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**LOSS-OF-COOLANT ACCIDENT ANALYSIS OF THE SAVANNAH
RIVER NEW PRODUCTION REACTOR DESIGN**

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ABSTRACT

This document contains the loss-of-coolant accident analysis of the representative design for the Savannah River heavy water new production reactor. Included in this document are descriptions of the primary system, reactor vessel, and loss-of-coolant accident computer input models, the results of the cold leg and hot leg loss-of-coolant accident analyses, and the results of sensitivity calculations for the cold leg loss-of-coolant accident.

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ABBREVIATIONS

AC	alternating current
C	Centigrade
CHF	critical heat flux
DBA	design basis accident
DC	direct current
delta-p	pressure difference
ECCS	emergency core cooling system
F	Fahrenheit
ft	feet
gpm	gallons per minute
hp	horsepower
HW-NPR	heavy water new production reactor
HWR	heavy water reactor
kPa	kilopascal
Li-Al	Lithium-Aluminum
LANL	Los Alamos National Laboratory
LOCA	loss-of-coolant accident
LOPA	loss-of-pumping accident
m	meters
MW	megawatts
NPR	new production reactor
P	pressure
psi	pounds per square inch
psia	pounds per square inch absolute
psid	pounds per square inch difference
rad	radians
s, sec	second
SNL	Sandia National Laboratories
SRS	Savannah River Site
TRAC	Transient Reactor Analysis Code
U-Al	Uranium-Aluminum
WSRC	Westinghouse Savannah River Company

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1. INTRODUCTION

Sandia National Laboratories (SNL), in support of Westinghouse Savannah River Company (WSRC), is evaluating computational tools for thermal-hydraulic analyses of the heavy water new production reactor (HW-NPR). The importance of this effort stems from a recent evaluation of accident analyses of the current generation of Savannah River Site (SRS) reactors, which found that, if operated at their design limit of 2500 MW, boiling and flow instability would occur in fuel channels under postulated large break loss-of-coolant accident (LOCA) conditions. Such complex thermal-hydraulic phenomena are extremely difficult to model with computer simulation to a high degree of confidence, and the uncertainty has led to a policy to reduce power to assure water-solid conditions are maintained for all postulated accidents.

The objective of this analysis is to provide expectations for behavior of the current HW-NPR system representative design under postulated large-break LOCA conditions. This analysis is a continuation of the study reported in Reference 1, in which the Savannah River HW-NPR representative design is modeled and in which a steady-state and loss-of-pumping accident analysis is performed. This LOCA study and the analysis in Reference 1 strive to provide assurance that all thermal-hydraulic phenomena expected in a HW-NPR during bounding design-basis accidents (DBAs) will be within the capabilities of computer simulation and that no previously unidentified safety issue will surface late in the design phase of the program. These studies are for the benefit of the heavy water reactor (HWR) development program and applicable NPR working groups.

The scope of this document includes modification and development of applicable computer models for simulating large-break LOCAs, performing scoping LOCA calculations for two postulated break locations in the primary system, and reporting results of the calculations. This report also includes a cold leg LOCA sensitivity study in which the effect of pressurizer/accumulator injection, variation of primary system pressure, and variation of reactor power are examined.

2. SYSTEM DESCRIPTION

In this section, the reactor vessel and primary cooling system of the heavy water new production reactor (HW-NPR) are described and the steps of the subsequent modeling process are detailed. The HW-NPR system configuration presented in this report includes improved safety design features intended to avoid or assist in mitigating severe accidents.

Section 2.1 provides a description of the assumptions made in the process of developing a HW-NPR representative design. Section 2.2 provides a description of the reactor vessel, fuel assemblies, and other major components comprising the primary cooling system.

2.1 Assumptions

Numerous assumptions were necessary to arrive at a representative design for the HW-NPR. Consideration was given to pertinent NPR design requirements and design limitations of the current SRS production reactor configuration in determining a representative design. The most influential design assumptions are given below, including a rationale where appropriate.

- a. Information from the Heavy Water Reactor Preliminary Design Report (ARD-88-10)², which describes the current generation of SRS reactors, applies in this report except where noted.
- b. The NPR vessel will have flow up through the core, unlike the current production reactor design. An upflow design aids the natural tendency of heated, more buoyant (lower density) fluid to seek a higher level, or rise within the core. A downflow design opposes this tendency.
- c. The NPR vessel will have an internal downcomer design, also unlike the current production reactor design. This allows a significant volume of fluid to reside upstream of the lower plenum, which will aid in keeping the core flooded during some accident scenarios.
- d. Each loop will have a pressure control and injection component that functions as both a pressurizer and an accumulator located downstream of the pump (between the pump and reactor vessel). This location enhances pressure stability in the core region since pressurizer/accumulator injection would be directly upstream of the reactor vessel. To avoid flow instability problems, maintaining system pressure in the core region immediately following an accident is critical. Decreasing pressure decreases the saturation temperature, which leads to voiding in the core and interruption of assembly coolant flow, or flow instability. An ECCS system will also provide injection to the vessel. However, ECCS injection, which has no effect on prevention of or recovery from flow instability in the first seconds of a LOCA, is not modeled in this study.

