ATTACHMENT 15

ANP-3384NP, Browns Ferry Units 1, 2, and 3 LOCA-ECCS Analysis MAPLHGR Limits for ATRIUM-10 Fuel (EPU) (Non-Proprietary)

ANP-3384NP
Revision 3

Browns Ferry Units 1, 2, and 3 LOCA-ECCS Analysis MAPLHGR Limits for ATRIUM-10 Fuel (EPU)

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AREVA Inc.

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Nature of Changes

Item	Page	Description and Justification
1.	7-1	Updated Reference 1 to the latest revision.

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Nomenclature

ADS	automatic depressurization system
ANS	American Nuclear Society
BWR	boiling water reactor
CFR	Code of Federal Regulations
CMWR	core average metal-water reaction
DEG	double-ended guillotine
ECCS	emergency core cooling system
EOB	end of blowdown
EPU	extended power uprate
HPCI	high-pressure coolant injection
LHGR	linear heat generation rate
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
MWR	metal-water reaction
NRC	Nuclear Regulatory Commission, U.S.
OLTP	original licensed thermal power
PCT	peak cladding temperature
RDIV	recirculation discharge isolation valve
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

Nomenclature *(Continued)*

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
TCD	thermal conductivity degradation

1.0 Introduction

The results of the loss-of-coolant accident emergency core cooling system (LOCA-ECCS) analyses for Browns Ferry Units 1, 2 and 3 are documented in this report. The purpose of the LOCA-ECCS analysis is to specify the maximum average planar linear heat generation rate (MAPLHGR) limit versus exposure for ATRIUM™-10* fuel and to demonstrate that the MAPLHGR limit is adequate to ensure that the LOCA-ECCS criteria in 10 CFR 50.46 are satisfied for operation at or below the limit. The report also documents the licensing basis peak cladding temperature (PCT) and corresponding local cladding oxidation from the metal water reaction (MWR) for ATRIUM-10 fuel used at Browns Ferry Units 1, 2, and 3.

The analyses documented in this report were performed with LOCA Evaluation Models developed by AREVA NP and approved for reactor licensing analyses by the U.S. Nuclear Regulatory Commission (NRC). The models and computer codes used by AREVA for LOCA analyses are collectively referred to as the EXEM BWR-2000 Evaluation Model. The EXEM BWR-2000 Evaluation Model and NRC approval are documented in Reference 2. A summary description of the LOCA analysis methodology is provided in Section 4.0.

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The application of the EXEM BWR Evaluation Model for the Browns Ferry LOCA break spectrum analysis is documented in Reference 1. The LOCA conditions evaluated in Reference 1 include break size, type, location, axial power shape, and ECCS single failure. The limiting LOCA break characteristics identified in Reference 1 are presented below:

Limiting LOCA Break Characteristics	
Location	recirculation discharge pipe
Type / size	split / 0.23 ft^2
Single failure	battery (DC) power, board A
Axial power shape	top-peaked
Initial State	102% power / **[ ]**

* ATRIUM is a trademark of AREVA Inc.

The LOCA break spectrum analysis documented in Reference 1 was based on a generic ATRIUM 10XM neutronic design at beginning of life EPU conditions. The ATRIUM 10XM break spectrum analysis is the basis for the limiting combination of break size, single failure, and power shape. This limiting combination and the limiting ATRIUM 10XM average core boundary conditions calculated are used to determine the PCT and MWR of the ATRIUM-10 fuel type at EPU conditions.

This calculation supports a limited amount of previously exposed ATRIUM-10 assemblies that may be included in a transition cycle with co-resident ATRIUM 10XM fuel type operating at EPU conditions. At EPU power, any ATRIUM-10 fuel would be in its third cycle of operation. [

] The heatup analyses were performed using the fluid conditions from the LOCA analysis based on the limiting ATRIUM 10XM average core with an ATRIUM-10 hot channel analysis.

Calculations assumed an initial core power of 102% of 3952 MWt as per NRC requirements. 3952 MWt corresponds to 120% of the original licensed thermal power (OLTP) and is referred to as extended power uprate (EPU).

2.0 Summary

The MAPLHGR limit was determined by applying the EXEM BWR-2000 Evaluation Model for the analysis of the limiting LOCA event. The exposure-dependent MAPLHGR limit for ATRIUM-10 fuel is shown in Figure 2.1. [

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The response of the reactor system and hot channel during the limiting LOCA analysis are presented in Section 5.0. The MAPLHGR analysis results for the limiting lattice design are presented in Section 5.0. The peak cladding temperature (PCT) and metal-water reaction (MWR) results for the ATRIUM-10 limiting lattice design are presented in Table 2.1.

The SLO analyses (Reference 1) support operation with an ATRIUM-10 MAPLHGR multiplier of 0.85 applied to the normal two-loop operation MAPLHGR limit. The results of these calculations confirm that the LOCA acceptance criteria in the Code of Federal Regulations (10 CFR 50.46) are met for operation at or below these limits.

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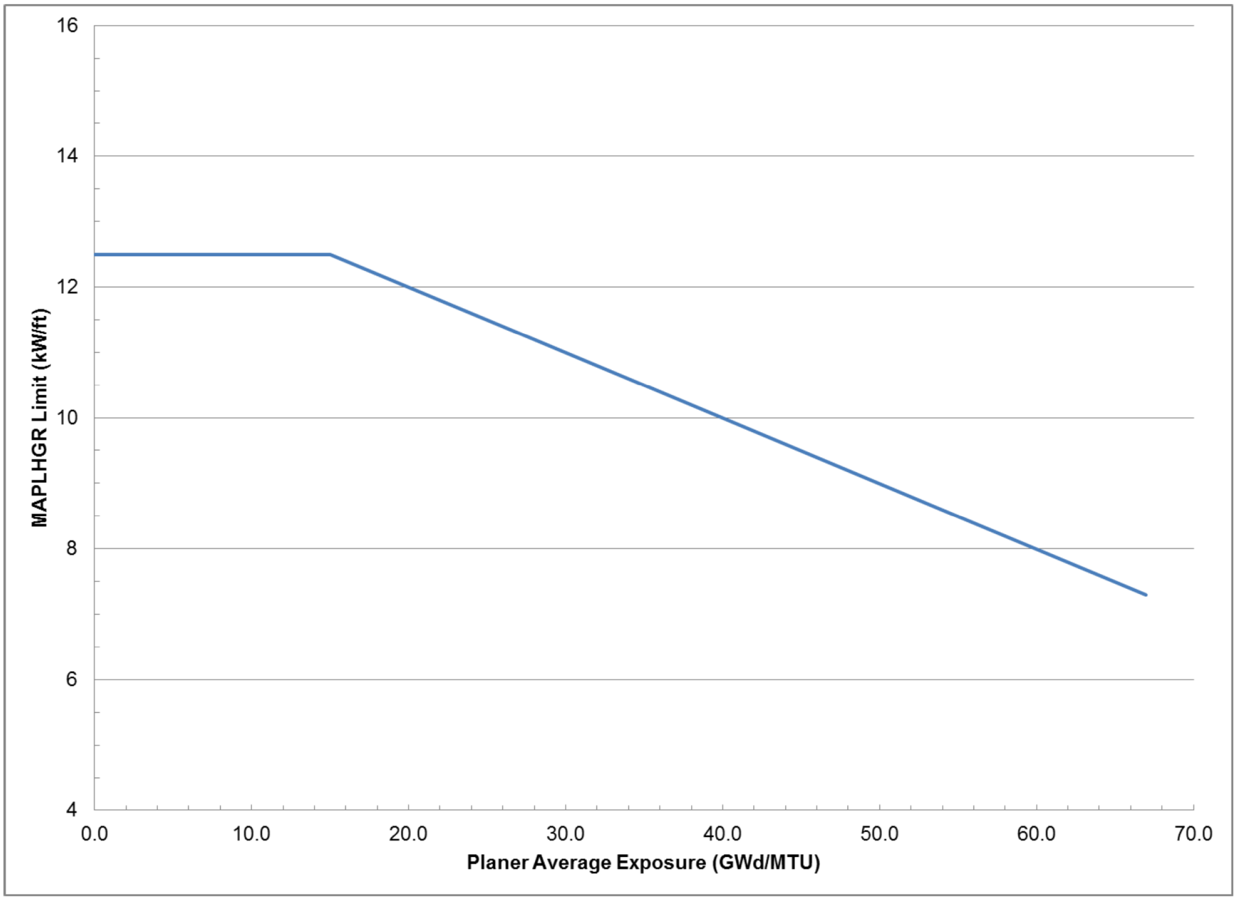
Table 2.1 LOCA Results for PCT Limiting Conditions

