ATTACHMENT 17

Browns Ferry Nuclear Plant (BFN)
Unit 1

Technical Specifications (TS) Change 473

AREVA Fuel Transition

LOCA Break Spectrum Analysis Report
(Non-Proprietary)

Attached is the non-proprietary version of the AREVA LOCA break spectrum analysis report for 105% OLTP conditions.

ANP-2908(NP)
Revision 0

Browns Ferry Units 1, 2, and 3 105% OLTP LOCA Break Spectrum Analysis

March 2010

AREVA NP Inc.

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Nature of Changes

Item	Page	Description and Justification
1.	All	This is the initial release.

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Nomenclature

ADS	automatic depressurization system
ADSVOOS	ADS valve out of service
ANS	American Nuclear Society
BOL	beginning of life
BWR	boiling water reactor
CFR	Code of Federal Regulations
CHF	critical heat flux
CLTP	current licensed thermal power (3458 MWt)
CMWR	core average metal-water reaction
DEG	double-ended guillotine
DG	diesel generator
ECCS	emergency core cooling system
EOB	end of blowdown
FFWTR	final feedwater temperature reduction
FHOOS	feedwater heaters out of service
FSAR	Final Safety Analysis Report
HPCI	high-pressure coolant injection
ICF	increased core flow
LOCA	loss-of-coolant accident
LPCI	low-pressure coolant injection
LPCS	low-pressure core spray
MAPLHGR	maximum average planar linear heat generation rate
MCPR	minimum critical power ratio
MELLLA	maximum extended load line limit analysis
MSIV	main steam isolation valve
MWR	metal-water reaction
NRC	Nuclear Regulatory Commission, U.S.
OLTP	original licensed thermal power
PCT	peak cladding temperature

Nomenclature *(Continued)*

RCIC	reactor core isolation cooling
RDIV	recirculation discharge isolation valve
RWCU	reactor water cleanup
SF-ADS	single failure of ADS
SF-ADS|IL	single failure of ADS, initiation logic
SF-ADS|SV	single failure of ADS, single valve
SF-BATT	single failure of battery (DC) power
SF-BATT|BA	single failure of battery (DC) power, board A
SF-BATT|BB	single failure of battery (DC) power, board B
SF-BATT|BC	single failure of battery (DC) power, board C
SF-DGEN	single failure of a diesel generator
SF-HPCI	single failure of the HPCI system
SF-LOCA	single failure of opposite unit false LOCA signal
SF-LPCI	single failure of a LPCI valve
SLO	single-loop operation
TLO	two-loop operation

1.0 Introduction

The results of a loss-of-coolant accident (LOCA) break spectrum analysis for Browns Ferry Units 1, 2, and 3 are documented in this report. The purpose of the break spectrum analysis is to identify the parameters that result in the highest calculated peak cladding temperature (PCT) during a postulated LOCA. The LOCA parameters addressed in this report include the following:

- Break location
- Break type (double-ended guillotine (DEG) or split)
- Break size
- Limiting emergency core cooling system (ECCS) single failure
- Axial power shape (top- or mid-peaked)

The analyses documented in this report were performed with LOCA Evaluation Models developed by AREVA NP* and approved for reactor licensing analyses by the Nuclear Regulatory Commission, U.S. (NRC). The models and computer codes used by AREVA for LOCA analyses are collectively referred to as the EXEM BWR-2000 Evaluation Model (References 1-4). The EXEM BWR-2000 Evaluation Model and NRC approval are documented in Reference 1. A summary description of the LOCA analysis methodology is provided in Section 4.0. The calculations described in this report were performed in conformance with 10 CFR 50 Appendix K requirements and satisfy the event acceptance criteria identified in 10 CFR 50.46.

The break spectrum analyses documented in this report were performed for a core composed entirely of ATRIUM™-10† fuel at beginning-of-life (BOL) conditions. Calculations assumed an initial core power of 102% of 3458 MWt, providing an analysis licensing basis power of 3527 MWt. The 2.0% increase reflects the maximum uncertainty in monitoring reactor power, as per NRC requirements. 3458 MWt corresponds to 105% of the original licensed thermal power (OLTP) and is referred to as the current licensed thermal power (CLTP). The limiting assembly in the core was assumed to be at a maximum average planar heat generation rate (MAPLHGR) limit of 12.5 kW/ft. The analyses assumed a generic ATRIUM-10 neutronic design. Other initial conditions used in the analyses are described in Section 4.0.

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* AREVA NP Inc. is an AREVA and Siemens company.

† ATRIUM is a trademark of AREVA NP.

This report identifies the limiting LOCA break characteristics (location, type, size, single failure, and axial power shape) that will be used in subsequent analyses to determine the LOCA-ECCS MAPLHGR limit versus exposure for ATRIUM-10 fuel used at Browns Ferry Units 1, 2, and 3. The value of PCT calculated for any given set of break characteristics is dependent on exposure and local fuel rod power peaking. Therefore, heatup analyses are performed to determine the PCT versus exposure for each ATRIUM-10 nuclear design in the core. The heatup analyses are performed each cycle using the limiting boundary conditions determined in the break spectrum analysis. The maximum PCT versus exposure from the heatup analyses are documented in the MAPLHGR limit report.

[ ]

The operating domain of the power/flow map of Reference 6 is applicable for the ATRIUM-10.

[ ] This report also presents results for single-loop operation (SLO).

2.0 Summary of Results

Based on analyses presented in this report, the limiting break characteristics are identified below.

Limiting LOCA Break Characteristics	
Location	recirculation discharge pipe
Type / size	split / 0.25 ft^2
Single failure	battery (DC) power, board B
Axial power shape	top-peaked
Initial State	[ ]

The SLO LOCA analyses support operation with an ATRIUM-10 MAPLHGR multiplier of 0.85 applied to the normal two-loop operation MAPLHGR limit.

A more detailed discussion of results is provided in Sections 6.0 – 8.0.

[

] The break characteristics identified in this report can be used in subsequent heatup analyses to determine the ATRIUM-10 MAPLHGR limit appropriate for a full core of ATRIUM-10 as well as transition cores.

Available ADS valves are presented in Tables 5.1 and 5.2. No additional valves are assumed out-of-service (ADSVOOS). [

]

[

] The conclusions of this report are applicable for operation with [ ], FHOOS, FFWTR, and SLO.

While the fuel rod temperatures in the limiting plane of the hot channel during a LOCA are dependent on exposure, the factors that determine the limiting break characteristics are primarily associated with the reactor system and are not dependent on fuel-exposure characteristics. Fuel parameters that are dependent on exposure (e.g., stored energy, local peaking) have an insignificant effect on the reactor system response during a LOCA. The limiting break characteristics determined using BOL fuel conditions are applicable for exposed fuel. Fuel exposure effects are addressed in heatup analyses performed to determine or verify MAPLHGR limits versus exposure for each ATRIUM-10 fuel design.

The break spectrum analysis was performed using the NRC-approved AREVA EXEM BWR-2000 LOCA methodology. All SER restrictions and ranges of applicability for the EXEM BWR-2000 methodology were reviewed prior to final documentation of the LOCA analysis to ensure compliance with NRC requirements and methodology limitations.

3.0 LOCA Description

3.1 *Accident Description*

The LOCA is described in the Code of Federal Regulations 10 CFR 50.46 as a hypothetical accident that results in a loss of reactor coolant from breaks in reactor coolant pressure boundary piping up to and including a break equivalent in size to a double-ended rupture of the largest pipe in the reactor coolant system. There is not a specifically identified cause that results in the pipe break. However, for the purpose of identifying a design basis accident, the pipe break is postulated to occur inside the primary containment before the first isolation valve.

For a BWR, a LOCA may occur over a wide spectrum of break locations and sizes. Responses to the break vary significantly over the break spectrum. The largest possible break is a double-ended rupture of a recirculation pipe; however, this is not necessarily the most severe challenge to the emergency core cooling system (ECCS). A double-ended rupture of a main steam line causes the most rapid primary system depressurization, but because of other phenomena, steam line breaks are seldom limiting with respect to the criteria of 10 CFR 50.46. Special analysis considerations are required when the break is postulated to occur in a pipe that is used as the injection path for an ECCS (e.g. core spray line). Although these breaks are relatively small, their existence disables the function of an ECCS. In addition to break location dependence, different break sizes in the same pipe produce quite different event responses, and the largest break area is not necessarily the most severe challenge to the event acceptance criteria. Because of these complexities, an analysis covering the full range of break sizes and locations is required.

Regardless of the initiating break characteristics, the event response is conveniently separated into three phases: the blowdown phase, the refill phase, and the reflood phase. The relative duration of each phase is strongly dependent upon the break size and location. The last two phases are often combined and will be discussed together in this report.

During the blowdown phase of a LOCA, there is a net loss of coolant inventory, an increase in fuel cladding temperature due to core flow degradation, and for the larger breaks, the core becomes fully or partially uncovered. There is a rapid decrease in pressure during the blowdown phase. During the early phase of the depressurization, the exiting coolant provides core cooling. Low-pressure core spray (LPCS) also provides some heat removal. The blowdown phase is defined to end when LPCS reaches rated flow.

In the refill phase of a LOCA, the ECCS is functioning and there is a net increase of coolant inventory. During this phase the core sprays provide core cooling and, along with low-pressure and high-pressure coolant injection (LPCI and HPCI), supply liquid to refill the lower portion of the reactor vessel. In general, the core heat transfer to the coolant is less than the fuel decay heat rate and the fuel cladding temperature continues to increase during the refill phase.

In the reflood phase, the coolant inventory has increased to the point where the mixture level reenters the core region. During the core reflood phase, cooling is provided above the mixture level by entrained reflood liquid and below the mixture level by pool boiling. Sufficient coolant eventually reaches the core hot node and the fuel cladding temperature decreases.

3.2 *Acceptance Criteria*

A LOCA is a potentially limiting event that may place constraints on fuel design, local power peaking, and in some cases, acceptable core power level. During a LOCA, the normal transfer of heat from the fuel to the coolant is disrupted. As the liquid inventory in the reactor decreases, the decay heat and stored energy of the fuel cause a heatup of the undercooled fuel assembly. In order to limit the amount of heat that can contribute to the heatup of the fuel assembly during a LOCA, an operating limit on the MAPLHGR is applied to each fuel assembly in the core.

The Code of Federal Regulations prescribes specific acceptance criteria (10 CFR 50.46) for a LOCA event as well as specific requirements and acceptable features for Evaluation Models (10 CFR 50 Appendix K). The conformance of the EXEM BWR-2000 LOCA Evaluation Models to Appendix K is described in Reference 1. The ECCS must be designed such that the plant response to a LOCA meets the following acceptance criteria specified in 10 CFR 50.46:

- The calculated maximum fuel element cladding temperature shall not exceed 2200°F.
- The calculated local oxidation of the cladding shall nowhere exceed 0.17 times the local cladding thickness.
- The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, except the cladding surrounding the plenum volume, were to react.
- Calculated changes in core geometry shall be such that the core remains amenable to cooling.
- After any calculated successful operation of the ECCS, the calculated core temperature shall be maintained for the extended period of time required by the long-lived radioactivity remaining in the core.

These criteria are commonly referred to as the peak cladding temperature (PCT) criterion, the local oxidation criterion, the hydrogen generation criterion, the coolable geometry criterion, and the long-term cooling criterion. A MAPLHGR limit is established for the ATRIUM-10 fuel type to ensure that these criteria are met. LOCA PCT results are provided in Sections 6.0 – 8.0 to determine the limiting LOCA event. LOCA analysis results demonstrating that the PCT, local oxidation, and hydrogen generation criteria are met are provided in the follow-on MAPLHGR report and cycle-specific heatup analyses. Cycle-specific heatup analyses are performed to demonstrate that the MAPLHGR limits versus exposure for the ATRIUM-10 fuel remains applicable for cycle-specific nuclear designs. Compliance with these three criteria ensures that a coolable geometry is maintained. Long-term coolability criterion is discussed in Section 9.0.

