ATTACHMENT 6

NON-PROPRIETARY VERSION OF RLBLOCA SUMMARY REPORT

The following pages provide the non-proprietary version of the technical summary reports provided by Framatome supporting this license amendment request.

ANP-3944NP, "Callaway Realistic Large Break LOCA Analysis with GAIA Fuel Design," Revision 1, dated October 2022

[NON-PROPRIETARY REPORT]

104 pages follow this cover sheet

Callaway Realistic Large Break LOCA Analysis with GAIA Fuel Design

ANP-3944NP
Revision 1

Licensing Report

October 2022

Nature of Changes

Item	Revision No	Section(s) or Page(s)	Description and Justification
1	1	All	Updated Proprietary notations throughout document.

Contents

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Criteria (c) and (d) defined in associated affidavit for this document apply to bracketed material on this page.

Nomenclature

Acronym	**Definition**
AO	Axial Offset
CFR	Code of Federal Regulations
CHF	Critical Heat Flux
CSAU	Code Scaling, Applicability and Uncertainty
CWO	Core-Wide Oxidation
ECCS	Emergency Core Cooling System
ECR	Equivalent Cladding Reacted
EM	Evaluation Model
EMDAP	Evaluation Model Development and Assessment Process
$F_{\Delta H}$	Nuclear Enthalpy Rise Factor/Radial Peaking Factor
F_Q	Total Peaking Factor/Global Peaking Factor
Framatome	Framatome Inc.
FSRR	Fuel Swell Rupture and Relocation
Gd_2O_3	Gadolinia or Gad
GDC	General Design Criteria
HHSI	High Head Safety Injection
HMP	High Mechanical Performance
HTC	Heat Transfer Coefficient
IGM	Intermediate GAIA Mixing Grid
IHSI	Intermediate Head Safety Injection
k(z)	Axial-Dependent Peaking Factor
LBLOCA	Large Break Loss-of-Coolant Accident
LCO	Limiting Condition of Operation
LHGR	Linear Heat Generation Rate
LHSI	Low Head Safety Injection

Acronym	**Definition**
LOCA	Loss-of-Coolant Accident
LOOP	Loss-of-Offsite Power
MLO	Maximum Local Oxidation
No-LOOP	No Loss of Offsite Power
NRC	U. S. Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
PCT	Peak Clad Temperature
PIRT	Phenomena Identification and Ranking Table
PWR	Pressurized Water Reactor
RAI	Request for Additional Information
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RLBLOCA	Realistic Large Break Loss of Coolant Accident
SE	Safety Evaluation
SG	Steam Generator
SI	Safety Injection
SIAS	Safety Injection Actuation Signal
SRM	Swelling and Rupture Model
TS	Technical Specification
UTL	Upper Tolerance Limit

1.0 INTRODUCTION

This report summarizes the Realistic Large Break Loss-of-Coolant Accident (RLBLOCA) analysis for Callaway Nuclear Plant Unit 1 (Callaway). The purpose of the RLBLOCA analysis is to support the Vendor Qualification Program (VQP) for Callaway with the Framatome GAIA fuel design. This analysis was performed in accordance with the U.S. Nuclear Regulatory Commission (NRC)-approved S-RELAP5-based methodology described in Reference 1 with the noted exception in Section 3.3.

Callaway is a 4-loop, Westinghouse-designed Pressurized Water Reactor (PWR). The Framatome GAIA fuel design with $M5_{Framatome}$ cladding for Callaway consists of a 17x17 array with GAIA and Intermediate GAIA Mixing (IGM) grids, a lower High Mechanical Performance (HMP) grid and an upper HMP grid. The fuel assembly includes a MONOBLOC guide tube design, $M5_{Framatome}$ fuel rod design and a GRIP lower nozzle. The fuel is standard UO_2 fuel with 2, 4, 6, and 8 weight-percent Gadolinia (Gd_2O_3) rods included.

The analysis assumes full-power operation at a core power level of 3636 MWt (including measurement uncertainty), a maximum-allowed total peaking factor (F_Q) of 2.50 (represents total peaking with an axial-dependent factor k(z) set to 1.0), a radial peaking factor ($F_{\Delta H}$) of 1.65 (includes uncertainty), and up to 5% steam generator (SG) tube plugging per SG. This analysis also addresses typical operational ranges or technical specification (TS) limits (whichever is applicable) with regard to [

]. The analysis explicitly analyzes fresh and once-burned fuel assemblies. The plant parameter specification for this analysis is provided in Table 4-1. The analysis uses the Fuel Swelling, Rupture, and Relocation (FSRR) model to determine if cladding rupture occurs and evaluate the consequences of FSRR on the transient response.

[ ]

2.0 SUMMARY OF RESULTS

The UTL results providing 95/95 simultaneous coverage from this evaluation meet the 10 CFR 50.46(b) criteria with a PCT of 1561°F, a maximum local oxidation of 2.35 percent and a total core-wide oxidation of 0.028 percent. The PCT of 1561°F occurred in a once-burned 2 weight-percent Gadolinia rod with an assembly burnup of 26.2 GWd/mtU. The results of the analysis demonstrate the adequacy of the ECCS to support the 10 CFR 50.46(b) (1-3) criteria (Reference 2).

3.0 DESCRIPTION OF ANALYSIS

3.1 *Acceptance Criteria*

The purpose of the analysis is to verify the adequacy of the Callaway ECCS by demonstrating compliance with the following 10 CFR 50.46(b) criteria (Reference 2):

1. The calculated maximum fuel element cladding temperature shall not exceed 2200°F.

2. The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation.

3. 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, excluding the cladding surrounding the plenum volume, were to react.

The final two criteria, coolable geometry and long-term cooling, are treated in separate plant-specific evaluations.

Note: The original 17% value in the second acceptance criterion for MLO was based on the usage of the Baker-Just correlation. For present reviews on ECCS Evaluation Model (EM) applications, the NRC staff imposed a limitation specifying that the equivalent cladding reacted (ECR) results calculated using the Cathcart-Pawel correlation are considered acceptable in conformance with 10 CFR 50.46(b)(2) if the ECR value is less than 13% (Section 3.3.3, NRC Final Safety Evaluation (SE) for EMF-2103(P) Rev. 3). The limitation is addressed in Table 3-1.

3.2 *Description of LBLOCA Event*

A Large Break Loss-of-Coolant Accident (LBLOCA) is initiated by a postulated rupture of the Reactor Coolant System (RCS) primary piping. The most challenging break location is in the cold leg piping between the reactor coolant pump and the reactor. The plant is assumed to be operating normally at full power prior to the accident and the break is assumed to open instantaneously. A worst case single-failure is also assumed to occur during the accident. The single-failure for this analysis, as defined in the EM, is the loss of one ECCS pumped injection train without the loss of containment spray.

The LBLOCA event is typically described in three phases: blowdown, refill, and reflood. Following the initiation of the break, the blowdown phase is characterized by a sudden depressurization from operating pressure down to the saturation pressure of the hot leg fluid. For larger cold leg breaks, an immediate flow reversal and stagnation occurs in the core due to flow out the break, which causes the fuel rods to pass through critical heat flux (CHF), usually within one second following the break. Following this initial rapid depressurization, the RCS depressurizes at a more gradual rate. Reactor trip and emergency injection signals occur when either the low pressure setpoint or the containment high-pressure setpoint are reached. However, for LBLOCA, reactor trip and scram are essentially inconsequential, as reactor shutdown is accomplished by moderator reactivity feedback. During blowdown, core cooling is supported by the natural evolution of the RCS flow pattern as driven by the break flow.

When the system pressure falls below the accumulator pressure, flow from the accumulator is injected into the cold legs ending the blowdown period and initiating the refill period. Once the system pressure falls below the respective shutoff heads of the safety injection systems and the system startup time delays are met, flow from the pumped safety injection systems is injected into the RCS. While some of the ECCS flow bypasses the core and goes directly out of the break, the downcomer and lower plenum gradually refill until the mixture in the lower head and lower plenum regions reaches the bottom of the active core and the reflood period begins. Core cooling is supported by the natural evolution of the RCS flow pattern as driven by the break flow and condensation in the RCS promoted by safety injection. Towards the end of the refill period, heat transfer from the fuel rods is relatively low, steam cooling and rod-to-rod radiation being the primary mechanisms of core heat removal.

Once the lower plenum is refilled to the bottom of the fuel rod heated length, refill ends and the reflood phase begins. Substantial ECCS fluid is retained in the downcomer during refill. This provides the driving head to move coolant into the core. As the mixture level moves up the core, steam is generated and liquid is entrained, providing cooling in the upper core regions. The two-phase mixture extends into the upper plenum and some liquid may de-entrain and flow downward back into the cooler core regions. The remaining entrained liquid passes into the steam generators where it vaporizes, adding to the steam that must be discharged through the break and out of the system. The difficulty of venting steam is, in general, referred to as steam binding. It acts to impede core reflood rates. With the initiation of reflood, a quench front starts to progress up the core. With the advancement of the quench front, the cooling in the upper regions of the core increases, eventually arresting the rise in fuel rod surface temperatures. Later the core is quenched and a pool cooling process is established that can maintain the cladding temperature near saturation, so long as the ECCS makes up for the core boil off.

