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Recent SCDAP/RELAP5 Improvements for BWR Severe Accident Simulations *†

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ABSTRACT

A new model for the SCDAP/RELAP5 severe accident analysis code that represents the control blade and channel box structures in a boiling water reactor (BWR) has been under development since 1991. This model accounts for oxidation, melting, and relocation of these structures, including the effects of material interactions between B_4C, stainless steel, and Zircaloy.

This paper describes improvements that have been made to the BWR control blade/ channel box model during 1994 and 1995. These improvements include new capabilities that represent the relocation of molten material in a more realistic manner and modifications that improve the usability of the code by reducing the frequency of code failures. This paper also describes a SCDAP/RELAP5 assessment calculation for the Browns Ferry Nuclear Plant design based upon a short-term station blackout accident sequence.

1. INTRODUCTION

Work began at Oak Ridge National Laboratory (ORNL) in 1991 to improve the SCDAP/RELAP5 code[1] for boiling water reactor (BWR) applications. SCDAP/RELAP5 is a best estimate analysis tool for light water reactor severe accident applications that has been developed primarily at Idaho National Engineering Laboratory (INEL). INEL is the sole institution responsible for maintaining the official version of SCDAP/RELAP5.

The efforts at ORNL have focussed on the development of a new SCDAP/RELAP5 model that represents the severe accident response of the control blade and channel box structures in a BWR.[2] Sketches of a typical BWR control blade and fuel assembly are shown in Figure 1. The control blade has four wings, each consisting of a stainless steel blade sheath that surrounds a row of small stainless steel absorber tubes

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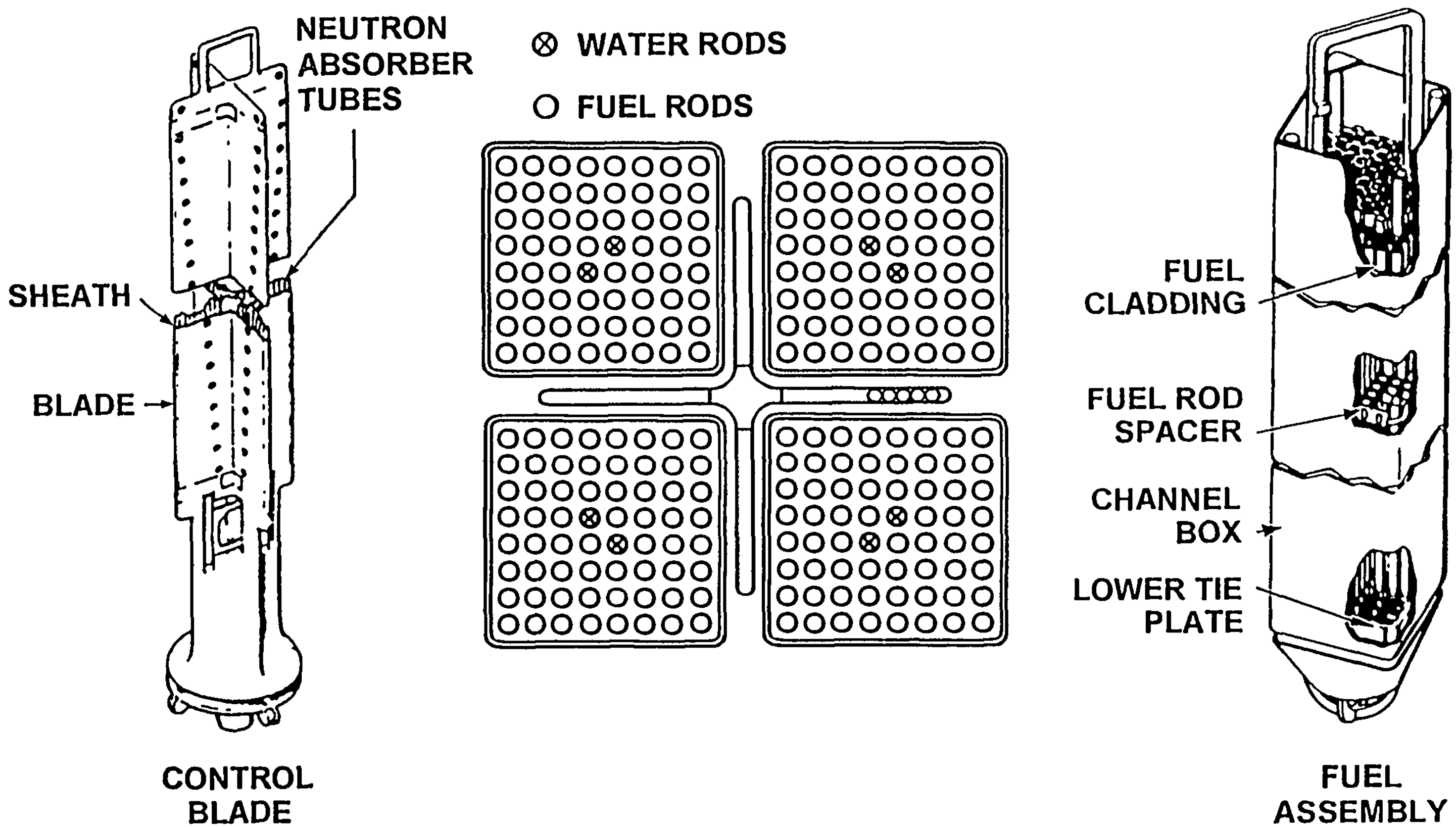

Figure 1. Sketch of typical BWR control blade and fuel assembly.

filled with B_4C powder. The fuel assembly consists of an array of fuel rods surrounded by a Zircaloy channel box. The control blade is located between four fuel assemblies.

The BWR control blade/channel box model was originally developed at ORNL as a result of posttest analyses of the CORA-16 and CORA-17 BWR experiments[3] and accounts for oxidation, melting, and relocation of these structures, including the effects of material interactions. The degradation process begins when the B_4C powder reacts with the stainless steel absorber tubes. This B_4C/stainless steel eutectic liquifies at a temperature of ~ 1505 K, which is lower than the melting temperature of pure stainless steel. Then, either by failing the stainless steel blade sheath or flowing through the water circulation holes in the sheath (shown in Figure 1), the molten B_4C/stainless steel mixture relocates downward and freezes to form a crust on the surface of the control blade. This crust builds up until a blockage spans the gap between the control blade and the channel box. The Zircaloy channel box reacts with the stainless steel in the blockage and melts at the stainless steel/Zircaloy liquefaction temperature of ~ 1523 K, which is much lower than the melting temperature of pure Zircaloy.

The basic features of the BWR control blade/channel box model within SCDAP/RELAP5 were described in a previous paper at the 1993 Water Reactor Safety Information Meeting.[4] Subsequently, several new capabilities and improvements have been added to the model and are described in this paper along with an example simulation of a short-term station blackout accident sequence for the Browns Ferry Nuclear Plant design. The improved BWR control blade/channel box model is available from INEL as a part of SCDAP/RELAP5, Mod3.1, Release D.

