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RAMONA-3B APPLICATION TO BROWNS FERRY ATWS*

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1. INTRODUCTION

The Anticipated Transient Without Scram (ATWS) is known to be a dominant accident sequence for possible core melt in a Boiling Water Reactor (BWR). A recent Probabilistic Risk Assessment (PRA) analysis[1] for the Browns Ferry nuclear power plant indicates that ATWS is the second most dominant transient for core melt in BWR/4 with Mark I containment. The most dominant sequence being the failure of long term decay heat removal function of the Residual Heat Removal (RHR) system.

Of all the various ATWS scenarios, the Main Steam Isolation Valve (MSIV) closure ATWS sequence is the most severe because of its high frequency of occurrence and its challenge to the residual heat removal system and containment integrity. Therefore, this transient has been, and continues to be, analyzed by various organizations using various computer codes[2-3]. However, most of the prior efforts have been carried out using point-kinetics codes.

The objective of this paper is to discuss two preliminary MSIV closure ATWS calculations done using the RAMONA-3B code and the work being done to create the necessary cross section sets for the Browns Ferry Unit 1 reactor. The RAMONA-3B code employs a three-dimensional neutron kinetics model coupled with one-dimensional, four equation, nonhomogeneous, nonequilibrium thermal hydraulics. To be compatible with 3-D neutron kinetics, the code uses parallel coolant channels in the core. It also includes a boron transport model and all necessary BWR components such as jet pump, recirculation pump, steam separator, steamline with safety and relief valves, main steam isolation valve, turbine stop valve, and turbine bypass valve. A summary of RAMONA-3B neutron kinetics and thermal hydraulics models is presented in the Appendix.

2. INPUT MODEL DESCRIPTION

2.1 Hydraulic Input

The input description for the Browns Ferry Unit 1 plant can be broken into two categories: geometrical and nuclear. The geometrical information was taken from an extensively quality assured input deck for RELAP5 previously generated at INEL[4]. This information was directly applicable to RAMONA-3B since both codes use 1-D hydraulics. Also, since RAMONA-3B was developed specifically for BWR analysis, the reactor system (Figure 1) in the code is already "preassembled" and the nodalization for this particular study is given in Table 1.

2.2 Neutronic Input

Two different neutronic core representations were used in this analysis. In the first case (i.e., the preliminary calculations) cross sections

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from the Peach Bottom turbine trip tests[5] were used. This was done in order to get immediate results while using a realistic core model to verify the transient scenarios, and to quality assure the input deck before the processing of actual plant cross sections was completed. However, it should be remembered that these calculations are preliminary and the final results can only be obtained with the actual Browns Ferry neutronic data.

The second core representation, which only recently became available, used the neutronic data corresponding to cycle 5 of Browns Ferry Unit 3 at 8876 MWD/MT (complete core description was provided by the Tennessee Valley Authority (TVA)). The analysis showed that 5 basic fuel types had to be calculated by CASMO[6] (a 2-D transport code) to supply the diffusion parameters required for the calculation of the state of the core as a function of exposure and void history. The grid used for each fuel type for each CASMO calculation set can be seen in Figure 2.

The overall scheme to process the cross sections can be seen in Figure 3. After CASMO has defined the diffusion parameters, BLEND2 (which has been developed at BNL) takes the exposures and void histories from the TVA process computer for each node and determines its cross section set in the BNL TWIGL Format[7]:

$$\Sigma(\alpha,T_F,\ T_m,\ F|E,V)= F\ (a\ +b\alpha\ +\ c\alpha^2)+(1-F)(a'+b'\alpha+c'\alpha^2)$$

$$+\ P(T_m\ -\ T_m^*)\ +\ R(\sqrt{T_F}-\sqrt{T_F^*}) \qquad (1)$$

α = instantaneous void
T_F = fuel temperature
T_m = moderator temperature
F = control fraction
E = exposure
V = void history
P = $\delta\Sigma/\delta T_m$
R = $\delta\Sigma/\delta\sqrt{T_F}$
a,b,c = coefficients used in rod controlled void feedback
a',b',c' = coefficients used in rod uncontrolled void feedback

Once each node is assigned its cross section set, BLEND2 does a constrained minimization process using the exposure and void history of each node to group members of a fuel type (a group is called a chain) using the criteria

$$|E_i - E_j| < \delta E \qquad (2)$$

$$|V_i - V_j| < \delta V \qquad (3)$$

where E_j, V_j, are the characteristics of a node arbitrarily chosen as a chain head and E_i, V_i are the properties of an element or chain head of another group to determine whether it falls within the criteria of acceptance determined by δE and δV. Several sweeps are conducted with increasing bounds of acceptance for δE and δV until BLEND2 arrives at a predetermined number of cross sections. Then the cross section sets are averaged and this average set number and location is printed out by BLEND2. The cross sections are automatically printed in the RAMONA-3B format.

3. RESULTS

Two preliminary calculations have been done using the Peach Bottom EOC-2 cross sections on a Browns Ferry reactor specific deck. These are called Transient 1 and 2 and are discussed here. Another calculation presented below uses the neutronic data of Browns Ferry EOC-5 to predict the reactor steady state condition supplied by TVA.