4.0 LOCA Analysis Description

The Evaluation Model used for the break spectrum analysis is the EXEM BWR-2000 LOCA analysis methodology described in Reference 1. The EXEM BWR-2000 methodology employs three major computer codes to evaluate the system and fuel response during all phases of a LOCA. These are the RELAX, HUXY, and RODEX2 computer codes. RELAX is used to calculate the system and hot channel response during the blowdown, refill and reflood phases of the LOCA. The HUXY code is used to perform heatup calculations for the entire LOCA, and calculates the PCT and local clad oxidation at the axial plane of interest. RODEX2 is used to determine fuel parameters (such as stored energy) for input to the other LOCA codes. The code interfaces for the LOCA methodology are illustrated in Figure 4.1.

A complete analysis for a given break size starts with the specification of fuel parameters using RODEX2 (Reference 4). RODEX2 is used to determine the initial stored energy for both the blowdown analysis (RELAX system and hot channel) and the heatup analysis (HUXY). This is accomplished by ensuring that the initial stored energy in RELAX and HUXY is the same or higher than that calculated by RODEX2 for the power, exposure, and fuel design being considered.

4.1 *Blowdown Analysis*

The RELAX code (Reference 1) is used to calculate the system thermal-hydraulic response during the blowdown phase of the LOCA. For the system blowdown analysis, the core is represented by an average core channel. The reactor core is modeled with heat generation rates determined from reactor kinetics equations with reactivity feedback and with decay heating as required by Appendix K of 10 CFR 50. The reactor vessel nodalization for the system blowdown analysis is shown in Figure 4.2. This nodalization is consistent with that used in the topical report submitted to the NRC (Reference 1).

The RELAX analysis is performed from the time of the break initiation through the end of blowdown (EOB). The system blowdown calculation provides the upper and lower plenum transient boundary conditions for the hot channel analysis.

Following the system blowdown calculation, another RELAX analysis is performed to analyze the maximum power assembly (hot channel) of the core. The RELAX hot channel calculation is used to calculate hot channel fuel, cladding, and coolant temperatures during the blowdown

phase of the LOCA. The RELAX hot channel nodalization is shown in Figure 4.3 for a top-peaked power shape, and in Figure 4.4 for a mid-peaked axial power shape. The hot channel analysis is performed using the system blowdown results to supply the core power and the system boundary conditions at the core inlet and exit. The initial average fuel rod temperature at the limiting plane of the hot channel is conservative relative to the average fuel rod temperature calculated by RODEX2 for operation of the ATRIUM-10 assembly at the MAPLHGR limit. The heat transfer coefficients and fluid conditions in the hot channel from the RELAX hot channel calculation are used as input to the HUXY heatup analysis.

4.2 *Refill / Reflood Analysis*

The RELAX code is used to compute the system and hot channel hydraulic response during the refill/reflood phase of the LOCA. The RELAX system and RELAX hot channel analyses continue beyond the end of blowdown to analyze system and hot channel responses during the refill and reflood phases. The refill phase is the period when the lower plenum is filling due to ECCS injection. The reflood phase is when some portions of the core and hot assembly are being cooled with ECCS water entering from the lower plenum. The purpose of the RELAX calculations beyond blowdown is to determine the time when the liquid flow via upward entrainment from the bottom of the core becomes high enough at the hot node in the hot assembly to end the temperature increase of the fuel rod cladding. This event time is called the time of hot node reflood. [

] The time when the core bypass mixture level rises to the elevation of the hot node in the hot assembly is also determined.

RELAX provides a prediction of fluid inventory during the ECCS injection period. Allowing for countercurrent flow through the core and bypass, RELAX determines the refill rate of the lower plenum due to ECCS water and the subsequent reflood times for the core, hot assembly, and the core bypass. The RELAX calculations provide HUXY with the time of hot node reflood and the time when the liquid has risen in the bypass to the height of the axial plane of interest (time of bypass reflood).

4.3 *Heatup Analysis*

The HUXY code (Reference 2) is used to perform heatup calculations for the entire LOCA transient and provides PCT and local clad oxidation at the axial plane of interest. The heat

generated by metal-water reaction (MWR) is included in the HUXY analysis. HUXY is used to calculate the thermal response of each fuel rod in one axial plane of the hot channel assembly. These calculations consider thermal-mechanical interactions within the fuel rod. The clad swelling and rupture models from NUREG-0630 have been incorporated into HUXY (Reference 3). The HUXY code complies with the 10 CFR 50 Appendix K criteria for LOCA Evaluation Models.

HUXY uses the end of blowdown time and the times of core bypass reflood and core reflood at the axial plane of interest from the RELAX analysis. [

]

Throughout the calculations, decay power is determined based on the ANS 1971 decay heat curve plus 20% as described in Reference 1. [

] used in the HUXY analysis. The principal results of a HUXY heatup analysis are the PCT and the percent local oxidation of the fuel cladding, often called the percent maximum local metal water reaction (%MWR).

4.4 *Plant Parameters*

The LOCA break spectrum analysis is performed using the plant parameters presented in Reference 6. Table 4.1 provides a summary of reactor initial conditions used in the break spectrum analysis. Table 4.2 lists selected reactor system parameters.

The break spectrum analysis is performed for a full core of ATRIUM-10 fuel. Some of the key ATRIUM-10 fuel parameters used in the break spectrum analysis are summarized in Table 4.3.

4.5 *ECCS Parameters*

The ECCS configuration is shown in Figure 4.5. Tables 4.4 – 4.8 provide the important ECCS characteristics assumed in the analysis. The ECCS is modeled as fill junctions connected to the appropriate reactor locations: LPCS injects into the upper plenum, HPCI injects into the upper downcomer, and LPCI injects into the recirculation lines.

The flow through each ECCS valve is determined based on system pressure and valve position. Flow versus pressure for a fully open valve is obtained by linearly interpolating the pump

capacity data provided in Tables 4.4 – 4.6. No credit for ECCS flow is assumed until ECCS pumps reach rated speed.

The ADS valves are modeled as a junction connecting the reactor steam line to the suppression pool. The flow through the ADS valves is calculated based on pressure and valve flow characteristics. The valve flow characteristics are determined such that the calculated flow is equal to the rated capacity at the reference pressure shown in Table 4.7.

In the AREVA LOCA analysis model, ECCS initiation is assumed to occur when the water level drops to the applicable level setpoint. No credit is assumed for the start of HPCI, LPCS, or LPCI due to high drywell pressure. [

]

The recirculation discharge isolation valve (RDIV) parameters are shown in Table 4.8.

Table 4.1 Initial Conditions*

Reactor power (% of rated)	102	102
Total core flow (% of rated)	[ ]	[ ]
Reactor power (MWt)	3527	3527
Total core flow (Mlb/hr)	[ ]	[ ]
[ ]		
Steam flow rate (Mlb/hr)	14.47	14.47
Steam dome pressure (psia)	1054	1054
Core inlet enthalpy (Btu/lb)	[ ]	[ ]
ATRIUM-10 hot assembly MAPLHGR (kW/ft)	12.5	12.5
[ ]		
ECCS fluid temperature (°F)	120	120
Axial power shape	Fig. 4.6	Fig. 4.7

* The AREVA calculated heat balance is adjusted to match the 100% power/100% flow values given in the plant parameters document (Reference 6). The model is then rebalanced based on AREVA heat balance calculations to establish these LOCA initial conditions at 102% of rated thermal power.

† [ ]

Table 4.2 Reactor System Parameters

Parameter	Value
Vessel ID (in)	251
Number of fuel assemblies	764
Recirculation suction pipe area (ft^2)	3.507
1.0 DEG suction break area (ft^2)	7.013
Recirculation discharge pipe area (ft^2)	3.507
1.0 DEG discharge break area (ft^2)	7.013

Table 4.3 ATRIUM-10 Fuel Assembly Parameters

Parameter	Value
Fuel rod array	10x10
Number of fuel rods per assembly	83 (full-length rods) 8 (part-length rods)
Non-fuel rod type	Water channel replaces 9 fuel rods
Fuel rod OD (in)	0.3957
Active fuel length (in) (including blankets)	149.45 (full-length rods) 90 (part-length rods)
Water channel outside width (in)	1.378
Fuel channel thickness (in)	0.075 (minimum wall) 0.100 (corner)
Fuel channel internal width (in)	5.278

Table 4.4 High-Pressure Coolant Injection Parameters

Parameter	Value
Coolant temperature (°F)	120
Initiating Signals and Setpoints	
Water level*	L2 (448 in)
High drywell pressure (psig)	2.6 (Not used)
Time Delays	
Time for HPCI pump to reach rated speed and injection valve wide open (sec)	35
Delivered Flow Rate Versus Pressure	
Vessel to Drywell ΔP (psid)	Flow Rate (gpm)
0	0
150	5000
1120	5000
1174	3600

* Relative to vessel zero.

Table 4.5 Low-Pressure Coolant Injection Parameters

Parameter	Value
Reactor pressure permissive for opening valves (psia)	350
Coolant temperature (°F)	120
Initiating Signals and Setpoints	
Water level*	L1 (372.5 in)
High drywell pressure (psig)	2.6 (Not used)
Time Delays	
Time for LPCI pumps to reach rated speed (max) (sec)†	44
LPCI injection valve stroke time (sec)	40

Delivered Flow Rate Versus Pressure

Vessel to Drywell ΔP (psid)	Flow Rate (gpm)	
	2 Pumps Into 1 Loop	4 Pumps Into 2 Loops
0	17,240	34,480
20	16,540	33,080‡
319.5	0	0

* Relative to vessel zero.

† Includes 13-second delay for diesel generator start. 2-second signal processing delay for water level trip L1 is assumed in parallel with diesel generator delay.

‡ Conservative value relative to specified value in Reference 6 (33,240 gpm). Modeling limitations require the more conservative value of either the specified 4 pumps into 2 loops flow or twice the specified 2 pumps into 1 loop flow be used.

Table 4.6 Low-Pressure Core Spray Parameters

Parameter	Value
Reactor pressure permissive for opening valves (psia)	350
Coolant temperature (°F)	120
Initiating Signals and Setpoints	
Water level*	L1 (372.5 in)
High drywell pressure (psig)	2.6 (Not used)
Time Delays	
Time for LPCS pumps to reach ADS permissive (max) (sec)†	40
Time for LPCS pumps to reach rated speed (max) (sec)†	43
LPCS injection valve stroke time (sec)	33

Delivered Flow Rate Versus Pressure

Vessel to Drywell ΔP (psid)	Flow Rate (gpm)	
	2 Pumps Into 1 Sparger	4 Pumps Into 2 Spargers
0	6,935	13,870
105	5,435	10,870
200	3,835	7,670
289	0	0

* Relative to vessel zero.

† Includes 13-second delay for diesel generator start. 2-second signal processing delay for water level trip L1 is assumed in parallel with diesel generator delay.

Table 4.7 Automatic Depressurization System Parameters

Parameter	Value
Number of valves installed	6
Number of valves available	6
Minimum flow capacity of available valves (Mlbm/hr at psig)	4.8 at 1125
Initiating Signals and Setpoints	
Water level*	L1 (372.5 in)
LPCS ready permissive†	L1 + 40 sec (max)
Time Delays	
Delay time (from ADS timer permissive to time valves are open) (sec)	120

* Relative to vessel zero.

† ADS timer initiation occurs after level trip L1 is met and LPCS pumps reach the ADS ready permissive (see Table 4.6).