2. DESCRIPTION OF BWR CONTROL BLADE/CHANNEL BOX MODEL

The control blade/channel box model is based on the nodal configuration shown in Figure 2. The actual control blade configuration of small absorber tubes inside a stainless steel blade sheath is converted into an equivalent slab geometry. At each axial elevation, three radial temperature nodes are used for the control blade while two temperature nodes represent the channel box wall. The surrounding dashed line in Figure 2 represents an adiabatic boundary. Because of thermal symmetry, the three control blade temperature nodes actually represent only half of a control blade divided along the centerline of the row of absorber tubes. The channel box wall is divided into two segments because one segment is adjacent to a control blade while the other is not.

The control blade and channel box interact with two RELAP5 hydraulic volumes: one for the interstitial region and the other for the fuel bundle region. The gap between the blade sheath and the absorber tubes is modeled. This gap communicates with the interstitial coolant volume through a series of holes in the blade sheath (shown in Figure 1). The gap results in two additional surfaces for stainless steel oxidation and also imposes an additional thermal resistance between the blade sheath and the absorber tubes. The portion of the interstitial coolant volume beyond the tip of the control blade provides an important path for molten control blade material to relocate downward onto the core plate.

A finite difference formulation is used to model the thermal responses of the control blade and channel box structures. Energy equations representing conduction and convection heat transfer in the radial direction are solved implicitly to determine new values for the five nodal temperatures at each axial elevation. Axial conduction, relocation/solidification, oxidation, and radiation heat transfer are computed

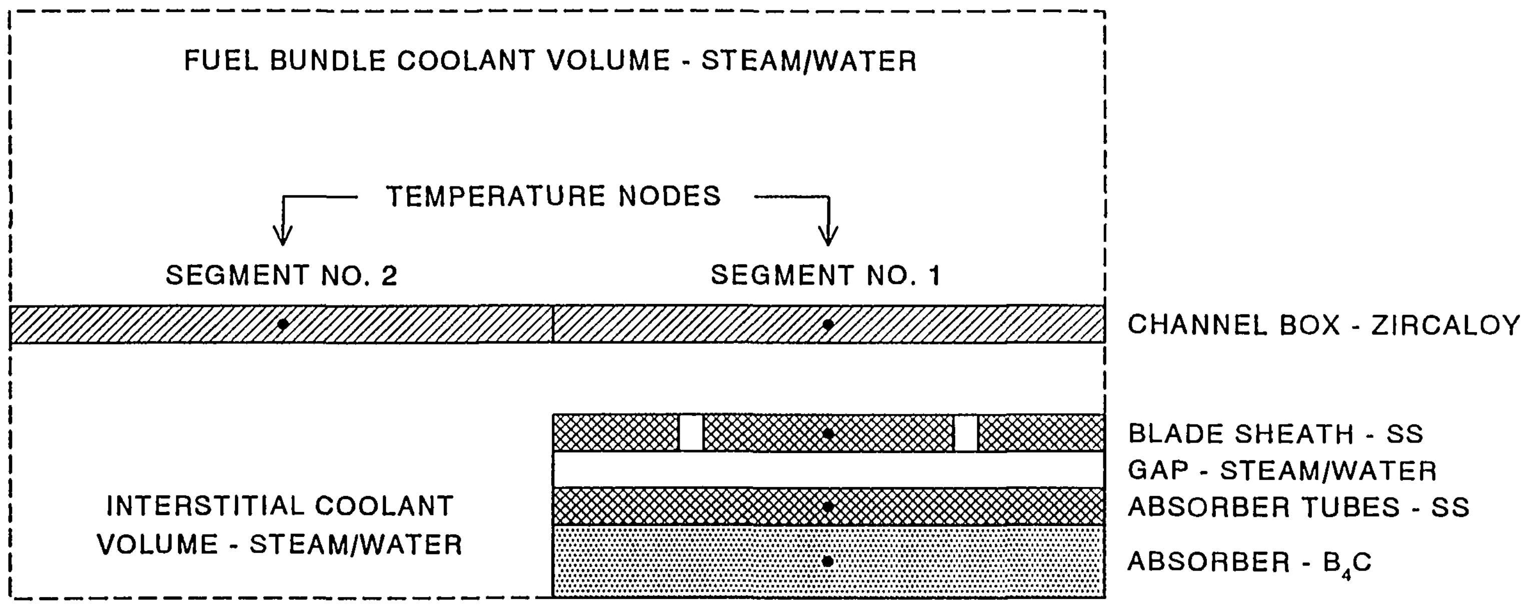

Figure 2. BWR control blade/channel box model with equivalent slab geometry and five temperature nodes at each axial elevation.

explicitly using previous timestep information and are included as constant heat transfer terms in the energy equations.

An approximate solution method is used to account for melting in the energy equations. At the end of each timestep, the new nodal temperatures calculated from the energy equations are compared with their associated melting points. If a nodal temperature is greater than its melting point and the node also contains solid material, then the nodal temperature is adjusted downward to equal its melting point, and the associated sensible heat is used to melt an appropriate amount of the solid material. This temperature adjustment is not made if the node does not contain any solid material (i.e., if only liquid material is present).

Material interactions between B_4C/stainless steel and stainless steel/Zircaloy are represented by using reduced melting temperatures. Eutectic liquefaction temperatures are used in the melting calculations rather than the melting temperatures of the pure materials. The liquefaction temperatures are ~ 1505 K for B_4C/stainless steel mixtures and ~ 1523 K for stainless steel/Zircaloy mixtures.

The relocation of molten material is assumed to be controlled by solidification rates. As molten material relocates downward over an underlying solid structure, it solidifies and transfers heat to the underlying structure. In the control blade/channel box model, molten material is allowed to relocate downward until it either solidifies, is diverted horizontally by a blockage, or moves past the bottom of the defined core. The effects of blockages on the relocation of molten material are described in more detail in Section 2.2.

The oxidation of Zircaloy, stainless steel, and B_4C is included in the control blade/channel box model. Based on steam availability and the reaction kinetics, oxidation heat generation rates and hydrogen production rates are calculated for each surface. Oxidation of stainless steel is based on a chemical composition of 74% iron, 18% chromium (which is the major contributor to total reaction energy), and 8% nickel.

Radiation calculations on the fuel bundle side of the channel box are performed in the normal manner by the SCDAP radiation model,[5] with the two segments of the channel box treated as independent surfaces. Radiation calculations on the interstitial side of the channel box are performed internally by the control blade/channel box model. These radiation calculations are activated whenever the local coolant void fraction exceeds a user-specified value.

All hydrodynamic parameters used in the control blade/channel box model are obtained from the RELAP5 data base. These parameters include steam flow rates, coolant properties (pressures, temperatures, void fractions), and convective heat transfer coefficients. Control blade/channel box parameters returned to SCDAP/RELAP5 include hydrogen generation rates, heat transfer rates from the structures to the coolant, and coolant flow area reductions caused by frozen crust formation.