Parameter	ATRIUM-10
Exposure (GWd/MTU)	0.0
Peak cladding temperature (°F)	2086
Local cladding oxidation (max %)	2.51
Planar average oxidation (max %)	1.29
Total hydrogen generated (% of total hydrogen possible)	<1.0 (<0.36)



Average Planar Exposure (GWd/MTU)	ATRIUM-10 MAPLHGR (kW/ft)
0	12.5
15	12.5
67	7.3

Figure 2.1 MAPLHGR Limit for ATRIUM-10 Fuel

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. The results of the Browns Ferry ATRIUM 10XM break spectrum calculations using EXEM BWR-2000 LOCA methodology are summarized in Reference 1 and were the basis for the identification of the limiting combination of break size, single failure, and power shape.

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. Later in the blowdown, core cooling is provided by lower plenum flashing as the system continues to depressurize. 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 2. 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 each ATRIUM-10 fuel type to ensure that these criteria are met. For jet pump BWRs, the most challenging criterion is that PCT must not exceed 2200°F.

LOCA analysis results demonstrating that the PCT, local oxidation, and hydrogen generation criteria are met are provided in Section 5.0. Compliance with these three criteria ensures that a coolable geometry is maintained. Compliance with the long-term coolability criterion is discussed in Reference 1.

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 2. 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 3). 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 2) 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 4.3. This nodalization is consistent with that used in the topical report submitted to the NRC (Reference 2).

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 4.4 for a top-peaked 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 results from the RELAX hot channel calculation used as input to the HUXY heatup analysis are heat transfer coefficients and fluid conditions in the hot channel.

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 4) 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 5). 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 2. [ ] are 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 reactor (%MWR). The core average metal-water reaction (CMWR) criterion of less than 1.0% can often be satisfied by demonstrating that the maximum planar average MWR calculated by HUXY is less than 1.0%.

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4.4.1 [ ]

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4.5 *Plant Parameters*

The LOCA break spectrum analysis is performed using the plant parameters presented in Reference 7. 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 10XM fuel. The ATRIUM-10 hot channel analysis is performed based on the limiting break characteristics determined in the break spectrum analysis. Some of the key fuel parameters used in the hot channel analysis are summarized in Table 4.3. A top-peaked axial power shape, shown in Figure 4.6, was identified as the most conservative power shape for the limiting break (Reference 1).

4.6 *ECCS Parameters*

The ECCS configuration is shown in Figure 4.5. Tables 4.4 – 4.8 provide the important ECCS characteristics assumed in the LOCA break spectrum 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 line.

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. For the break spectrum analyses, no credit for ECCS flow is assumed until ECCS pumps reach rated speed.

The automatic depressurization system (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. [

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

The potentially limiting single failures of the ECCS are provided in Section 5.0 of Reference 1. Table 4.9 shows these failures and gives the ECCS systems that are available for each assumed failure.

Table 4.1 Initial Conditions*

Reactor power (% of rated)	102
Total core flow (% of rated)	[ ]
Reactor power (MWt)	4031
Total core flow (Mlb/hr)	[ ]
[ ]	
Steam flow rate (Mlb/hr)	16.85
Steam dome pressure (psia)	1053
Core inlet enthalpy (Btu/lb)	[ ]
ATRIUM-10 hot assembly MAPLHGR (kW/ft) ‡	12.5
[ ]	
ECCS fluid temperature (°F)	120
Axial power shape	Fig. 4.6

* The AREVA calculated heat balance is adjusted to match the 100% power/100% flow values given in the plant parameters document (Reference 7). The model is then rebalanced based on AREVA heat balance calculations to establish these LOCA initial conditions at 102% of rated thermal power.

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‡ The ATRIUM 10XM input used to generate the axial power shape in the average core RELAX calculation is provided in Table 4.1 of Reference 1.

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	ATRIUM-10*
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) 95 (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

* The ATRIUM 10XM input used in the average core RELAX calculation is provided in Table 4.3 of Reference 1.

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 ADS permissive (max) (sec) †	32‡	
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.

‡ Analyses assume the larger delay from LPCS (40 sec) for ADS permissive. Refer to Table 4.6.

§ Conservative value relative to specified value in Reference 7 (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 Sparger
0	6,785	13,570
105	5,285	10,570
200	3,685	7,370
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.

‡ Flow rates have been reduced to account for leakages associated with the Lower Sectional Hardware mod as discussed in GE Safety Communication SC 10-05. This further reduction applies only to the configuration at Unit 3 and can be conservatively applied to Units 1 and 2.

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)
Time Delays	
Delay time (from ADS initiating signal to time valves are opened)† (sec)	120

* Relative to vessel zero.

† ADS timer initiation occurs after L1 set point trip. ADS valves are opened after the timer has elapsed and LPCS or LPCI pumps reach the ADS ready permissive. Analyses assume the longer delay from LPCS for 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 (sec)	36

Table 4.9 ECCS Single Failure

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

* 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 values 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.