3.3 *Description of Analytical Models*

The NRC-approved RLBLOCA methodology is documented in EMF-2103(P)(A) *Realistic Large Break LOCA Methodology for Pressurized Water Reactors* (Reference 1). The methodology follows the Code Scaling, Applicability and Uncertainty (CSAU) evaluation methodology (Reference 3) and the requirements of the Evaluation Model Development and Assessment Process (EMDAP) documented in Reference 4. The CSAU method outlines an approach for defining and qualifying a best-estimate thermal-hydraulic code and quantifies the uncertainties in a Loss-of-Coolant Accident (LOCA) analysis.

The Framatome RLBLOCA methodology evaluation model for event response of the primary and secondary systems and the hot fuel rod used in this analysis is based on the use of two computer codes.

- COPERNIC for computation of the initial fuel stored energy, fission gas release, and the transient fuel-cladding gap conductance.
- S-RELAP5 for the thermal-hydraulic system calculations (includes ICECON for containment response).

The methodology (Reference 1) has been reviewed and approved by the NRC to perform LBLOCA analyses. However, a difference from the approved Reference 1 LBLOCA methodology was included in this analysis, as described below. This difference has been presented in recent NRC-approved RLBLOCA analyses (References 5 and 6).

The governing two-fluid (plus non-condensable) model with conservation equations for mass, energy, and momentum transfer is used. The reactor core is modeled in S-RELAP5 with heat generation rates determined from reactor kinetics equations (point kinetics) with reactivity feedback, and with actinide and decay heat.

The two-fluid formulation uses a separate set of conservation equations and constitutive relations for each phase. The effects of one phase on the other are accounted for by interfacial friction and heat and mass transfer interaction terms in the equations. The conservation equations have the same form for each phase; only the constitutive relations and physical properties differ.

The modeling of plant components is performed by following guidelines developed to ensure accurate accounting for physical dimensions and that the dominant phenomena expected during the LBLOCA event are captured. The basic building blocks for modeling are hydraulic volumes for fluid paths and heat structures for heat transfer. In addition, special purpose components exist to represent specific components such as the Reactor Coolant Pumps (RCPs) or the SG separators. All geometries are modeled at the resolution necessary to best resolve the flow field and the phenomena being modeled within practical computational limitations.

The analysis considers blockage effects due to clad swelling and rupture as well as increased heat load due to fuel relocation in the ballooned region of the cladding in the prediction of the hot fuel rod PCT.

A typical calculation using S-RELAP5 begins with the establishment of a steady-state initial condition with all loops intact. The input parameters and initial conditions for this steady-state calculation are chosen to reflect plant technical specifications or to match measured data. Additionally, the COPERNIC code provides initial conditions for the S-RELAP5 fuel models. Specific parameters are discussed in Section 3.6.

Following the establishment of an acceptable steady-state condition, the transient calculation is initiated by introducing a break into one of the loops. The evolution of the transient through blowdown, refill, and reflood is computed continuously using S-RELAP5. Containment pressure is calculated by the ICECON module within S-RELAP5.

A detailed assessment of the S-RELAP5 computer code was made through comparisons to experimental data. These assessments were used to develop quantitative estimates of the ability of the code to predict key physical phenomena in a PWR LBLOCA. The final step of the best-estimate methodology is to combine all the uncertainties related to the code and plant parameters and estimate values for the first three criteria of 10 CFR 50.46(b) with a probability of at least 95 percent with 95 percent confidence. The steps taken to derive the uncertainty estimate are summarized below:

1. Base Plant Input File Development

   First, base COPERNIC and S-RELAP5 input files for the plant (including the containment input file) are developed. The code input development guidelines documented in Appendix A of Reference 1 are applied to ensure that model nodalization is consistent with the model nodalization used in the code validation.

2. Sampled Case Development

   The statistical approach requires that many “sampled” cases be created and processed. For every set of input created, each “key LOCA parameter” is randomly sampled over a range established through code uncertainty assessment or expected operating limits (provided by plant technical specifications or data). Those parameters considered "key LOCA parameters" are listed in Table A-6 of Reference 1. This list includes both parameters related to LOCA phenomena, based on the PIRT provided in Reference 1, and to plant operating parameters. The uncertainty ranges associated with each of the model parameters are provided in Table A-7 of Reference 1.

3. Determination of Adequacy of ECCS

   The RLBLOCA methodology uses a non-parametric statistical approach to determine that the first three criteria of 10 CFR 50.46(b) are met with a probability higher than 95 percent with 95 percent confidence.

3.4 *GDC-35 Limiting Condition Determination*

General Design Criteria (GDC)-35 requires that a system be designed to provide abundant core cooling with suitable redundancy such that the capability is maintained in either the LOOP or No-LOOP conditions. [

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3.5 *Overall Statistical Compliance to Criteria*

[

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3.6 *Plant Description*

The plant analyzed is the Callaway Nuclear Plant Unit 1, Westinghouse-designed PWR, which has four loops, each with a hot leg, a U-tube steam generator, and a cold leg with an RCP. The Reactor Coolant System (RCS) includes one pressurizer connected to a hot leg. The ECCS provides injection to each of the four loops via the centrifugal charging/high head safety injection (HHSI) system, SI/intermediate head safety injection (IHSI) system, residual heat removal (RHR)/low head safety injection (LHSI) system, and accumulators. The RLBLOCA transients are of sufficiently short duration that the switchover to sump cooling water for ECCS pumped injection does not need to be considered.

The RCS, reactor vessel, pressurizer, and ECCS are explicitly modeled in the S-RELAP5 model. For the LOCA analysis, IHSI and HHSI are modeled as a combined system and identified as HHSI. For each RCS loop, the ECCS model includes an injection connection to the cold leg for the accumulator, another connection for HHSI, and another connection for LHSI. The ECCS injection connections to the cold leg pipes are downstream of the RCP discharge. The ECCS pumped injection is modeled as a table of flow versus backpressure. Also modeled is the secondary-side steam generator that is instantaneously isolated (closed main steam isolation valve and feedwater trip) at the time of the break. The primary and secondary coolant systems for Callaway were nodalized consistent with code input guidelines in Appendix A of Reference 1. System nodalization details are shown in Figure 3-1 through Figure 3-3.

The results used to demonstrate compliance with the 10 CFR 50.46(b) criteria are only applicable to the Framatome fuel product. However, the analysis includes considerations for the mixed core scenario. [

]

As described in Section 3.3, many parameters associated with LBLOCA phenomenological uncertainties and plant operation ranges are sampled. Values for process or operational parameters, including ranges of sampled process parameters, and fuel design parameters used in this analysis are given in Table 4-1. Table 4-2 presents a summary of the uncertainties used in the analysis. Two parameters (refueling water storage tank temperature and diesel start time) are set at conservative bounding values for all calculations. The passive heat sinks and material properties used in the containment input model are provided in Table 4-3.

3.7 *Safety Evaluation Limitations*

Except for the differences noted in Section 3.3, the RLBLOCA analysis for Callaway presented herein is consistent with the submitted RLBLOCA methodology documented in EMF-2103(P)(A), Revision 3 (Reference 1). The limitation and conditions from the NRC SE (Reference 1) for EMF-2103(P)(A), Revision 3, are addressed in Table 3-1.

Table 3-1
EMF-2103(P)(A), Revision 3, SE Limitations Evaluation

	Limitations (Sub-sections of Section 4.0 in Ref. 1)	Response
1	This EM was specifically reviewed in accordance with statements in EMF-2103, Revision 3. The NRC staff determined that the EM is acceptable for determining whether plant-specific results comply with the acceptance criteria set forth in 10 CFR 50.46(b), paragraphs (1) through (3). AREVA did not request, and the NRC staff did not consider, whether this EM would be considered applicable if used to determine whether the requirements of 10 CFR 50.46(b)(4), regarding coolable geometry, or (b)(5), regarding long-term core cooling, are satisfied. Thus, this approval does not apply to the use of SRELAP5-based methods of evaluating the effects of grid deformation due to seismic of LOCA blowdown loads, or for evaluating the effects of reactor coolant system boric acid transport. Such evaluations would be considered separate methods.	This analysis applies only to the acceptance criteria set forth in 10 CFR 50.46(b), paragraphs (1) through (3).
2	EMF-2103, Revision 3, approval is limited to application for 3-loop and 4-loop Westinghouse-designed nuclear steam supply systems (NSSSs), and to Combustion Engineering-designed NSSSs with cold leg ECCS injection, only. The NRC staff did not consider model applicability to other NSSS designs in its review.	Callaway is a 4-loop Westinghouse-designed NSSS with cold leg ECCS injection.
3	The EM is approved based on models that are specific to AREVA proprietary M5 fuel cladding. The application of the model to other cladding types has not been reviewed.	The analysis supports operation with $M5_{Framatome}$ cladding.
4	Plant-specific applications will generally be considered acceptable if they follow the modeling guidelines contained in Appendix A to EMF 2103, Revision 3. Plant-specific licensing actions referencing EMF 2103, Revision 3, analyses should include a statement summarizing the extent to which the guidelines were followed, and justification for any departures.	The modeling guidelines contained in Appendix A of EMF-2103(P)(A), Revision 3 (Reference 1) were followed completely for the analysis described in this report.