2.1 RECENT MODEL IMPROVEMENTS

Several new capabilities and improvements have been added to the BWR control blade/channel box model during 1994 and 1995. These improvements include: (1) implementation of timestep repetition, (2) modifications that improve execution times for BWR simulations, (3) replacement of hard-wired material property correlations with equivalent correlations from the MATPRO library of material properties,[6] (4) modifications that allow molten control blade/channel box material to spread radially into the fuel bundle, and (5) modifications that allow molten control blade/channel box material to relocate downward into the lower plenum.

Improvements 1 and 2 have minimal impact on the calculated results for BWR simulations, but they reduce the frequency of water property failures and improve the usability of the code. The timestep repetition modifications (improvement 1) make the control blade/channel box model consistent with logic implemented previously at INEL for the other SCDAP core models and involves saving all necessary variables at the beginning of a timestep so the calculations for the timestep can be repeated if necessary.

The use of MATPRO material properties (improvement 3) makes the code easier to maintain by eliminating redundancy. Improvements 4 and 5 are new capabilities that represent the relocation of molten material during core degradation in a more realistic manner. The relocation modeling is explained in greater detail in Section 2.2.

2.2 RELOCATION LOGIC

During a BWR severe accident, core degradation begins with control blade liquefaction at a temperature of ~ 1505 K. A molten B_4C/stainless steel mixture relocates downward and solidifies at a lower elevation to form a blockage in the interstitial volume between the control blade and channel box segment No. 1 (see Figure 2). The relocation logic in the control blade/channel box model allows for horizontal movement of molten material when a blockage inhibits downward movement. For a blockage between the control blade and channel box segment No. 1, the horizontal relocation logic proceeds in the following order:

(1) If channel box segment No. 1 has failed, molten control blade material relocates radially through the opening in the channel box wall and down the fuel bundle side of the channel box.

(2) If the interstitial volume beyond the tip of the control blade is open, any remaining molten material relocates laterally from segment No. 1 to segment No. 2.

(3) Any remaining molten material pools up on top of the interstitial blockage.

Molten control blade material that relocates radially through the opening in the channel box wall can freeze to form a second blockage in the fuel bundle volume between channel box segment No. 1 and the first row of fuel rods. When this second blockage exists, molten material relocates radially into the fuel bundle region and down the outer surface of adjacent fuel or simulator rods. The recent model improvements represent this radial spreading into the fuel bundle by transferring the molten material from the control blade/channel box model to the SCDAP fuel or simulator rod models.

Downward-moving molten material that does not solidify before reaching the lowest elevation of the control blade or channel box surfaces relocates below the bottom of the defined core. If a COUPLE finite element mesh[7] is defined by the user to represent lower plenum debris, then the recently added lower plenum relocation logic transfers this molten control blade/channel box material directly into the lower plenum debris bed. (This relocating material does not interact with the structures in the core plate region because SCDAP/RELAP5 does not currently include a model to represent the behavior of the core plate during a severe accident.)

3. ASSESSMENT CALCULATION FOR BROWNS FERRY NUCLEAR PLANT DESIGN

A SCDAP/RELAP5 input deck has been prepared for the Browns Ferry Nuclear Plant design based upon a short-term station blackout (STSB) accident sequence. This STSB simulation was developed by modifying an existing Browns Ferry input deck representing a large-break loss of coolant accident (LOCA) sequence obtained from INEL. The STSB input deck has been used to perform assessment calculations for the purpose of testing the BWR control blade/channel box model improvements described in this paper. A description of the input deck is provided in this section along with some calculated results from the Browns Ferry STSB accident simulation.

A nodalization diagram for the Browns Ferry reactor coolant system is shown in Figure 3. The reactor coolant system is represented from the feedwater inlet to the turbine inlet and includes the reactor pressure vessel, two recirculation loops, the feedwater piping, the control rod drive (CRD) cooling water, and the steam piping. The recirculation loop on the left-hand side of Figure 3 was originally used at INEL to represent the initiating pipe break for a LOCA sequence. For the STSB accident sequence, both recirculation loops remain intact (i.e., the valves between volumes 200 and 998 and volumes 210 and 999 are closed). The Browns Ferry containment is not represented explicitly. Rather, the safety/relief valves (SRVs) discharge into volume 561, which is assumed to remain at a constant pressure of 0.31 MPa (45 psia).

A more detailed nodalization diagram of the lower half of the reactor pressure vessel is shown in Figure 4 with hydraulic volumes represented by open boxes and solid structures represented by shaded boxes. The active core is divided into four radial rings and thirteen axial levels. The center ring of the core (right-hand side of Figure 4) represents ~55% of the fuel assemblies, while the other three rings near the periphery of the core (where the power density is progressively lower) each represent ~15% of the fuel assemblies. The lower 1.07 m (3.5 ft) of the active core is divided into seven axial levels, while the upper 2.74 m (9.0 ft) of the active core is divided into six axial levels.

In each of the four radial rings, there is one set of volumes that represents the coolant flow inside the fuel assemblies (volumes 320, 321, 322, and 323) and another set of volumes that represents the coolant flow outside the fuel assemblies in the interstitial region surrounding the control blades (volumes 370, 324, 325, and 326). The interstitial volumes outside the fuel assemblies are connected by cross-flow junctions that allow coolant to flow horizontally between the center and the periphery of the core. Note that for simplicity, these cross-flow junctions are illustrated in Figure 4 only at axial node 13. Similar cross-flow junctions also exist at axial nodes 1 through 12.

The primary coolant flow through the core (~90% of total) is from the lower plenum (volumes 292, 293, and 294) through the fuel support pieces and lower tie plates (volumes 300, 301, 302, and 303) and into the fuel assemblies. The remaining core coolant flow (~10% of total) is through the interstitial region. Coolant enters the interstitial region by either flowing through holes machined in the lower tie plates, leaking past the core plate, or flowing through the control rod guide tubes (volumes 340 and 350) from the CRD cooling water pumps.

The jet pumps are represented by volumes 260, 265, and 270 (first recirculation loop) and volumes 261, 266, and 271 (second recirculation loop). In a BWR, the upper mixing sections of the jet pumps are connected to the lower diffuser sections by mechanical slip fits for easy removal during maintenance.