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

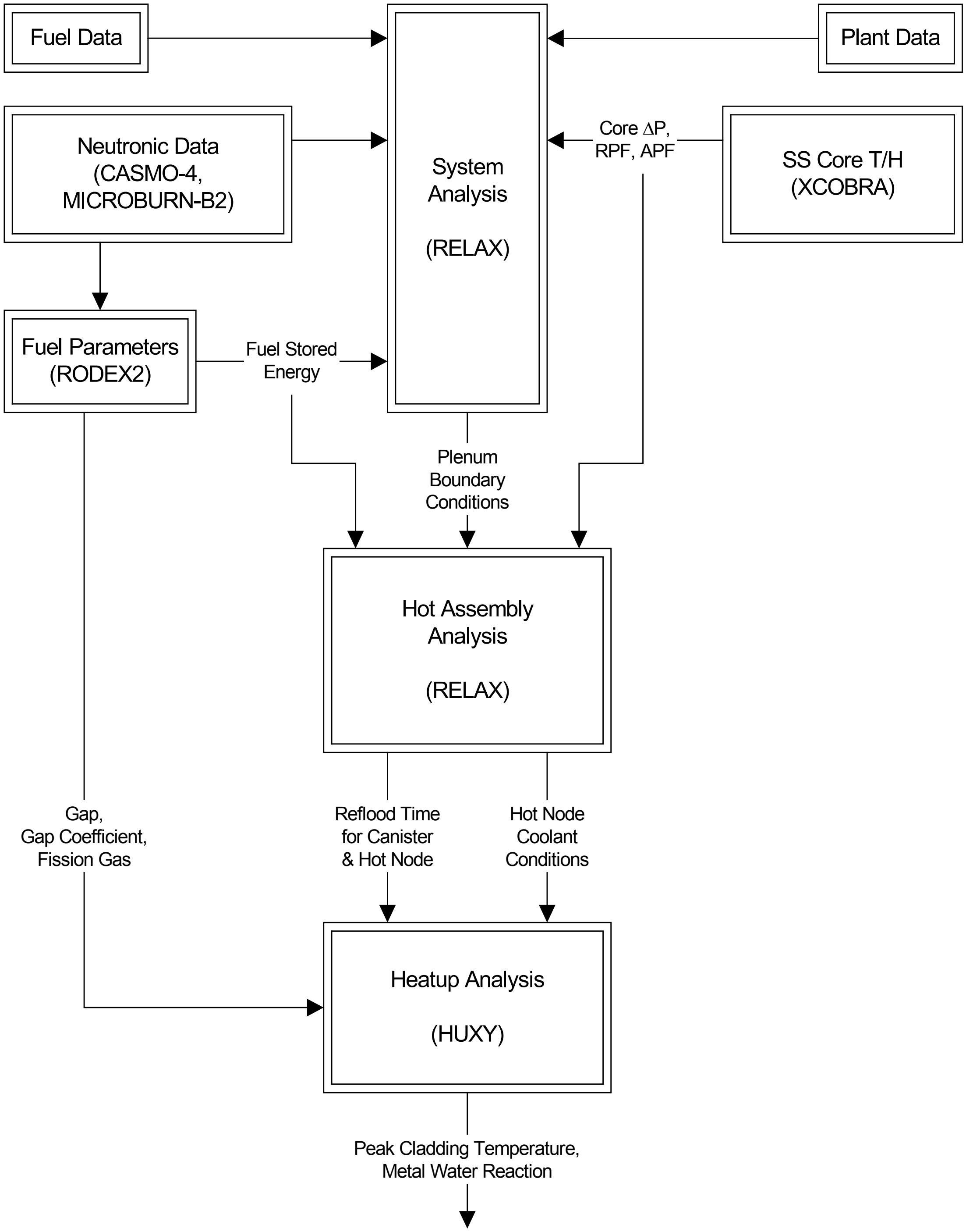

Figure 4.1 Flow Diagram for EXEM BWR-2000 ECCS Evaluation Model

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Figure 4.2 [

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Figure 4.3 RELAX System Blowdown Model