- e. The top of the heat exchangers will be below the top of the upper plenum. This is to ensure that no positive head exists at the top of the reactor vessel, which allows the top of the vessel to be removed for refueling or maintenance without spilling inventory.
- f. The elevation of the pump and pressurizer/accumulator will be approximately the same as that of the heat exchangers. This reduces the overall length of loop piping, brings the pump above lower floor levels that could flood and affect the pump's performance, minimizes the possibility of siphoning effects on lower elevation piping, and simplifies the overall loop piping.
- g. The fuel assemblies will be positioned as far below the top of the downcomer annulus as possible. In this configuration, since the water level in the downcomer always remains above that of the fuel assemblies, positive gravity head will tend to keep the fuel covered with water.
- h. The moderator system and the primary system are fully separate to minimize the effects of severe accidents. If the primary and moderator systems were coupled and if fuel were to melt and relocate, it could enter the moderator space and add to the reactivity.

2.2 System Components

In this section, the components that comprise the NW-NPR primary cooling system, including the fuel assemblies and reactor vessel, will be briefly described. Information on these components related to the thermal-hydraulic modeling process will be included.

2.2.1 Reactor Vessel

The reactor vessel will incorporate several design changes from the current production reactor design. The two most important changes are the upflow direction of coolant flow and the addition of a downcomer within the vessel. All dimensions of the vessel and pertinent elevations within the core are given in Figure 2-1.

2.2.2 Fuel Assemblies

The proposed NPR reactor vessel design contains 438 fuel assemblies. Assuming a total reactor maximum power of 2500 MW, each fuel assembly will have a power of approximately 5.7 MW. The fuel assemblies are modeled as a five tube design (2 Li-Al target tubes and 3 U-Al fuel tubes) rather than the four tube design (2 Li-Al target tubes and 2 U-Al fuel tubes) currently utilized in the production reactors. Figure 2-2 shows the cross-section of the proposed HW-NPR fuel assembly.

Notes:

Drawing is not to scale.

Dimensions are in meters.

Downcomer wall is 5.1 cm (2") thick.

Lower core support plate/lower shield

is 25.4 cm (10") thick.

Downcomer annulus is 10.2 cm (4").

Figure 2-1. Reactor Vessel Dimensions.

Fuel Assembly Dimensions (cm)

Flow Tube OD	- 13.051	Middle Fuel Tube OD Clad	- 8.235
ID	- 12.797	OD Core	- 8.082
Outer Target Tube OD Clad	- 12.141	ID Core	- 7.755
OD Core	- 11.989	ID Clad	- 7.602
ID Core	- 11.882	Inner Fuel Tube OD Clad	- 5.766
ID Clad	- 11.730	OD Core	- 5.613
Outer Fuel Tube OD Clad	- 10.566	ID Core	- 5.372
OD Core	- 10.414	ID Clad	- 5.220
ID Core	- 10.198	Inner Target Tube OD Clad	- 3.988
ID Clad	- 10.046	OD Core	- 3.835
		ID Core	- 2.934
		ID Clad	- 2.781

Figure 2-2. NPR MK-22 Assembly Cross-section and Dimensions.

2.2.3 Primary Cooling System

The primary cooling system functions to remove heat from the reactor vessel to a secondary system. The four loops in the primary system exit the vessel from the upper plenum and return to the top of the downcomer. The vessel and a characteristic loop of the representative design are shown in Figure 2-3. Each loop contains three heat exchangers, a centrifugal pump, and a pressurizer/accumulator. ECCS inlet lines also enter each primary loop on the cold leg near the reactor vessel, although ECCS injection will not be included in the primary system model for this study because it has no effect on flow instability during a large-break LOCA. All primary system hot and cold leg piping consists of 0.508 meter (20 inch) inner diameter lines. The moderator volume and flow are separate from the primary system. Dimensions of primary cooling system components are given in Figure 2-4. Other important primary system operational parameters are given in Table 2.1.

Table 2.1

Primary System Parameters	
Total Primary System Flow	8.96 m ³ /s (142,000 gpm)
Total Secondary Flow	11.51 m ³ /s (182,400 gpm)
Total Primary Water Volume	128.7 m ³ (34,000 gal)
Pressurizer/Accumulator Pressure	1207 kPa (175 psi)

Heat Exchangers

Each of the four primary cooling system loops is equipped with three counterflow heat exchangers in parallel to remove heat generated in the core. The secondary system coolant flows on the shell side to remove primary system heat. As previously mentioned, the elevation of the heat exchangers is slightly below that of the top of the reactor vessel to allow removing the vessel head without spilling primary coolant inventory. All pertinent heat exchanger parameters are given in Figure 2-5.

Pumps

The primary cooling system contains four pumps, one for each loop, to deliver sufficient coolant to the reactor vessel during operating conditions. Each pump's operational characteristics are given in Table 2.2.

Drawing not to scale

Figure 2-3. NPR Primary Cooling System Representative Design

Dimensions in meters.

Figure 2-4. Dimensions of NPR Primary Cooling System.

HEAT EXCHANGER MODEL

Operational Characteristics:

primary flow = $0.747 \text{ m}^3/\text{s}$ (11862 gpm)
secondary flow = $0.960 \text{ m}^3/\text{s}$ (15200 gpm)
inlet temp (primary) = 106.7°C (224.0°F)
outlet temp (primary) = 43.3°C (110.0°F)
inlet temp (secondary) = 22.2°C (72.0°F)
outlet temp (secondary) = 71.0°C (159.8°F)
pressure drop (primary) = 515 kPa (74.7 psi)

Figure 2-5. Heat Exchanger Parameters.