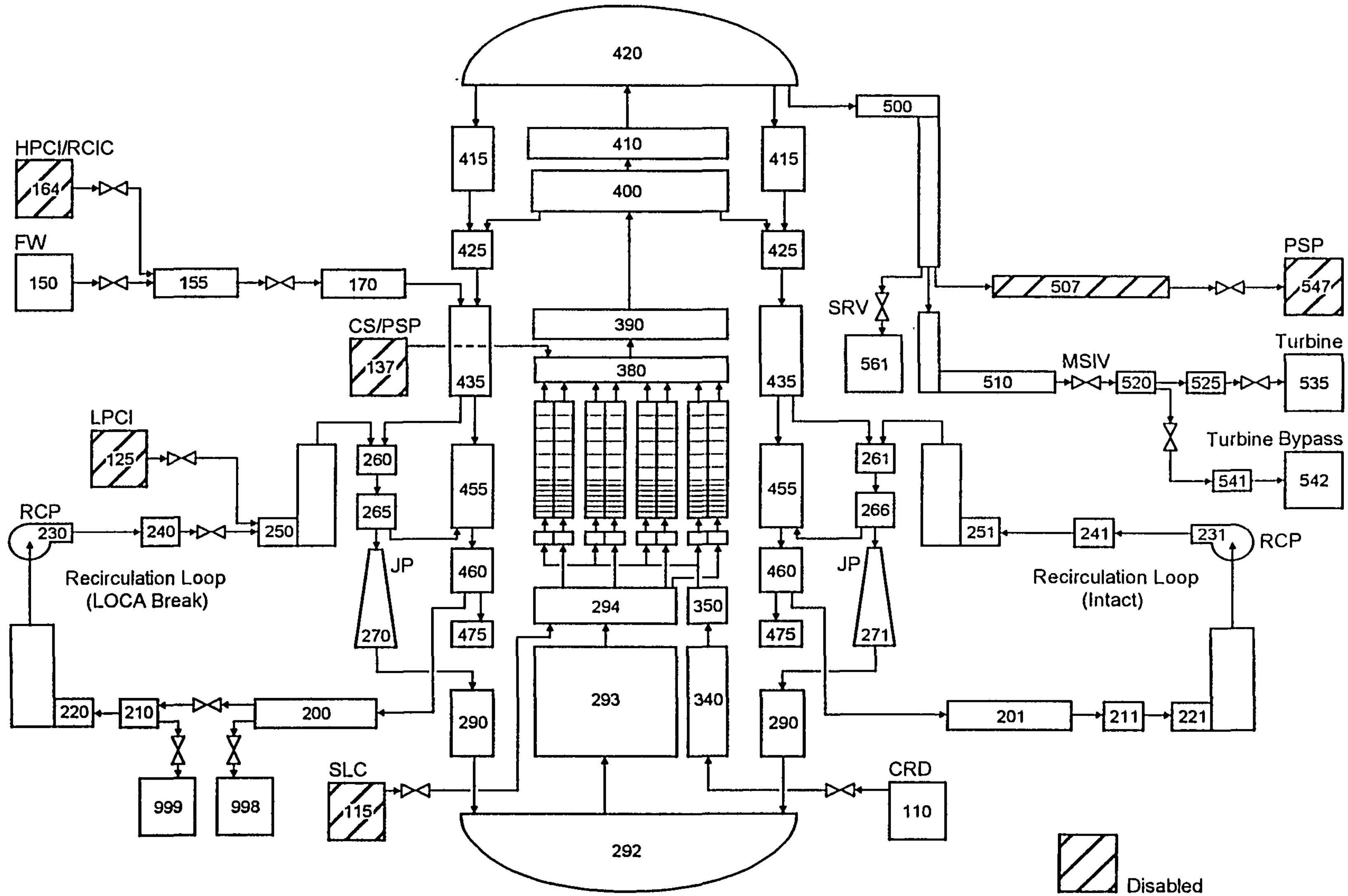

Figure 3. Nodalization diagram of Browns Ferry reactor coolant system.

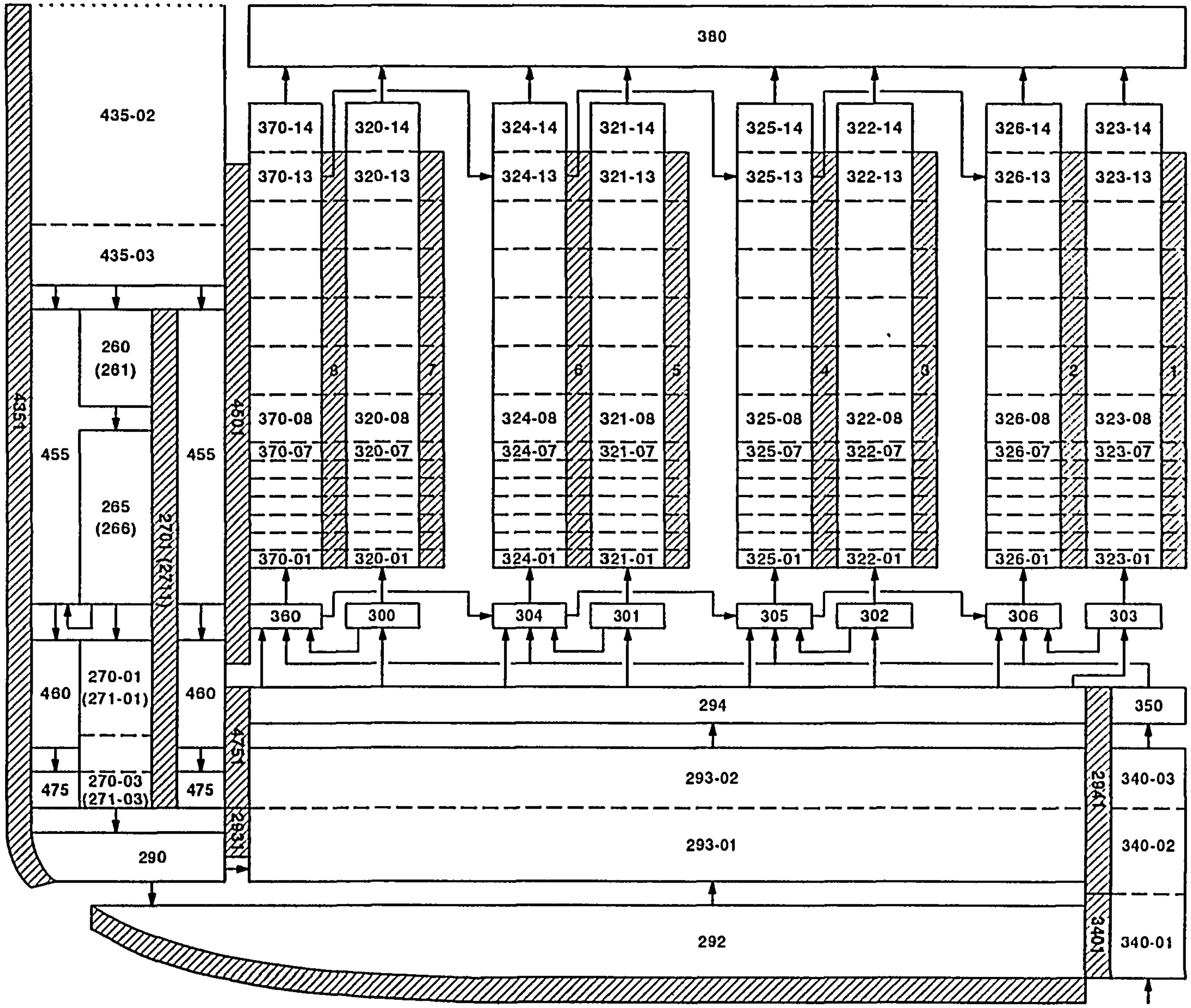

Figure 4. Nodalization diagram of Browns Ferry lower reactor vessel.