Table 4.8 Recirculation Discharge Isolation Valve Parameters

Parameter	Value
Reactor pressure permissive for closing valves – analytical (psia)	215
RDIV stroke time after pressure permissive (sec)	36

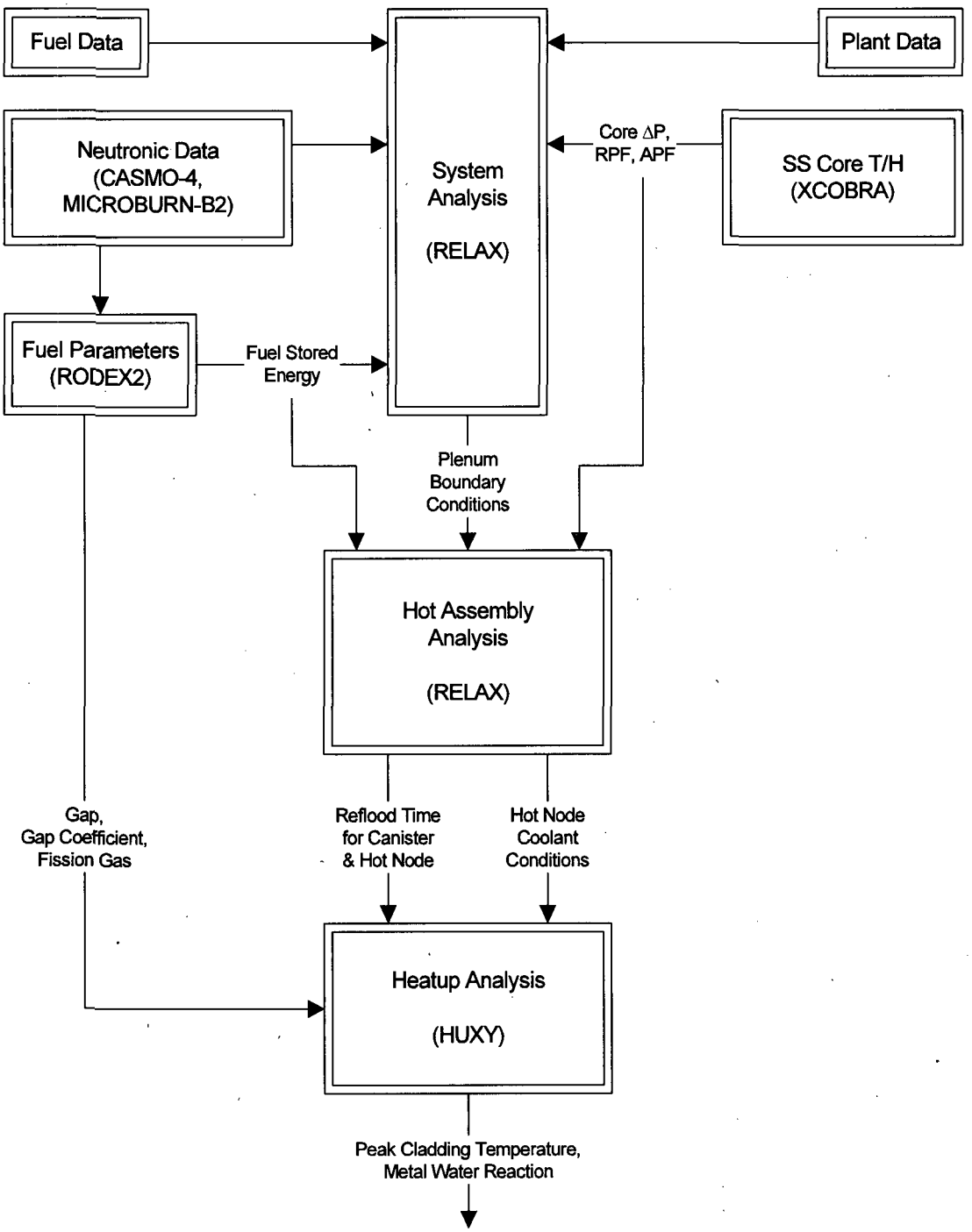

Figure 4.1 Flow Diagram for EXEM BWR-2000 ECCS Evaluation Model

Figure 4.2 RELAX System Blowdown Model

Figure 4.3 RELAX Hot Channel Blowdown Model Top-Peaked Axial

Figure 4.4 RELAX Hot Channel Blowdown Model Mid-Peaked Axial

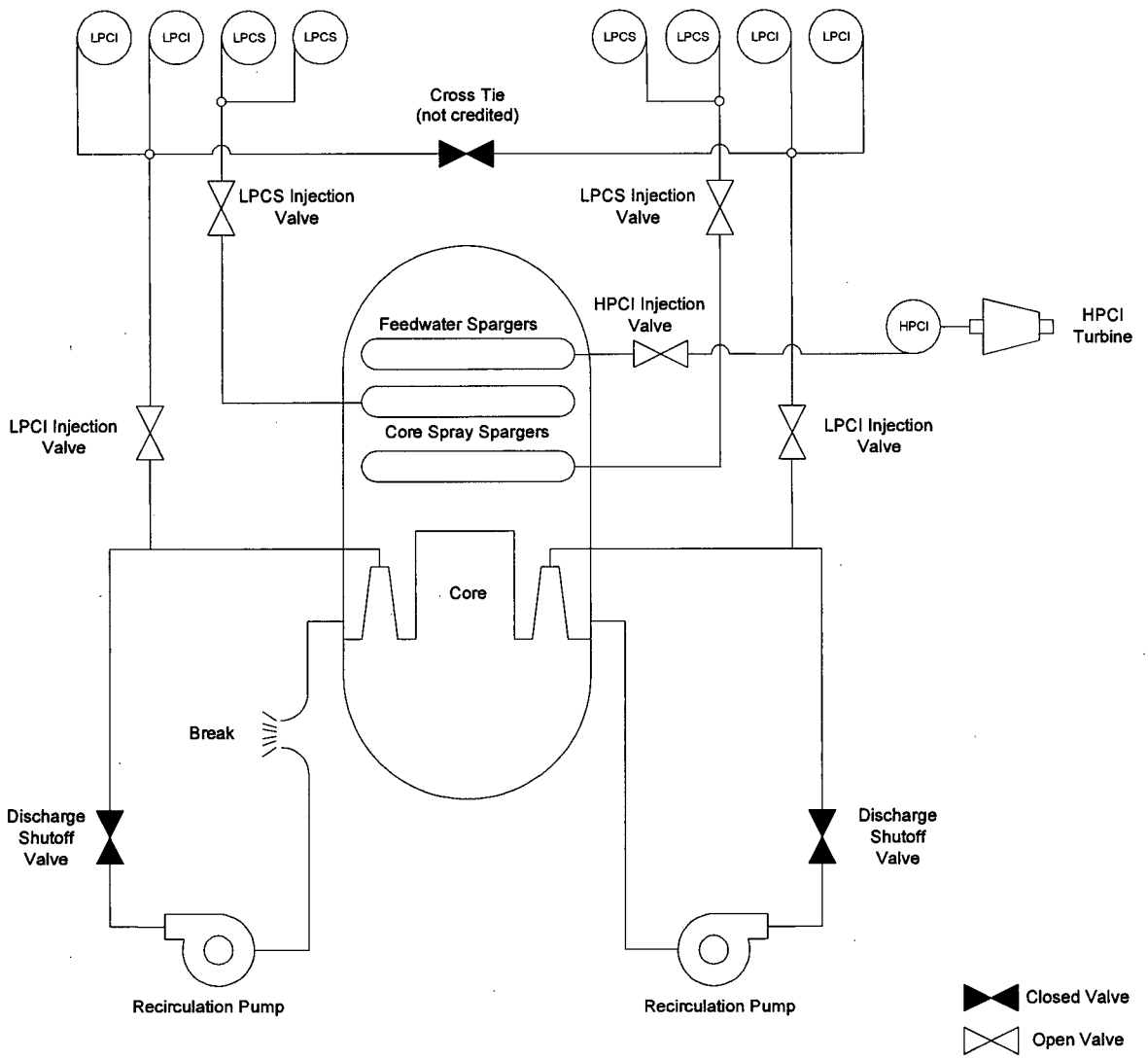

Figure 4.5 ECCS Schematic

Figure 4.6 Axial Power Distributions at 102% Power / [ ]

Figure 4.7 Axial Power Distributions at 102% Power/ [ ]

5.0 Break Spectrum Analysis Description

The objective of this LOCA analyses is to ensure that the limiting break location, break type, break size, and ECCS single failure are identified. The LOCA response scenario varies considerably over the spectrum of break locations. Potential break locations have been separated into two groups: recirculation line breaks and non-recirculation line breaks. The basis for the break locations and potentially limiting single failures analyzed in this report is described in the following sections.

5.1 *Limiting Single Failure*

Regulatory requirements specify that the LOCA analysis be performed assuming that all offsite power supplies are lost instantaneously and that only safety grade systems and components are available. In addition, regulatory requirements also specify that the most limiting single failure of ECCS equipment must be assumed in the LOCA analysis. The term "most limiting" refers to the ECCS equipment failure that produces the greatest challenge to event acceptance criteria. The limiting single failure can be a common power supply, an injection valve, a system pump, or system initiation logic. The equipment identified (Reference 6) that may produce a limiting single failure (SF) is shown below:

- Backup battery power (SF-BATT)
  - Board A (SF-BATT|BA)
  - Board B (SF-BATT|BB)
  - Board C (SF-BATT|BC)
- Opposite unit false LOCA signal (SF-LOCA)
- Low-pressure coolant injection valve (SF-LPCI)
- Diesel generator (SF-DGEN)
- High-pressure coolant injection system (SF-HPCI)
- Automatic depressurization system (SF-ADS)
  - Failure of initiation logic (SF-ADS|IL)
  - Failure of single ADS valve (SF-ADS|SV)

The single failures and the available ECCS for each failure assumed in this analysis are summarized in Table 5.1 for recirculation line breaks and in Table 5.2 for non-recirculation line breaks. Other potential failures are not specifically considered because they all result in as much or more ECCS capacity.

5.2 *Recirculation Line Breaks*

The response during a recirculation line LOCA is dependent on break size. ADS operation is an important emergency system for small breaks. The ADS assists in depressurizing the reactor system, and thereby reduces the time required to reach rated LPCS and LPCI flow. For large breaks, rated LPCS and LPCI flow is generally reached before or shortly after the time when the ADS valves open so the ADS system is not required to mitigate the LOCA. The availability of ADS for each single failure is presented in Table 5.1. Table 5.1 clearly demonstrates that a single failure of the battery for board A or board B results in the least amount of ECCS capacity and is therefore limiting. It should be noted that SF-LOCA and SF-DGEN are identical.

Large recirculation line break analyses are performed for breaks in both the discharge and suction side of the recirculation pump. It is generally expected that the pump suction side break will be more severe due to the more rapid blowdown. The two largest flow resistances in the recirculation piping are the recirculation pump and the jet pump nozzle. For breaks in the discharge piping, there is a major flow resistance in both flow paths to the break. For breaks in the suction piping, there is only one major flow resistance due to the recirculation pump between the break and reactor vessel. As a result, suction side breaks allow the coolant to exit more rapidly and generally result in more severe events than discharge side breaks (if ECCS capacity is equal). Both suction and discharge recirculation pipe breaks are considered in the break spectrum analysis.

Two break types (geometries) are considered for the recirculation line break. The two types are the double-ended guillotine (DEG) break and the split break.

For a DEG break, the piping is assumed to be completely severed resulting in two independent flow paths to the containment. The DEG break is modeled by setting the break area (at both ends of the pipe) equal to the full pipe cross-sectional area and varying the discharge coefficient between 1.0 and 0.4. The range of discharge coefficients is used to cover uncertainty in the actual geometry at the break. Discharge coefficients below 0.4 are unrealistic and not considered in the EXEM BWR-2000 methodology.

A split type break is assumed to be a longitudinal opening or hole in the piping that results in a single break flow path to the containment. Appendix K of 10 CFR 50 defines the cross-sectional area of the piping as the maximum split break area required for analysis.

