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PROPRIETARY INFORMATION – WITHHOLD UNDER 10 CFR 2.390
UPON REMOVAL OF ENCLOSURES 4 AND 6, THIS LETTER IS DECONTROLLED

10 CFR 50.90

March 6, 2020
Serial: RA-20-0032

ATTN: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Shearon Harris Nuclear Power Plant, Unit 1
Docket No. 50-400 / Renewed License No. NPF-63

Subject: License Amendment Request to Reduce the Minimum Required Reactor Coolant System Flow Rate and Update the List of Analytical Methods Used in the Determination of Core Operating Limits

Ladies and Gentlemen:

Pursuant to 10 CFR 50.90, Duke Energy Progress, LLC (Duke Energy), hereby requests a revision to the Technical Specifications (TS) for the Shearon Harris Nuclear Power Plant, Unit 1 (HNP). Duke Energy is proposing changes to TS 3/4.2.5, "DNB Parameters," and TS 6.9.1.6, "Core Operating Limits Report," in support of analysis development for HNP Cycle 24 and the introduction of reload batches of Framatome, Inc. (Framatome) GAIA fuel assemblies. HNP TS 3/4.2.5 would be revised to reflect a lower minimum Reactor Coolant System (RCS) flow rate, whereas TS 6.9.1.6.2 would reflect the incorporation of the Framatome topical report EMF-2103(P)(A), Revision 3, "Realistic Large Break LOCA {Loss-of-Coolant Accident} Methodology for Pressurized Water Reactors". HNP TS 6.9.1.6.2 will also be revised to reflect the removal of analytical methods no longer applicable for the determination of HNP core operating limits. As part of this license amendment request, Duke Energy is providing an updated HNP Small Break LOCA analysis reflecting the proposed lower minimum RCS flow rate and featuring Framatome GAIA fuel.

The proposed changes have been evaluated in accordance with 10 CFR 50.91(a)(1) using criteria in 10 CFR 50.92(c), and it has been concluded that the proposed changes involve no significant hazards consideration. Enclosure 1 of this license amendment request provides Duke Energy's evaluation of the proposed changes. Enclosure 2 provides a copy of the proposed HNP TS changes. Enclosure 3 provides the affidavits from Framatome, supporting the request for withholding information in Enclosures 4 and 6 from public disclosure. Enclosures 4 and 5 provide the proprietary and non-proprietary reports summarizing the Realistic Large Break LOCA analysis, respectively. Enclosures 6 and 7 provide the proprietary and non-proprietary reports summarizing the HNP Small Break LOCA analysis, respectively.

Approval of the proposed license amendment is requested within twelve months of acceptance. The amendment shall be implemented by HNP prior to the startup of HNP Cycle 24 (Spring 2021).

In accordance with 10 CFR 50.91, a copy of this application, with non-proprietary enclosures, is being provided to the designated North Carolina officials.

This document contains no new regulatory commitments.

Please refer any questions regarding this submittal to Art Zaremba, Manager – Nuclear Fleet Licensing, at (980) 373-2062.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on March 6, 2020.

Sincerely,

John R. Dills
Plant Manager
Harris Nuclear Plant

Enclosures:

1. Evaluation of the Proposed Changes
2. Proposed HNP Technical Specification Changes
3. Affidavits for Withholding of Proprietary Information (Framatome, Inc.)
4. ANP-3767P, Revision 0, "Harris Nuclear Plant Unit 1 Realistic Large Break LOCA Analysis with GAIA Fuel Design" (Proprietary)
5. ANP-3767NP, Revision 0, "Harris Nuclear Plant Unit 1 Realistic Large Break LOCA Analysis with GAIA Fuel Design" (Non-Proprietary)
6. ANP-3766P, Revision 0, "Harris Nuclear Plant Unit 1 Small Break LOCA Analysis with GAIA Fuel Design" (Proprietary)
7. ANP-3766NP, Revision 0, "Harris Nuclear Plant Unit 1 Small Break LOCA Analysis with GAIA Fuel Design" (Non-Proprietary)

cc: (All with Enclosures unless otherwise noted)