Figure 4.4 RELAX Hot Channel Blowdown Model
Top-Peaked Axial

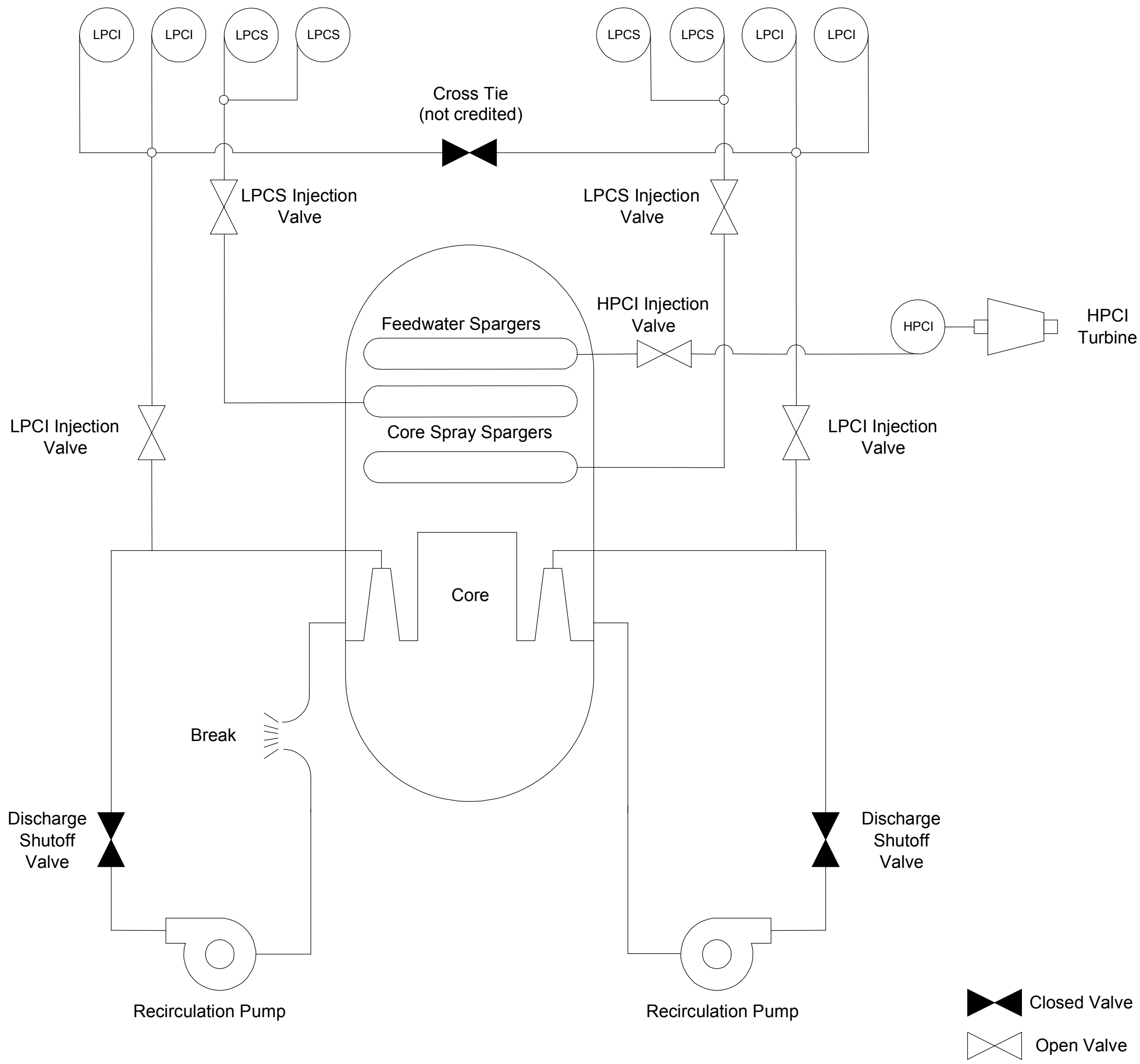

Figure 4.5 ECCS Schematic

**Figure 4.6 Axial Power Distribution
for Limiting LOCA Case in RELAX Calculation**

5.0 MAPLHGR Analysis Results

An exposure-dependent MAPLHGR limit for ATRIUM-10 fuel is obtained by performing HUXY heatup analyses using results from the limiting LOCA analysis case identified in Reference 1. The break characteristics for the limiting analysis are summarized in Section 1.0. Table 5.1 shows event times for the analysis. The response of the reactor system is shown in Figures 5.1 – 5.18. In the MAPLHGR analysis, the ATRIUM-10 fuel rod stored energy is set to be bounding at all exposures and the RELAX hot channel peak power node is modeled at the highest MAPLHGR, which is 102% of 12.5 kW/ft for the ATRIUM-10 fuel.

Table 5.2 shows the MAPLHGR analysis results for the ATRIUM-10 fuel. The HUXY model of the ATRIUM-10 fuel is applied to obtain these results as described in Section 4.3. The HUXY analysis is performed at 5 GWd/MTU exposure intervals for exposures between 0 and 65 GWd/MTU and an ending exposure of 67 GWd/MTU. The HUXY MAPLHGR input is consistent with the data in Figure 2.1. Exposure-dependent ATRIUM-10 fuel rod data is provided from RODEX2 results and includes gap coefficient, hot gap thickness, cold gap thickness, gas moles, fuel rod plenum length, and spring relaxation time. This data is provided as a function of linear heat generation rate at each exposure analyzed.

The ATRIUM-10 limiting PCT is 2086°F at the 0.0 GWd/MTU exposure. The corresponding maximum local cladding oxidation at the PCT limiting exposure is 2.51%. Analysis results show the CMWR is less than 1.0% total hydrogen generated.

Figure 5.19 shows the rod surface temperature of the ATRIUM-10 PCT rod as a function of time for the limiting break. The maximum temperature of 2086°F occurs at 426.5 seconds. These results demonstrate the acceptability of the ATRIUM-10 MAPLHGR limit shown in Figure 2.1.

5.1 *Thermal Conductivity Degradation*

The RODEX2 code was approved by the NRC in the early 1980s. At that time, thermal conductivity degradation with burnup was not well characterized by irradiation or post-irradiation testing. As a result, fuel codes at that time did not account for thermal conductivity degradation (TCD). In the past 20 years, requests to the NRC have been made for commercial fuel operation to increasingly higher burnup levels. This has resulted in renewed interest in the degree and nature of burnup-induced TCD. Interactions between AREVA and the NRC

(References 10-12) on this topic are summarized in this section, describing how TCD is addressed in the AREVA analyses for Browns Ferry.

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The impact of TCD is included in the results summarized in Table 5.2. The assessment for Browns Ferry ATRIUM-10 fuel shows that the PCT calculated at 0.0 GWd/MTU, which is not affected by TCD, is still the highest exposure-dependent PCT when the effects of TCD are included.

Table 5.1 Event Times for Limiting Break
0.23 ft^2 Split Pump Discharge SF-BATT|BA
Top-Peaked Axial 102% Power

Event	Time (sec)
Initiate break	0.0
Initiate scram	0.6
Low-Low liquid level, L2 (448 in)	30.4
Low-Low-Low liquid level, L1 (372.5 in)	49.8
Jet pump uncovers	75.0
Recirculation suction uncovers	124.4
Lower Plenum Flashes	167.6
LPCS high-pressure cutoff	258.4
LPCS valve pressure permissive	248.0
LPCS valve starts to open	250.0
LPCS valve fully open	283.0
LPCS permissive for ADS timer	78.8
LPCS pump at rated speed	81.8
LPCS flow starts	258.4
RDIV pressure permissive	287.7
RDIV starts to close	289.7
RDIV fully closed	325.7
Rated LPCS flow	353.2
ADS valves open	171.8
Blowdown ends	353.2
Bypass reflood	486.9
Core reflood	426.5*
PCT	426.5*

* Core reflood and time of PCT are the only event times determined in the ATRIUM-10 hot channel RELAX calculation. All others are determined in the average core RELAX calculation.

Table 5.2 ATRIUM-10 MAPLHGR
Analysis Results

Average Planar Exposure (GWd/MTU)	MAPLHGR (kW/ft)	PCT (°F)	Local Cladding Oxidation (%)
0.0	**12.50**	**2086**	**2.51**
5.0	12.50	2051	2.22
10.0	12.50	2025	1.88
15.0	12.50	2040	3.11
20.0	12.00	1961	1.51
25.0	11.50	2052	3.29
30.0	11.00	1937	2.84
35.0	10.50	1907	2.62
40.0	10.00	1871	2.31
45.0	9.50	1802	1.83
50.0	9.00	1734	1.43
55.0	8.50	1673	0.99
60.0	8.00	1605	0.44
65.0	7.50	1562	0.34
67.0	7.30	1546	0.31
CMWR is <1.0% at all exposures.			