The rate of reactor vessel depressurization is slower for intermediate and small breaks (break area $\leq$ 1.0 ft^2) compared to large break LOCAs. The HPCI and ADS will assist in reducing the reactor vessel pressure to the pressure where the LPCI and LPCS flows start.

The break spectrum analyses in the intermediate and small break region consider break sizes between 1.0 ft^2 and 0.07 ft^2. Break sizes and single failures are analyzed for both suction and discharge recirculation line breaks.

Section 6.0 provides a description and result summary for large, intermediate, and small breaks in the recirculation line.

5.3 *Non-Recirculation Line Breaks*

In addition to breaks in the recirculation line, breaks in other reactor coolant system piping must be considered in the LOCA break spectrum analysis. Although the recirculation line large break results in the largest coolant inventory loss, it does not necessarily result in the most severe challenge to event acceptance criteria. The double-ended rupture of a main steam line is expected to result in the fastest depressurization of the reactor vessel. Special consideration is required when the postulated break occurs in ECCS piping. Although ECCS piping breaks are small relative to a recirculation pipe DEG break, this break disables an ECCS system and therefore, increases the postulated break severity. Table 5.2 summarizes the available ECCS components of the potentially limiting single failures. The following sections address potential LOCAs due to breaks in non-recirculation line piping.

5.3.1 Main Steam Line Breaks

A steam line break inside containment is assumed to occur between the reactor vessel and the inboard main steam line isolation valve (MSIV) upstream of the flow limiters. The break results in high steam flow out of the broken line and into the containment. Prior to MSIV closure, a steam line break also results in high steam flow in the intact steam lines as they feed the break via the steam line manifold. A large steam line break inside containment results in a rapid depressurization of the reactor vessel. Because of the rapid depressurization, the primary

mitigation for the event is the low pressure ECCS. Initially the break flow will be high quality steam; however, the rapid depressurization produces a water level swell that results in liquid discharge at the break. For steam line breaks, the largest break size results in the most level swell and liquid loss out of the break. [

]

A small steam line break will result in a slower depressurization of the reactor vessel. Because the reactor vesssel does not quickly depressurize, mitigation of the event is initially by HPCI and/or ADS, followed by the low pressure ECCS. For the smallest break sizes, HPCI and ADS are significant systems for the overall mitigation of the event. HPCI could become degraded from a steam line split break or completely disabled from a guillotine break of the HPCI steam supply line (Figure 5.1). Because a steam line break could disable HPCI, no credit for HPCI is assumed for the steam line break analyses. The limiting single failure analyzed from Table 5.2 for the steam line break/HPCI steam supply line break is SF-BATT|BB. SF-BATT|BA is non-limiting since it has the same available ECCS as SF-BATT|BB with the addition of ADS. Analyses confirm that the steam line break/HPCI steam supply line break is non-limiting relative to the recirculation line breaks, refer to Section 7.

5.3.2 Feedwater Line Breaks

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The above comparison of a feedwater line break to a recirculation line break is not applicable if one of the available ECCS systems is HPCI. Because the HPCI injection line is connected to the feedwater line, as shown in Figure 5.1, a break in the feedwater line could result in a full or partial loss of HPCI injection, which could increase the severity of the event relative to a recirculation line break. No credit for HPCI is assumed for the feedwater line break analyses (HPCI flow is not modeled). The limiting single failure from Table 5.2 is analyzed for the feedwater line break is SF-BATT|BB. SF-BATT|BA is non-limiting since it has the same available ECCS as SF-BATT|BB with the addition of ADS. For this analysis, a spectrum of break sizes is evaluated. Because the break flow is choked, the largest break area is assumed to be the minimum area of the feedwater line (the spectrum encompasses the size of the HPCI injection line). Analyses confirm that the feedwater line break is non-limiting relative to the recirculation line breaks, refer to Section 7.

5.3.3 HPCI Line Breaks

Figure 5.1 shows the location of the HPCI injection line and HPCI steam supply line.

The HPCI injection line is connected to the feedwater line outside of the containment. As discussed in Section 5.3.2, a break in the feedwater line could result in a full or partial loss of HPCI injection. The limiting single failure from Table 5.2 is analyzed for the feedwater or HPCI injection line break is SF-BATT|BB. SF-BATT|BA is non-limiting since it has the same available ECCS as SF-BATT|BB with the addition of ADS. For this analysis, a spectrum of break sizes is evaluated. Because the break flow is choked, the largest break area is assumed to be the minimum area of the feedwater line (the spectrum encompasses the size of the HPCI injection line). For the analysis, the break flow is not assumed to be terminated by an isolation or check valve closure; therefore, a break in the HPCI injection line is assumed to be equivalent to a break of the same size in the feedwater line inside containment. Analyses confirm that the HPCI injection line break via a feedwater line break is non-limiting relative to the recirculation line breaks, refer to Section 7.

The HPCI steam supply line is connected to the main steam line inside of containment.

[

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5.3.4 LPCS Line Breaks

A break in the LPCS line is expected to have many characteristics similar to [ ] However, some characteristics of the LPCS line break are unique and are not addressed in other LOCA analyses. Two important differences from other LOCA analyses are that the break flow will exit from the region inside the core shroud and the break will disable one LPCS system. The LPCS line break is assumed to occur just outside the reactor vessel. [ ]

5.3.5 LPCI Line Breaks

The LPCI injection lines are connected to the larger recirculation discharge lines. A break in a LPCI line would result in the partial or full loss of LPCI injection. As shown in Table 5.2, the limiting single failures are SF-BATT|BA and SF-BATT|BB. Because LPCI is assumed to be fully lost from the broken line, the available ECCS is equivalent to the recirculation discharge breaks. [ ]

5.3.6 RCIC Line Breaks

The RCIC discharges to the feedwater line; [ ]

The steam supply to the RCIC turbine comes from the main steam line from the reactor vessel; [ ]

5.3.7 RWCU Line Breaks

The RWCU extraction line is connected to a recirculation suction line with an additional connection to the vessel bottom head. [

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The RWCU return line is connected to the feedwater line; [

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5.3.8 Instrument Line Breaks

[

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Table 5.1 Available ECCS for Recirculation Line Break LOCAs

Assumed Failure	Systems* † Remaining		Disposition
	Recirculation‡ Suction Break	Recirculation Discharge Break	
SF-BATT|BA	6 ADS, 1 LPCS, 2 LPCI	6 ADS, 1 LPCS	Analyze
SF-BATT|BB	HPCI, 1 LPCS, 2 LPCI, Manual ADS§	HPCI, 1 LPCS, Manual ADS	Analyze
SF-BATT|BC**	4 ADS, HPCI, 1 LPCS, 3 LPCI	4 ADS, HPCI, 1 LPCS, 1 LPCI	Bounded by SF-BATT|BB
SF-LOCA	6 ADS, HPCI, 1 LPCS, 2 LPCI	6 ADS, HPCI, 1 LPCS	Bounded by SF-BATT|BA and SF-BATT|BB
SF-LPCI	6 ADS, HPCI, 2 LPCS, 2 LPCI	6 ADS, HPCI, 2 LPCS	Bounded by SF-BATT|BA and SF-BATT|BB
SF-DGEN	6 ADS, HPCI, 1 LPCS, 2 LPCI	6 ADS, HPCI, 1 LPCS	Bounded by SF-BATT|BA and SF-BATT|BB
SF-HPCI	6 ADS, 2 LPCS, 4 LPCI	6 ADS, 2 LPCS, 2 LPCI	Bounded by SF-BATT|BA
SF-ADS|IL	HPCI, 2 LPCS, 4 LPCI, Manual ADS	HPCI, 2 LPCS, 2 LPCI, Manual ADS	Bounded by SF-BATT|BB
SF-ADS|SV	5 ADS, HPCI, 2 LPCS, 4 LPCI	5 ADS, HPCI, 2 LPCS, 2 LPCI	Bounded by SF-BATT|BB

* Each LPCS means operation of two core spray pumps in a system. It is assumed that both pumps in a system must operate to take credit for core spray cooling or inventory makeup. Furthermore, 2 LPCI refers to two LPCI pumps into one loop, 3 LPCI refers to two LPCI pumps into one loop and one LPCI pump into one loop. 4 LPCI refers to four LPCI pumps into two loops, two per loop.

† 4 ADS, 5 ADS and 6 ADS means the number of ADS valves available for automatic activation.

‡ Systems remaining, as identified in this table for recirculation suction line breaks, are applicable to other non-ECCS line breaks. For a LOCA from an ECCS line break, the systems remaining are those listed for recirculation suction breaks, less the ECCS in which the break is assumed.

§ Manual ADS means 4 ADS valves are available for to be opened manually. The analyses assumes the 4 valves are opened 10 minutes after the break occurs.

** Unit 3 systems remaining. Conservative for Units 1 and 2.

Table 5.2 Available ECCS for Non-Recirculation Line Break LOCAs

Non-Recirculation Line Break	Assumed Failure	Available ECCS*	Disposition
HPCI† Feedwater Line† Steam Line†	SF-BATT|BA	6 ADS, 1 LPCS, 2 LPCI	Bounded by SF-BATT|BB
	SF-BATT|BB	1 LPCS, 2 LPCI, Manual ADS	Analyze
	SF-BATT|BC	4 ADS, 1 LPCS, 3 LPCI	Bounded by SF-BATT|BB
	SF-LOCA	6 ADS, 1 LPCS, 2 LPCI	Bounded by SF-BATT|BB
	SF-LPCI	6 ADS, 2 LPCS, 2 LPCI	Bounded by SF-BATT|BB
	SF-DGEN	6 ADS, 1 LPCS, 2 LPCI	Bounded by SF-BATT|BB
	SF-HPCI	6 ADS, 2 LPCS, 4 LPCI	Bounded by SF-BATT|BB
	SF-ADS|IL	2 LPCS, 4 LPCI, Manual ADS	Bounded by SF-BATT|BB
	SF-ADS|SV	5 ADS, 2 LPCS, 4 LPCI	Bounded by SF-BATT|BB
LPCS	SF-BATT|BA	6 ADS, 2 LPCI	Analyze
	SF-BATT|BB	HPCI, 2 LPCI, Manual ADS	Analyze
	SF-BATT|BC	4 ADS, HPCI, 3 LPCI	Bounded by SF-BATT|BB
	SF-LOCA	6 ADS, HPCI, 2 LPCI	Bounded by SF-BATT|BA and SF-BATT|BB
	SF-LPCI	6 ADS, HPCI, 1 LPCS, 2 LPCI	Bounded by SF-BATT|BA and SF-BATT|BB
	SF-DGEN	6 ADS, HPCI, 2 LPCI	Bounded by SF-BATT|BA and SF-BATT|BB
	SF-HPCI	6 ADS, 1 LPCS, 4 LPCI	Bounded by SF-BATT|BA
	SF-ADS|IL	HPCI, 1 LPCS, 4 LPCI, Manual ADS	Bounded by SF-BATT|BB
	SF-ADS|SV	5 ADS, HPCI, 1 LPCS, 4 LPCI	Bounded by SF-BATT|BB

* Refer to footnotes of Table 5.1 for additional information.

† Assumes no credit for HPCI flow.