L. Dudes, USNRC Region II – Regional Administrator
J. Zeiler, USNRC Senior Resident Inspector – HNP
T. Hood, USNRC NRR Project Manager – HNP
W. L. Cox, III, Section Chief, NC DHSR (NC) (Without Enclosures 4 and 6)

ENCLOSURE 1

EVALUATION OF THE PROPOSED CHANGES

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1
DOCKET NO. 50-400
RENEWED LICENSE NUMBER NPF-63

24 PAGES PLUS THE COVER

Evaluation of the Proposed Changes

License Amendment Request to Reduce the Minimum Required Reactor Coolant System Flow Rate and Update the List of Analytical Methods Used in the Determination of Core Operating Limits

1.0 Summary Description

Pursuant to 10 CFR 50.90, Duke Energy Progress, LLC (Duke Energy), hereby requests a revision to the Technical Specifications (TS) for the Shearon Harris Nuclear Power Plant, Unit 1 (HNP). Duke Energy is proposing changes to TS 3/4.2.5, "DNB Parameters," and TS 6.9.1.6, "Core Operating Limits Report," in support of analysis development for HNP Cycle 24 and the introduction of reload batches of Framatome, Inc. (Framatome) GAIA fuel assemblies. HNP TS 3/4.2.5 would be revised to reflect a lower minimum Reactor Coolant System (RCS) flow rate, whereas TS 6.9.1.6.2 would reflect the incorporation of the Framatome topical report EMF-2103(P)(A), Revision 3, "Realistic Large Break LOCA {Loss-of-Coolant Accident} Methodology for Pressurized Water Reactors" (Reference 1.11). HNP TS 6.9.1.6.2 will also be revised to reflect the removal of analytical methods no longer applicable for the determination of HNP core operating limits. As part of this license amendment request, Duke Energy is providing an updated HNP Small Break LOCA (SBLOCA) analysis reflecting the proposed lower minimum RCS flow rate and featuring Framatome GAIA fuel.

2.0 Detailed Description

2.1 Current Technical Specification Requirements

RCS Flow Rate

HNP TS 3.2.5.c requires that the total RCS flow rate in Mode 1 be maintained greater than or equal to both 293,540 gallons per minute (gpm) and the limit specified in the Core Operating Limits Report (COLR). The 293,540 gpm value also appears in the title of Figure 2.1-1 of the TS Index as well as TS page 2-2. The content of Figure 2.1-1 was relocated from the TS to the COLR when adopting Technical Specification Task Force (TSTF)-339, as approved by the NRC with the issuance of License Amendment No. 161 by letter dated November 6, 2017 (Reference 1.4).

Realistic Large Break LOCA (RLBLOCA) Analysis

HNP TS 6.9.1.6.2.f specifies the Large Break Loss-of-Coolant Accident (LBLOCA) methodology reference as ANP-3011(P), Revision 1 (Reference 1.1). The reference describes the plant-specific implementation of the generic LBLOCA methodology from EMF-2103(P)(A), Revision 0 (Reference 1.2). The plant-specific methodology was approved by the NRC with the issuance of License Amendment 138 per letter dated May 30, 2012 (Reference 1.3). HNP TS 6.9.1.6.2.f identifies the associated specifications as 3.2.1 – Axial Flux Difference; 3.2.2 – Heat Flux Hot Channel Factor; and 3.2.3 – Nuclear Enthalpy Rise Hot Channel Factor.