Limitations (Sub-sections of Section 4.0 in Ref. 1)		Response
5	The response to RAI 15 indicates that the fuel pellet relocation packing factor is derived from data that extend to currently licensed fuel burnup limits (i.e., rod average burnup of [ ]). Thus, the approval of this method is limited to fuel burnup below this value. Extension beyond rod average burnup of [ ] would require a revision or supplement to EMF-2103, Revision 3, or plant-specific justification.	The analysis burnups applied in this analysis do not exceed the rod average burnup of [ ].
6	The response to RAI 15 indicates that the fuel pellet relocation packing factor is derived from currently available data. Should new data become available to suggest that fuel pellet fragmentation behavior is other than that suggested by the currently available database, the NRC may request AREVA to update its model to reflect such new data.	The analysis uses the approved EMF-2103(P)(A), Revision 3 (Reference 1) relocation packing factor application. [ ]
7	The regulatory limit contained in 10 CFR 50.46(b)(2), requiring cladding oxidation not to exceed 17 percent of the initial cladding thickness prior to oxidation, is based on the use of the Baker-Just oxidation correlation. To account for the use of the Cathcart-Pawel correlation, this limit shall be reduced to 13 percent, inclusive of pre-transient oxide layer thickness.	The MLO UTL is less than 13% (Table 4-4). [ ]

	Limitations (Sub-sections of Section 4.0 in Ref. 1)	Response
8	In conjunction with Limitation 7 above, Cathcart-Pawel oxidation results will be considered acceptable, provided plant-specific [ ] If second-cycle fuel is identified in a plant-specific analysis, whose [ ], the NRC staff reviewing the plant-specific analysis may request technical justification or quantitative assessment, demonstrating that [ ]	All second cycle fuel rod [ ]
9	The response to RAI 13 states that all operating ranges used in a plant-specific analysis are supplied for review by the NRC in a table like Table B-8 of EMF-2103, Revision 3. In plant-specific reviews, the uncertainty treatment for plant parameters will be considered acceptable if plant parameters are [ ], as appropriate . Alternative approaches may be used, provided they are supported with appropriate justification.	[ ]
10	[ ]	[ ] were not used in this analysis.

Limitations (Sub-sections of Section 4.0 in Ref. 1)		Response
11	Any plant submittal to the NRC using EMF-2103, Revision 3, which is not based on the first statistical calculation intended to be the analysis of record must state that a re-analysis has been performed and must identify the changes that were made to the evaluation model and/or input in order to obtain the results in the submitted analysis.	The present analysis is the first statistical application of EMF-2103, Revision 3 for this plant.

Figure 3-1
Primary System Noding

[ ]

Figure 3-2
Secondary System Noding

[ ]

Figure 3-3
Reactor Vessel Noding

Criteria (c) and (d) defined in associated affidavit for this document apply to bracketed material on this page.

4.0 RLBLOCA ANALYSIS

4.1 *RLBLOCA Results*

[ ]

For a simultaneous coverage/confidence level of 95/95, the UTL values, [ ], are a PCT of 1561°F, an MLO of 2.35 percent, and a CWO of 0.028 percent. The fraction of total hydrogen generated was not directly calculated; however, it is conservatively bounded by the calculated total core wide percent oxidation, which is well below the 1 percent limit.

A summary of the major input parameters for the demonstration case is provided in Table 4-5. The sequence of event times for the demonstration case is provided in Table 4-6. The heat transfer parameter ranges for the demonstration case are provided in Table 4-7. Table 4-8 [ ].

The analysis scatter plots for the case set are shown in Figure 4-1 through Figure 4-5. Figure 4-1 shows linear scatter plots of the key parameters sampled for all cases. These figures illustrate the parameter ranges used in the analysis. Visual examination of the linear scatter plots demonstrates that the spread and coverage of all of the values used is appropriate and within the uncertainty ranges listed in Table 4-2. Appendix A provides a listing of all the sampled input values for each case. Key results such as the PCT and event timings are also listed for the case set.

Figure 4-2 and Figure 4-3 show PCT scatter plots versus the time of PCT and versus break size, respectively. The scatter plots for the maximum local oxidation and total core-wide oxidation are shown in Figure 4-4 and Figure 4-5, respectively.

Figure 4-2 shows about 26% of cases have PCT during the blowdown phase (PCT time less than ~30 seconds). The next cluster of PCTs occurs during the early to late reflood period. Blowdown PCT cases are dominated by rapid RCS depressurization and stored energy content. Reflood PCT cases are dominated by decay heat removal capacity. In general, plants with high pressure accumulators inject early in the transient when the break flow is still high. The high pressure and high break flow drive some of this fluid to bypass the core, retarding the progression of the core reflood. This results in cases with PCTs in the reflood phase of the transient.

The high PCT cases in the upper part of Figure 4-2 are mainly influenced by the area of the break. This is demonstrated in Figure 4-3 which shows a general increasing trend in PCT with break size. From all sampled parameters, the break size is a dominant effect on PCT because of its influence on the rate of primary depressurization.

Figure 4-4 shows a correlation of MLO with PCT. Since the MLO includes the pre-transient oxidation, the MLO is not only a function of cladding temperature but also of time in cycle (burnup). [

] The CWO also shows a strong correlation to PCT as demonstrated in Figure 4-5, as higher PCT cases would have higher oxidation throughout the core.

The demonstration case is a reflood peak case with a PCT timing of 76 seconds. Figure 4-6 through Figure 4-17 show key parameters from the S-RELAP5 calculations for the demonstration case. The transient progression for the demonstration case follows that described in Section 3.2.

[

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4.2 *Conclusions*

This report describes and provides results from the RLBLOCA analysis for the Callaway VQP with the Framatome GAIA fuel design. The application of the Framatome RLBLOCA methodology involves developing input decks, executing the simulations that comprise the uncertainty analysis, retrieving PCT, MLO, and CWO information and determining the simultaneous UTL results for the criteria. [

] The UTL results providing a 95/95 simultaneous coverage/confidence level from this evaluation meet the 10 CFR 50.46(b) criteria with a PCT of 1561°F, a MLO of 2.35 percent and a CWO of 0.028 percent.

Table 4-1
RLBLOCA Analysis - Plant Parameter Values and Ranges

Plant Parameter		Parameter Value
1.0	**Plant Physical Description**	
	1.1 Fuel	
	a) Cladding outside diameter	0.374 in.
	b) Cladding inside diameter	0.329 in.
	c) Pellet outside diameter	0.3225 in.
	d) Initial Pellet density	[ ]
	e) Active fuel length	144 in.
	f) Gd_2O_3 concentrations	2, 4, 6, 8 weight-percent
	1.2 RCS	
	a) Flow resistance	Analysis
	b) Pressurizer location	[ ]
	c) Hot assembly location	Anywhere in core
	d) Hot assembly type	17x17
	e) SG tube plugging	5 percent
2.0	**Plant Initial Operating Conditions**	
	2.1 Reactor Power	
	a) Analyzed reactor power	3636[1] MWt
	b) F_Q	2.50[1]
	c) $F_{\Delta H}$	1.65[1]
	2.2 Fluid Conditions	
	a) Total Loop flow	139.3 Mlbm/hr $\leq$ M $\leq$ 162.6 Mlbm/hr
	b) RCS average temperature	581.75°F $\leq$ T $\leq$ 592.7°F
	c) Upper head temperature	~RCS Cold Leg Temperature[2]
	d) Pressurizer pressure	2189.3 psia $\leq$ P $\leq$ 2279.3 psia
	e) Pressurizer liquid level	13.5 percent $\leq$ L $\leq$ 95.5 percent
	f) Accumulator pressure	616.3 psia $\leq$ P $\leq$ 662.3 psia
	g) Accumulator liquid volume	810.2 $ft_3 \leq$ V $\leq$ 889.6 ft_3
	h) Accumulator temperature	50°F $\leq$ T $\leq$ 120°F[3]
	i) Accumulator resistance fL/D	As-built piping configuration
	j) Accumulator boron	2300 ppm

[1] Includes measurement uncertainty.

[2] Upper head temperature will change based on sampling of RCS temperature.

[3] Coupled with containment temperature.

Criteria (c) and (d) defined in associated affidavit for this document apply to bracketed material on this page.