Table 2.2

NPR Pump Parameters	
Power	3150 hp
Rated Head	91.4 m (300 ft)
Rated Speed	104.7 rad/sec
Rated Flow	2.25 m ³ /s (35,587 gpm)
Pump Delta-P	888 kPa (128.8 psi)

Pressurizer/Accumulator

Each loop contains a pressurizer/accumulator component located between the pump and the reactor vessel on the cold leg. These components function to maintain pressure and volume control in the primary system by responding to pressure fluctuations. Each is capable of injecting 11.4 m³ (3000 gal) of water into the cold leg near the inlet to the vessel.

3. MODEL DESCRIPTION AND APPLICATION

The following sections describe the development of a complete TRAC steady-state model for the reactor vessel and primary cooling system and the transient models for the LOCA scenarios. Prominent features of the models are given first, followed by a description of the components included in each model.

3.1 Steady-State Model Features

In order to simulate the LOCAs, a steady-state model of the complete reactor vessel and primary cooling system was developed and run to provide initial conditions for the start of the transient. Given below are significant features of the HW-NPR system TRAC model:

- the reactor vessel is modeled as a three-dimensional component which represents the upper plenum, lower plenum, internal downcomer, and reactor core,
- the fuel and loop components are modeled as one-dimensional components,
- the fuel assemblies are equally divided among core regions,
- pump models utilize existing Savannah River homologous pump curves, and
- the three parallel heat exchangers in each loop are lumped into one equivalent heat exchanger model.

3.2 Steady-State Model Components

The complete steady-state model contains 129 components, which include 80 ROD (heat structure), 28 PIPE, 4 STGEN (steam generator), 4 PUMP, 4 TEE, 4 FILL (boundary condition component), 4 BREAK (boundary condition component) components and 1 VESSEL component. These components are described in detail in Reference 3. The division of the reactor vessel, fuel assemblies and pipe components into separate TRAC component models, and further into the appropriate number of cells, was completed with the intention of including enough detail to accurately model all important processes without creating a computationally cumbersome (and costly) model. In the following subsections, the modeling process for each of the individual components of the primary cooling system is described.

3.2.1 Reactor Vessel

The reactor vessel is modeled using the TRAC 3-dimensional VESSEL component. The 3-dimensional nature of this component is important to model the reactor flow patterns of many postulated transients and accidents. The reactor vessel model is divided in the radial, axial

and circumferential directions. Figure 3-1 shows the radial (5 segments) and axial (7 segments) cell division pattern for the vessel. Four 90-degree circumferential divisions are used to define quadrant numbers 1, 2, 3, and 4. These numbering conventions greatly simplify the process of assigning numbers to all components associated with the reactor vessel and fuel assemblies, as well as the coolant loops.

As seen in a cross-sectional view of the reactor vessel (Figure 3-2), the radial and circumferential segmentation creates 20 areas or vessel regions. The radius of each radial segment in the core region was chosen such that the innermost 16 areas (i.e., vessel regions 1 to 16 in Figure 3-2), representing the fuel assembly locations, are equal. These areas are numbered 01 to 04 from quadrant 1 to quadrant 4, respectively, at the cell center. This pattern continues outward to the outermost radial segment representing areas 17 to 20.

This segmentation of the reactor volume allows the modeler to conveniently assign certain reactor areas to segmented volumes. The lowest axial segment of the reactor (axial segment 1 in Fig. 3-1) represents the lower plenum, including the lower core support plate. The highest axial segment (axial segment 7 in Fig. 3-1) represents the upper plenum. The vessel outlet connections to each of the four loops are in the upper plenum. The internal downcomer is represented by the outermost radial segment of the reactor, areas 17 to 20 in Figure 3-2. Only axial segments 2 to 5 (see Figure 3-1) of this outermost radial segment model the downcomer. The four loop cold legs enter the vessel downcomer at axial segment 5. The active core region of the model is contained within the four innermost radial segments, vessel regions 01 to 16 in Fig. 3-2. The active core region includes only axial segments 2 through 5, since axial segments 1 and 7 represent the lower and upper plenums, respectively, and axial segment 6 represents the upper core shield region. These regions are graphically represented in Figure 3-3.

3.2.2 Fuel Assemblies

The fuel assemblies are modeled using the ROD and PIPE components of TRAC. Each of the 438 assemblies contains 3 U-Al fuel tubes and 2 Li-Al target tubes. The fuel tubes and target tubes are modeled using ROD (heat structure) components, which have a power producing option. However, this option was only utilized for the fuel tubes since the target tubes produce only negligible heat. The flow channels between the fuel and target tubes were modeled using PIPE components. As a simplification, only one PIPE component was used to represent all the coolant flow channels in each fuel assembly modeled. This is shown graphically in Figure 3-4. The flow areas for all channels within the assembly were lumped together to form one flow area in the PIPE component, with the exception of the channels inside the inner target tube and outside the outer target tube, which are very low flow areas. The equivalent hydraulic diameter for the PIPE component was calculated to be four times the flow area divided by the wetted perimeter.

Figure 3-1. VESSEL Model Radial and Axial Cell Division.

Figure 3-2. VESSEL Model Radial and Circumferential Cell Division.

Figure 3-3. NPR VESSEL Model Regions.

Figure 3-4. TRAC Fuel Assembly Components.