COLR List

HNP TS 6.9.1.6.2 lists twenty-seven analytical methods approved for use to determine the core operating limits for HNP. Twenty analytical methods are denoted as References "a" to "n" and "p" to "u", and seven analytical methods are denoted as Reference Group "o", "Mechanical Design Methodologies". The latest additions to the first set were References p to u for the Duke

Energy Thermal-Hydraulic, Nuclear Design, and Safety Analysis methods, as approved by the NRC with the issuance of License Amendment Nos. 148, 157 and 164, by letters dated March 8, 2016, May 18, 2017, and April 10, 2018, respectively (References 1.5 to 1.7). The latest addition to Reference Group "o" was for the Mechanical Design code COPERNIC, as approved by the NRC with the issuance of License Amendment No. 171 by letter dated April 29, 2019, as corrected by letter dated October 18, 2019 (References 1.8 and 1.9).

2.2 Reason for Proposed Changes

RCS Flow Rate
The proposed change would revise HNP TS 3.2.5.c (and two associated TS citations) to provide additional operating margin for the minimum RCS flow rate.

RLBLOCA Analysis
The proposed change would revise the HNP TS 6.9.1.6.2.f LBLOCA methodology reference to reflect the analysis completed to support operation with the Framatome GAIA fuel design. This fuel design will be implemented at HNP beginning with Cycle 24 in Spring 2021.

COLR List
The proposed change would revise and consolidate the HNP TS 6.9.1.6.2 COLR reference list to remove analytical methods that will no longer be used to determine the core operating limits, as these methods were replaced upon transitioning to NRC-approved Duke Energy methods. It would also remove the extraneous content from HNP TS 6.9.1.6.2 that cross-references the TS 6.9.1.6.2 COLR methods to the TS 6.9.1.6.1 COLR parameters. The proposed removal of the cross-reference material in HNP TS 6.9.1.6.2 would be in alignment with the structure provided in the Improved Standard Technical Specifications (ISTS) of NUREG-1431, Revision 4, "Standard Technical Specifications – Westinghouse Plants" (ADAMS Accession No. ML12100A222).

2.3 Description of Proposed Changes

RCS Flow Rate
The proposed change would revise HNP TS 3.2.5.c to decrease the minimum RCS flow rate from 293,540 gpm to 290,000 gpm. The title of Figure 2.1-1 in the TS Index and on TS page 2-2 would be revised accordingly. The content of TS Figure 2.1-1 was previously relocated to the COLR and replaced by a note that is not affected by the proposed change.

RLBLOCA Analysis
The proposed change would revise the HNP TS 6.9.1.6.2.f LBLOCA methodology reference to replace the current plant-specific methodology with the generic methodology.

Remove:

f. ANP-3011(P), "Harris Nuclear Plant Unit 1 Realistic Large Break LOCA Analysis," Revision 1, as approved by NRC Safety Evaluation dated May 30, 2012.

Add:

b. EMF-2103(P)(A), “Realistic Large Break LOCA Methodology for Pressurized Water Reactors,” approved version as specified in the COLR.

The generic methodology, as described in EMF-2103(P)(A), Revision 3 (Reference 1.11), was approved by the NRC per Safety Evaluation dated June 17, 2016 (Reference 1.12), and was utilized in the generation of the LBLOCA analysis for HNP Cycle 24 (Reference 1.13). Consistent with the HNP licensing basis, the proposed reference citation excludes the revision number, with the approved revision number to be specified in the COLR.

COLR List

The proposed change would revise the HNP TS 6.9.1.6.2 COLR reference list to remove the analytical methods listed below and renumber the remaining methods. The Reference “o” elements were numbered below to clarify the reference citations. The proposed change would also delete the extraneous content that cross-references the TS 6.9.1.6.2 COLR methods to their respective TS 6.9.1.6.1 COLR parameters, as aligned with the content for ISTS 5.6.3 in NUREG-1431, Revision 4.

a. XN-75-27(P)(A), “Exxon Nuclear Neutronics Design Methods for Pressurized Water Reactors,” approved version as specified in the COLR.

b. ANF-89-151(P)(A), “ANF-RELAP Methodology for Pressurized Water Reactors: Analysis of Non-LOCA Chapter 15 Events,” approved version as specified in the COLR.

c. XN-NF-82-21(P)(A), “Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations,” approved version as specified in the COLR.

