^2	density of homogenized target fissile material plus cladding, g/cm ³
$C(^{235}U)^2$	$C(^{235}U)^2$	concentration of ^{235}U in moderator (reflecting entrained particles), atom/barn-cm

$C(Gd)^3$	$C(Gd)^3$	concentration of gadolinium in moderator, atom/barn-cm

^a. Superscript denotes maximum power of term in correlations.

TABLE 2

Regime Grouping of the State Variables
for Fuel and Target Correlations

<i>Regime</i>	<i>State Variables</i>	
	<i>Fuel</i>	<i>Target</i>
<u>NORMAL</u>	E_f	E_t
	E_t	E_f
	Li_c	Li_c
	T_M	T_M
	T_{cf}	T_{ct}
	T_{mf}	T_{mt}
<u>WEAK</u>	E_f	E_f
	E_t	E_t
	Li_c	Li_c
	ρ_{cf}	ρ_{cf}
		ρ_{ct}
<u>STRONG</u>	E_f	E_t
	$C(D)$	$C(D)$
	$C(^{235}U)$	$C(^{235}U)$
	ρ_f	ρ_f
	ρ_t	ρ_t
<u>SCRAMINK</u>	E_f	E_t
	$C(Gd)$	$C(Gd)$

TABLE 3

Base Values of State Variables

<i>Regime - State Variable</i>	<i>Base Value</i>
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WEAK ρ_{cf}, ρ_{ct}

Liquid phase density (g/cm^3) at coolant saturation temperature within each cell type.

STRONG $C(D)$

Concentration of deuterium (atoms/barn-cm) in moderator liquid phase at saturation temperature.

 $C(^{235}\text{U})$

0.0 (atoms/barn-cm).

 ρ_f

3.0826 g/cm^3 , the effective density of all fissile and cladding material in the fuel assembly.

 ρ_t

16.365 g/cm^3 , the effective density of all fissile and cladding material in the target assembly.

SCRAMINK $C(\text{Gd})$

0.0 (atoms/barn-cm).

FIGURE 1. Plan View of a Typical Mixed Charge Lattice for an SRP Production Reactor

FIGURE 2. Schematic Diagram of SRP Production Reactor Showing Coolant Flow

FIGURE 3. Overall Data Flow Structure for the Neutronic Data Base

FIGURE 4. NORMAL Regime Correlations for Control Zone Fuel Assemblies (moderator temperature)

FIGURE 5. WEAK Regime Correlations for Control Zone Fuel Assemblies (coolant density)

Curve	$C(^{235}\text{U})$
A	O = 10.0000E-16
B	= 10.0000E-06
C	+ = 20.0000E-06
D	= 30.0000E-06
E	X = 40.0000E-06

FIGURE 6. STRONG Regime Correlations for Control Zone Fuel Assemblies (deuterium concentration)

FIGURE 7. STRONG Regime Correlations for Control Zone Fuel Assemblies (fuel density)

FIGURE 8. SCRAMINK Regime Correlations for Control Zone Fuel Assemblies (GdNO_3 in moderator)

FIGURE 9. Material Buckling in the Control Zone Supercell
vs. State Variables for Combined NORMAL and
WEAK Regimes

FIGURE 10. Material Buckling in the Control Zone Supercell
vs. State Variables for Combined NORMAL, WEAK,
and STRONG Regimes