As described in the previous section, the active core region of the reactor vessel model is contained in axial segments 2 to 5 of the inner four radial segments, which represents 16 regions (see 'reactor core' in Figure 3-3). Each of these 16 vessel regions contains one representative assembly (i.e., one PIPE component and five ROD components), numbered by appending a '4' to the area number (01 to 16), resulting in the 16 vessel assemblies, 401 to 416, shown in Figure 3-5. The five ROD components in each assembly represent the three fuel tubes and two target tubes. The 438 actual fuel assemblies that reside within this region were divided equally among the 16 vessel regions that comprise the active core, since each of these regions has an equal cross-sectional area. This division resulted in 27.38 fuel assemblies for each of the 16 vessel regions. This modeling scheme results in 5 ROD components in each of the 16 vessel regions, or 80 ROD components in the vessel model. These ROD components are numbered 501 to 580 in ascending order with each vessel region (501 to 505 for region 01, 506 to 510 for region 02, etc.), as shown in Figure 3-6. The TRAC code contains effective multiplying factors to allow one fuel assembly model to represent 27.38 actual fuel assemblies.

The fuel and target tubes are nodalized as shown in Figure 3-7. Fifteen equally-spaced axial cells and three radial nodes in the tube wall were utilized. The flow channel in each assembly is modeled with 20 axial cells. Fifteen of these cells have the same length as that of the fuel and target tubes, representing the active core region. The remaining five axial cells represent sections of the flow channel above and below the fuel and target material. The cells of the fuel and target tubes (RODs) are coupled to the appropriate nodes of the flow channels (PIPE components) to model the core heat transfer. The axial power shape utilized in the model of the core region is taken from DPST-89-292⁴ and is shown in Figure 3-8. Given in this figure are the fuel tube power fractions used in the TRAC input files. Table 3.1 gives the properties of aluminum, lithium-aluminum, uranium-aluminum, and stainless steel used in the TRAC heat transfer model.⁵

The 16 fuel assembly models, consisting of PIPE and ROD components, are connected to the VESSEL component in the core region previously described (see Section 3.2.1). The outlet of the uppermost cell of each flow channel is connected to the lower side of the appropriate volume of the upper plenum, represented by the highest axial segment of the VESSEL model. The inlet of the lowermost cell of each flow channel is connected to the upper side of the appropriate volume of the lower plenum, represented by the bottom axial segment of the VESSEL model. The space between the 16 flow channels in the core region model represents the moderator coolant fluid. This fluid and the structural components of the reactor core are assumed to receive six percent of the total reactor power, or 150 MW, in the form of gamma ray heating.

Figure 3-5. Fuel Assembly Numbers Within VESSEL Model.

Figure 3-6. Heat Structure Numbers Within VESSEL Model.

AXIAL NODES

RADIAL NODES

Figure 3-7. Fuel Tube and Target Tube Nodalization.

FUEL TUBE POWER FRACTIONS:
 Inner - 0.154
 Middle - 0.3886
 Outer - 0.4574

Figure 3-8. Fuel Tube Axial Power Shape.

REF: DPST-89-292

Table 3.1

Metal Properties Used in TRAC (SI Units).

ALUMINUM PROPERTIES

temp (c)	temp (k)	density	specific ht	conductivity
0.000000e+00	2.731500e+02	2.709489e+03	8.799064e+02	2.376412e+02
1.000000e+02	3.731500e+02	2.688659e+03	9.127124e+02	2.380540e+02
2.000000e+02	4.731500e+02	2.667830e+03	9.523942e+02	2.368006e+02
3.000000e+02	5.731500e+02	2.647000e+03	9.989518e+02	2.338809e+02
4.000000e+02	6.731500e+02	2.626170e+03	1.052385e+03	2.292950e+02
5.000000e+02	7.731500e+02	2.605340e+03	1.112694e+03	2.230429e+02
6.000000e+02	8.731500e+02	2.584510e+03	1.179879e+03	2.151245e+02
7.000000e+02	9.731500e+02	2.563680e+03	1.253940e+03	2.055399e+02
8.000000e+02	1.073150e+03	2.542850e+03	1.334876e+03	1.942891e+02

LI-ALUMINUM PROPERTIES

temp (c)	temp (k)	density	specific ht	conductivity
0.000000e+00	2.731500e+02	2.705157e+03	8.962193e+02	2.409149e+02
1.000000e+02	3.731500e+02	2.686568e+03	9.306894e+02	2.409149e+02
2.000000e+02	4.731500e+02	2.668150e+03	9.704774e+02	2.409149e+02
3.000000e+02	5.731500e+02	2.649899e+03	1.015583e+03	2.409149e+02
4.000000e+02	6.731500e+02	2.631814e+03	1.066007e+03	2.409149e+02
5.000000e+02	7.731500e+02	2.613894e+03	1.121749e+03	2.409149e+02
6.000000e+02	8.731500e+02	2.596136e+03	1.182809e+03	2.409149e+02
7.000000e+02	9.731500e+02	2.578538e+03	1.249187e+03	2.409149e+02
8.000000e+02	1.073150e+03	2.561099e+03	1.320882e+03	2.409149e+02

U-ALUMINUM PROPERTIES

temp (c)	temp (k)	density	specific ht	conductivity
0.000000e+00	2.731500e+02	2.755831e+03	7.267006e+02	2.215775e+02
1.000000e+02	3.731500e+02	2.739353e+03	7.540605e+02	2.235314e+02
2.000000e+02	4.731500e+02	2.723007e+03	7.874366e+02	2.227456e+02
3.000000e+02	5.731500e+02	2.706791e+03	8.268287e+02	2.192202e+02
4.000000e+02	6.731500e+02	2.690703e+03	8.722371e+02	2.129551e+02
5.000000e+02	7.731500e+02	2.674742e+03	9.236615e+02	2.039504e+02
6.000000e+02	8.731500e+02	2.658908e+03	9.811021e+02	1.922060e+02
7.000000e+02	9.731500e+02	2.643198e+03	1.044559e+03	1.777220e+02
8.000000e+02	1.073150e+03	2.627612e+03	1.114032e+03	1.604983e+02