**Figure 5.1 Limiting Break
Upper Plenum Pressure**

**Figure 5.2 Limiting Break
Total Break Flow Rate**

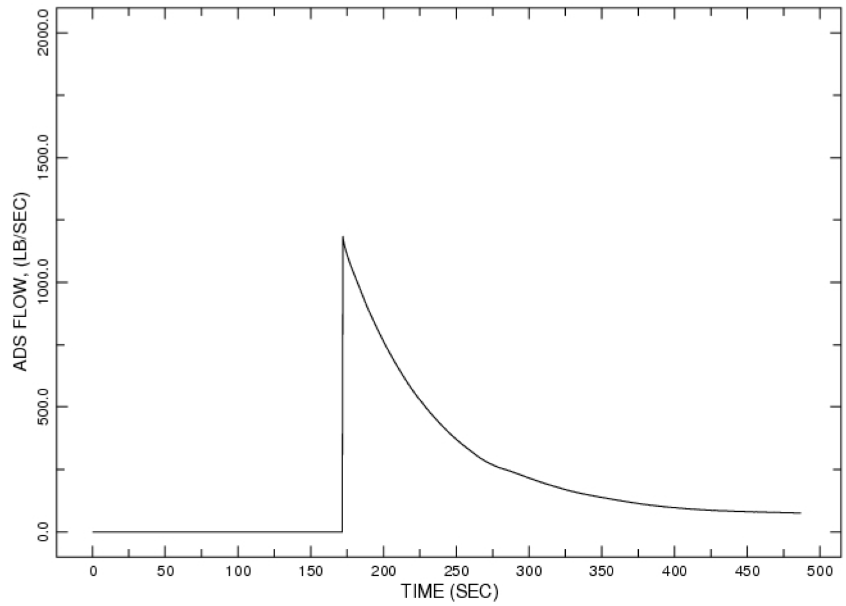

Figure 5.3 Limiting Break
ADS Flow Rate

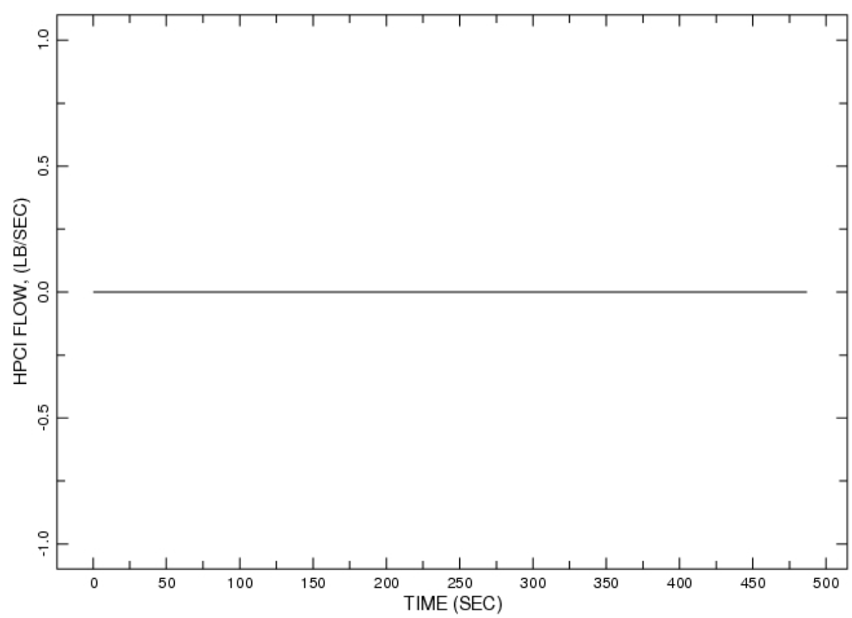

Figure 5.4 Limiting Break
HPCI Flow Rate

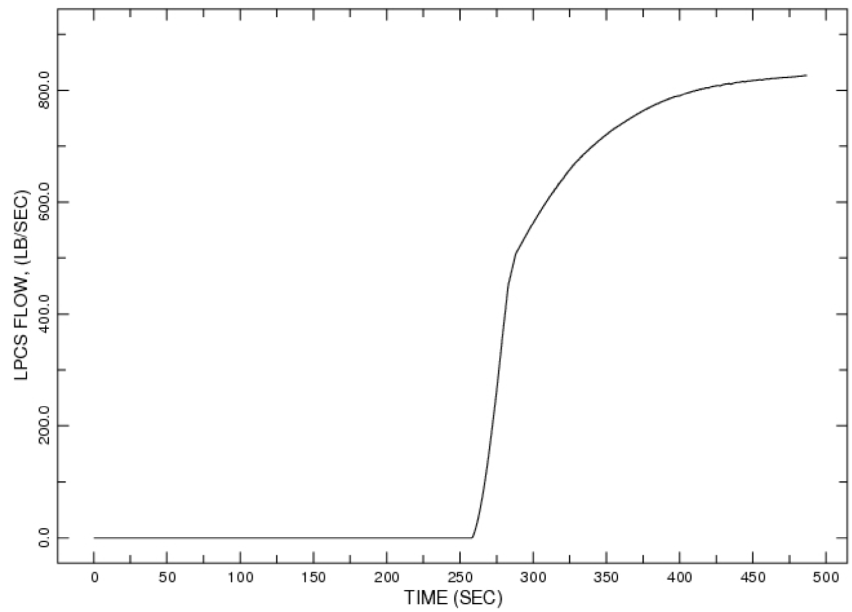

Figure 5.5 Limiting Break
LPCS Flow Rate

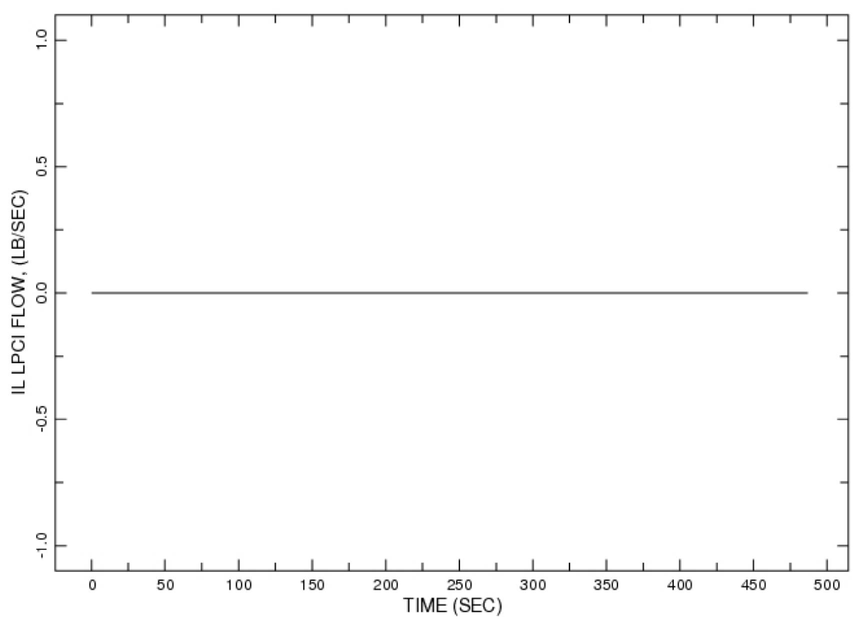

Figure 5.6 Limiting Break
Intact Loop LPCI Flow Rate

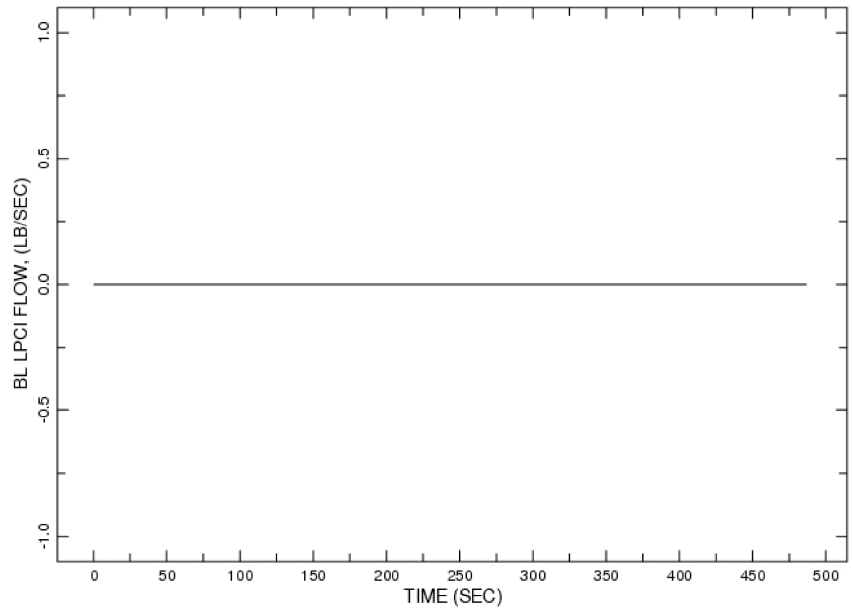

Figure 5.7 Limiting Break
Broken Loop LPCI Flow Rate

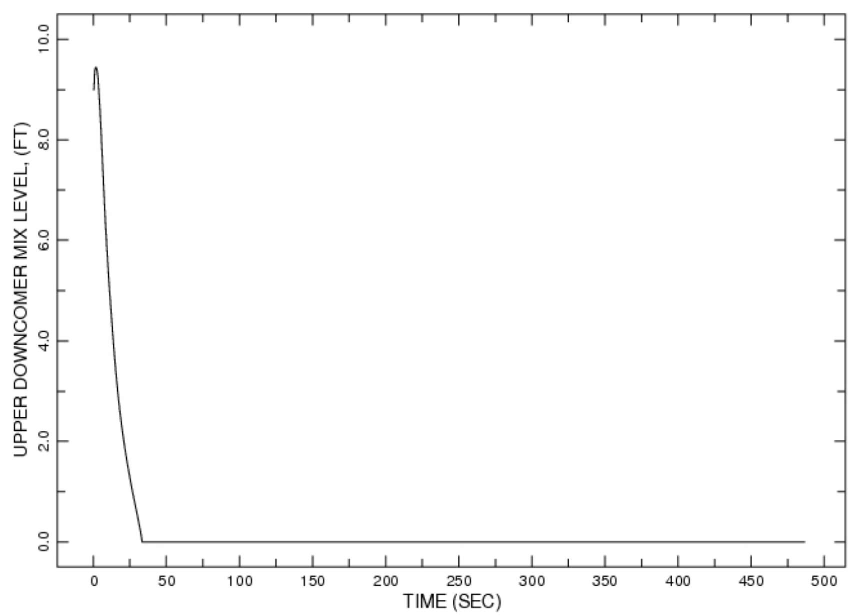

Figure 5.8 Limiting Break
Upper Downcomer Mixture Level

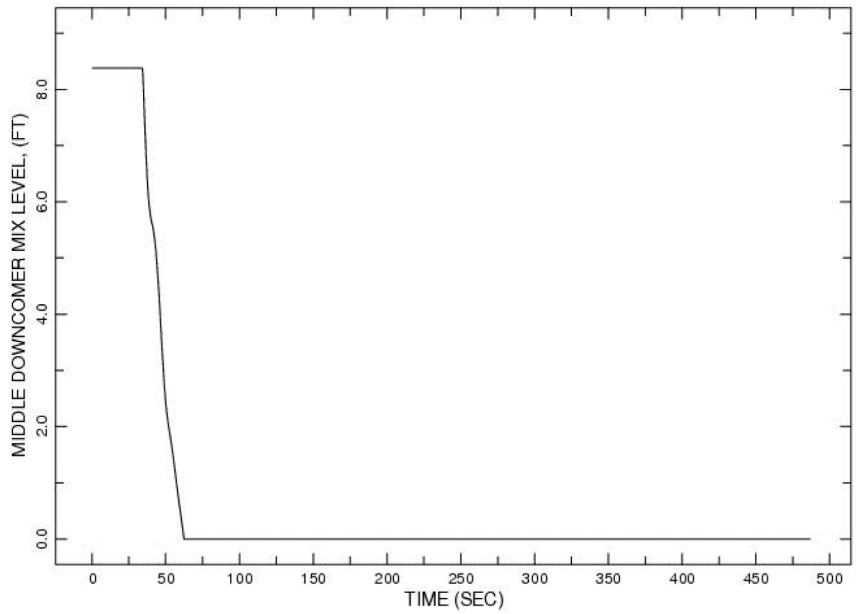

Figure 5.9 Limiting Break
Middle Downcomer Mixture Level