Table 4-1
RLBLOCA Analysis - Plant Parameter Values and Ranges

(Continued)

	Plant Parameter	Parameter Value
3.0	**Accident Boundary Conditions**	
	a) Break location	Cold leg pump discharge
	b) Break type	Double-ended guillotine or split
	c) Break size (each side, relative to cold leg pipe area)	[ ]
	d) ECCS pumped injection temperature	100°F
	e) HHSI pump delay	17 s (No-LOOP) 29 s (LOOP)
	f) LHSI pump delay	32 s (No-LOOP) 44 s (LOOP)
	g) Initial containment pressure	14.3 psia
	h) Initial containment temperature	50°F ≤ T ≤ 120°F
	i) Containment sprays delay	0 s
	j) Containment spray water temperature	37°F
	k) LHSI Flow	

RCS Cold Leg Pressure (psia)	Broken Loop Flow (gpm)	Total Intact Loops Flow (gpm)
14.3	932.1	2848.4
34.3	865.1	2306.1
54.3	792.0	1744.9
74.3	711.0	1196.1
94.3	618.5	826.9
114.3	509.1	375.8
134.3	0.0	0.0

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Table 4-1
RLBLOCA Analysis - Plant Parameter Values and Ranges

(Continued)

Plant Parameter	Parameter Value

l) IHSI and HHSI Combined Flow

RCS Cold Leg Pressure (psia)	Broken Loop Flow (gpm)	Total Intact Loops Flow (gpm)
14.3	245.3	742.8
34.3	244.2	737.8
54.3	243.1	731.3
74.3	242.0	725.5
94.3	240.9	720.4
114.3	239.8	713.9
134.3	238.0	708.1
154.3	236.9	703.1
174.3	235.4	696.5
194.3	233.9	690.7
214.3	232.4	685.9
614.3	199.2	553.8

Table 4-2
Statistical Distribution Used for Process Parameters

[ ]	Lower Value	Upper Value	[ ]
	2189.3	2279.3	
	13.5	95.5	
	810.2	889.6	
	616.3	662.3	
	50	120	
	2.5	2.7	
	139.3	162.6	
	581.75	592.7	

Table 4-3
Passive Heat Sinks and Material Properties in Containment Geometry

Heat Sink	Surface Area, ft^2	Thickness, ft	Material
Heat Structure #1	64919.0	0.021	Carbon Steel
		4.0	Concrete
Heat Structure #2	34129.0	0.021	Carbon Steel
		3.0	Concrete
Heat Structure #3	13538.0	1.5	Concrete
		0.021	Carbon Steel
		10.0	Concrete
Heat Structure #4	8564.0	1.0	Concrete
Heat Structure #5	43497.0	2.0	Concrete
Heat Structure #6	17061.0	2.5	Concrete
Heat Structure #7	7821.0	0.021	Carbon Steel
		2.0	Concrete
Heat Structure #8	8708.0	0.021	Stainless Steel
		2.0	Concrete
Heat Structure #9	8081.0	0.0001083	Zinc Coating
		0.005	Carbon Steel
		2.0	Concrete
Heat Structure #10	186183.0	0.0001083	Zinc Coating
		0.0104	Carbon Steel
Heat Structure #11	17746.0	0.0104	Carbon Steel
Heat Structure #12	114205.0	0.0208	Carbon Steel
Heat Structure #13	49101.0	0.0417	Carbon Steel
Heat Structure #14	31372.0	0.0833	Carbon Steel
Heat Structure #15	5631.0	0.1667	Carbon Steel
Heat Structure #16	8355.0	0.3333	Carbon Steel
Heat Structure #17	503.0	0.6667	Carbon Steel
Heat Structure #18	9726.0	0.0833	Carbon Steel
Heat Structure #19	35760.0	0.0104	Stainless Steel
Heat Structure #20	10885.0	0.0417	Stainless Steel

Heat Sink Material	Thermal Conductivity Btu/hr-ft-°F	Volumetric Heat Capacity Btu/ft3-°F
Concrete	1.2	30.0
Carbon Steel	30.0	54.0
Stainless Steel	10.0	60.0
Zinc Coating	65.0	41.0

Table 4-4
Compliance with 10 CFR 50.46(b)

UTL for 95/95 Simultaneous Coverage/Confidence		
Parameter	**Value**	**Case Number**
PCT (°F)	1561	[ ]
MLO (%)	2.35	[ ]
CWO (%)	0.028	[ ]

Characteristics of Case Setting the PCT UTL	
PCT (°F)	1561
PCT Rod Type	Once-Burned 2 weight-percent Gadolinia Rod
Time of PCT (s)	76.23
Elevation within Core (ft)	9.82
Local Maximum Oxidation (%)	1.90
Total Core-Wide Oxidation (%)	0.032
PCT Rod Rupture Time (s)	N/A
Rod Rupture Elevation within Core (ft)	N/A

[ ]

Table 4-5
Summary of Major Parameters for the Demonstration Case

Parameter	Value
Core Power (MWt)	3636
Time in Cycle (hrs)	2125
Limiting Rod Assembly Burnup (GWd/mtU)	26.2
Limiting Rod Equivalent, F_Q	2.48
Limiting Rod Radial Peak, $F_{\Delta H}$	1.65
Limiting Rod Axial Offset	0.295
Break Type	Guillotine
Break Size (ft^2/side)	3.4202
[ ]	[ ]

Table 4-6
Calculated Event Times for the Demonstration Case

Event	Time (sec)
Break Opens	0.0
RCP Trip	0.0
SIAS Issued	0.9
Start of Broken Loop Accumulator Injection	7.9
Start of Intact Loop Accumulator Injection [ ]	16.7, 16.7 and 16.7
SI Available	29.9
Broken Loop HHSI Delivery Began	29.9
Intact Loop HHSI Delivery Began [ ]	29.9, 29.9 and 29.9
Beginning of Core Recovery (Beginning of Reflood)	34.1
Broken Loop Accumulator Emptied	42.5
LHSI Available	44.9
Broken Loop LHSI Delivery Began	44.9
Intact Loop LHSI Delivery Began [ ]	44.9, 44.9 and 44.9
Intact Loop Accumulator Emptied [ ]	46.1, 46.3 and 45.6
PCT Occurred	76.2
Transient Calculation Terminated	782.6

Criteria (c) and (d) defined in associated affidavit for this document apply to bracketed material on this page.

Table 4-7
Heat Transfer Parameters for the Demonstration Case

Time (s)	[					]
LOCA Phase	Early Blowdown	Blowdown[1]	Refill	Reflood	Quench	Long Term Cooling[2]
Heat Transfer Mode	[					
Heat Transfer Correlations						
Maximum LHGR (kW/ft)						
Pressure (psia)						
Core Inlet Mass Flux (lbm/s-ft^2)						
Vapor[4] Reynolds Number						
Liquid Reynolds Number						
Vapor Prandtl Number						
Liquid Prandtl Number						
Vapor[5] Superheat (°F)						]

[1] End of blowdown considered as beginning of refill.

[2] Quench to End of Transient.

[3] [ ]

[4] Not important in pre-CHF heat transfer.

[5] Vapor superheat is meaningless during blowdown and system depressurization.

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Table 4-8
Fuel Rod Rupture Ranges of Parameters

[ ]

Figure 4-1
Scatter Plot Key Parameters

Figure 4-1
Scatter Plot Key Parameters (continued)

[ ]

Figure 4-2
PCT versus PCT Time Scatter Plot

[ ]

Figure 4-3
PCT versus Break Size Scatter Plot

[ ]

Criteria (c) and (d) defined in associated affidavit for this document apply to bracketed material on this page.

Figure 4-4
Maximum Local Oxidation versus PCT Scatter Plot

[ ]

Figure 4-5
Total Core Wide Oxidation versus PCT Scatter Plot

Criteria (c) and (d) defined in associated affidavit for this document apply to bracketed material on this page.

Figure 4-6
Demonstration Case - Peak Cladding Temperature (Independent of Elevation)