3.1 Transient 1

The transient scenario is presented in Table 2; it is seen that the operator assumes level and pressure control according to the Emergency Procedural Guidelines (EPGs). In the course of this study a question about effects of condensation on the ECCS cold water jets was raised: the condensation starts after the feedwater sparger nozzles are uncovered and the water is injected into predominantly steam environment. This issue may become important because the reactor power depends strongly on the water subcooling at the core inlet. Figure 4 shows the total reactor power history for two <u>bounding</u> calculations: one with high core inlet subcooling (no condensation on the ECCS water jets) and the second with low core inlet subcooling (high-equilibrium-condensation rate on the ECCS water jets). The corresponding core inlet subcooling histories are shown in Figure 5. The difference in the reactor power predicted in these two calculations was found significant enough to initiate a developmental work on the appropriate condensation model. Another calculation with this model will be performed later for the Browns Ferry, Unit 3, EOC-5 core fuel condition.

In Figure 6 the system pressure is plotted along with the depressurization curve imposed by the EPG's to ensure that the heat capacity limit of the pressure suppression pool will not be exceeded.

As mentioned before, one of the major features of RAMONA-3B is the three-dimensional, time-dependent, neutron kinetics and power calculation. Comparisons between the axial and radial power distributions at various times show a stronger variation in the axial direction. Figure 7 shows the axial power distributions as calculated by RAMONA-3B at various times. It should be emphasized that the strong shifts in power distribution found in the RAMONA-3B calculation cannot be predicted using a point kinetics core model.

3.2 Transient 2

The scenario for Transient 2 is presented in Table 3. Although it is recognized that the probability of a recirculation pump trip failure is very remote, it was deemed necessary to analyze this transient in order to identify the timings and the characteristics of the events. In Figure 8 it can be seen that the power returns to about 80% of rated power which allows the SRV's to control the pressure in the vessel since the core steam generation rate is below the SRV valves capacity (the system pressure is shown in Figure 9). It can easily be recognized that with the large amount of steam leaving the vessel, the vessel water level is dropping extremely fast, which will potentially lead to core damage.

3.3 TVA State Point Calculation

TVA supplied the axial power distribution along the fuel description, exposures and void histories for Browns Ferry Unit 3 Cycle 5, and the necessary boundary conditions to test RAMONA-3B performance in simulating the reactor steady state condition.

The point chosen for analysis was at E = 8876 MWD/MT because the EOC-5 is the most reactive state of the core since the control rods are almost completely withdrawn. The BLEND2 code was run twice: once to get a set of 13 cross sections which had the final selection criteria of ΔE=5?91 MWD/MT and ΔV = 0.39, and a second run of 20 cross section sets with a final selection criteria of ΔE = 4233 MWD/MT and ΔV = 0.31. The simulation was done using a 1/8 core representation (101 channels with 24 axial levels = 2424 neutronic nodes) with 9 hydraulic channels (i.e., 216 hydraulic nodes). It should be noted that the RAMONA-3B code user must finalize a core which supplies the correct thermal-hydraulic response. This concept justifies the averaging technique used to find the cross sections and the coarse hydraulic noding used in the core. The code's results obtained by using the above cross section sets and the TVA supplied boundary conditions are found in Figure 10.

4. CONCLUSIONS

Three different predictions have been presented in this report. The first two are preliminary Peach Bottom 2 EOC-2 calculations involving MSIV closure ATWS: one uses level and pressure control to reduce the steaming to the PSP; the second analyzes the situation when the recirculation pumps do not trip during an ATWS. The third calculation uses the Browns Ferry Unit 3, EOC-5 neutronic data to calculate a static point supplied by TVA. The neutronic data from BF EOC-5 will be used to rerun both of the preliminary transients (i.e., Transients 1 and 2).

Transient 1 which used the level and pressure control procedures, showed that the modeling of the feedwater spargers is very important once they are uncovered since condensation on cold water controls the subcooling at the core inlet. The predictions showed that the results for the total reactor power range between 14 and 28 percent depending upon whether there is complete condensation or none at all, respectively. Hence, the geometry of the spargers has been analyzed and a model for the condensation developed.

Transient 2 demonstrates the case when the recirculation pumps do not trip. The calculations showed that after 15 seconds the reactor reaches a power of about 80 percent of rated power so that the SRV's could maintain the RPV below the pressure failure point. However, due to the fact that the water level is dropping quickly, core damage seems inevitable. Both results are considered preliminary since the transients were run with the Peach Bottom EOC-2 cross sections; the BF EOC-5 cross sections are being created for the final calculations.

The third calculation demonstrates the ability of transforming the raw plant data to a consistent set of cross sections for the RAMONA-3B 3-D neutronics calculations. The reproduction of the TVA static point validates the method and also supplies the starting point to recalculate the two transients discussed in this paper with consistent neutronic data. Development of the BLEND2 code at BNL has automated the 3-D cross sections generation procedure. This will help in the future to generate the 1-D cross sections for use in both the RAMONA-3B and TRAC-BD1 codes. Also, the reactivity edits capability in RAMONA-3B, can be used to derive the point kinetics reactivity feedback coefficients based on the 3-D set of cross sections.