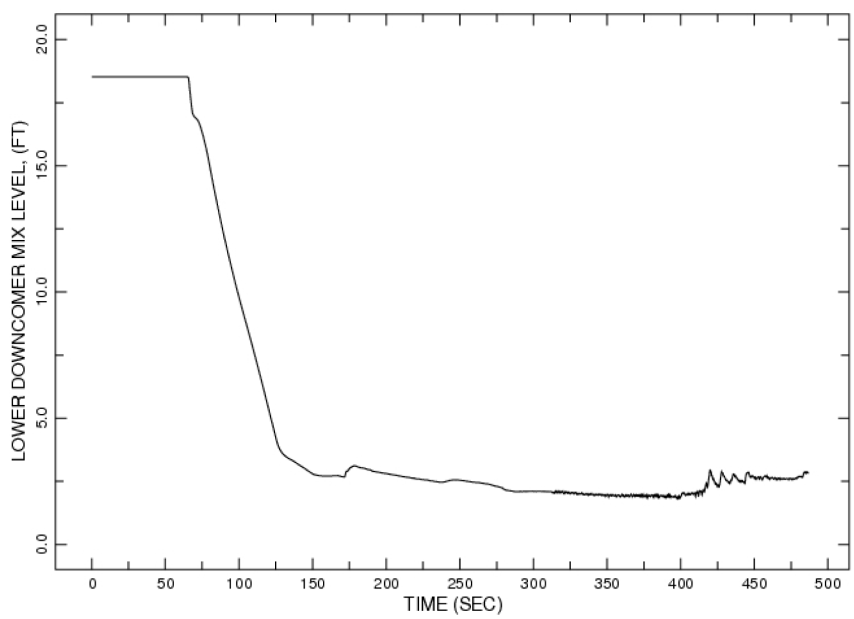

Figure 5.10 Limiting Break
Lower Downcomer Mixture Level

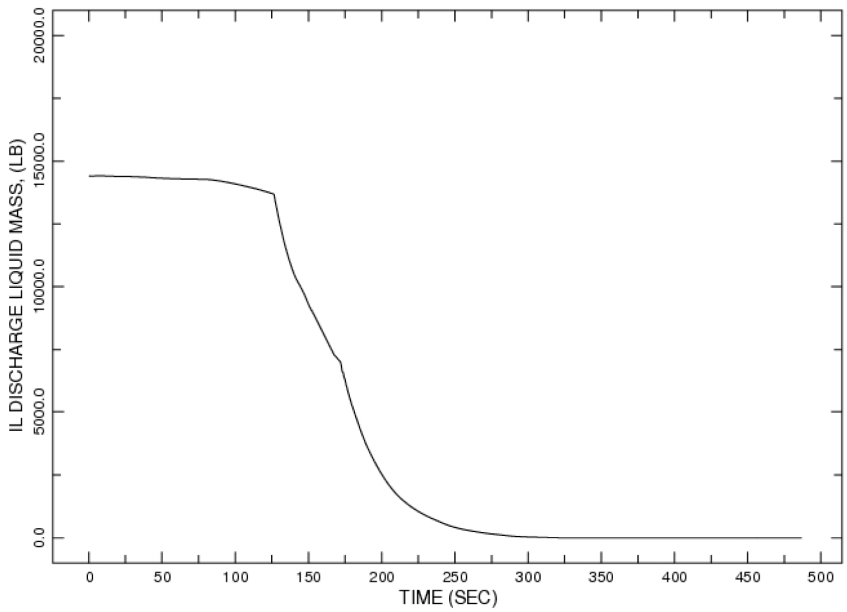

Figure 5.11 Limiting Break
Intact Loop Discharge Line Liquid Mass

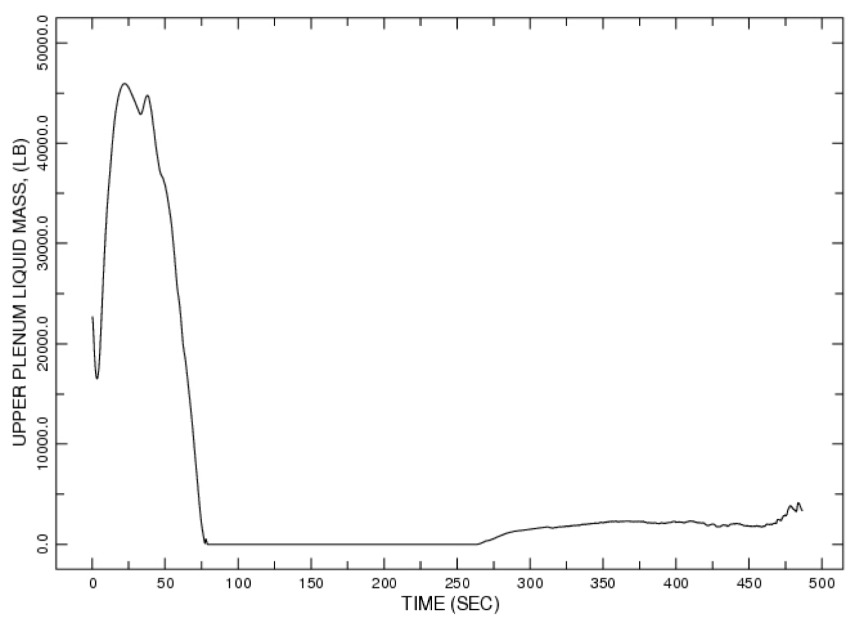

Figure 5.12 Limiting Break
Upper Plenum Liquid Mass

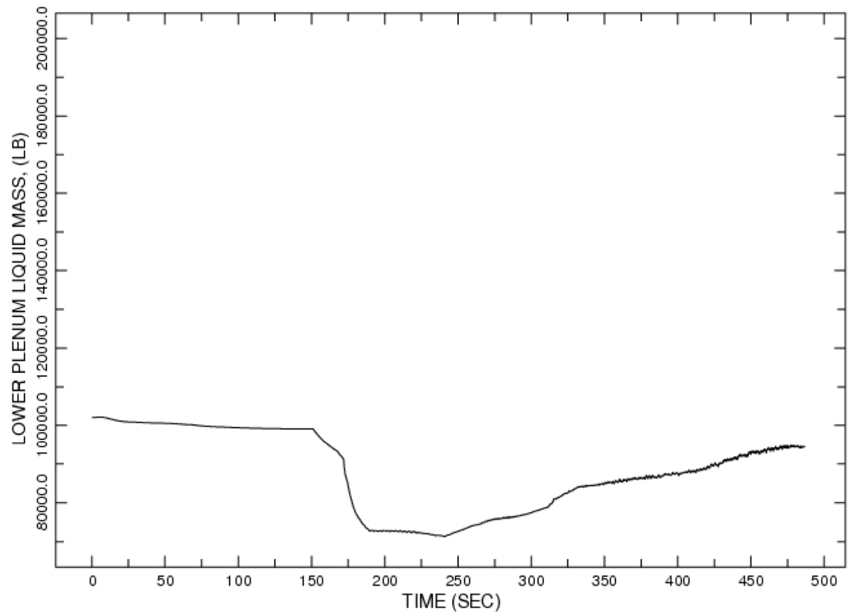

Figure 5.13 Limiting Break
Lower Plenum Liquid Mass

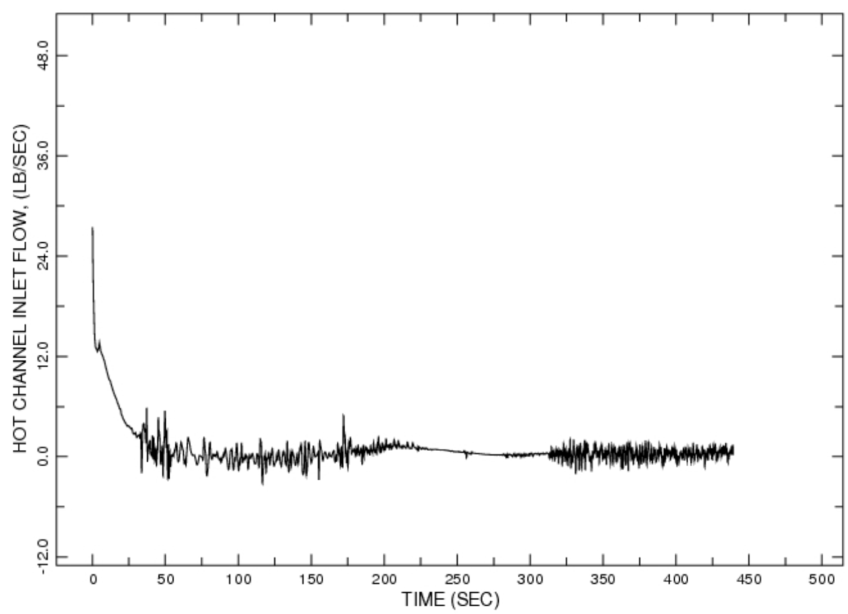

Figure 5.14 Limiting Break
Hot Channel Inlet Flow Rate

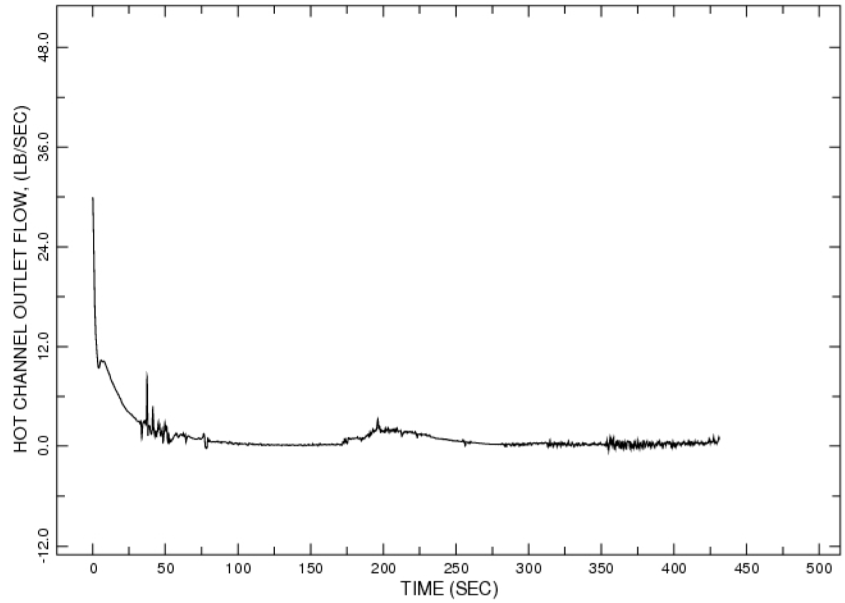

Figure 5.15 Limiting Break
Hot Channel Outlet Flow Rate

Figure 5.16 Limiting Break
Hot Channel Coolant Temperature
at the Limiting Node

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Figure 5.17 Limiting Break