STAINLESS STEEL (304L) PROPERTIES

temp (c)	temp (k)	density	specific ht	conductivity
0.000000e+00	2.731500e+02	8.027523e+03	3.829839e+02	1.477054e+01
1.000000e+02	3.731500e+02	7.982849e+03	3.977586e+02	1.628763e+01
2.000000e+02	4.731500e+02	7.938176e+03	4.125333e+02	1.773059e+01
3.000000e+02	5.731500e+02	7.893502e+03	4.273081e+02	1.909943e+01
4.000000e+02	6.731500e+02	7.848828e+03	4.420828e+02	2.039414e+01
5.000000e+02	7.731500e+02	7.804154e+03	4.568576e+02	2.161472e+01
6.000000e+02	8.731500e+02	7.759481e+03	4.716323e+02	2.276118e+01
7.000000e+02	9.731500e+02	7.714807e+03	4.864071e+02	2.383352e+01
8.000000e+02	1.073150e+03	7.670133e+03	5.011818e+02	2.483172e+01

REF: DPST-87-692

3.2.3 Primary Cooling System

The primary cooling system model consists of, for each loop, TRAC components for the pump (PUMP), heat exchangers (STGEN), pressurizer/accumulator (BREAK or ACCUM), and loop piping (PIPE). These components are shown for one coolant loop in Figure 3-9. The coordinates for the component positions with respect to the middle of the hot leg outlet from the reactor vessel, which give both length and elevation, are given in Figure 3-10. Components are numbered in each loop using two digit identifiers, the first number of which is the loop number in which the component resides (1X for loop 1, 2X for loop 2, etc.). The following subsections describe the modeling of each of these components.

Heat Exchanger

As previously mentioned, each of the four coolant loops has three heat exchangers. The three heat exchangers per loop were lumped into one per loop for a total of four heat exchangers in the primary system, modeled as one-dimensional STGEN components. The STGEN component is formed by representing the primary and secondary flow channels as PIPE components and representing the heat exchanger structure as a ROD (heat structure) component with no power production. The flow channels are modeled with twelve cells for both the primary and secondary sides. The ROD component contains the same mesh cell spacing and the same number of cells as the PIPE component to facilitate thermal-hydraulic coupling of the mesh cells. The active heat transfer region of the STGEN component is represented by the middle ten cells. The heat transfer in these components is one-dimensional, between the ROD and PIPE mesh cells (i.e., no axial conduction). The STGEN components are numbered 12, 22, 32 and 42 for loops 1 to 4, respectively. Figure 3-11 shows the heat exchanger component.

Pump

Each pump in the primary cooling system is modeled using the TRAC one-dimensional PUMP component. The pump model must consist of at least two cells, with the pressure rise applied between the first and second cells. The pump model in this analysis contains four cells. Homologous pump head curves for current Savannah River production reactors are used for the NPR pumps in this model. The version of TRAC used in this analysis contains no pump cavitation model. The PUMP components are numbered 14, 24, 34 and 44 for loops 1 to 4, respectively. Figure 3-12 shows the pump model.

Pressurizer/Accumulator

Each primary coolant loop will have a component to perform both the function of a pressurizer and an accumulator. This component will

Figure 3-9. TRAC Components in Primary Cooling Loop.

Figure 3-10. Component (x,y) Positions.

Figure 3-11. Heat Exchanger Noding.

Figure 3-12. Pump Noding.

regulate the primary system pressure to a constant value and provide emergency coolant injection. It will be located on each loop between the pump and the cold leg inlet to the reactor vessel. The pressurizer is modeled using a TEE and a BREAK component. The TEE is an eight-cell, one-dimensional pipe with a two-cell vertical branch, or side pipe. The side pipe has a BREAK component connected to it to model the pressurizer action. The BREAK component is a constant-pressure device that is set at 175 psia to compensate for pressure fluctuations. The numbering for the TEE components is 15, 25, 35 and 45 for loops 1 to 4, respectively, and for the BREAK components is 150, 250, 350 and 450 for loops 1 to 4, respectively.

Loop Piping

All piping for the primary cooling system is 0.508 meters (20 inches) in inner diameter and follows the same numbering convention as for other loop components. Each loop has a seven-cell hot leg outlet pipe from the reactor and a twelve-cell cold leg pipe following the heat exchanger. For loops one to four, the hot legs are numbered 11, 21, 31 and 41, and the cold leg piping sections (between the heat exchanger and pump) 13, 23, 33 and 43, respectively.

3.3 Transient Model Features

In order to simulate a LOCA using TRAC, a run must be executed, using the model described in Section 3.2, for a sufficient length of time to allow steady-state conditions to exist in the primary system. The final state of this run is used for initial conditions for the transient, or LOCA, case. Steady-state testing of the models used in this analysis is described in Reference 1. The reactor scram, or decay heat, curve, utilized in both LOCA models, is given in Figure 3-13. This curve is input to the model as a control function and the scram is assumed to occur at the instant of the break. The following sections describe the development of the cold leg and hot leg LOCA input models.

3.3.1 Cold Leg LOCA

The break in the primary system pipe for this accident was simulated to occur on the cold leg of loop 1, near the inlet to the vessel internal downcomer, as shown in Figure 3-14. The LOCA input model contains BREAK components on both openings in the broken pipe, which are set to atmospheric pressure to allow unrestricted flow out of the opening. The primary and secondary pumps remain fully operational throughout the accident.