The leakage through these slip joints is represented by junctions that connect the jet pumps (bottom of volumes 265 and 266) to the annulus region (volume 455).

The SCDAP components representing the core structures are shown in Figure 4 as shaded boxes with labels 1 through 8. A summary of these SCDAP components is provided in Table 1. All fuel rods and control blade/channel boxes within each core radial ring are represented by single representative SCDAP components. The RELAP5 heat structures representing the pressure vessel wall and the other internal structures are shown in Figure 4 as shaded boxes with 4-digit labels. The lower vessel wall (not labeled in Figure 4) and any lower plenum debris are represented by a COUPLE finite element mesh. There are 18 elements representing the vessel wall and 36 elements (initially filled with water) representing the lower plenum debris volumes.

The radial and axial power profiles used for the Browns Ferry STSB simulation are shown in Figures 5 and 6. Note that the peripheral fuel assemblies produce only one third as much power as the center fuel assemblies. Also note that the peak power in the center fuel assemblies is at a low elevation in the core.

The initial condition for the Browns Ferry STSB simulation is steady-state operation at full reactor power (3293 MWt). The STSB accident sequence is caused by a loss of off-site AC power combined with failure of the emergency diesel generators and is initiated at time zero by: (1) loss of AC power to the recirculation pumps, (2) loss of AC power to the CRD cooling water pumps, (3) initiation of main steam isolation valve (MSIV) closure, (4) initiation of scram, and (5) loss of the turbine-driven feedwater pumps and initiation of feedwater coast-down. Throughout the duration of the STSB accident sequence, all sources of cooling water for the core are unavailable.

After the feedwater trip signal at time zero, the turbine-driven feedwater pumps briefly continue to supply coolant to the vessel during coast-down. In the STSB simulation, the feedwater flow continues at 100% for 11.78 s after the trip signal and then reduces linearly to zero by 35 s after the trip signal. The total

Table 1. Summary of SCDAP components for Browns Ferry simulation

Number	Type	Description
1	Fuel rod	432 fuel assemblies at center of core (ring 1)
2	Control blade/channel box	108 control blades at center of core (ring 1)
3	Fuel rod	104 fuel assemblies (ring 2)
4	Control blade/channel box	26 control blades (ring 2)
5	Fuel rod	128 fuel assemblies (ring 3)
6	Control blade/channel box	32 control blades (ring 3)
7	Fuel rod	100 fuel assemblies at periphery of core (ring 4)
8	Control blade/channel box	19 control blades at periphery of core (ring 4)

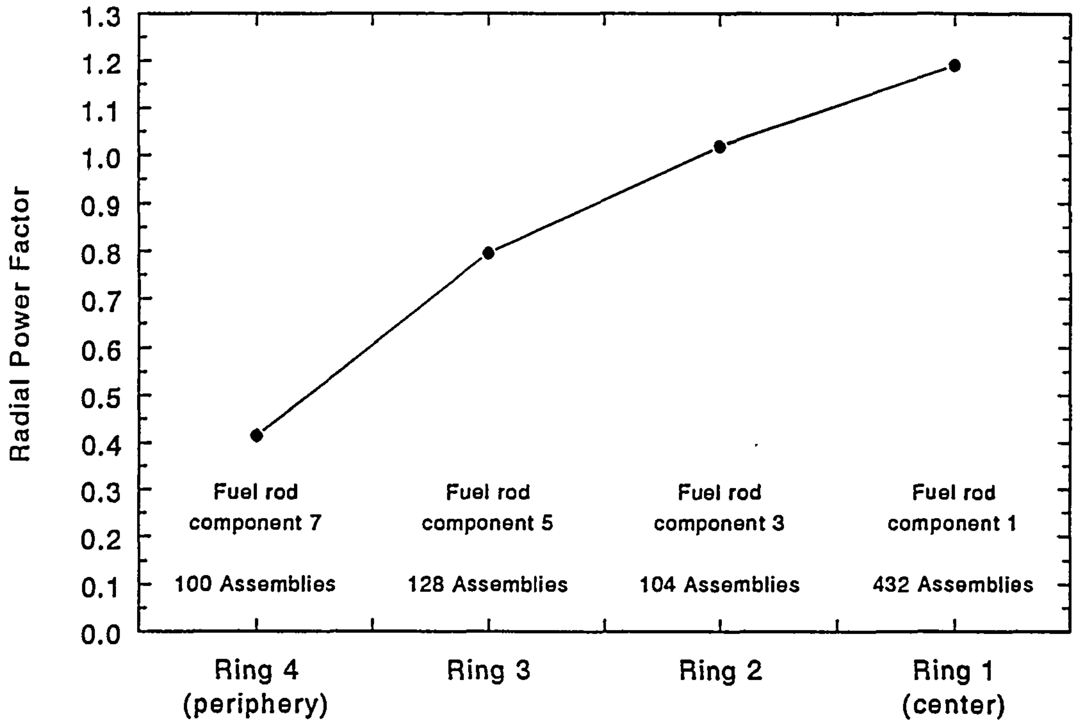

Figure 5. Browns Ferry power profile for 4 radial rings.

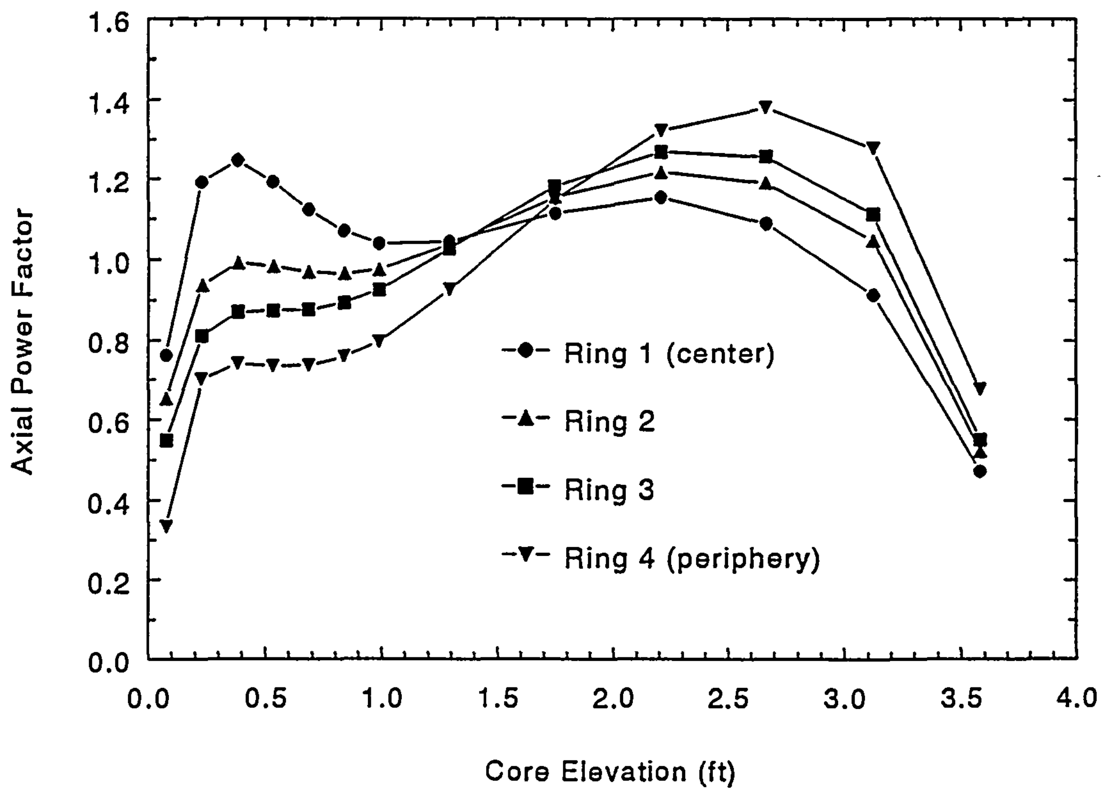

Figure 6. Browns Ferry power profiles for 13 axial nodes.