Hot Channel Quality at the Limiting Node

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Figure 5.18 Limiting Break
Hot Channel Heat Transfer Coefficient
at the Limiting Node

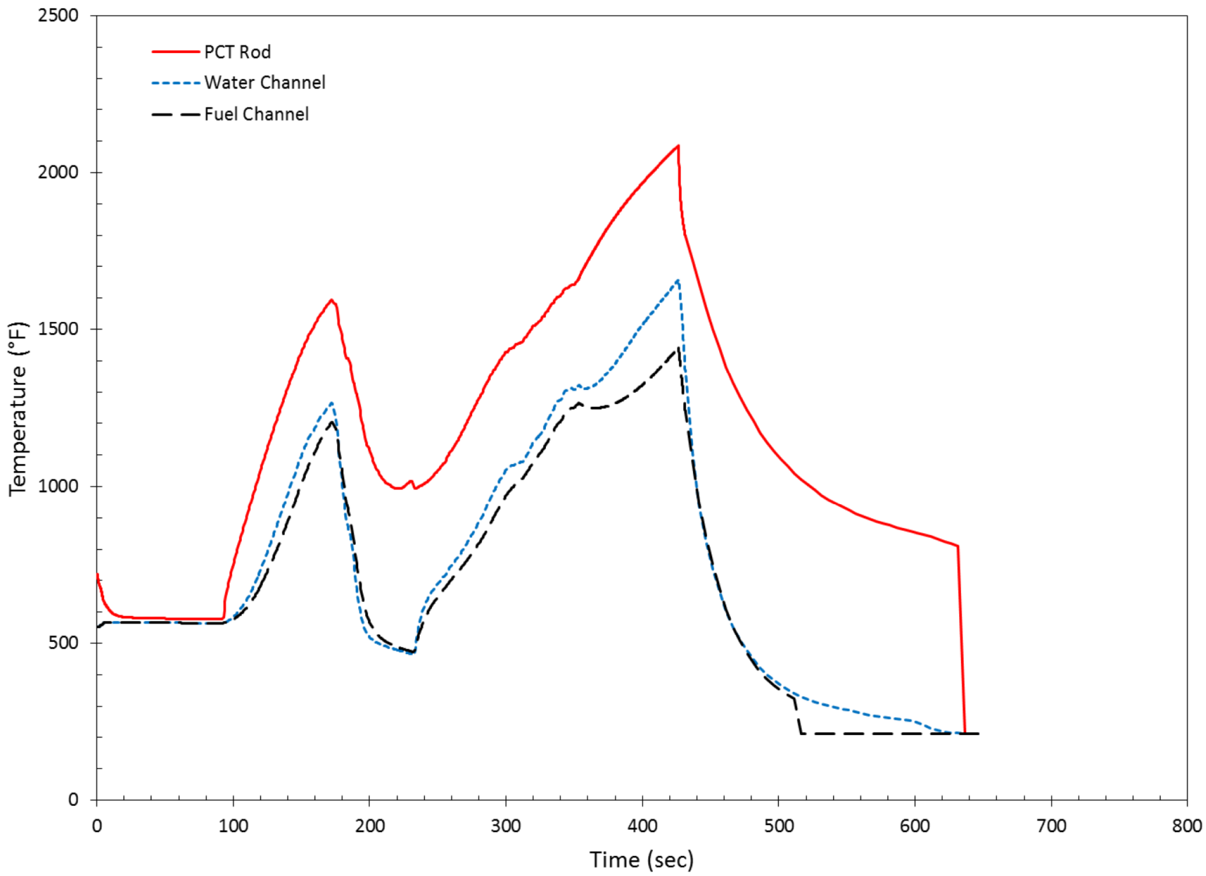

Figure 5.19 Limiting Break Cladding Temperatures

6.0 Conclusions

The EXEM BWR-2000 Evaluation Model was applied to determine the ATRIUM-10 MAPLHGR limit for Browns Ferry at EPU conditions. The following conclusions were made from the analyses presented.

- The acceptance criteria of the Code of Federal Regulations (10 CFR 50.46) are met for operation at or below the ATRIUM-10 MAPLHGR limit given in Figure 2.1.
  - Peak PCT < 2200°F (2086°F).
  - Local cladding oxidation thickness < 0.17 (0.0251)
  - Total hydrogen generation < 0.01. The hot channel MWR was determined to be 0.0036. The core-wide metal-water reaction (CMWR) is less than the hot channel MWR. It is concluded that CMWR would be less than 0.01 total hydrogen generation.
  - 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 (Reference 1).
- The MAPLHGR limit is applicable for ATRIUM-10 fuel in ATRIUM 10XM fuel transition cores.
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7.0 References

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13. BAW-10247PA Revision 0, *Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors*, AREVA NP, February 2008.

APPENDIX A ***SUPPLEMENTAL INFORMATION***

Table A.1 Computer Codes Used for MAPLHGR Limit Analysis