3.3.2 Hot Leg LOCA

The break in the primary system pipe for this accident was simulated to occur on the hot leg pipe of loop 1, near the outlet of the upper plenum, as shown in Figure 3-15. BREAK components were applied to the

Figure 3-13. Reactor Power Decay Curve.

Figure 3-14. Cold Leg Break Location.

Figure 3-15. Hot Leg Break Location.

pipe openings as in the previous case. The downstream pipe opening, however, was initialized with air since the downstream pump suction will rapidly draw air into the system from this opening. Also as before, all primary and secondary pumps remain operational during this accident.

3.4 Application

SNL currently uses Version 5.3 of TRAC-PF1/MOD2, released by Los Alamos National Laboratory (LANL). Manuals, assessment reports, and other documentation for this version of the code are currently being developed by LANL and will be issued shortly. TRAC has been extensively applied to the analysis of anticipated and design basis accidents of nuclear power plants with much success for many years. The code has also been benchmarked against data from numerous experimental facilities involving a wide range of conditions and scale lengths. No modification has been made to the TRAC version received from LANL for use in this study.

The TRAC version released to SNL is currently undergoing assessment by LANL. SNL ran test problems at LANL using their validated release version of TRAC and verified those results with results from the same test problems ran at SNL. In this manner, SNL ensured the code was copied to the SNL site correctly.

Two areas of concern regarding TRAC modeling are worthy of mention. The current version of TRAC is known to overpredict the onset of critical heat flux (CHF) in the fuel assemblies (i.e., predicts CHF more often than it would occur). Future study and modification to the CHF correlations in TRAC should give the user more confidence in predicting CHF. However, since TRAC is conservative in predicting CHF, the user can be confident CHF is unlikely to occur if a TRAC simulation did not predict it. Another potential limitation of TRAC concerns pump cavitation, for which TRAC contains no model.

4. RESULTS

The steady-state and transient results of the cold leg and hot leg LOCA analyses, including those for the cold leg sensitivity analysis, are presented in this section. These results include plots of variables of interest for each LOCA calculation and a discussion of the progression of each accident in terms of primary system parameters. Section 4.1 presents the system parameters at steady-state conditions and Sections 4.2 and 4.3 present LOCA calculation information for the cold leg and hot leg, respectively. Section 4.4 discusses the cold leg sensitivity analysis.

4.1 Steady-State

Prior to executing the LOCA calculations, a steady-state model was run to generate operating conditions from which to start the LOCA. Table 4.1 gives the values of primary system parameters at steady-state conditions.

Table 4.1

STEADY-STATE SYSTEM PARAMETERS

TOTAL REACTOR POWER	2500 MW (2350 MW to primary coolant)
AVERAGE ASSEMBLY FLOW RATE	0.021 m ³ /s (325 gpm)
PRIMARY SYSTEM FLOW RATE	8.99 m ³ /s (142,406 gpm)
SECONDARY SYSTEM FLOW RATE	11.5 m ³ /s (182,403 gpm)
PRESSURIZER PRESSURE	1207 kPa (175 psi)
UPPER PLENUM PRESSURE	938 kPa (136 psi)
DELTA-P ACROSS PRIMARY PUMPS	889 kPa (129 psi)
DELTA-P ACROSS HEAT EXCHANGER	517 kPa (75 psi)
HEAT EXCHANGER PRIMARY INLET TEMP	107°C (224°F)
HEAT EXCHANGER PRIMARY OUTLET TEMP	44°C (111°F)
HEAT EXCHANGER SECONDARY INLET TEMP	22°C (72°F)
HEAT EXCHANGER SECONDARY OUTLET TEMP	71°C (160°F)

4.2 Cold Leg LOCA

The first seconds of this LOCA are characterized by rapid system depressurization and high coolant flow out of both break openings. The close proximity of this break location to the lower plenum causes the lower plenum pressure to decrease rapidly, and hence, decrease the driving force for flow through the assemblies. However, sufficient pressure difference is maintained to avoid flow instability problems. The lower plenum pressure and assembly pressure drop are given in Figures 4-1 and 4-2. The mass flow rates out of both break locations reach a maximum in less than one second and slowly decrease throughout the remainder of the accident. The vessel-side and pump-side break mass flows are shown in Figures 4-3 and 4-4.

The pressure spike, observed at the end of the calculation in Figures 4-1 and 4-2 (also noticeable in later saturation temperature plots), is caused by the emptying of the pressurizer/accumulators in loops 2, 3, and 4 approximately 8 seconds into the accident. The loop 1 pressurizer/accumulator has no effect on the system pressure due to the location of the break. The surge of air entering the system after the last volume of water is forced out of each pressurizer/accumulator creates the pressure spike evident in the figures.

Observed throughout the accident was steady injection from the pressurizer/accumulators, uniform reduction of coolant flow through the fuel assemblies, and steady or decreasing fuel and fluid temperatures. The loop 1 pressurizer/accumulator (3000 gallons each) discharged in 7 seconds, as shown in Figure 4-5. The assembly flow rate, following a sharp decline in the first two seconds, leveled off at approximately half of the steady-state value, as seen in Figure 4-6. This decreased fuel cooling ability is reflected in the temperatures in the system. The sharply reduced coolant flow in the first second of the LOCA allowed the fuel temperature to rise before gradually decreasing, as seen in Figure 4-7. The coolant fluid at the outlet of the assemblies remained subcooled through the LOCA, although the degree of subcooling was very small at one point, as seen in Figure 4-8. Results indicated that CHF was not reached and significant voiding did not occur at any point in the fuel assemblies, although nonidealities could produce localized heat flux variations which would affect fuel assembly subcooling. Since the CHF correlations in TRAC are conservative, no problem is anticipated. The coolant in the hot leg of loop 1 reached saturation for a portion of the accident (see Figure 4-9) due to the reduced pressure (and hence, reduced saturation temperature) in loop 1 piping caused by greater pump suction (i.e., the pump pushes only against atmospheric pressure).