Table 5.2 (cont.) Available ECCS for Non-Recirculation Line Break LOCAs

Non-Recirculation Line Break	Assumed Failure	Available ECCS*	Disposition
LPCI	SF-BATT|BA	6 ADS, 1 LPCS	Equivalent to recirculation pump discharge breaks (SF-BATT|BA). Additional analysis not required.
	SF-BATT|BB	HPCI, 1 LPCS, Manual ADS	Equivalent to recirculation pump discharge breaks (SF-BATT|BB). Additional analysis not required.
	SF-BATT|BC	4 ADS, HPCI, 1 LPCS, 1 LPCI	Bound by SF-BATT|BB
	SF-LOCA	6 ADS, HPCI, 1 LPCS	Bounded by SF-BATT|BA and SF-BATT|BB
	SF-LPCI	6 ADS, HPCI, 2 LPCS	Bounded by SF-BATT|BA and SF-BATT|BB
	SF-DGEN	6 ADS, HPCI, 1 LPCS	Bounded by SF-BATT|BA and SF-BATT|BB
	SF-HPCI	6 ADS, 2 LPCS, 2 LPCI	Bounded by SF-BATT|BA
	SF-ADS|IL	HPCI, 2 LPCS, 2 LPCI, Manual ADS	Bounded by SF-BATT|BB
	SF-ADS|SV	5 ADS, HPCI, 2 LPCS, 2 LPCI	Bounded by SF-BATT|BB

* Refer to footnotes of Table 5.1 for additional information.

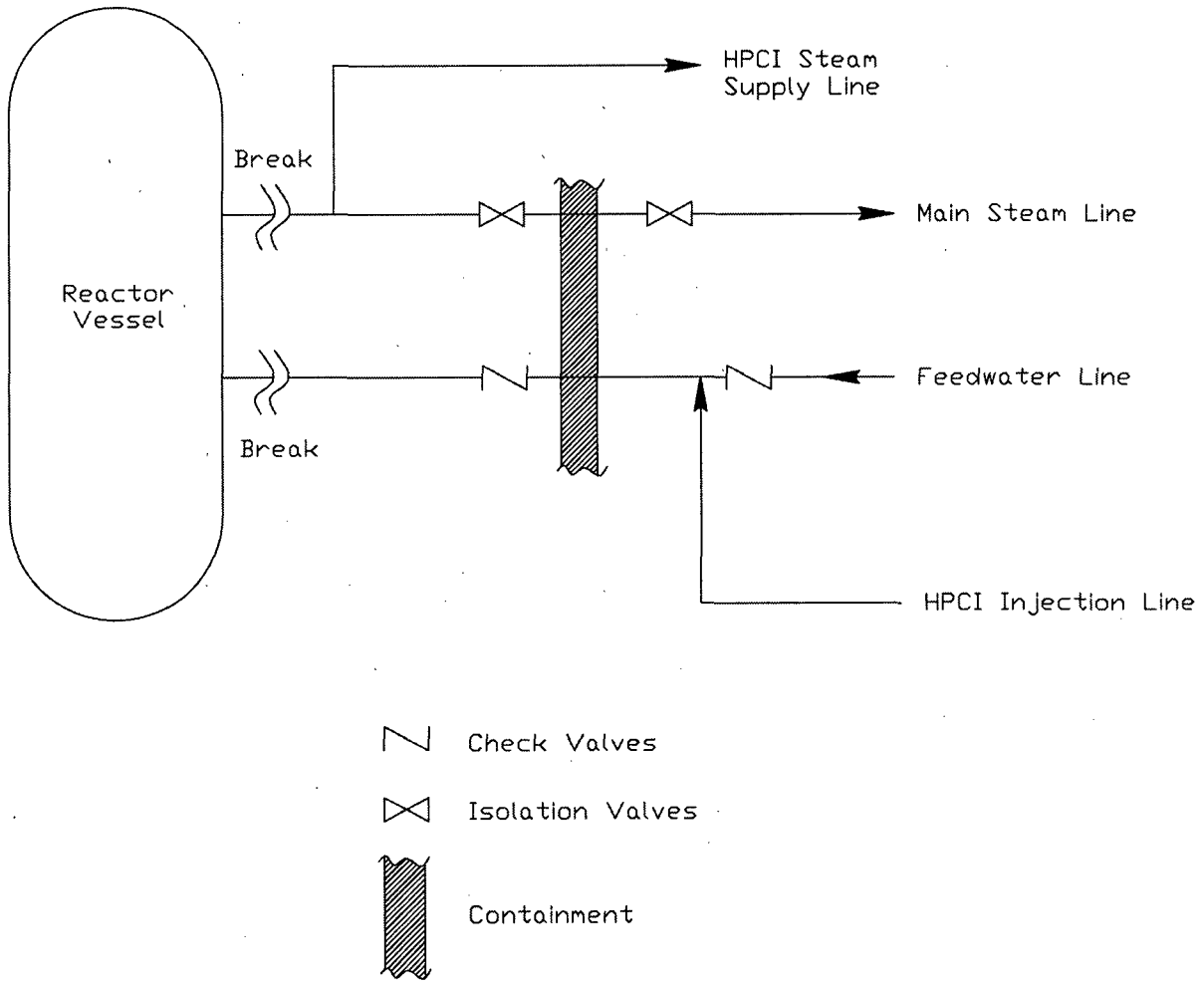

Figure 5.1 Steam, Feedwater and HPCI Lines

6.0 Recirculation Line Break LOCA Analyses

The largest diameter recirculation system pipes are the suction line between the reactor vessel and the recirculation pump and the discharge line between the recirculation pump and the riser manifold ring. LOCA analyses are performed for breaks in both of these locations with consideration for both DEG and split break geometries. The break sizes considered included DEG breaks with discharge coefficients from 1.0 to 0.4 and split breaks with areas ranging between the full pipe area to 0.07 ft^2. As shown in Table 5.1, the limiting single failures considered in the recirculation line break analyses are SF-BATT|BA and SF-BATT|BB.

[

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6.1 *Limiting Break Analysis Results*

The analyses demonstrate that the limiting (highest PCT) recirculation line break is the 0.25 ft^2 split break in the pump discharge piping for SF-BATT|BB and a top-peaked axial power shape. The limiting PCT is 1973°F. The key results and event times for this limiting break are provided in Tables 6.1 and 6.2, respectively. Figures 6.1 – 6.18 provide plots of key parameters from the RELAX system and hot channel blowdown analyses. A plot of cladding temperature versus time in the hot assembly from the HUXY heatup analysis is provided in Figure 6.19. Table 6.3 presents the detailed break spectrum PCT.

The limiting PCT for single failure SF-BATT|BA is bound by the limiting PCT for single failure SF-BATT|BB. For SF-BATT|BA, the highest PCT occurred for a recirculation split line break of 0.5 ft^2 in the pump discharge piping and a top-peaked axial power shape. The PCT is 1809°F. The key results and event times for this limiting break are provided in Table 6.4 and 6.5, respectively. Table 6.6 presents the detailed break spectrum PCT.

The results of the break analyses are discussed in the following sections. [

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[

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6.2 *Break Location Analysis Results*

Table 6.7 shows that the maximum PCT calculated for a recirculation line break occurs in the pump discharge piping.

6.3 *Break Geometry and Size Analysis Results*

Recirculation line break PCT results versus break geometry (DEG or split) and size are presented in Tables 6.3 and 6.6. The maximum PCT calculated for a recirculation line break occurs for a 0.25 ft^2 split break for SF-BATT|BB.

For large breaks, depressurization occurs quickly; therefore, HPCI and ADS do not significantly mitigate the event. Depressurization to activate the low pressure ECCS occurs before ADS activation. Both SF-BATT|BA and SF-BATT|BB have LPCS to mitigate the event. SF-BATT|BB also has HPCI; therefore, the largest break sizes are performed only for SF-BATT|BA since it has the least available ECCS.

The other differences in break sizes between Table 6.3 and Table 6.6 are due to the size of the limiting break and the resolution needed to determine the limiting break size. Table 6.3 consists of 0.01 ft^2 intervals near the limiting size of 0.25 ft^2. The limiting break for Table 6.6 is 0.5 ft^2; an interval of 0.05 ft^2 provides the resolution needed to determine the limiting break size.

6.4 *Limiting Single-Failure Analysis Results*

As previously mentioned, SF-BATT|BA and SF-BATT|BB are the limiting single failures based on available ECCS.

6.5 *Axial Power Shape Analysis Results*

The results in Table 6.3 show that the top-peaked axial power shape is limiting compared to the mid-peaked shape analyses for the limiting break size.

Table 6.1 Results for Limiting TLO Recirculation Line Break 0.25 ft^2 Split Pump Discharge SF-BATT|BB Top-Peaked Axial 102% Power

PCT	1973°F
Maximum local cladding oxidation	1.84%
Maximum planar average MWR	1.29%

Table 6.2 Event Times for Limiting TLO Recirculation Line Break 0.25 ft^2 Split Pump Discharge SF-BATT|BB Top-Peaked Axial 102% Power

Event	Time (sec)
Initiate break	0.0
Initiate scram	0.5
Low-Low liquid level, L2 (448 in)	30.4
Low-Low-Low liquid level, L1 (372.5 in)	52.3
Jet pump uncovers	82.0
Recirculation suction uncovers	513.7
Lower plenum flashes	97.5
HPCI rated flow pressure	70.0
HPCI valve fully open	65.4
HPCI flow starts	65.4
LPCS high-pressure cutoff	352.6
LPCS valve pressure permissive	324.7
LPCS valve starts to open	326.7
LPCS valve fully open	359.7
LPCS permissive for ADS timer	81.3
LPCS pump at rated speed	84.3
LPCS flow starts	352.6
RDIV pressure permissive	444.1
RDIV starts to close	446.1
RDIV fully closed	482.1
Rated LPCS flow	639.7
ADS valves open	600.0
Blowdown ends	639.7
Bypass reflood	589.4
PCT	376.0

Table 6.3 TLO Recirculation Line Break Spectrum Results for 102% Power SF-BATT|BB

Table 6.4 Results for TLO Recirculation Line Break 0.5 ft^2 Split Pump Discharge SF-BATT|BA Top-Peaked Axial 102% Power

PCT	1809°F
Maximum local cladding oxidation	0.94%
Maximum planar average MWR	0.48%

Table 6.5 Event Times for TLO Recirculation Line Break 0.5 ft^2 Split Pump Discharge SF-BATT|BA Top-Peaked Axial 102% Power

Event	Time (sec)
Initiate break	0.0
Initiate scram	0.5
Low-Low liquid level, L2 (448 in)	17.0
Low-Low-Low liquid level, L1 (372.5 in)	28.6
Jet pump uncovers	38.7
Recirculation suction uncovers	61.8
Lower plenum flashes	71.4
LPCS high-pressure cutoff	201.8
LPCS valve pressure permissive	193.2
LPCS valve starts to open	195.2
LPCS valve fully open	228.2
LPCS permissive for ADS timer	57.5
LPCS pump at rated speed	60.5
LPCS flow starts	201.8
RDIV pressure permissive	223.3
RDIV starts to close	225.3
RDIV fully closed	261.3
Rated LPCS flow	275.3
ADS valves open	177.6
Blowdown ends	275.3
Bypass reflood	409.9
Core reflood	332.8
PCT	332.8

Table 6.6 TLO Recirculation Line Break Spectrum Results for 102% Power SF-BATT|BA

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Table 6.7 Summary of TLO Recirculation Line Break Results Highest PCT Cases