amount of coolant injected during the feedwater pump coast-down is 39,310 kg (86,660 lbm), which is about 14% of the normal vessel inventory.

Automatic actuation of the SRVs is represented during the initial slow boil-off; manual operation of the SRVs by reactor operators as specified in the Emergency Procedure Guidelines is not represented during this time period. The individual SRVs open and close at different pressures. The first SRV opens when the vessel pressure increases to 7.688 MPa (1115 psia) and remains open until the vessel pressure declines to 6.991 MPa (1014 psia). Other SRVs would also open if the pressure continued to increase.

The calculated response of the vessel pressure is illustrated in Figure 7. The vessel pressure is controlled by the automatic SRV actuations until the Automatic Depressurization System (ADS) is manually initiated as specified in the Emergency Procedure Guidelines, which occurs when the fuel bundle water level in the center radial ring (see Figure 8) reaches one-third of the active fuel height. The vessel depressurizes until it equilibrates with the containment pressure [represented here as a constant 0.31 MPa (45 psia)]. The vessel water inventory flashes during the depressurization, stabilizing in the lower plenum at a level well below the core.

Core degradation occurs under "dry" conditions with very little steam available for oxidation reactions. The calculated hydrogen generation of 210 kg shown in Figure 9 is much less than the theoretical value of ~3000 kg that would be generated by complete oxidation of all Zircaloy from the cladding and channel boxes and all stainless steel from the control blades.

The calculated times of important core degradation events are listed in Table 2. Control blade liquefaction is calculated to begin in core radial ring 1 at 120 min after scram when the control blade temperatures reach 1505 K. This molten control blade material relocates downward and solidifies at

Table 2. Calculated event times for Browns Ferry STSB accident

Description of Event	Time After Scram (min)			
Water level at top of active fuel	59			
ADS initiation, water level at 1/3 active fuel	91			
	Ring 4 (periphery)	Ring 3	Ring 2	Ring 1 (center)
First control blade liquefaction (1505 K)	168	133	125	120
First control blade radial spreading into fuel bundle	260	170	152	131
First fuel cladding relocation (2200 K)	256	166	150	134
Molten ceramic pool (2844 K) slumps into lower plenum		259	259	259
Creep rupture failure of lower vessel wall	300			

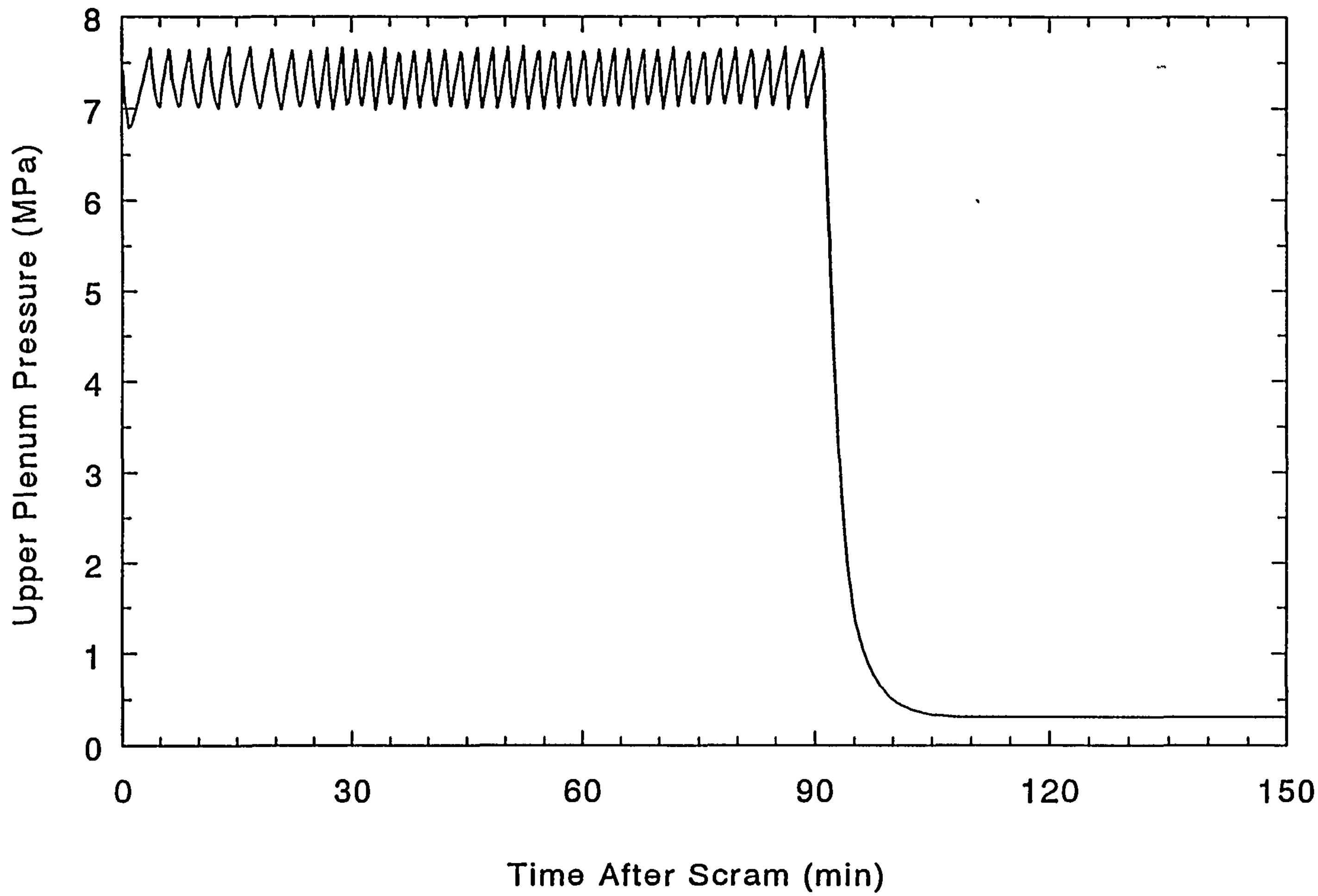

Figure 7. Calculated vessel pressure for Browns Ferry STSB accident.