4.3 Hot Leg LOCA

The system response to this LOCA is very different from that of the cold leg LOCA. The break location, near the upper plenum, causes the pressure drop across the assemblies to increase for the first several

Figure 4-1. Lower Plenum Pressure.

Figure 4-2. Assembly Delta-P.

Figure 4-3. Vessel-Side Break Mass Flow Rate.

Figure 4-4. Pump-Side Break Mass Flow Rate.

Figure 4-5. Loop 1 Accumulator Water Level.

Figure 4-6. Assembly 401 Outlet Mass Flow Rate.

cold leg break
175 psi

Figure 4-7. Assembly 401 Maximum Inner Fuel Temperature.

Figure 4-8. Assembly 401 Outlet Fluid Temperature.

Figure 4-9. Loop 1 Hot Leg Fluid Temperature.

seconds, due to the abrupt pressure loss in the upper plenum. Figures 4-10 and 4-11 show the upper plenum pressure and the assembly pressure drop. Significant inventory is lost only out of the vessel-side break opening. A small amount of coolant is lost from the heat exchanger-side opening before it is pumped dry. Flow rates for both break openings are given in Figures 4-12 and 4-13.

The enhanced lower plenum to upper plenum pressure difference in the initial seconds of this LOCA created a surge of increased coolant flow in the assemblies, shown in Figure 4-14. The flow then gradually decreases for the remainder of the accident. The initial surge of coolant in the assemblies is reflected in the fuel and coolant fluid temperature profiles, given in Figures 4-15 and 4-16. Both profiles show a rapid cooling trend from the start of the accident. A similar trend is seen in the loop coolant flow, shown in Figure 4-17 for loop 2. The pressurizer/accumulators uniformly discharged during the accident (see Figure 4-18) in 15 seconds, compared to an 8 second discharge for the cold leg LOCA. This is due to higher pressure in the lower plenum for the hot leg break.

The oscillations observed in the pressure and mass flow curves for the hot leg LOCA, which were not observed in the cold leg LOCA results, are caused by the action of the pumps. The pressure throughout the primary system decreases immediately following the accident initiation, most markedly in the upper plenum and in the loops due to the close proximity of the break. Pump suction in the cold leg piping upstream from the pumps creates saturated conditions. At this point, the pumps experience a liquid-vapor mixture. The oscillatory behavior observed in the figures results from rapid pressure fluctuations caused by attempts to pump a 2-phase mixture. As it experiences increasing amounts of vapor, the pump's performance degrades and less suction is created which allows some vapor to collapse in the liquid. The pump performance then increases, due to less vapor at the pump inlet, which creates greater suction (and lower pressure) and more vapor. Increased vapor again decreases the pump performance, and the cycle begins to repeat. This pattern of continued void, pressure, and mass flow oscillation continues throughout the transient. A pump cavitation model would lessen this oscillatory effect. Some uncertainty exists here as to when severe pump degradation begins and to what extent the pumps assist in core cooling during the first critical seconds of the LOCA. It is believed that the effect of the accumulator injection is primarily responsible for maintaining successful core cooling. However, further study is required to resolve the pump cavitation issue and its effect on core flow.

Problems were encountered for both LOCA calculations with pressurizer/accumulator discharge. The high rate of flow combined with the abrupt area change at the component exit produced some computational difficulty. The TRAC area-change model was applied to all component junctions of this type, which significantly reduced the difficulty.

Figure 4-10. Upper Plenum Pressure.

hot leg break
175 psi

Figure 4-11. Assembly Delta-P.

hot leg break
175 psi

Figure 4-12. Vessel-Side Break Mass Flow Rate.

hot leg break
175 psi

Figure 4-13. Heat Exchanger-Side Break Mass Flow Rate.

hot leg break
175 psi

Figure 4-14. Assembly 401 Mass Flow Rate.

hot leg break
175 psi

Figure 4-15. Assembly 401 Maximum Middle Fuel Temperature.

hot leg break
175 psi

Figure 4-16. Assembly 401 Outlet Fluid Temperature.

hot leg break
175 psi

Figure 4-17. Loop 2 Hot Leg Fluid Temperature.

hot leg break
175 psi

Figure 4-18. Loop 2 Accumulator Water Level.

The cold leg LOCA was observed to give more severe results than the hot leg LOCA in terms of margin to saturated conditions in the core. This is primarily due to the location of the break and its effect on the coolant flow in the core and through the loops. The cold leg break location causes the lower plenum pressure to decrease rapidly, which decreases the assembly pressure difference, and hence, the mass flow through the assemblies. The hot leg break location has a similar effect on the upper plenum, but this reduction in pressure causes the assembly pressure difference to increase, and thus increase flow through the assemblies for the first few seconds. Also, coolant is lost from both cold leg break openings, while only one opening loses coolant from the hot leg break.

4.4 Cold Leg LOCA Sensitivity

The results of the sensitivity cases on the cold leg LOCA are provided below.