REFERENCES

1. Mays, S. E., et al, "Interim Reliability Evaluation Program: Analysis of the Browns Ferry, Unit 1, Nuclear Plant," NUREG/CR-2802, EGG-2119, July 1982

2. Aronson, A., et al., "Status Report on BNL Calculations of ATWS in BWR's," BNL-17608, RP1022, 1973.

3. General Electric Company, "Assessment of BWR Mitigation of ATWS," Volume II, (NUREG-0460 Alternate No. 3), NEDO-24222, February 1981.

4. Wayne, C. Jouse, "Sequence Matrix for the Analysis of an ATWS in a BWR/4; Phenomena, Systems and Operation of Browns Ferry Nuclear Plant Unit 1," Draft, May 1984.

5. Wulff, W., et al., "A Description and Assessment of RAMONA-3B MOD.0 CYCLE 4: A Computer Code With Three-Dimensional Neutronic Kinetics for BWR System Transients," NUREG/CR-3664, BNL-NUREG-51476, January 1984.

6. Ahlin, Ake, et al., "A Fuel Assembly Burnup Program," AE-RF-76-4158, June 1978.

7. Diamond, D. J., Ed. "BNL-TWIGL, A Program For Calculating Rapid LWR Core Transient," BNL-NUREG-21925, Brookhaven National Laboratory, 1976.

8. Borresen, S., "A Simplified, Coarse-Mesh, Three-Dimensional Diffusion Scheme for Calculating the Gross Power Distribution in a Boiling Water Reactor," Nuclear Science and Engineering, 44, pp. 37-43, 1971.

Table 1 List of Nodes Used in RAMONA-3B Components

Hydraulics

Component	Number of Nodes
Downcomer 1	6
Downcomer 2	6
Lower Plenum 1	2
Lower Plenum 2	3
Core	12
Riser	5

Table 2 Transient 1 Scenario

Sequence of events

- MSIV closure in 5 sec
- Failure to Scram
- Feedwater flow ceases in 8 sec
- Recirculation pumps trip at high pressure
- Downcomer level hits Lo-Lo level (12.1 m) and HPCI and RCIC ramp up to 24.7 ℓ/s in 25 sec

150 sec operator takes control

- Operator follows depressurization line
- Water level is dropped to TAF and maintained
- HPCI suction is shifted from CST to PSP at high PSP water level (high level of 4.6 m)

Table 3 Transient 2 Scenario

Sequence of events

- MSIV closure in 5 sec
- Failure to Scram
- Feedwater flow ceases in 8 sec
- HPCI and RCIC are activated on Lo-Lo level (11.1 m) - 25 sec ramp to 24.7 ℓ/s
- Recirculation pumps fail to trip on high pressure signal

Table 4 TVA State Point at E = 8876 MWD/MT

	TVA	R-3B (13 cross-sections)	R-3B (20 cross-sections)
Power (MW_t)	3198	3198	3198
System Pressure (MPa)	6.9154 (1003 psi)	6.9154	6.9154
K_{eff}	1.00	1.0015	1.0054
Core Average Void	0.39	0.41	0.40
Core Flow (kg/s)	1.29 E+4 (102.19 Mlb/hr)	1.29 E+4	1.29 E+4

TVA also supplied:

- control rod pattern
- axial power distribution
- nodal relative power distribution from the TVA process computer