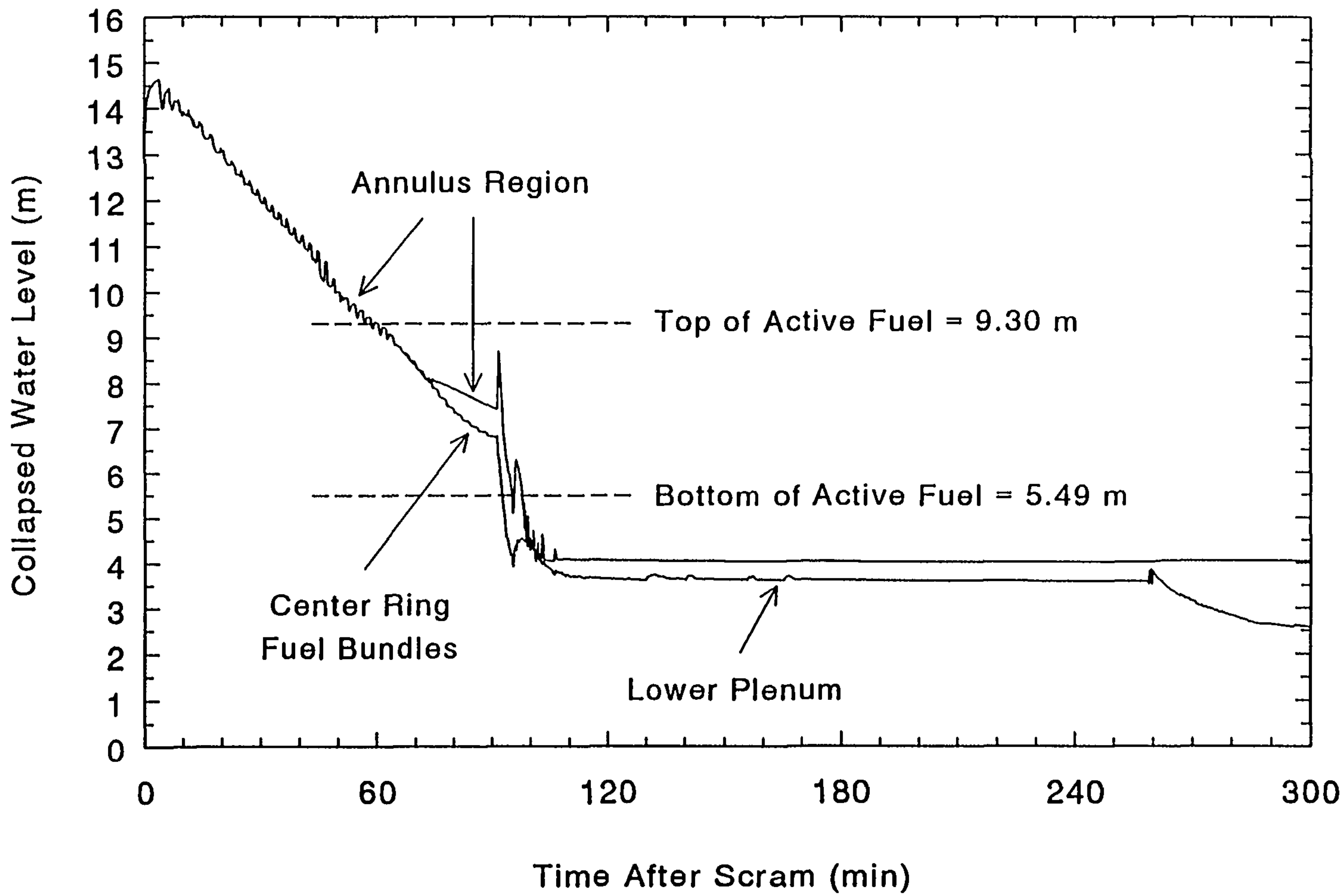

Figure 8. Calculated vessel water levels for Browns Ferry STSB accident.

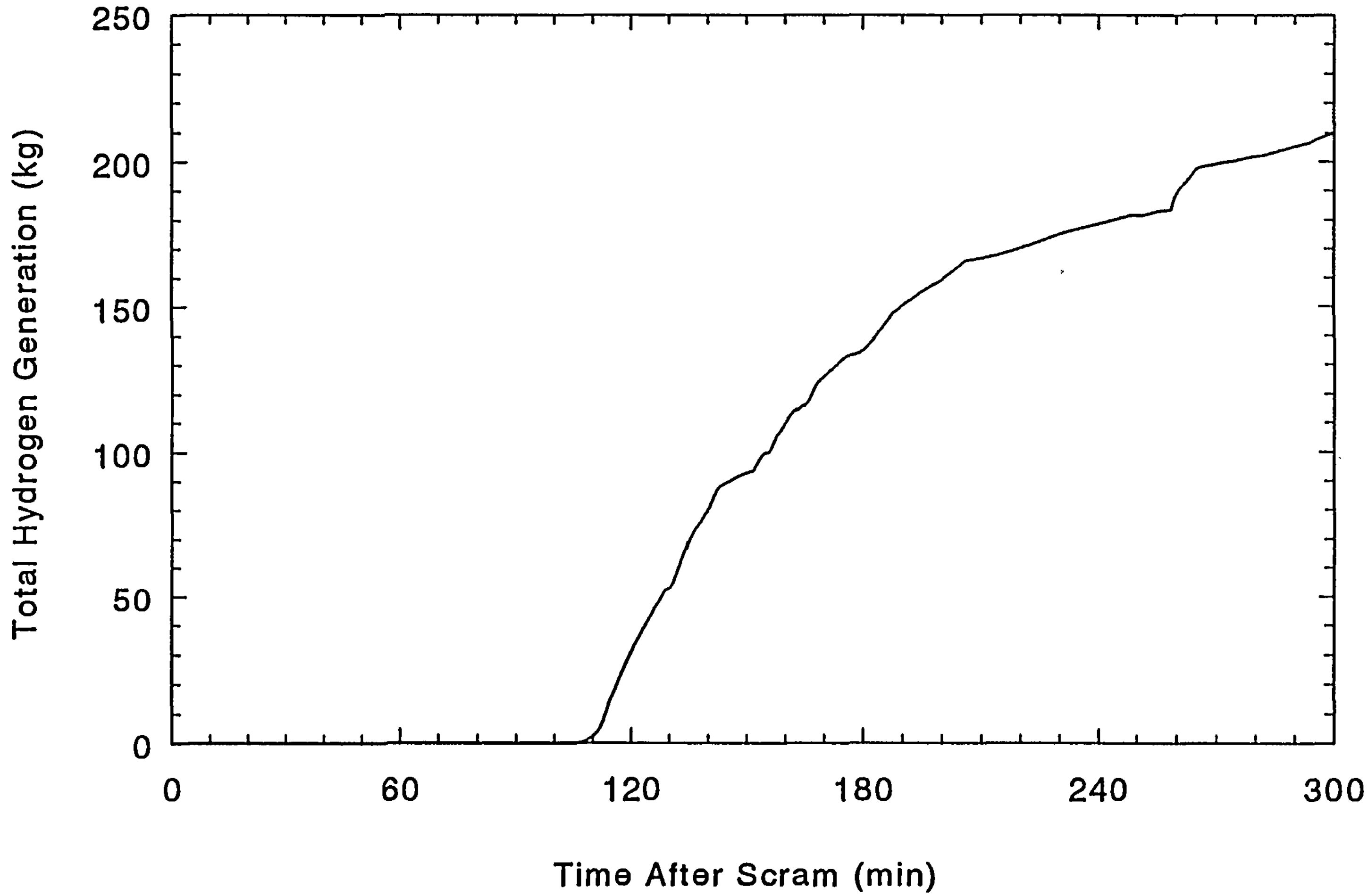

Figure 9. Calculated hydrogen generation for Browns Ferry STSB accident.