4.4.1 Pressurizer/Accumulator

The pressurizer/accumulator provides pressure control and volume control in the primary system. This component is critical to avoiding flow instability in the fuel assembly flow channels during a large-break LOCA. The cold leg LOCA model was run with and without the pressurizer/accumulator components to observe their effect on the primary system response. The results of this exercise, given in Figure 4-19, dramatically show the effect of this component. Without the pressurizer/accumulators, the pressure difference across the assemblies is not maintained and the flow decreases, significant voiding begins, and fuel melt conditions occur.

4.4.2 Primary System Pressure

To determine the effect of primary system pressure, the cold leg LOCA model was run at the higher system pressure of 250 psi and the results were compared to those from the 175 psi cold leg LOCA case. The higher pressure case clearly responded more favorably to the LOCA conditions. In both cases (175 and 250 psi), the primary system is opened to atmospheric pressure. The higher pressure case has the higher difference between system pressure and atmospheric pressure, and hence, higher flows throughout the system. This is seen comparatively in Figures 4-20 to 4-22, which show the assembly delta-p, assembly mass flow rate, and loop 2 mass flow rate. The higher mass flow rates in the primary system decrease the fuel and fluid temperatures, as shown in Figures 4-23 to 4-26. Figure 4-27 shows the degree of subcooling in the assemblies and illustrates the additional cooling ability of the higher pressure case. The disadvantages of the higher pressure case are the higher flow rate out of the break openings and the shorter discharge time of the pressurizer/accumulators, which could impact timing of the ECCS

Figure 4-19. Cold Leg Pressurizer/Accumulator Sensitivity - Assembly Delta-P.

Figure 4-20. Cold Leg Primary Pressure Sensitivity - Assembly Delta-P.

Figure 4-21. Cold Leg Primary Pressure Sensitivity - Assembly 401 Mass Flow Rate.

Figure 4-22. Cold Leg Primary Pressure Sensitivity - Loop 2 Mass Flow Rate.

Figure 4-23. Cold Leg Primary Pressure Sensitivity - Assembly 401 Maximum Inner Fuel Temperature.

Figure 4-24. Cold Leg Primary Pressure Sensitivity - Assembly 401 Maximum Middle Fuel Temperature.

Figure 4-25. Cold Leg Primary Pressure Sensitivity - Assembly 401 Maximum Outer Fuel Temperature.

Figure 4-26. Cold Leg Primary Pressure Sensitivity - Assembly 401 Effluent Temperature.

Figure 4-27. Cold Leg Primary Pressure Sensitivity - Assembly 401 Effluent Subcooling.

injection. The higher pump-side and vessel-side break flow rates for the higher pressure case are seen in Figures 4-28 and 4-29. The rapid flow rate decline at the end of both curves in Figure 4-28 is the effect of emptying the pressurizer/accumulators, which occurs earlier for the higher pressure case.

4.4.3 Assembly Power

The influence of the assembly power level on the transition to flow instability in the fuel channels was examined by performing a TRAC sensitivity analysis of a single fuel assembly model. The pressure responses of the upper and lower plenums for the cold leg LOCA were input as boundary conditions to a single fuel assembly. The power level was increased incrementally from the rated full power (5.71 MW per assembly) and the fluid behavior in the fuel channel was observed. The results of this exercise are shown in Figure 4-30. Between the power levels of 5.71 and 6.28 MW per assembly, no flow instability was observed. Between power levels of 6.28 and 7.13 MW per assembly, flow instability with recovery to stable assembly flow was observed. At power levels at and above 7.13 MW per assembly, flow instability without recovery was observed.

Figure 4-28. Cold Leg Primary Pressure Sensitivity - Pump-side Break Mass Flow Rate.

cold leg break

Figure 4-29. Cold Leg Primary Pressure Sensitivity - Vessel-side Break Mass Flow Rate.

Figure 4-30. Cold Leg Assembly Power Sensitivity.

5. CONCLUSIONS

The results of the LOCA analyses reported in this document indicate that the HW-NPR representative design can safely maintain coolant flow through the assemblies during a large-break LOCA occurring on the cold or hot leg of one loop of the primary system. The TRAC code exhibits full capability to simulate such accidents.

Although flow instability was avoided in both the cold leg and the hot leg LOCAs, the degree of assembly fluid subcooling for the cold leg LOCA was significantly less. This is because the upper plenum pressure for the cold leg LOCA decreases rapidly, due to the close proximity to the break location, which causes the pressure difference between the upper and lower plenums, and hence, the mass flow rates, to decrease more rapidly. Therefore, the assembly fluid for the cold leg case is hotter.

Uncertainties in this analysis associated with TRAC include pump cavitation and CHF calculation, although the CHF correlations in TRAC are known to be conservative. The hot leg LOCA calculation exhibits pumping of liquid-vapor mixtures in all loops. Further study is required to fully investigate and resolve these uncertainties.

The sensitivity cases involving the pressurizer/accumulator components, the primary system pressure, and the assembly power gave useful results. The pressure and volume control action of the pressurizer/accumulator components was shown to prevent flow instability. The results of varying the primary system pressure show that more safety margin exists during a large-break cold leg LOCA at higher primary system pressures. However, the primary system at higher pressure loses inventory more rapidly and the pressurizer/accumulator components discharge faster. The power level sensitivity indicated that flow instability with recovery occurs at assembly power levels between 10 and 25 percent above the rated power of 5.71 MW per assembly. Power levels over 25 percent above the rated power resulted in flow instability and fuel melt conditions.

6. REFERENCES

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