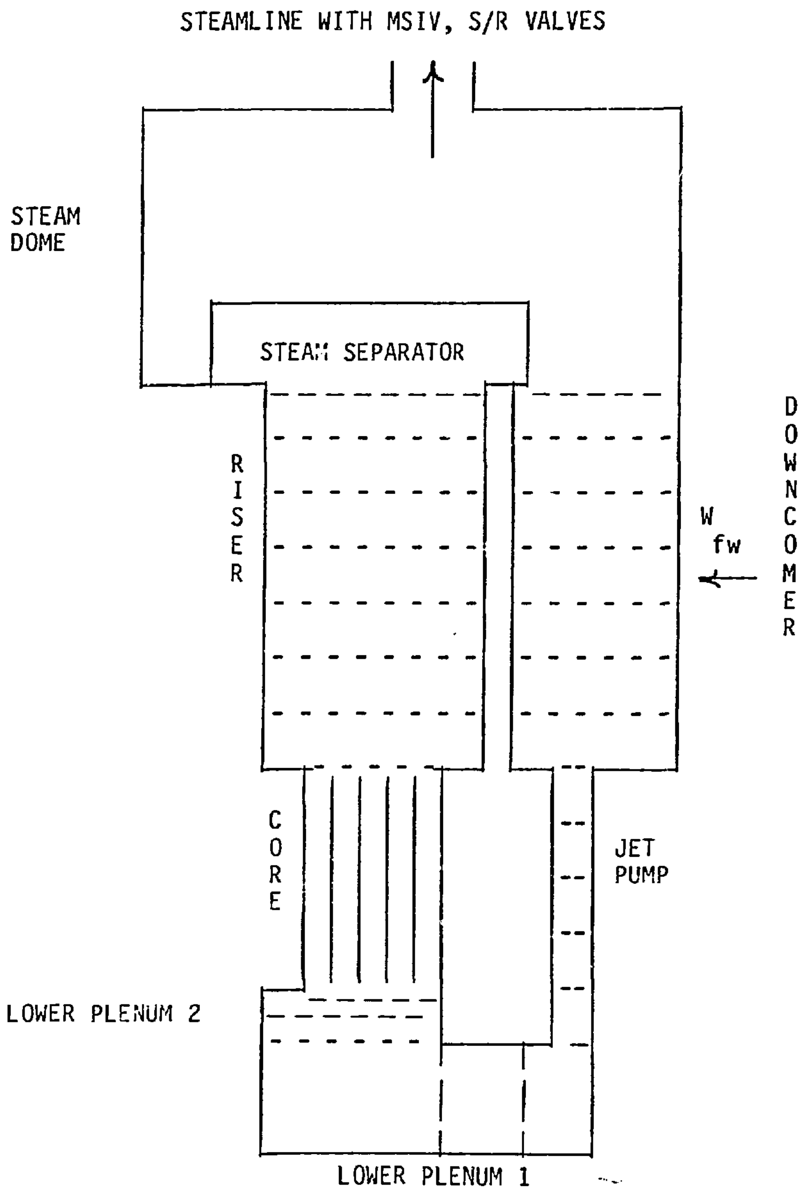

Figure 1 RAMONA-3B Representation of a BWR Vessel

Exposure in MWD/kgU

VOID HISTORY V,%	INSTANTANEOUS VOID-COEFFICIENT α , %	0	1		2		3		4		5		6		7		8	9	10	12.5	15	17.5	20	22.5	25	27.5	30
	0	xxx	x	x	x	x	x	x	x	x	x	x	x	x	x	x	x	x	x	x	x	x	x	x	x	x	x
	40/70-			x							x								x				x				x
0	0-Doppler	x									x								x				x				x
	0-Control Rod	x				x					x								x				x				x
	0-TM	x									x								x				x				x
	40/70-Control Rod	x				x					x								x				x				x
	40-	xxx	x	x	x	x	x	x	x	x	x	x	x	x	x	x	x	x	x	x	x	x	x	x	x	x	x
	0/70-			x							x								x				x				x
40	40-Doppler	x									x								x				x				x
	40-Control Rod	x				x					x								x				x				x
	40-TM	x									x								x				x				x
	0/70-Control Rod	x				x					x								x				x				x
	70-	xxx	x	x	x	x	x	x	x	x	x	x	x	x	x	x	x	x	x	x	x	x	x	x	x	x	x
	0/40-			x							x								x				x				x
70	70-Doppler	x									x								x				x				x
	70-Control Rod	x				x					x								x				x				x
	70-TM	x									x								x				x				x
	0/40-Control Rod	x				x					x								x				x				x

TOTAL # POINT/PER FUEL TYPE - 198.