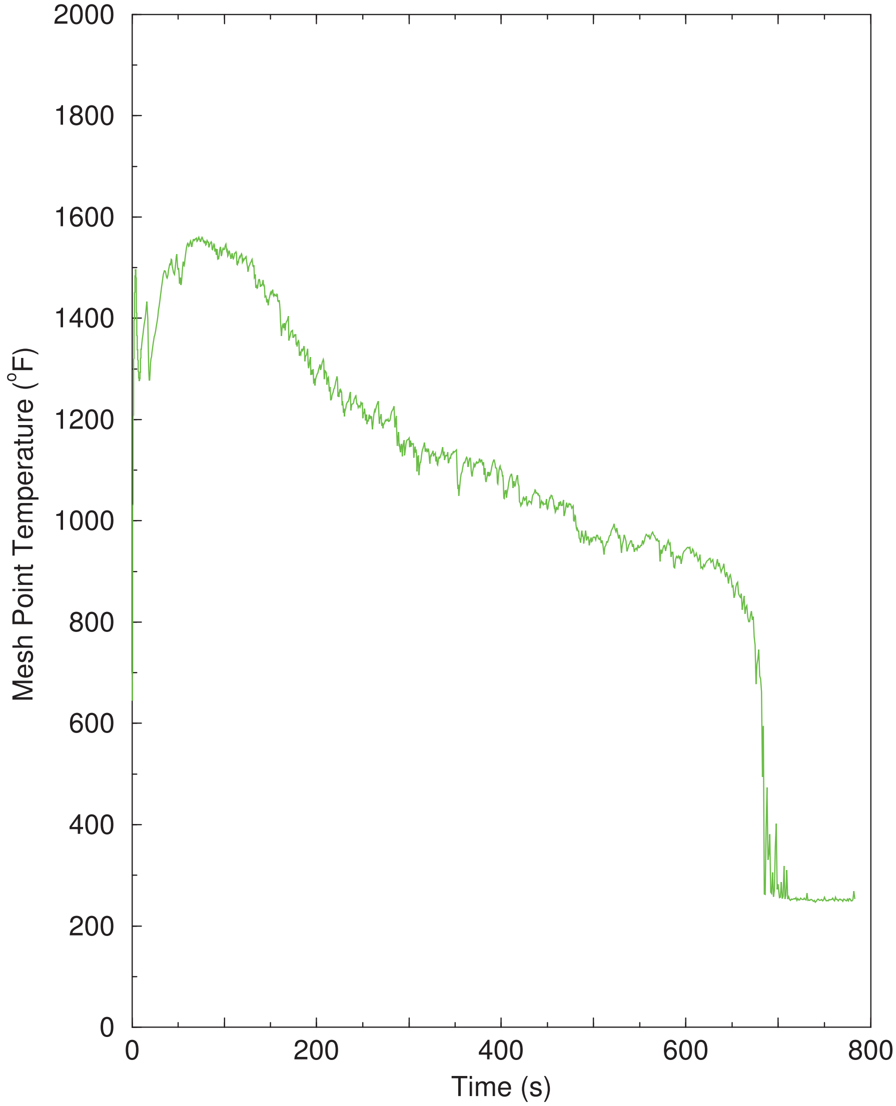

Figure 4-7
Demonstration Case - Break Flow

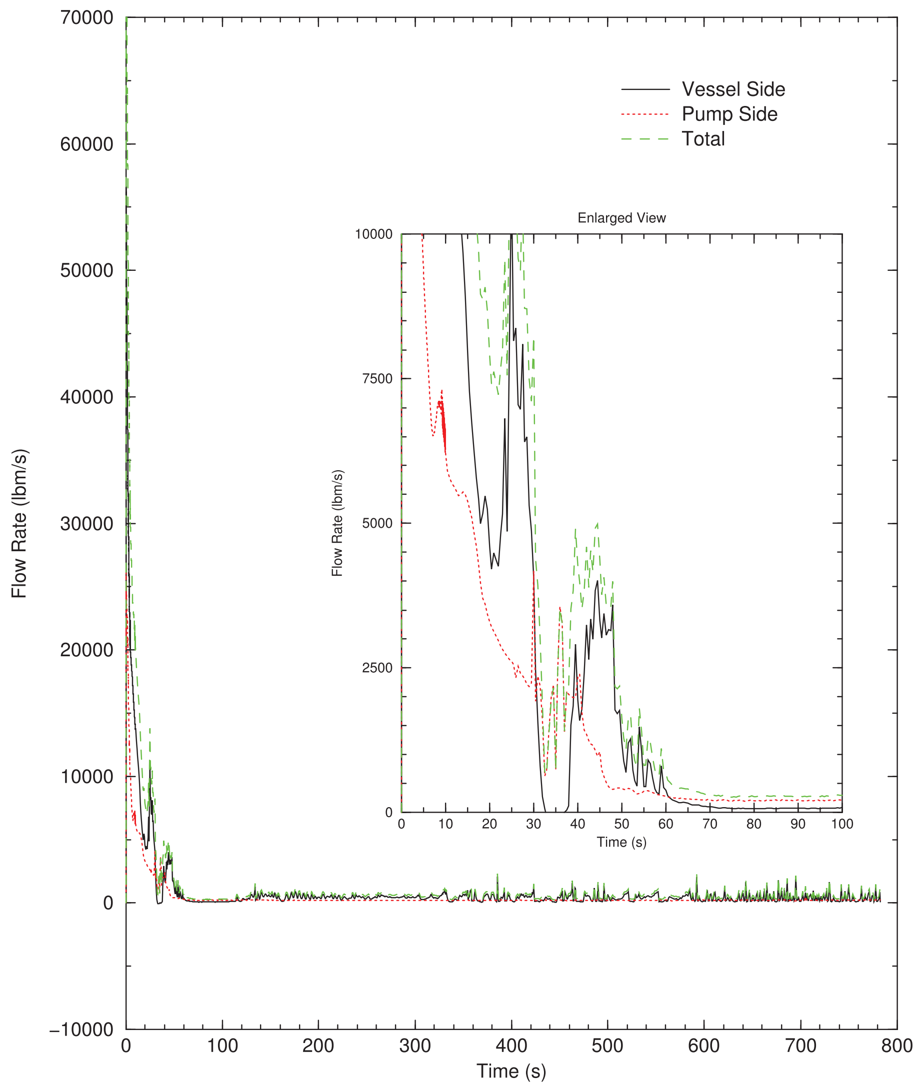

Figure 4-8
Demonstration Case - Core Inlet Mass Flux

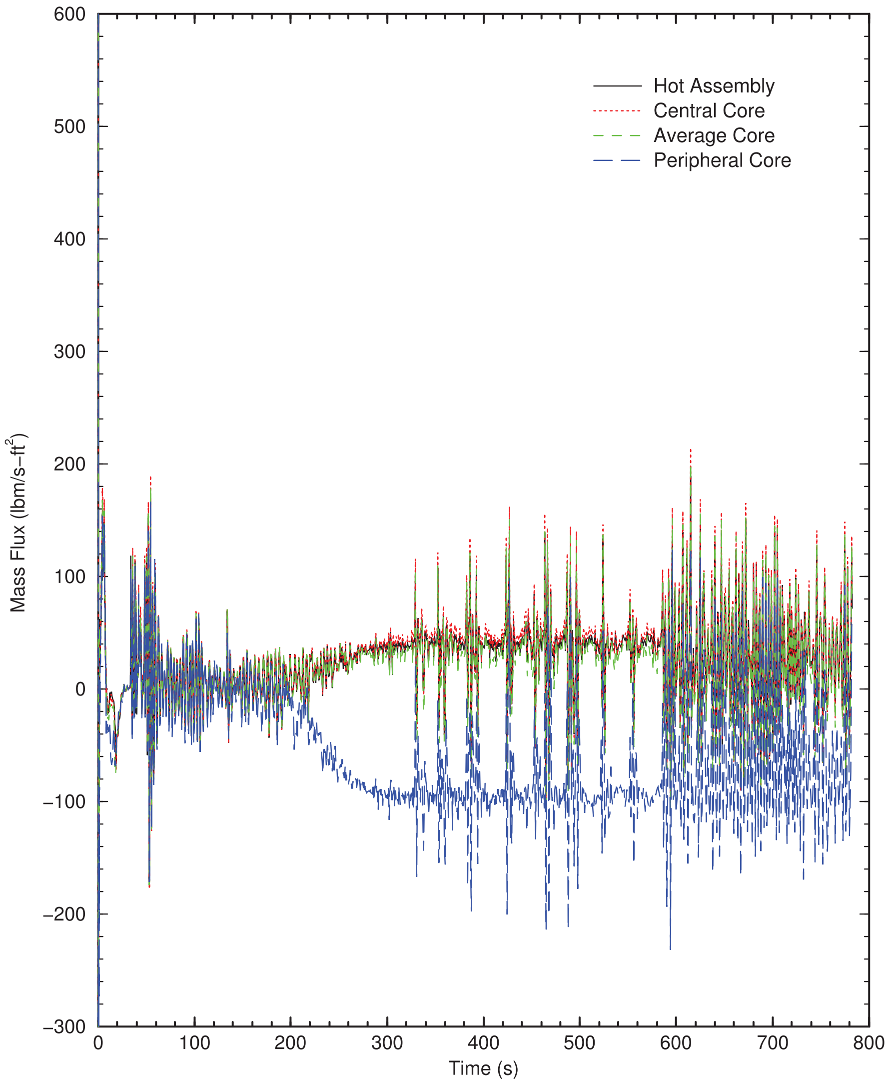

Figure 4-9