Figure 6.1 Limiting TLO Recirculation Line Break Upper Plenum Pressure

Figure 6.2 Limiting TLO Recirculation Line Break Total Break Flow Rate

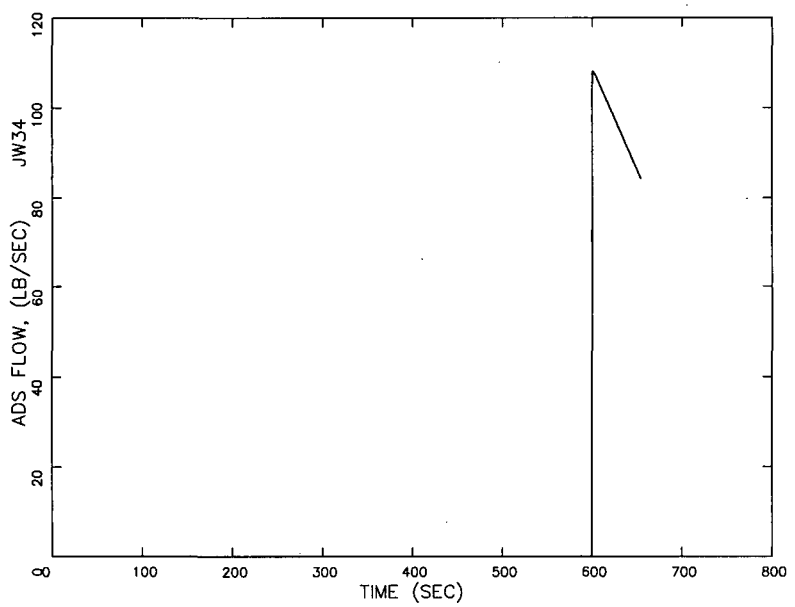

Figure 6.3 Limiting TLO Recirculation Line Break ADS Flow Rate

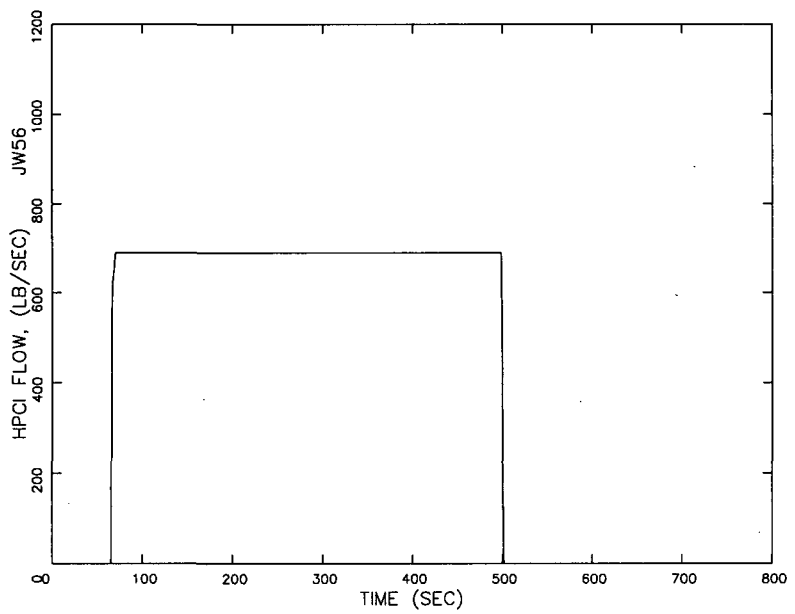

Figure 6.4 Limiting TLO Recirculation Line Break HPCI Flow Rate

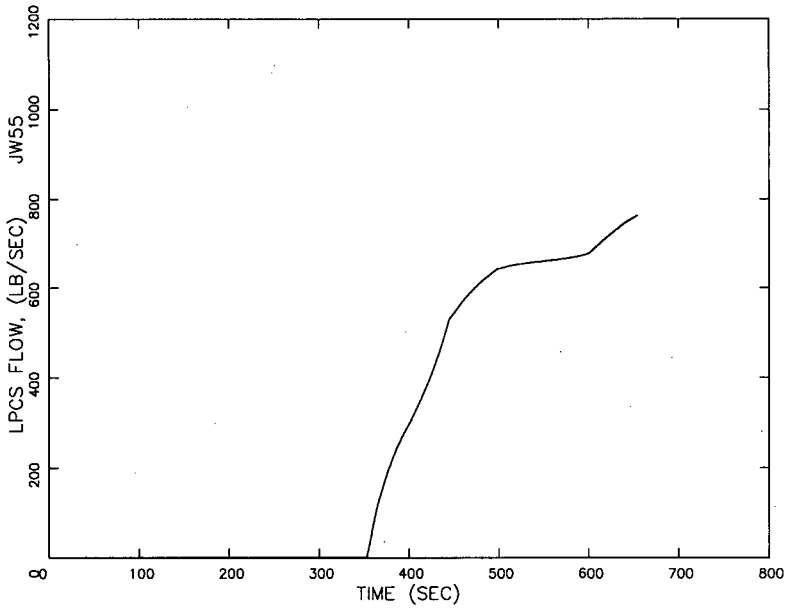

Figure 6.5 Limiting TLO Recirculation Line Break LPCS Flow Rate

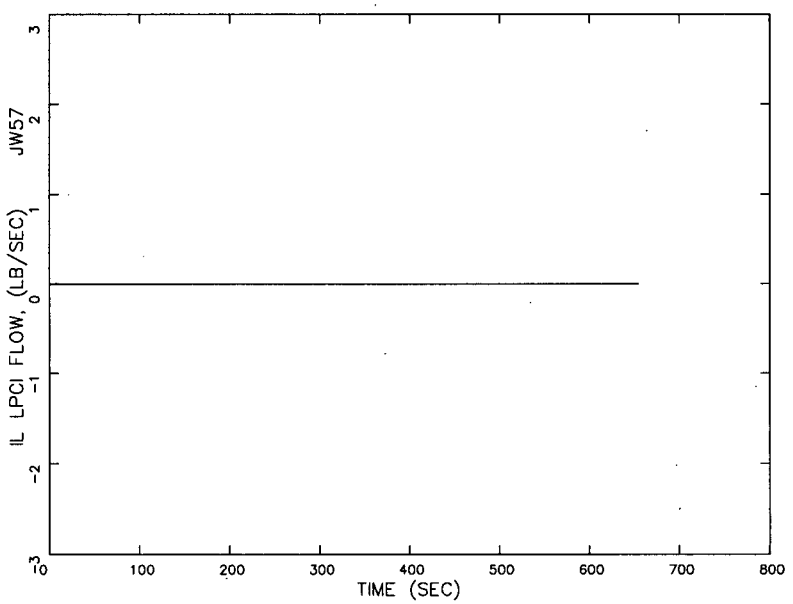

Figure 6.6 Limiting TLO Recirculation Line Break Intact Loop LPCI Flow Rate

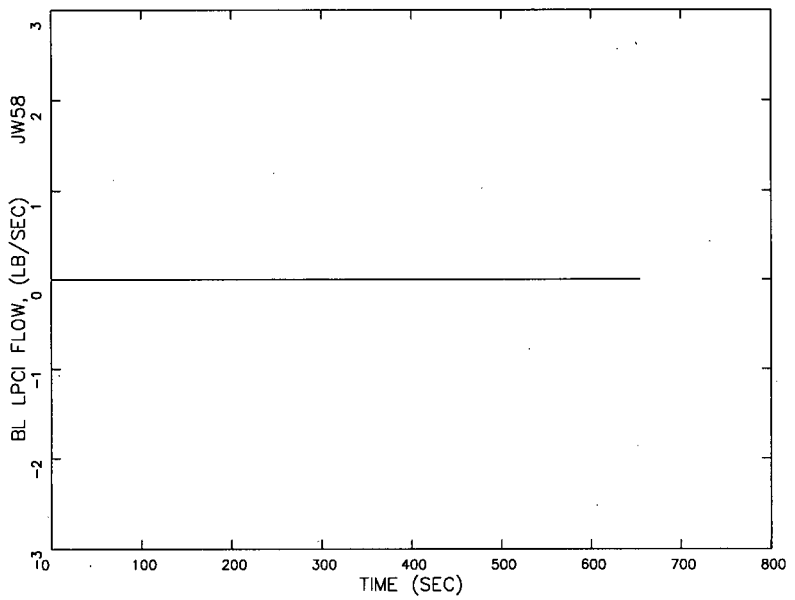

Figure 6.7 Limiting TLO Recirculation Line Break Broken Loop LPCI Flow Rate

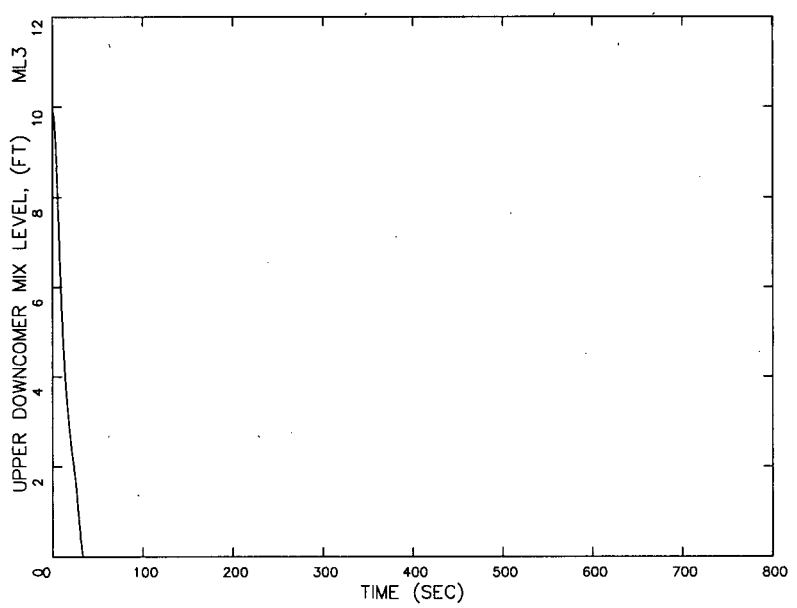

Figure 6.8 Limiting TLO Recirculation Line Break Upper Downcomer Mixture Level

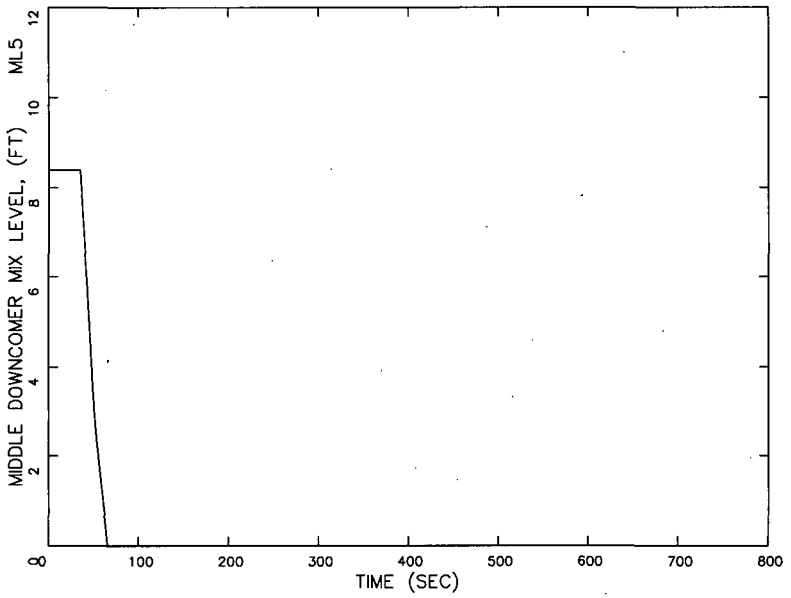

Figure 6.9 Limiting TLO Recirculation Line Break Middle Downcomer Mixture Level

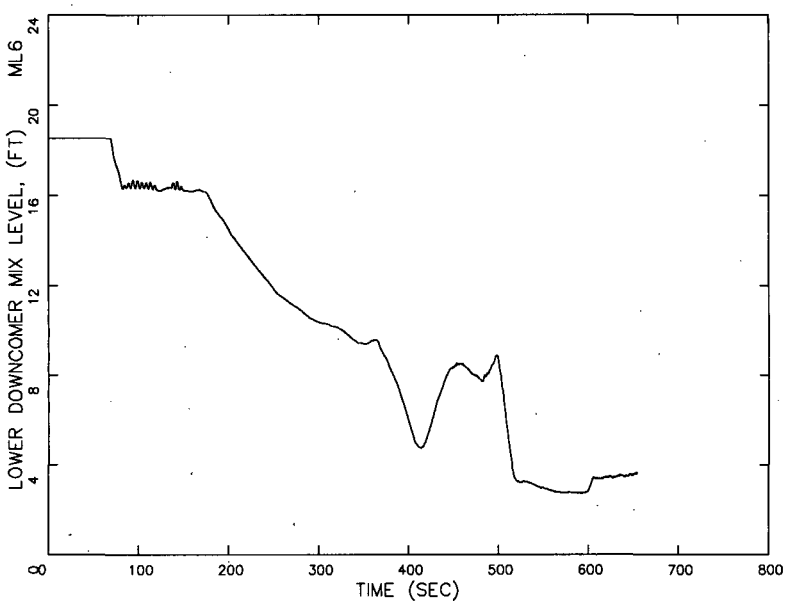

Figure 6.10 Limiting TLO Recirculation Line Break Lower Downcomer Mixture Level

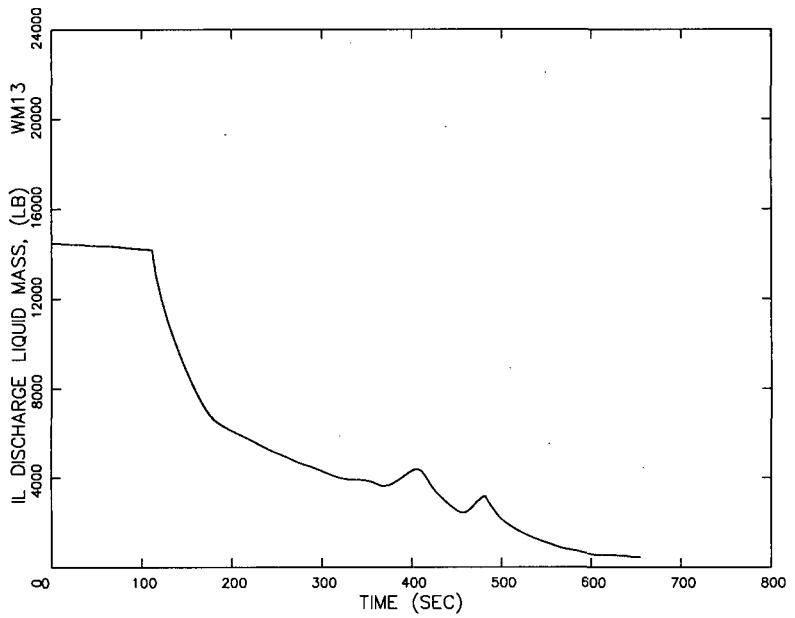

Figure 6.11 Limiting TLO Recirculation Line Break Intact Loop Discharge Liquid Mass

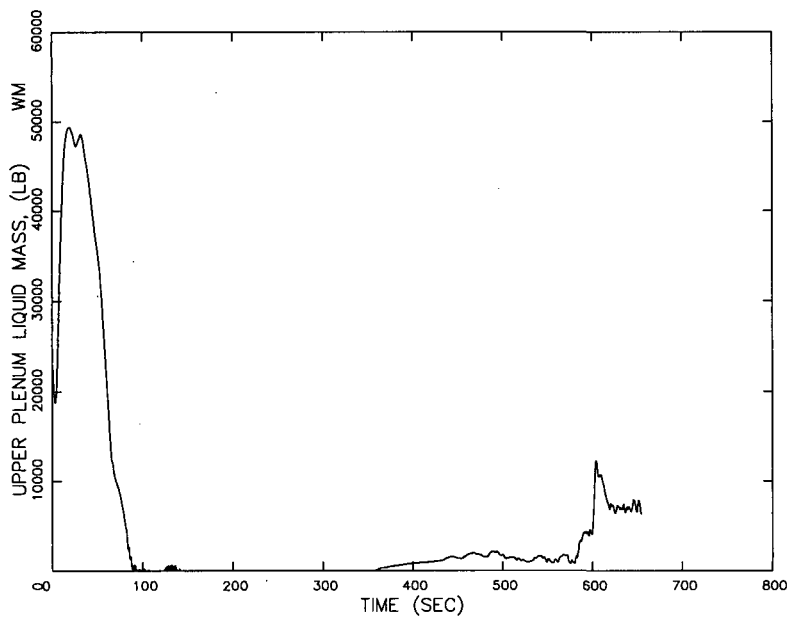