Figure 2 TVA - Browns Ferry 3 Cycle 5 CASMO State Points for Each Fuel Type

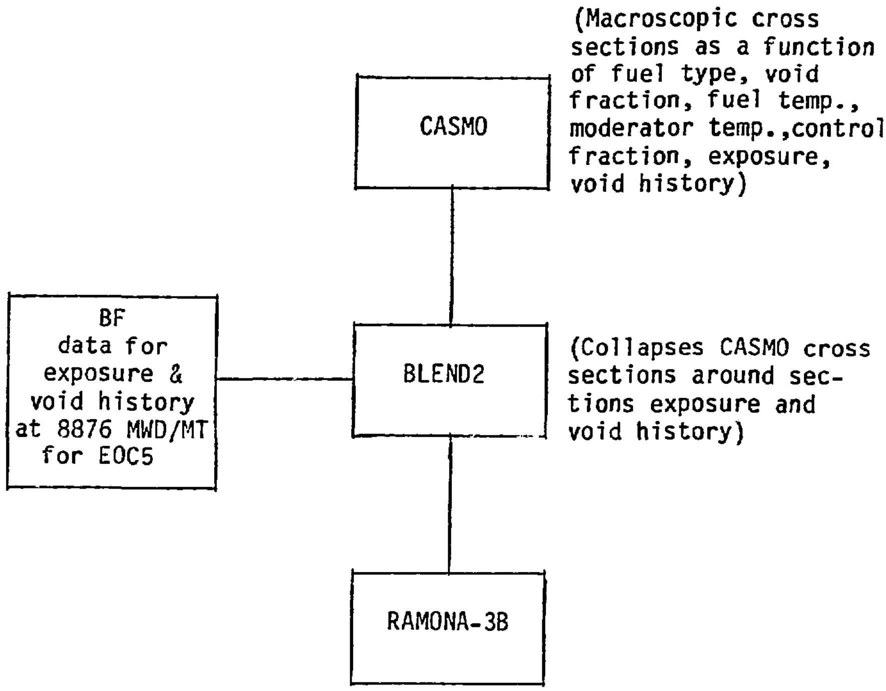

Figure 3 Generation of Browns Ferry (BF) EOC-5 Cross Sections

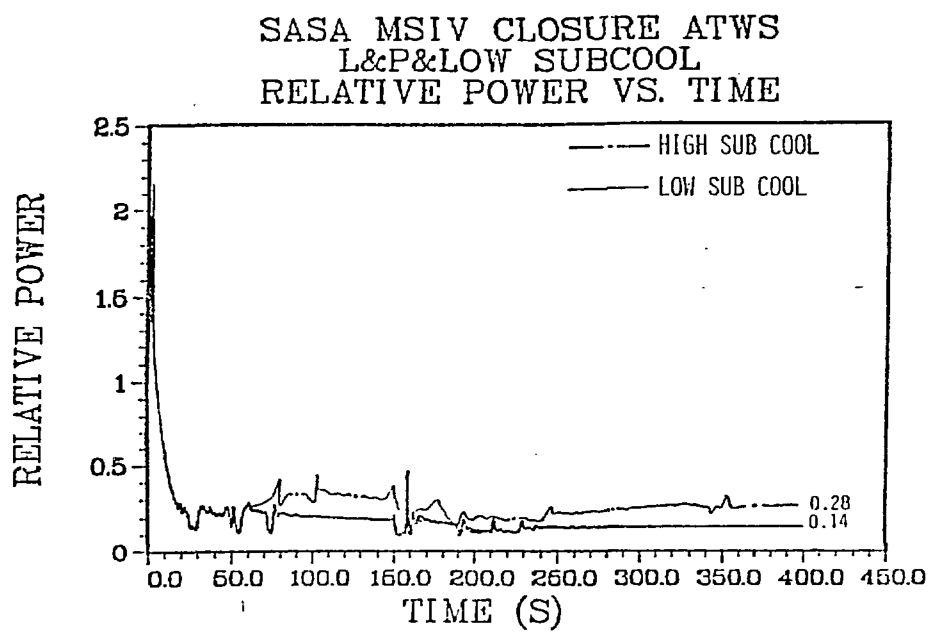

Figure 4 Relative Power

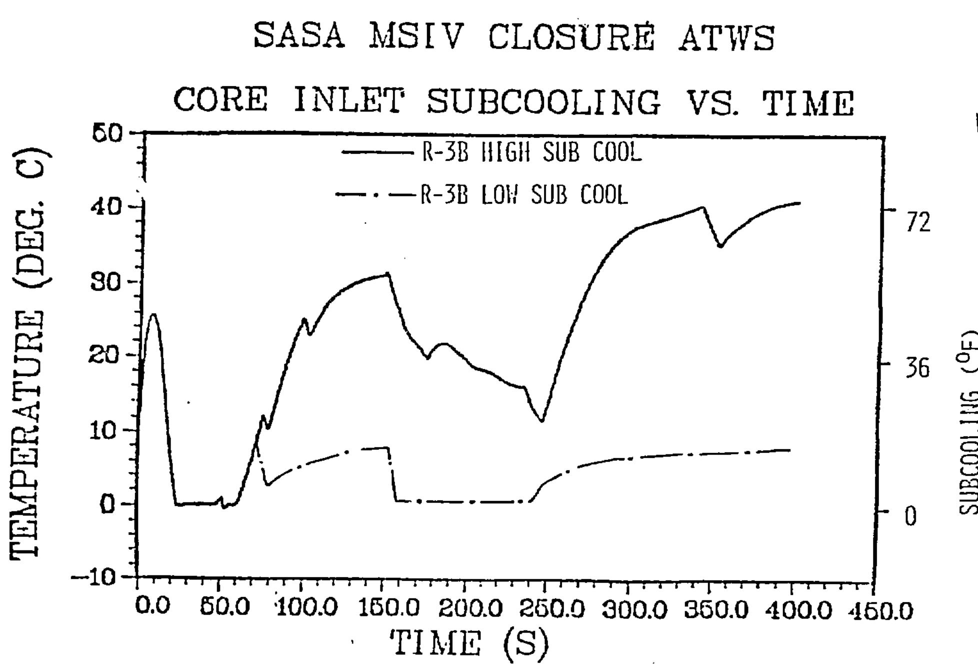

Figure 5 Core Inlet Subcooling

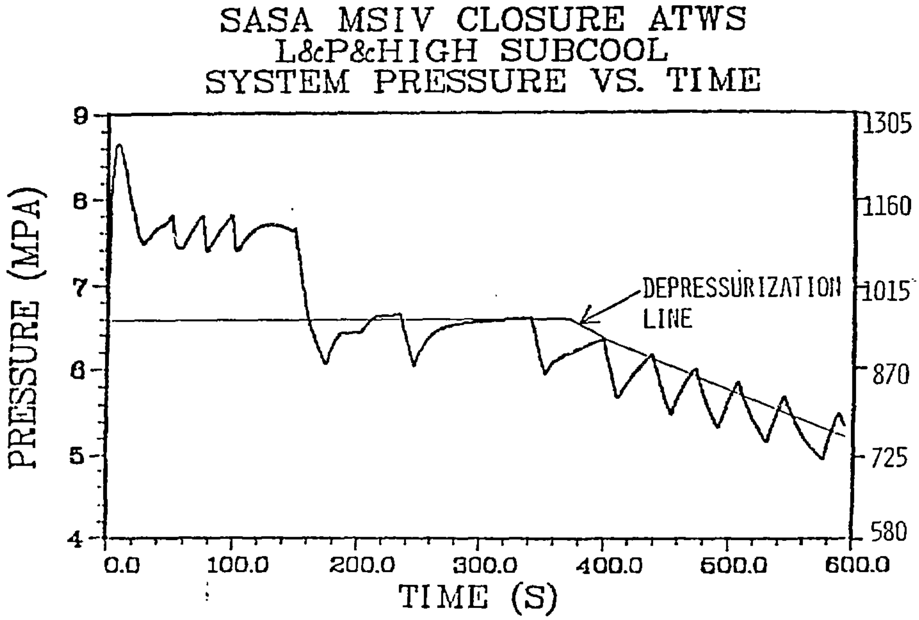

Figure 6 System Pressure

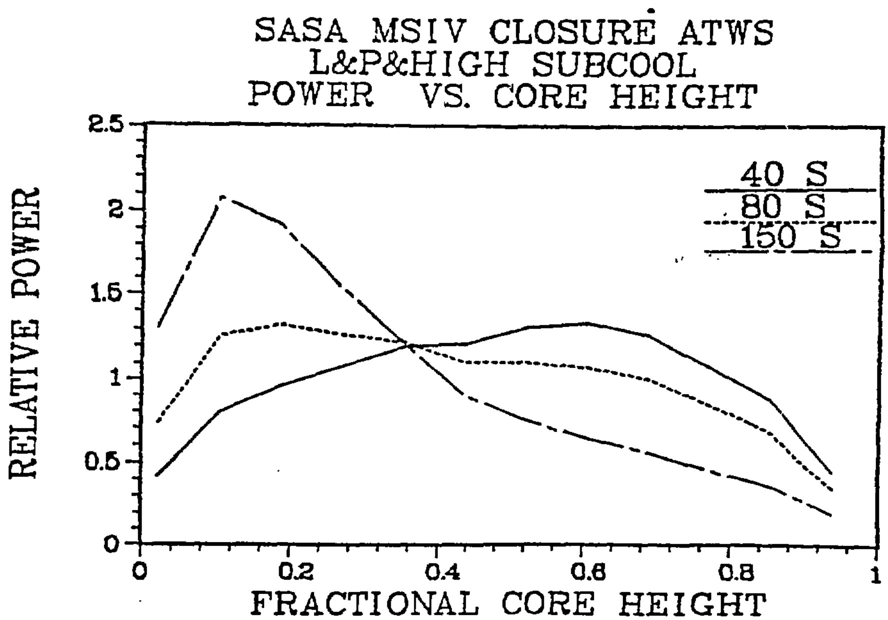

Figure 7 Relative Power Calculated at Different Times

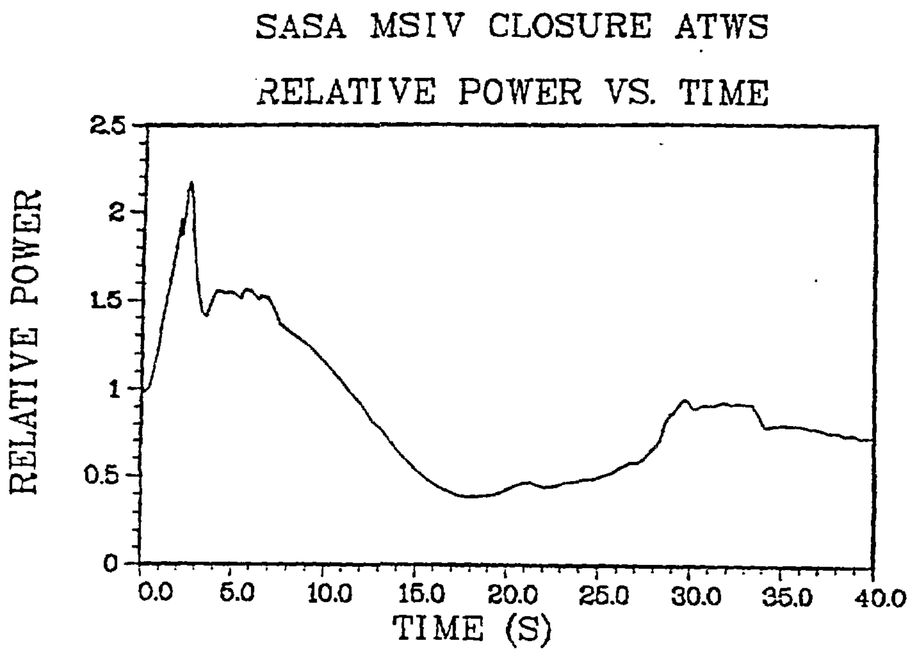

Figure 8 Relative Power

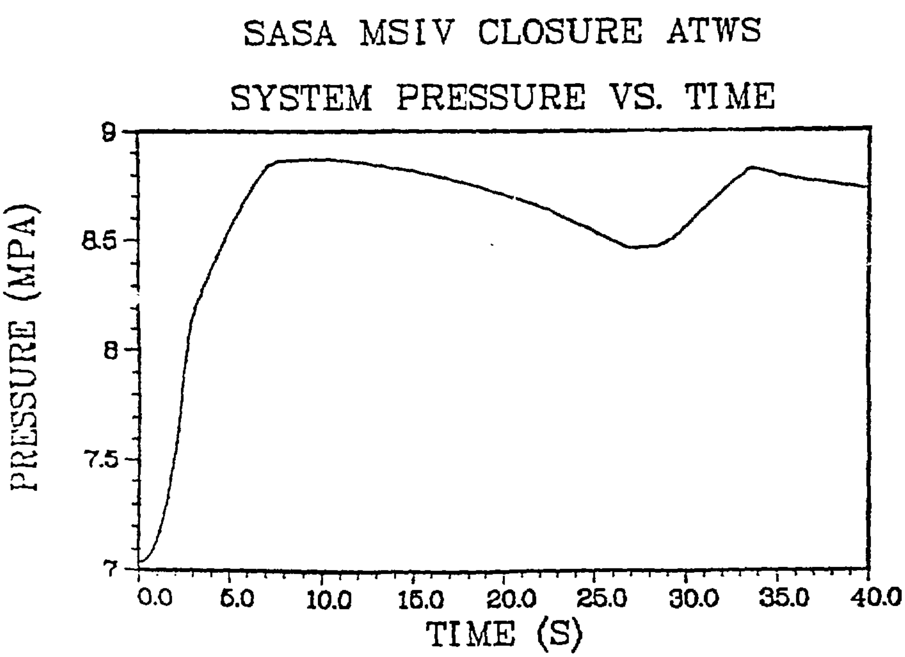

Figure 9 System Pressure

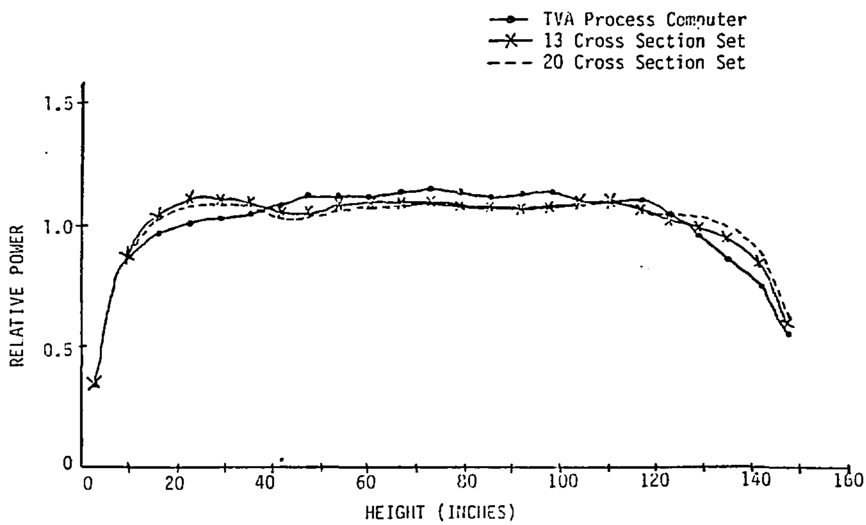

Figure 10 Steady State Predictions with Two Different Cross Section Sets

APPENDIX

SUMMARY OF RAMONA-3B MODELS

Neutron kinetics and thermal hydraulics are the two major parts of the RAMONA-3B code. Heat conduction in the fuel rod links these two parts.