Demonstration Case - Core Outlet Mass Flux

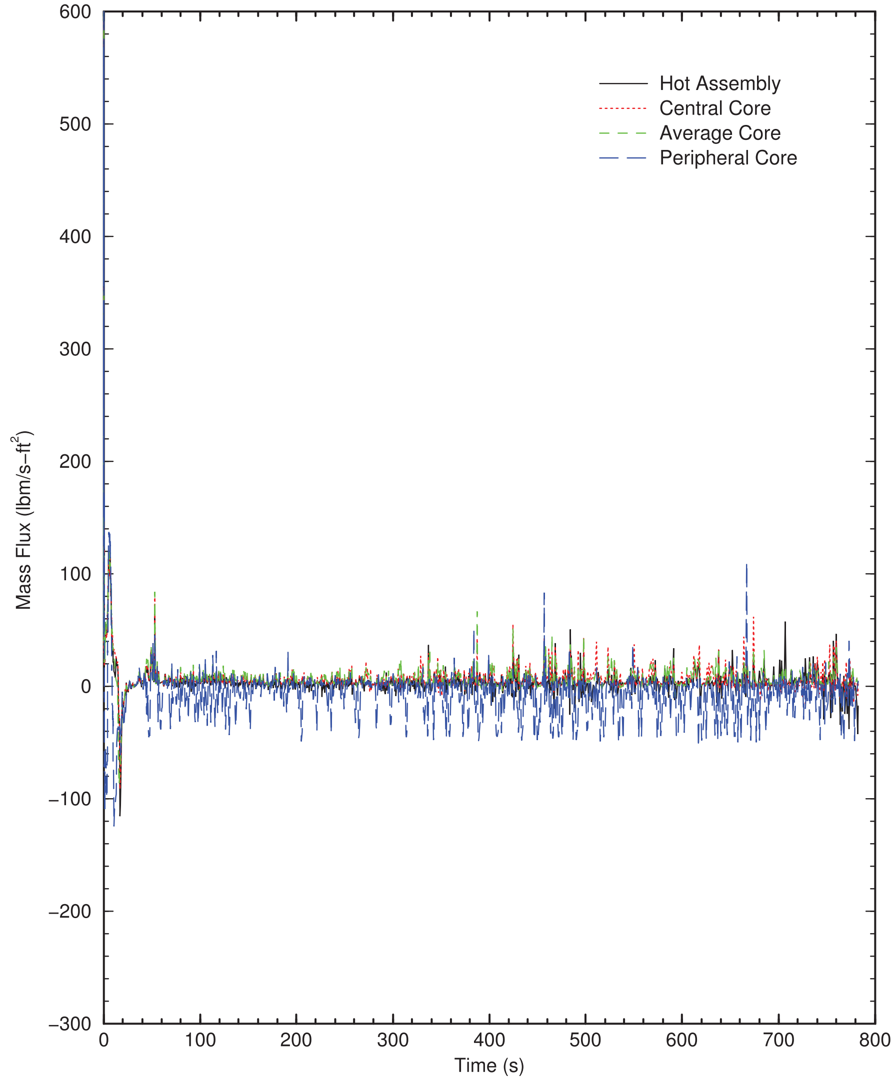

Figure 4-10
Demonstration Case - Void Fraction at RCS Pumps

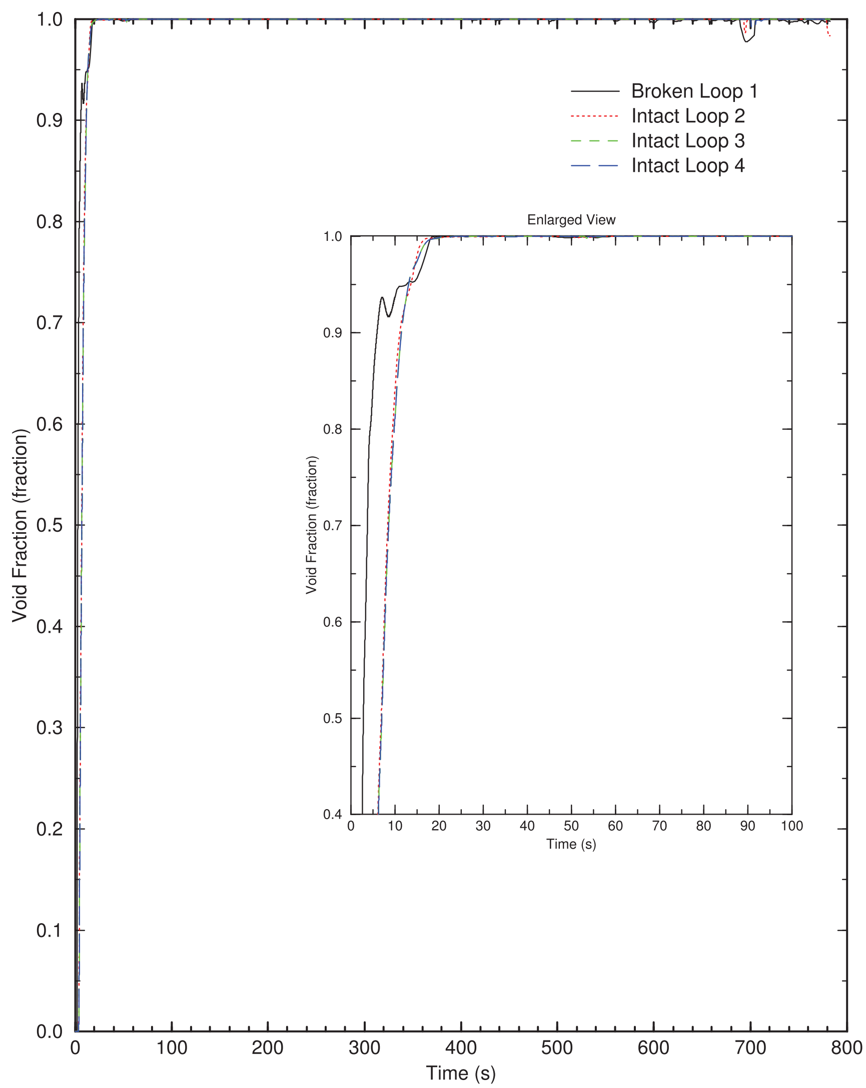

Figure 4-11
Demonstration Case - ECCS Flows (Includes Accumulator, HHSI and LHSI)