Figure 6.12 Limiting TLO Recirculation Line Break Upper Plenum Liquid Mass

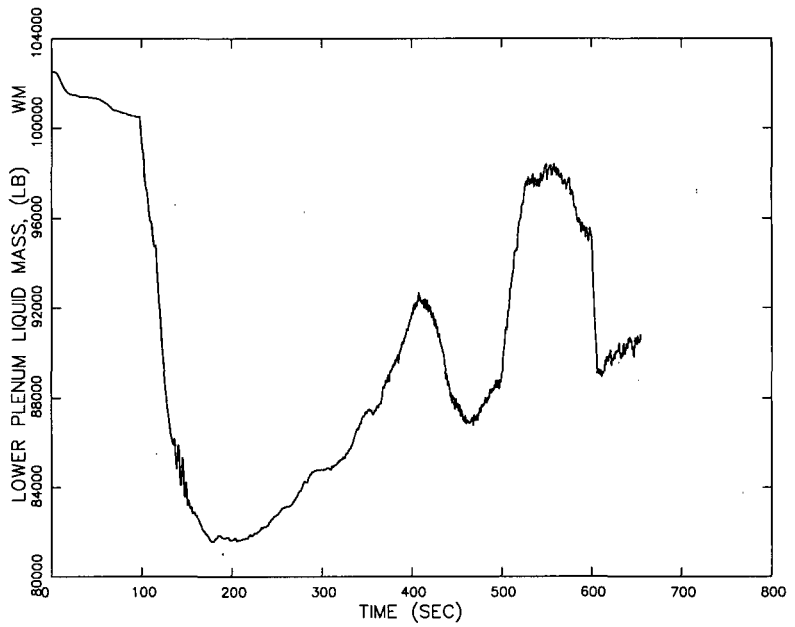

Figure 6.13 Limiting TLO Recirculation Line Break Lower Plenum Liquid Mass

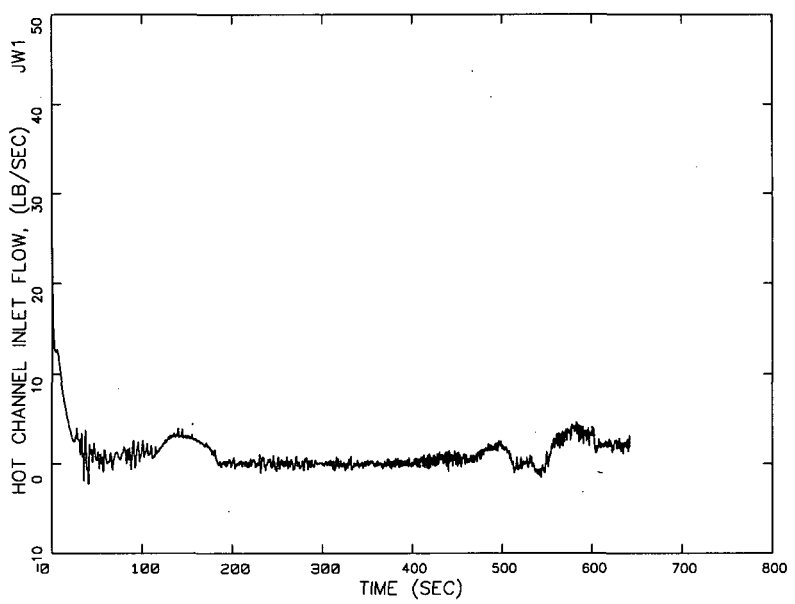

Figure 6.14 Limiting TLO Recirculation Line Break Hot Channel Inlet Flow Rate

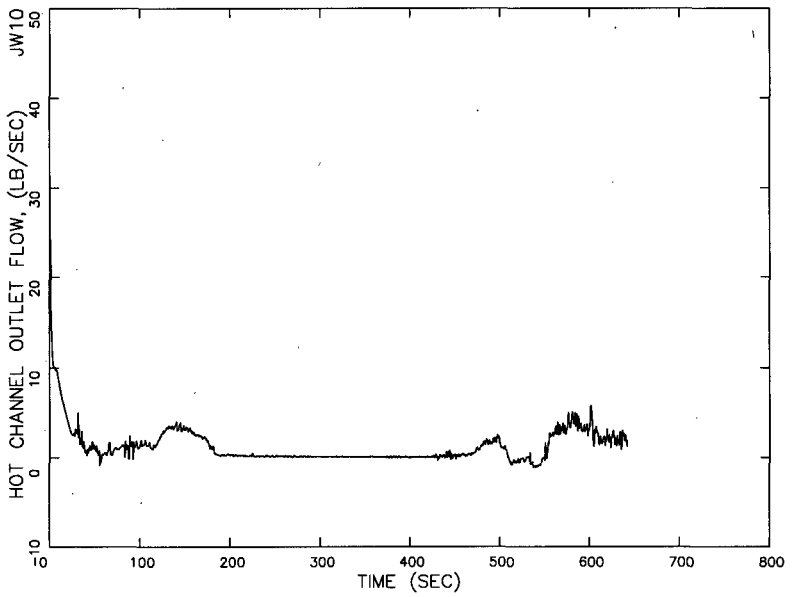

Figure 6.15 Limiting TLO Recirculation Line Break Hot Channel Outlet Flow Rate

Figure 6.16 Limiting TLO Recirculation Line Break Hot Channel Coolant Temperature at the Hot Node at EOB

Figure 6.17 Limiting TLO Recirculation Line Break Hot Channel Quality at the Hot Node at EOB

Figure 6.18 Limiting TLO Recirculation Line Break Hot Channel Heat Transfer Coef. at the Hot Node at EOB

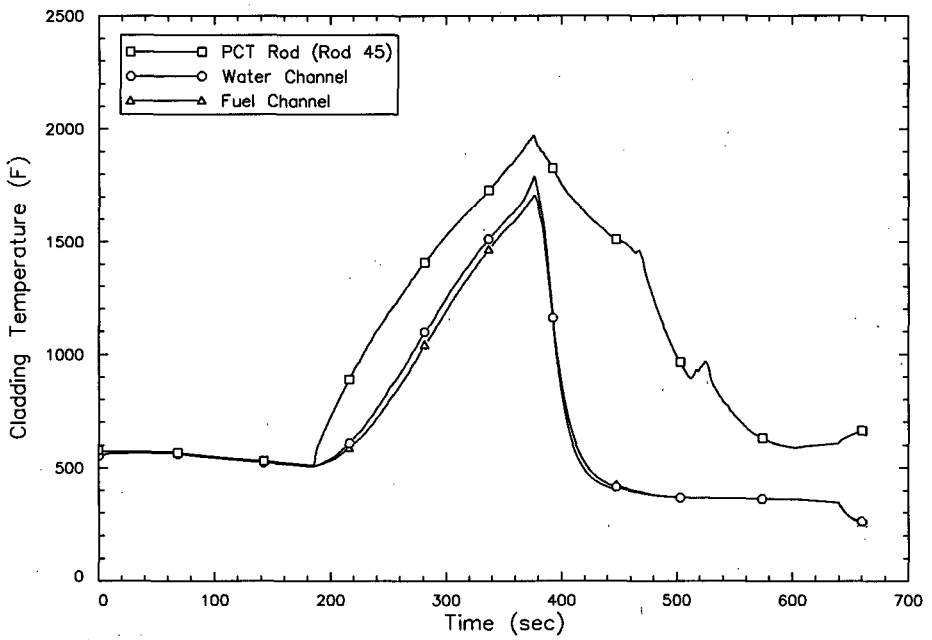

Figure 6.19 Limiting TLO Recirculation Line Break Cladding Temperatures

7.0 Non-Recirculation Line LOCA Analysis

LOCA analyses are performed for breaks in the steam, feedwater and LPCS lines. Breaks in other non-recirculation lines are less limiting for the reasons discussed in Section 5.3. Note that for LPCS line break cases with no core spray available, the HUXY heatup analysis is performed with the core spray heat transfer coefficients equal to 0.0.