Neutron Kinetics

The neutron kinetics model of RAMONA-3B starts from the following two-group, three-dimensional, time-dependent diffusion equations:

Fast Neutrons:

$$\frac{1}{v_1}\frac{\partial \phi_1}{\partial t} = \nabla \cdot D_1 \nabla \phi_1 - \Sigma_{21}\phi_1 - \Sigma_{a1}\phi_1 + (1-\beta)[\nu_1 \Sigma_{f1}\phi_1 + \nu_2 \Sigma_{f2}\phi_2] + \sum_{i=1}^{I} \lambda_i c_i$$

Thermal Neutrons:

$$\frac{1}{v_2}\frac{\partial \phi_2}{\partial t} = \nabla \cdot D_2 \nabla \phi_2 + \Sigma_{21}\phi_1 - \Sigma_{a2}\phi_2$$

Delayed Precursors:

$$\frac{\partial c_i}{\partial t} = \beta_i \left[\nu_1 \Sigma_{f1} + \nu_2 \Sigma_{f2}\phi_2 \right] - \lambda_i c_i$$

$$i = 1 \text{ to } I \text{ where } I = 6$$

However, in RAMONA-3B, it is assumed that the thermal neutron leakage term, i.e., $\nabla \cdot D_2 \nabla \phi_2$, can be either neglected or assumed to be constant. Thus, RAMONA-3B actually uses the well-known 1-1/2 group, coarse mesh diffusion model[8]. The boundary conditions at the core periphery are specified with parameters related to the extrapolation length for the fast flux and the albedo for the thermal flux.

The three-dimensional power generation is the sum of prompt and delayed energy deposition rates. The prompt component is proportional to the instantaneous fission rate, whereas the delayed energy deposition rate is calculated from the 1979 ANS Standard 5.1 for decay heat from fission products. The cross section dependence on fuel and moderator temperatures, void fraction and boron concentration is taken into account in the neutron kinetics calculation. The effect of control rod movement is also accounted for.