e. EMF-84-093(P)(A), “Steam Line Break Methodology for PWRs,” approved version as specified in the COLR.

g. XN-NF-78-44(NP)(A), “A Generic Analysis of the Control Rod Ejection Transient for Pressurized Water Reactors,” approved version as specified in the COLR.

h. ANF-88-054(P)(A), “PDC-3: Advanced Nuclear Fuels Corporation Power Distribution Control for Pressurized Water Reactors and Application of PDC-3 to H. B. Robinson Unit 2,” approved version as specified in the COLR.

i. EMF-92-081(P)(A), “Statistical Setpoint/Transient Methodology for Westinghouse Type Reactors,” approved version as specified in the COLR.

l. EMF-96-029(P)(A), “Reactor Analysis Systems for PWRs,” approved version as specified in the COLR.

n. EMF-2310(P)(A), “SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors,” approved version as specified in the COLR.

o.1. XN-NF-81-58(P)(A), “RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model,” approved version as specified in the COLR.

o.2. ANF-81-58(P)(A), "RODEX2 Fuel Rod Thermal Mechanical Response Evaluation Model," approved version as specified in the COLR.

o.3. XN-NF-82-06(P)(A), "Qualification of Exxon Nuclear Fuel for Extended Burnup," approved version as specified in the COLR.

o.4. ANF-88-133(P)(A), "Qualification of Advanced Nuclear Fuels' PWR Design Methodology for Rod Burnups of 62 GWd/MTU," approved version as specified in the COLR.

o.5. XN-NF-85-92(P)(A), "Exxon Nuclear Uranium Dioxide/Gadolinia Irradiation Examination and Thermal Conductivity Results," approved version as specified in the COLR.

o.6. EMF-92-116(P)(A), "Generic Mechanical Design Criteria for PWR Fuel Designs," approved version as specified in the COLR.

3.0 Technical Evaluation

3.1 RCS Flow Rate

This section describes the evaluation of the proposed decrease in the HNP TS 3.2.5.c minimum RCS flow rate from 293,540 gpm to 290,000 gpm. Since TS 3.2.5.c is the Departure-from-Nucleate-Boiling (DNB) parameter specification, this change impacts FSAR DNB analyses that model a minimum initial RCS flow rate (primarily Final Safety Analysis Report (FSAR) Chapter 15 transients). The evaluations are based on the Cycle 23 DNB analyses of record and the applicable NRC-approved Duke Energy methodology reports (References 1.10, 1.17 and 1.18). These analyses of record are contained within Amendment 63 of the HNP FSAR, which is due to be submitted to the NRC in May 2020. Any updates to the analyses of record in future FSAR amendments will account for the proposed reduction in the TS minimum RCS flow rate.

Many FSAR Chapter 15 DNB analyses of record utilize the Duke Energy Statistical Core Design (SCD) methodology described in Reference 1.18. The SCD methodology is used to assess compliance with the short-term core cooling acceptance criteria. Analyses performed using the SCD methodology assume an RCS flow rate equal to the surveillance limit associated with the precision heat balance for RCS flow per the response to NRC request for additional information (RAI) 16b associated with DPC-NE-3009-P-A, Revision 0, "FSAR / UFSAR Chapter 15 Transient Analysis Methodology" (Reference 1.20). The precision heat balance is performed following cycle startup to satisfy Surveillance Requirement (SR) 4.2.5.2. The measurement uncertainty associated with the precision heat balance is ± 2.2% flow. This measurement uncertainty is included in the HNP-specific Statistical Design Limit (SDL) utilized in analyses performed with the SCD methodology (Reference 1.18, Table I-5).

The surveillance limit for SR 4.2.5.2 is set to the TS minimum RCS flow rate plus an allowance for flow measurement uncertainty. The surveillance limit is currently calculated as 1.022 × 293,540 gpm = 299,998 gpm. The proposed reduction in minimum RCS flow rate from 293,540 gpm to 290,000 gpm will allow the surveillance limit to be reduced to 1.022 × 290,000 gpm = 296,380 gpm. FSAR DNB analyses of record performed with the SCD methodology must therefore consider a total RCS flow rate of 296,380 gpm. The existing SDL remains applicable for this flow rate. FSAR DNB analyses of record which do not use the SCD methodology must consider the TS minimum RCS flow rate of 290,000 gpm.