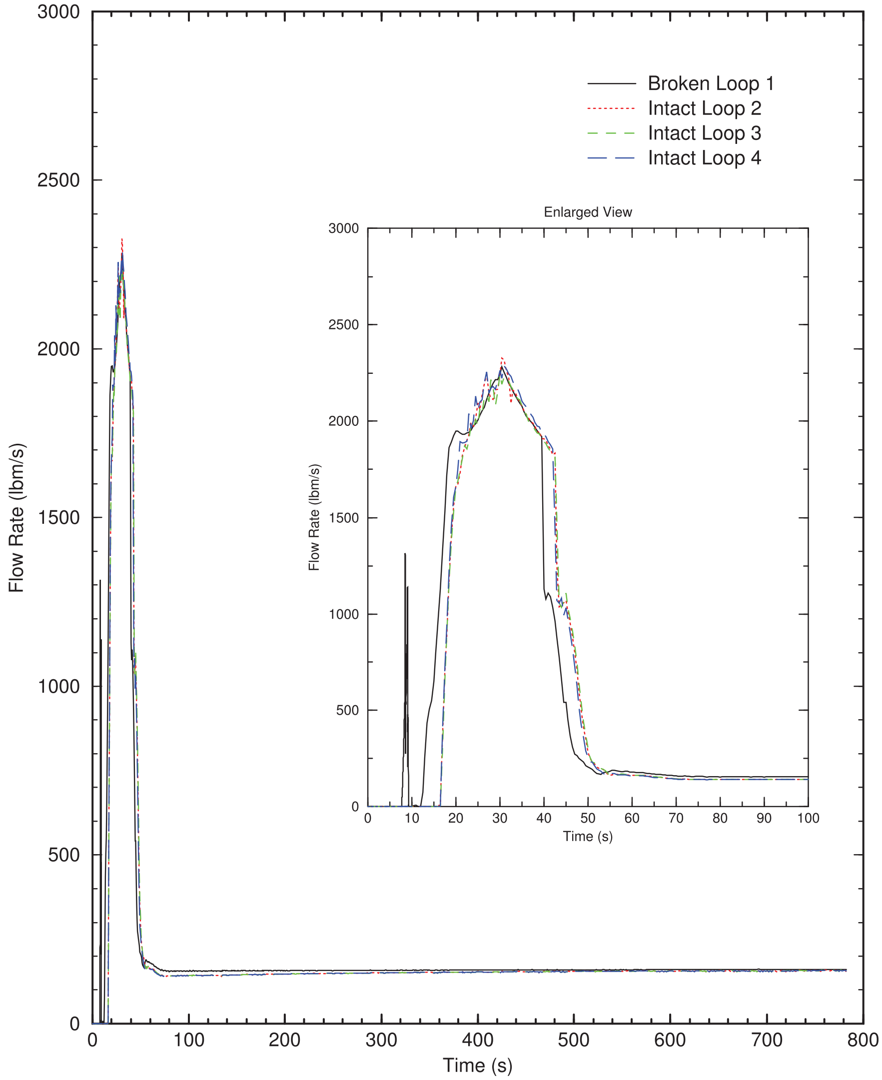

Figure 4-12
Demonstration Case - Upper Plenum Pressure

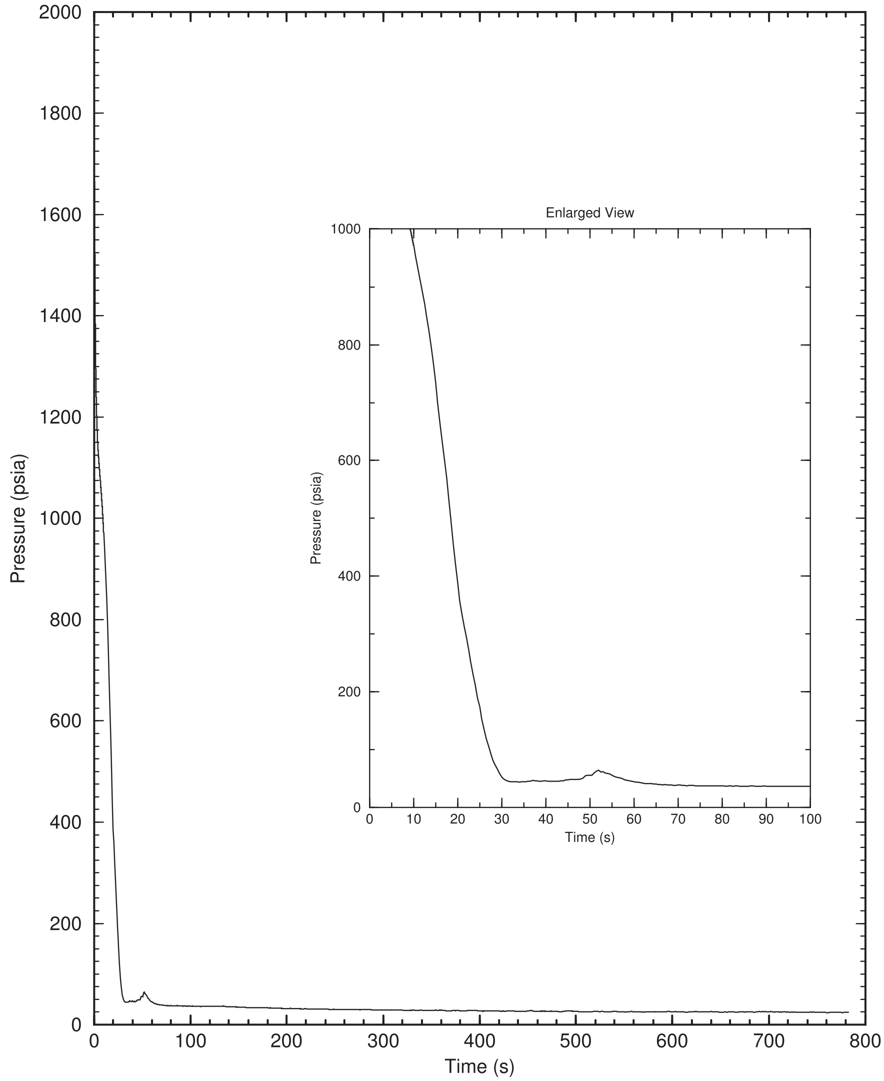

Figure 4-13
Demonstration Case - Collapsed Liquid Level in the Downcomer

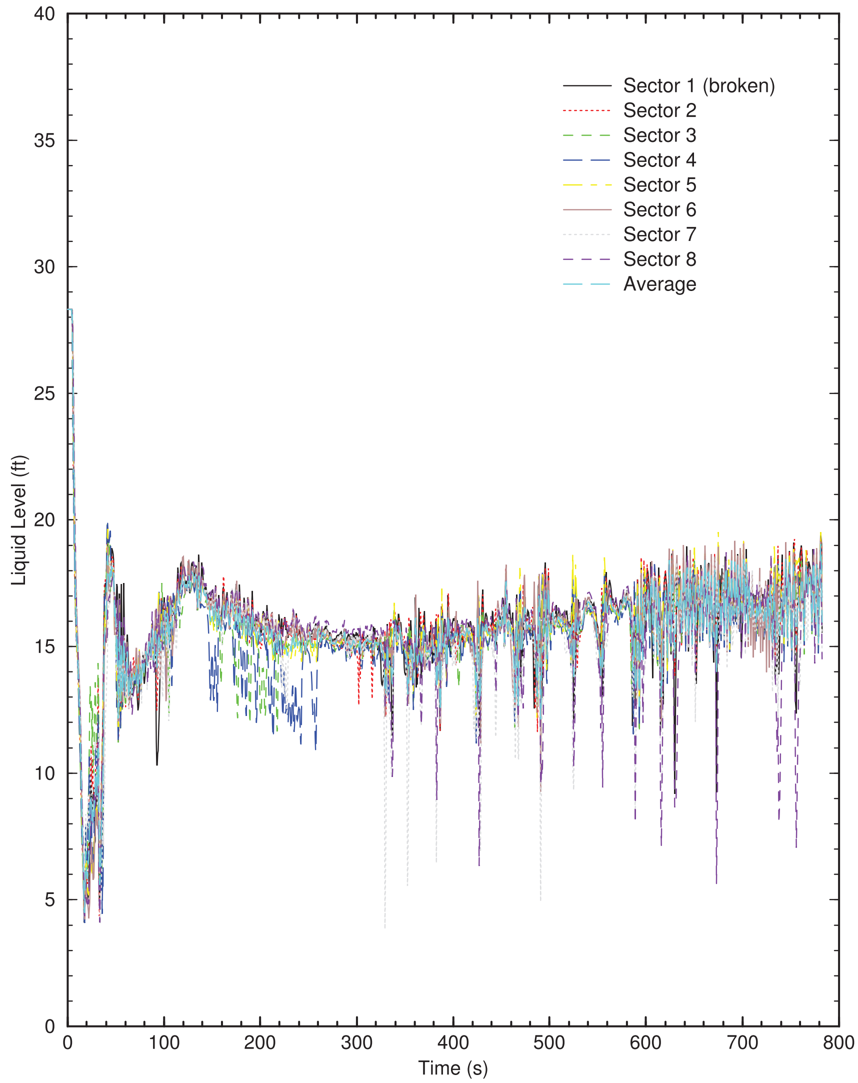

Figure 4-14
Demonstration Case - Collapsed Liquid Level in the Lower Plenum

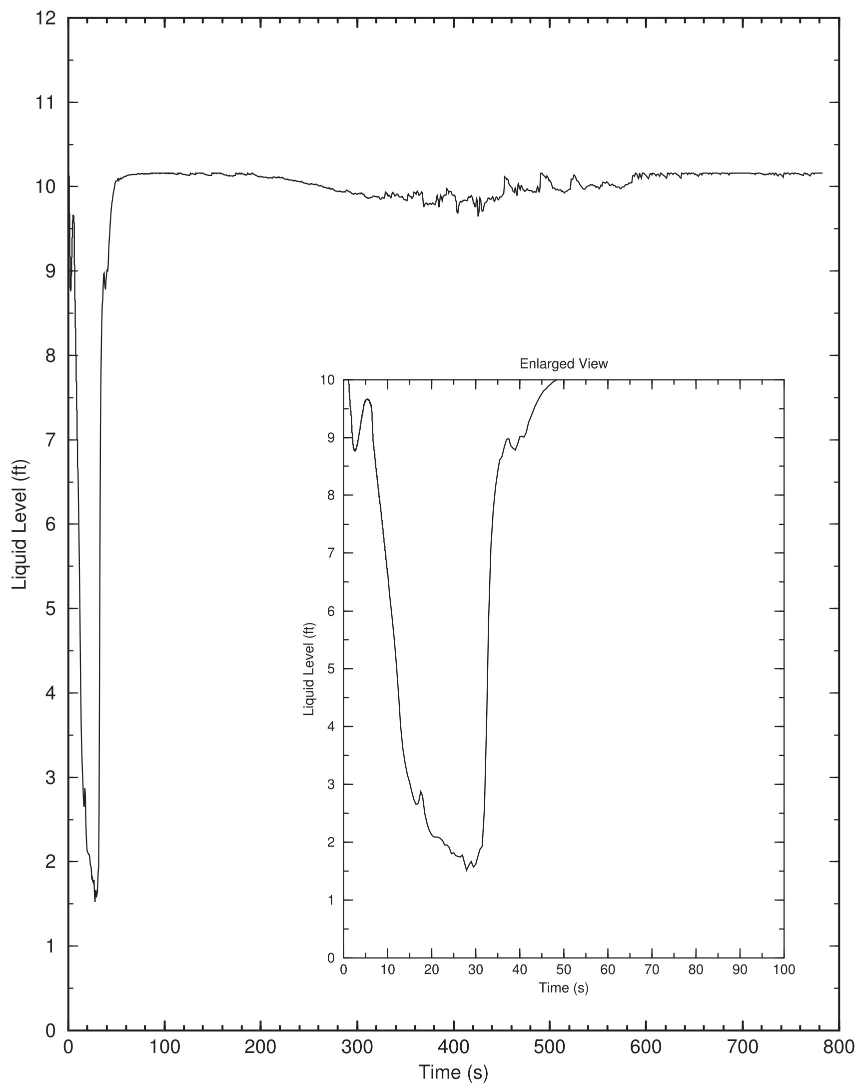

Figure 4-15
Demonstration Case – Core Collapsed Liquid Level

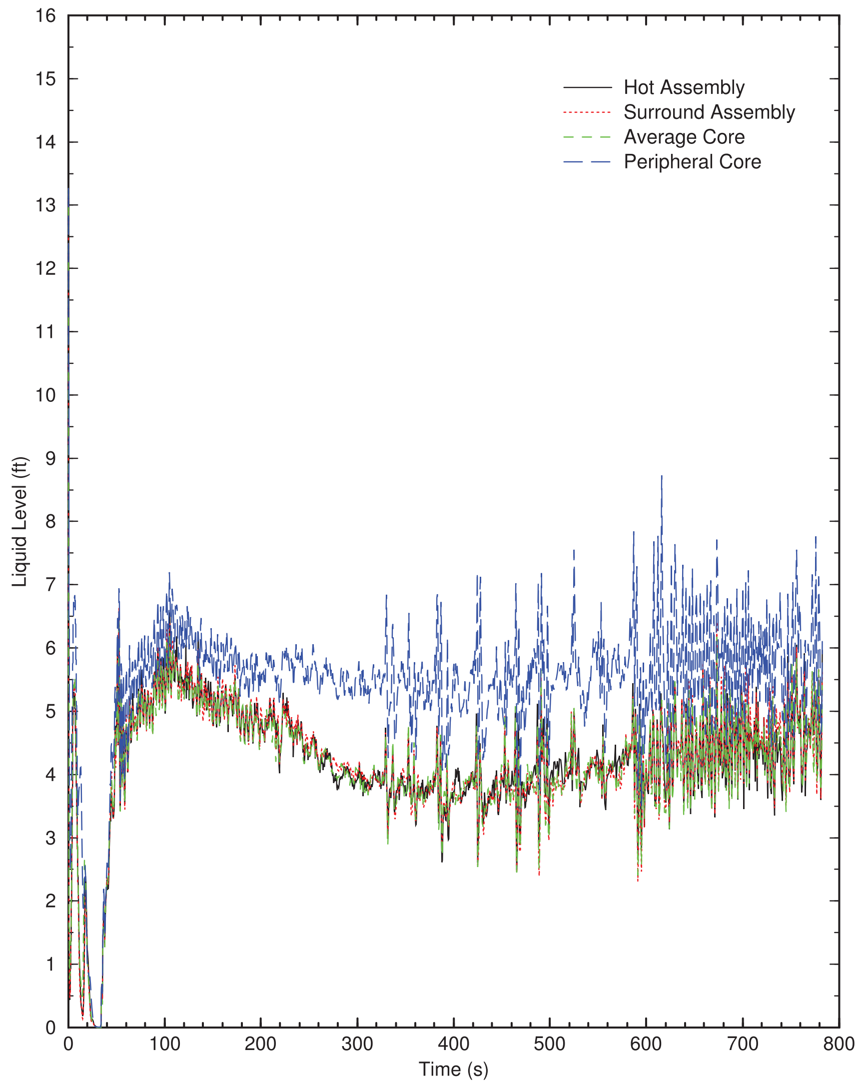

Figure 4-16
Demonstration Case - Containment and Loop Pressures

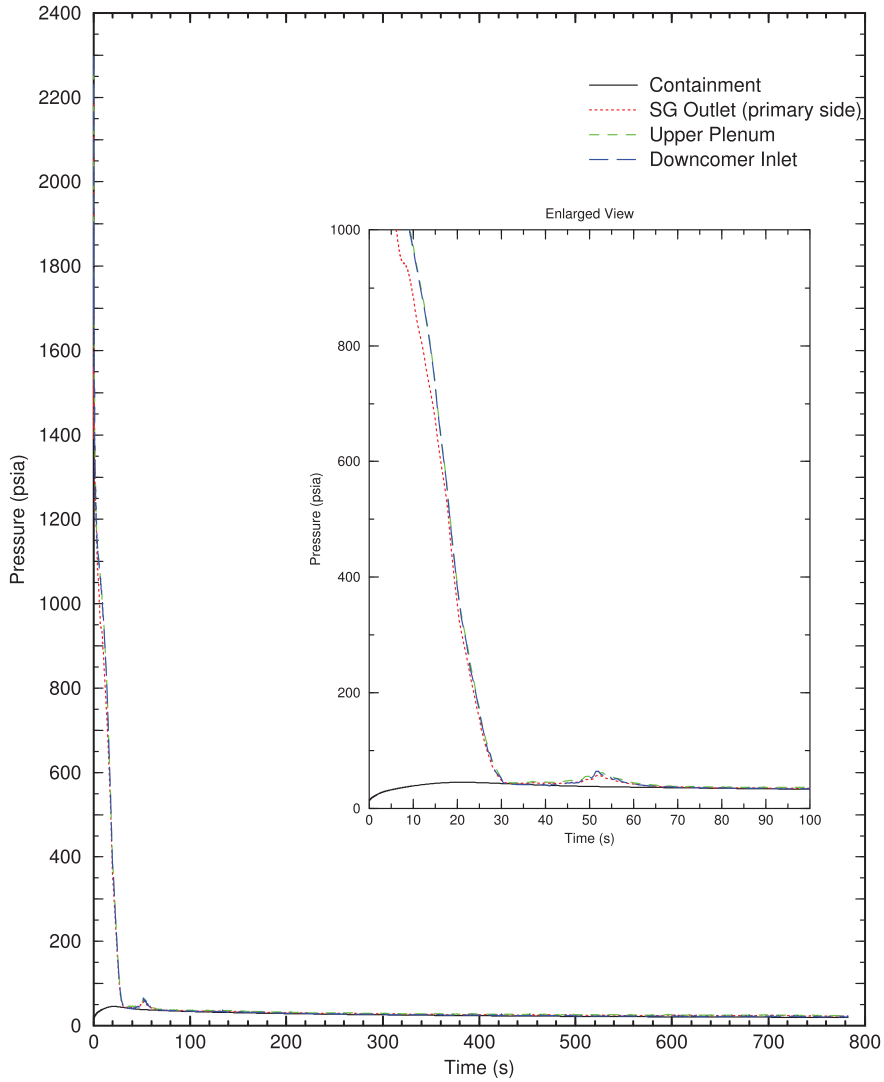

Figure 4-17
Demonstration Case - Pressure Differences between Upper Plenum and Downcomer

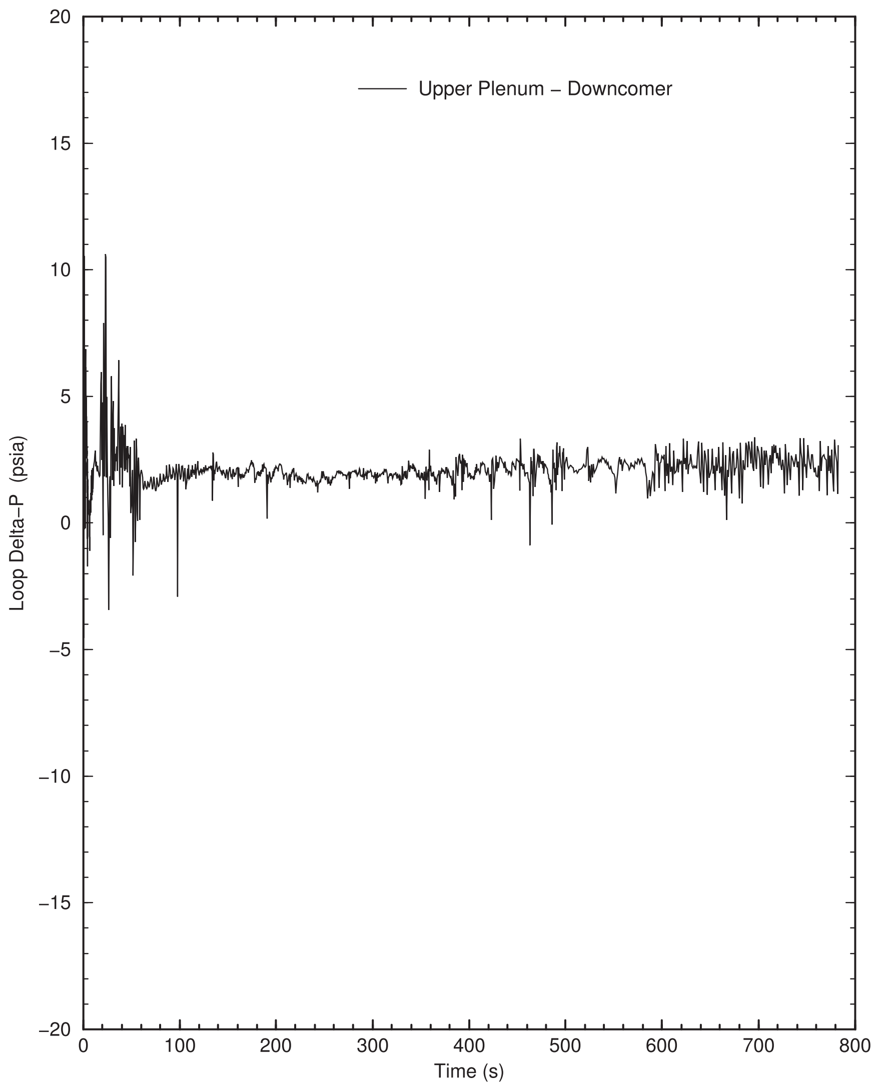

Figure 4-18

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[ ]

5.0 REFERENCES

1. EMF-2103(P)(A) Revision 3, *Realistic Large Break LOCA Methodology for Pressurized Water Reactors*, Framatome, June 2016.
2. Code of Federal Regulations, Title 10, Part 50, Section 46, "Acceptance Criteria For Emergency Core Cooling Systems For Light-Water Nuclear Power Reactors," August 2007.
3. NUREG/CR-5249, "Quantifying Reactor Safety Margins, Application of Code Scaling, Applicability, and Uncertainty Evaluation Methodology to a Large Break, Loss-of-Coolant Accident," U.S. NRC, December 1989.
4. Regulatory Guide 1.203, "Transient and Accident Analysis Methods," U.S. NRC, December 2005.