lower elevations in the core. At 131 min, extensive blockages have formed around the control blades in core radial ring 1, and molten control blade material is calculated to begin spreading radially into the fuel bundles. Some of the molten control blade material also relocates below the bottom of the core and slumps directly into the lower plenum to form a debris bed (see Figure 10). Rapid quenching of the debris as it falls through the water in the lower plenum is not represented in this calculation, but slow heat transfer from the top surface of the debris bed to the lower plenum water is represented after formation of the debris bed.

The first fuel cladding relocation is calculated to begin in core radial ring 1 at 134 min (see Table 2) when the cladding oxide layer fails at a user-specified temperature of 2200 K. As the core continues to heatup, the fuel rods form a large molten ceramic pool that occupies three of the four core radial rings. At 259 min, this entire molten ceramic pool has reached a temperature of 2844 K and is calculated to slump into the lower plenum (see Figure 10). After this fuel rod debris reaches the lower plenum, the associated decay heat causes the vessel wall temperatures to increase until creep rupture failure of the vessel wall is calculated at 300 min.

4. SUMMARY AND CONCLUSIONS

During the past several years, improvements have been made to the BWR control blade/channel box model in SCDAP/RELAP5. Relocation of molten material is now represented in a more realistic manner by accounting for radial spreading into the fuel bundle and downward movement into the lower plenum. Also, the frequency of execution failures has been reduced by implementing timestep repetition logic for the control blade/channel box model.

SCDAP/RELAP5 assessment calculations have been performed using an input deck that represents a short-term station blackout accident sequence for the Browns Ferry Nuclear plant design. Now that the BWR control blade/channel box model is more mature, these Browns Ferry simulations execute to the time of lower vessel wall failure without water property failures or user intervention to manually reduce the timestep.

Although significant progress has been made to improve SCDAP/RELAP5 for BWR applications, several limitations still exist that will require further development efforts. New or improved models are needed to represent (1) degradation of upper plenum structures (development work is currently in progress), (2) accumulation of debris on and failure of BWR core plates, and (3) quenching of debris as it falls through water in the lower plenum.

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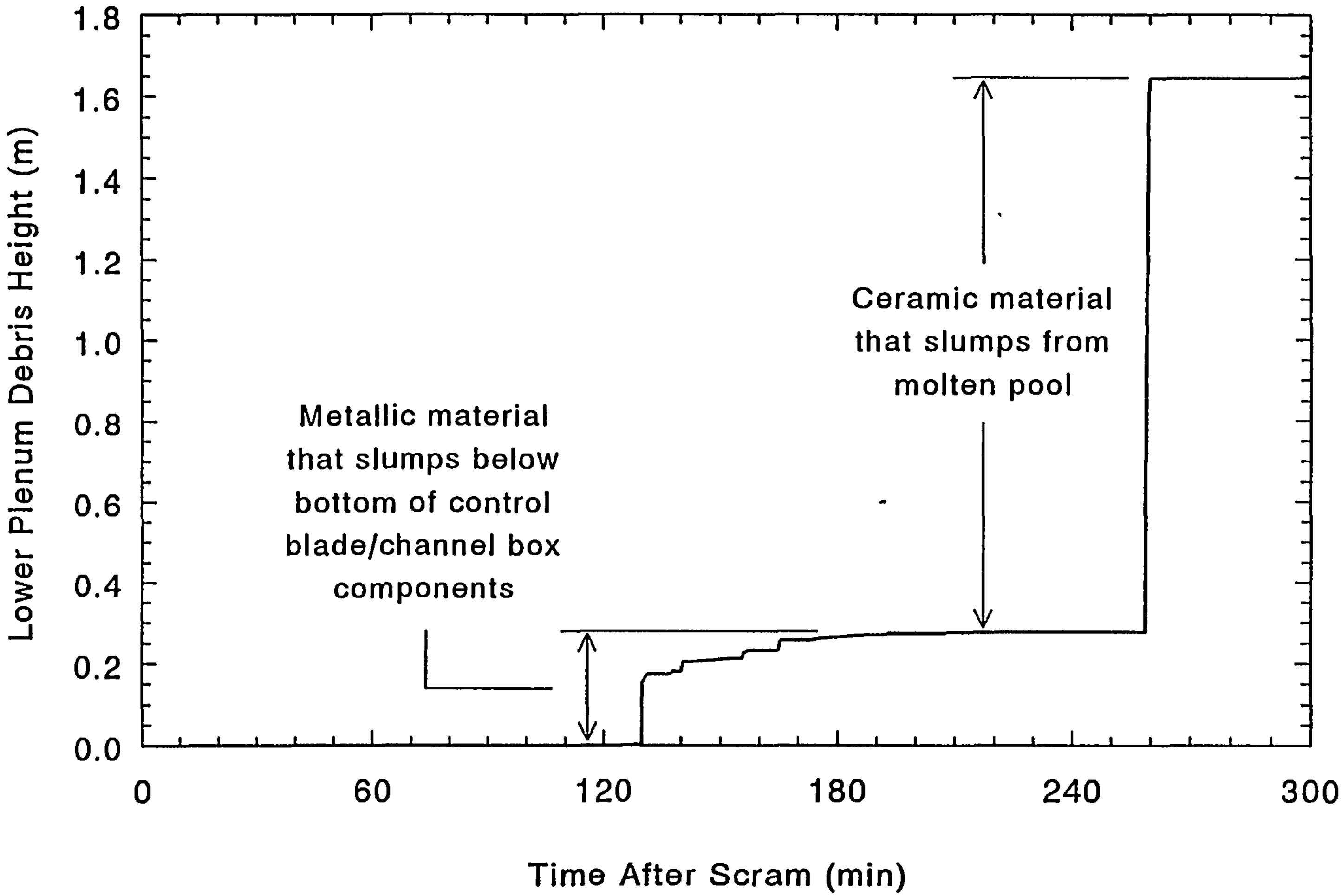

Figure 10. Calculated lower plenum debris height for Browns Ferry STSB accident.

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