7.1 *Limiting Non-Recirculation Line Break Results*

The results of this analysis indicate that the limiting non-recirculation line break is the 0.4 ft^2 DEG break in the LPCS line with SF-BATT|BA and a top-peaked axial power shape. The PCT for the limiting ECCS line break is 1580°F and maximum local cladding oxidation is 0.23%.

The other analyzed breaks; LPCS line, feedwater line and steam line with SF-BATT|BB are non-limiting. For most analyses, the core remained covered throughout the blowdown period and PCT is less than 1000°F.

For the feedwater line break, a spectrum of break sizes is evaluated and assumed to be as large as the minimum flow area of the feedwater line. As shown in Table 5.2, SF-BATT|BA is non-limiting since it has the same available ECCS as SF-BATT|BB with the addition of ADS. HPCI flow is not modeled. Depending on the break size, HPCI flow could fully block or partially block the break flow. HPCI flow would effectively change the break size for the limiting PCT. The limiting PCT is determined by performing a detailed break spectrum analysis as shown in Table 7.1 without credit for HPCI. The break sizes effectively represent the area in excess of that which would divert all of the HPCI flow out of the break. If HPCI flow is modeled, the limiting break size could change, but the feedwater line breaks would remain non-limiting. The analysis also does not credit LPCI, which is conservative. For the analysis, the break flow is not assumed to be terminated by an isolation or check valve closure; therefore, a break in the HPCI injection line is assumed to be equivalent to a break of the same size in the feedwater line inside containment. The results confirm that the feedwater line break is non-limiting relative to the recirculation line breaks.

For the steam line break, a spectrum of break sizes is evaluated. As shown in Table 5.2, SF-BATT|BA is non-limiting since it has the same available ECCS as SF-BATT|BB with the addition of ADS. In addition to not crediting HPCI, the analysis also does not credit LPCI, which is conservative. The largest break analyzed is the diameter of the HPCI steam supply line.

Larger steam line breaks are non-limiting as discussed in Section 5.3.1. The results confirm that the steam line break is non-limiting relative to the recirculation line breaks.

For the LPCS line break, [

] Analyses are performed for both SF-BATT|BA and SF-BATT|BB as shown in Table 5.2.

The key event times for the limiting non-recirculation break are provided in Table 7.1. Table 7.2 presents PCT results for the non-recirculation line breaks. The 1580°F maximum PCT for non-recirculation line breaks is lower than the maximum PCT for recirculation line breaks of 1973°F. Therefore, non-recirculation line breaks are non-limiting.

Table 7.1 Event Times for Limiting Non-Recirculation Line Break 0.4 ft^2 Double-Ended Guillotine SF-BATT|BA Top-Peaked Axial 102% Power

Event	Time (sec)
Initiate break	0.0
Initiate scram	0.5
Low-Low liquid level, L2 (448 in)	34.1
Low-Low-Low liquid level, L1 (372.5 in)	78.8
Jet pump uncovers	179.2
Recirculation suction uncovers	N.A.
Lower plenum flashes	57.0
LPCI high-pressure cutoff	205.8
LPCI valve pressure permissive	189.4
LPCI valve starts to open	191.4
LPCI valve fully open	231.4
LPCI pump at rated speed	111.8
LPCI flow starts	205.8
RDIV pressure permissive	263.1
RDIV starts to close	265.1
RDIV fully closed	301.1
Rated LPCS pressure	319.4
ADS valves open	227.8
Blowdown ends	319.4
Bypass reflood	367.1
Core reflood	345.5
PCT	345.5

Table 7.2 Non-Recirculation Line Break Spectrum Results

[ ]

* Analyses for feedwater line break assumed no HPCI flow and encompass the size of the HPCI injection line. Refer to Section 7.1 for additional information.

† Cases with PCT of <1000°F have little or no core uncovery.

‡ Steam line breaks encompassed the size of the HPCI steam supply line.

8.0 Single-Loop Operation LOCA Analysis

During SLO the pump in one recirculation loop is not operating. A break may occur in either loop, but results from a break in the inactive loop would be similar to those from a two-loop operation break. If a break occurs in the inactive loop during SLO, the intact active loop flow to the reactor vessel would continue during the recirculation pump coastdown period and would provide core cooling similar to the flow occurring in breaks during two-loop operation. System response would be similar to that resulting from an equal-sized break during two-loop operation. A break in the active loop during SLO results in a more rapid loss of core flow and earlier degraded core conditions relative to those from a break in the inactive loop. Therefore, only breaks in the active recirculation loop are analyzed.

A break in the active recirculation loop during SLO results in an earlier loss of core heat transfer relative to a similar break occurring during two-loop operation. This occurs because there is immediate loss of jet pump drive flow. Therefore, fuel rod surface temperatures increase faster in an SLO LOCA relative to a normal operation LOCA. Also, the early loss of core heat transfer results in higher stored energy in the fuel rods at the start of the heatup. The increased severity of an SLO LOCA can be reduced by applying an SLO multiplier to the two-loop MAPLHGR limits. [

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8.1 *SLO Analysis Modeling Methodology*

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8.2 *SLO Analysis Results*

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The SLO SF-BATT|BA analyses are performed with a 0.85 multiplier applied to the TLO MAPLHGR limit resulting in an SLO MAPLHGR limit of 10.625 kW/ft. The analyses are performed at BOL ATRIUM-10 fuel conditions. The limiting SLO LOCA is the 0.45 ft^2 split pump discharge line break with SF-BATT|BA and a mid-peaked axial power shape. The PCT for this case is 1673°F. Other key results and event times for the limiting SLO LOCA are provided in Tables 8.1 and 8.2, respectively.

Table 8.3 shows the spectrum of SLO analyses and the PCT for each case. A comparison of the limiting SLO and the limiting TLO results is provided in Table 8.4. The results in Table 8.4 show that the limiting TLO LOCA results bound the limiting SLO results when a 0.85 multiplier is applied to the TLO MAPLHGR limit.

Table 8.1 Results for Limiting SLO Recirculation Line Break 0.45 ft^2 Split Pump Discharge SF-BATT|BA Mid-Peaked Axial [ ]

PCT	1673°F
Maximum local cladding oxidation	0.45%
Maximum planar average MWR	0.21%

Table 8.2 Event Times for Limiting SLO Recirculation Line Break 0.45 ft^2 Split Pump Discharge SF-BATT|BA Mid-Peaked Axial [ ]

Event	Time (sec)
Initiate break	0.0
Initiate scram	0.5
Low-Low liquid level, L2 (448 in)	19.0
Low-Low-Low liquid level, L1 (372.5 in)	31.5
Jet pump uncovers	41.3
Recirculation suction uncovers	65.8
Lower plenum flashes	79.5
LPCS high-pressure cutoff	213.3
LPCS valve pressure permissive	204.9
LPCS valve starts to open	206.9
LPCS valve fully open	239.9
LPCS permissive for ADS timer	60.5
LPCS pump at rated speed	63.5
LPCS flow starts	213.3
RDIV pressure permissive	234.5
RDIV starts to close	236.5
RDIV fully closed	272.5
Rated LPCS flow	285.3
ADS valves open	180.5
Blowdown ends	285.3
Bypass reflood	377.4
Core reflood	337.9
PCT	337.9

Table 8.3 SLO Recirculation Line Break Spectrum Results for [ ] SF-BATT|BA

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Table 8.4 Single- and Two-Loop Operation PCT Summary

Operation	Limiting Case	PCT (°F)
Single-loop	0.45 ft^2 split pump discharge mid-peaked SF-BATT|BA	1673
Two-loop	0.25 ft^2 split pump discharge top-peaked SF-BATT|BB	1973

9.0 Long-Term Coolability

Long-term coolability addresses the issue of reflooding the core and maintaining a water level adequate to cool the core and remove decay heat for an extended time period following a LOCA. For non-recirculation line breaks, the core can be reflooded to the top of the active fuel and be adequately cooled indefinitely. For recirculation line breaks, the core will initially remain covered following reflood due to the static head provided by the water filling the jet pumps to a level of approximately two-thirds core height. Eventually, the heat flux in the core will not be adequate to maintain a two-phase water level over the entire length of the core. Beyond this time, the upper third of the core will remain wetted and adequately cooled by core spray. Maintaining water level at two-thirds core height with one core spray system operating is sufficient to maintain long-term coolability as demonstrated by the NSSS vendor (Reference 7).

10.0 Conclusions

The major conclusions of this LOCA break spectrum analysis are:

- The limiting recirculation line break is a 0.25 ft^2 split break in the pump discharge piping with single failure SF-BATT and a top-peaked axial shape for two-loop operation.
- The limiting break analysis identified above satisfies all acceptance criteria specified in 10 CFR 50.46. The analysis is performed in accordance with 10 CFR 50.46 Appendix K requirements.
  - Peak PCT < 2200°F (1973°F).
  - Local cladding oxidation thickness < 0.17 (0.0184).
  - Total hydrogen generation < 0.01. The break spectrum analysis determined a maximum planar average MWR of 0.0129. The core-wide metal-water reaction (CMWR) is significantly less than the maximum planar average MWR. It is concluded that CMWR would be less than 0.01 total hydrogen generation. CMWR is explicitly evaluated in the follow-on MAPLHGR report.
  - Coolable geometry, satisfied by meeting peak PCT, local cladding oxidation, and total hydrogen generation criteria.
  - Core long-term cooling, satisfied by concluding core flooded to top of active fuel or core flooded to the jet pump suction elevation with one core spray operating.
- Breaks in the non-recirculation lines are less limiting than the most severe break in the recirculation line.
- The MAPLHGR limit multiplier for SLO is 0.85 for ATRIUM-10 fuel. This multiplier ensures that a LOCA from SLO is less limiting than a LOCA from two-loop operation.

The limiting break characteristics determined in this report can be referenced and used in future Browns Ferry Units 1, 2, and 3 LOCA analyses to establish the MAPLHGR limit versus exposure for ATRIUM-10 fuel.

11.0 References

1. EMF-2361(P)(A) Revision 0, *EXEM BWR-2000 ECCS Evaluation Model,* Framatome ANP, May 2001.

2. XN-CC-33(A) Revision 1, *HUXY: A Generalized Multirod Heatup Code with 10 CFR 50 Appendix K Heatup Option Users Manual*, Exxon Nuclear Company, November 1975.

3. XN-NF-82-07(P)(A) Revision 1, *Exxon Nuclear Company ECCS Cladding Swelling and Rupture Model*, Exxon Nuclear Company, November 1982.

4. XN-NF-81-58(P)(A) Revision 2 and Supplements 1 and 2, *RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model*, Exxon Nuclear Company, March 1984.

5. EMF-2292(P)(A) Revision 0, *ATRIUM™-10: Appendix K Spray Heat Transfer Coefficients*, Siemens Power Corporation, September 2000.

6. ANP-2912(P) Revision 0, *Browns Ferry Units 1, 2, and 3 105% OLTP LOCA Parameters Document,* AREVA NP Inc., March 2010.

7. NEDO-20566A, *Analytical Model for Loss-of-Coolant Analysis in Accordance with 10 CFR Appendix K,* General Electric Company, September 1986.

Appendix A Computer Codes

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Appendix B TLO Recirculation Line Break, 1.0 DEG Supplemental Information

Appendix C TLO Recirculation Line Break, 0.45 ft^2 Split Supplemental Information

Appendix D TLO Recirculation Line Break, 0.07 ft^2 Split Supplemental Information

Appendix E Non-Recirculation Line Break, 0.4 ft^2 LPCS Line Supplemental Information

Appendix F Non-Recirculation Line Break, 0.32 ft^2 Feedwater Line Supplemental Information

Appendix G Non-Recirculation Line Break, 0.01 ft^2 Steam Line Break Supplemental Information