5. NRC Letter from Siva P. Lingam (NRC) to Maria L. Lacal (Arizona Public Service Company), "Palo Verde Nuclear Generating Station, Units 1, 2, And 3 – Nonproprietary, Issuance Of Amendment Nos. 212, 212, and 212 to Revise Technical Specifications to Support The Implementation Of Framatome High Thermal Performance Fuel (EPID L-2018-LLA-0194)", (NRC ADAMS Accession Number ML20031C947), March 4, 2020.
6. NRC Letter from Michael Mahoney (NRC) to Kim Maza (Shearon Harris Nuclear Power Plant), "Shearon Harris Nuclear Power Plant, Unit 1 – Issuance of Amendment No. 185 Regarding Reduction of Reactor Coolant System Minimum Flow Rate and Update to the Core Operating Limits Report (EPID L 2020 LLA 0040)", (NRC ADAMS Accession Number ML21047A470), April 8, 2021.

APPENDIX A [ ] SUMMARY OF KEY INPUT AND OUTPUT PARAMETERS

The following tables contain the sampled input values for all the cases analyzed. Key results are also included in columns 2 through 6 in Table A-1 for the case set. In all cases, the core power is 3636 MWt (including uncertainty).

Table A-1
Summary of Key Input and Output Parameters, Part 1

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Table A-2
Summary of Key Input and Output Parameters, Part 2

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Table A-3
Summary of Key Input and Output Parameters, Part 3

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