Heat Conduction in Fuel Rod

Thermal energy storage and heat conduction in the fuel elements (pellet, gas gap and cladding) are computed using the following discrete-parameter model:

$$\rho c \frac{\partial T}{\partial t} = \nabla \cdot k \nabla T + q'''$$

Thermal-Hydraulics

The reactor vessel thermal-hydraulics model of RAMONA-3B starts from the following one-dimensional, four-equation model:

Vapor Mass:

$$\frac{\partial}{\partial t}(\alpha\rho_g) + \nabla \cdot (j_g\rho_g) = \Gamma_v$$

Mixture Mass or Volumetric Flux:

$$\nabla \cdot j_m = \frac{\rho_\ell - \rho_g}{\rho_\ell \rho g}\Gamma_v - \left[\frac{\alpha}{\rho_g}\frac{D_g\rho_g}{Dt} + \frac{(1-\alpha)}{\rho_\ell}\frac{D_\ell\rho_\ell}{Dt}\right]$$

Mixture Momentum:

$$\frac{\partial G_m}{\partial t} + \frac{\partial}{\partial Z}\left[\alpha\rho_g v_g^2 + (1-\alpha)\rho_\ell v_\ell^2\right] = -\frac{\partial p}{\partial Z} - g\rho_m - f_\ell \phi_m^2 \frac{G_m|G_m|}{2\rho_\ell D_h}$$

Mixture Energy:

$$\frac{\partial}{\partial t} = \left[\alpha\rho_g u_g + (1-\alpha)\rho_\ell u_\ell\right] + \frac{\partial}{\partial Z}\left[\alpha\rho_g v_g h_g + (1-\alpha)\rho_\ell v_\ell h_\ell\right] = \frac{q_w'}{A} + q_\ell'''(1-\alpha)$$

Two further simplifications are made before the above set of equations is solved in RAMONA-3B. First, the mass and energy equations for the entire reactor vessel are combined (along with equations of state) to yield an equation for the average vessel pressure, which is a function of time, but not of space. Second, the momentum equations are integrated through each of the parallel core channels to obtain n-number of closed-contour integral momentum equations. These two simplifications reduce the computational burden of RAMONA-3B without significant loss in accuracy.

The code uses a slip model of the form:

$$v_g = S\, v_\ell + v_0$$

to calculate the relative velocity between the vapor and liquid phases. Non-equilibrium vapor generation and condensation are accounted for through appropriate correlations. However, the vapor phase is assumed to be at saturation, while the liquid phase can be either subcooled, saturated or superheated. Appropriate correlations are also used for wall friction, form losses and wall heat transfer, including post-CHF regime.

The code also uses the following boron transport equation:

$$\frac{\partial}{\partial t}[\rho_\ell(1-\alpha)c_B] + \nabla\cdot(\rho_\ell c_B j_\ell) = S_B$$

It should be noted that boron is assumed to move with the liquid velocity, and no boron stratification is allowed.

The code also employs models for typical BWR components, namely, jet pump, recirculation pump and steam separator. For the steam line, it uses the mass and momentum equations with the assumption of adiabatic process. It should be noted that in the steam line, pressure is a function of both time and space. Therefore, the acoustic effects in the steam line due to valve closure and/or opening are taken into account.

Solution Method

In RAMONA-3B, all partial differential equations are first transformed into ordinary differential equations. The initial or steady state conditions are then obtained by setting the time derivatives to zero, and iterating to obtain the eigenvalues of the system of equations. For the transient calculation, different methods are used for the different parts of the code. Specifically, the Gauss-Seidel iteration is used to integrate the prompt or fast neutron equations, explicit integration for delayed neutron equations, an iterative predictor-corrector method for heat conduction, the explicit first-order Euler method for vessel thermal-hydraulics, and finally, the fourth-order Runge-Kutta Simpson for the steam line dynamics. The neutron kinetics and fuel heat conduction equations are integrated with a master time step, whereas the thermal-hydraulic equations use a sub-step.

Nomenclature

A	Flow cross-sectional area, m^2
c_i	Delayed neutron precursor concentration, m^{-3}
c_B	Boron concentration per unit liquid mass
D	Diffusion coefficient, m
D_h	Hydraulic diameter, m
f	Friction factor
G	Mass flux, kg/m^2s
g	Acceleration due to gravity, m/s^2
h	Specific enthalpy, J/kg
j	Volumetric flux density, m/s
k	Thermal conductivity, W/m-K
p	Pressure, Pa
q'	Heat transfer per unit length, W/m
q'''	Heat generation or deposition per unit volume, W/m^3
S	Slip parameter
S_B	Source term for boron, kg/m^3s
T	Temperature, K
t	Time, s

u	Specific internal energy, J/kg
v	Velocity, m/s
v_o	Bubble rise velocity, m/s
Z	Axial coordinate, m
α	Void fraction
β	Total delayed neutron fraction
Γ_v	Vapor generation rate, kg/m^{3}s
λ	Decay constant for delayed neutrons, s^{-1}
Σ	Macroscopic neutron cross section, m^{-1}
ϕ	Neutron flux, m^{-2} s^{-1}
ϕ_m	Two-phase multiplier
ρ	Density, kg/m^3
ν	Mean number of neutrons in fast or thermal group

Subscripts

1	Fast-group neutrons
2	Thermal-group neutrons
a	Absorption
f	Fission
g	Vapor (saturated)
i	Index for delayed precursors
ℓ	Liquid
m	mixture