The FSAR transient analyses were divided into three categories for evaluation. The evaluations are described below.

3.1.1. Category 1: Transient Not Applicable, Bounded, or Insensitive to RCS Flow

Category 1 includes FSAR transients for which RCS flow rate is not an important parameter, based on one of the following criteria:

1. The FSAR transient does not apply to HNP.

2. The FSAR transient is bounded by another FSAR transient, as determined by the evaluations provided in the response to RAI 14 and RAI 31 associated with DPC-NE-3009-P-A (References 1.26 and 1.20, respectively).

3. The FSAR transient has been determined to be insensitive to RCS flow. This determination is made if: the event does not involve a Nuclear Steam Supply System (NSSS) transient; the event was determined to be insensitive to RCS flow in Section 5 of Reference 1.10; or the determination is otherwise supported by the analysis of record.

Table 1 summarizes the evaluations for Category 1. These transients are not affected by the proposed reduction in TS minimum RCS flow rate. Table 1 includes non-DNB analyses, which are not bound by TS 3.2.5.c. Nevertheless, the evaluation considered the impact of the flow reduction on these non-DNB transients.

3.1.2. Category 2: Transients Bounded by Current RCS Flow Assumption

Category 2 includes FSAR transients for which the current analysis of record has accounted for the proposed reduction in RCS flow rate. A reduction in the TS minimum RCS flow rate from 293,540 gpm to 290,000 gpm will have no impact on any DNB analysis which assumes an RCS flow rate of ≤ 290,000 gpm. Since this is a DNB parameter limit, non-DNB-related analyses can be performed at higher RCS flows. However, for this evaluation, those analyses are treated as constrained by the TS 3.2.5.c flow limit. A reduction in the TS minimum RCS flow rate from 293,540 gpm to 290,000 gpm will also have no impact on SCD analyses which assume an RCS flow rate of ≤ 296,380 gpm or analyses for which maximum RCS flow rate is conservative. The Small Break LOCA (SBLOCA) analysis to support operation with the GAIA fuel design beginning in Cycle 24, as described in Section 3.1.2.1 below, reflects the reduced RCS flow rate.

Table 2 summarizes the evaluations for Category 2 and the assumed RCS flow rates. Review of Table 2 concludes that both the DNB and non-DNB transients have accounted for the proposed reduction in the TS minimum RCS flow rate.

3.1.2.1 FSAR Section 15.6.5.3 – Small Break LOCA Transient

Framatome Reports ANP-3766P (proprietary) and ANP-3766NP (non-proprietary), Revision 0 (Reference 1.14), describe the SBLOCA analysis completed to support operation with the GAIA fuel design beginning in Cycle 24, utilizing the reduced TS minimum RCS flow rate. The SBLOCA analysis utilized the NRC-approved methodology specified in HNP TS 6.9.1.6.2.m, EMF-2328(P)(A), “PWR Small Break LOCA Evaluation Model, S-RELAP5 Based” (Reference 1.15), and supplemented in EMF-2328(P)(A), Revision 0, Supplement 1(P)(A), Revision 0 (Reference 1.16). The proprietary and non-proprietary analysis reports are provided as Enclosures 6 and 7, respectively.

The SBLOCA analysis considered a spectrum of cold leg breaks with equivalent diameters ranging from 1.0 inch to 8.7 inches. As required by the methodology, supporting analyses were completed considering delayed reactor coolant pump trip, attached piping breaks, and sensitivity to reduced Emergency Core Cooling System (ECCS) fluid temperature. The analysis supports the proposed decrease in the TS minimum RCS flow rate from 293,540 gpm to 290,000 gpm.

The following table provides the results for Peak Cladding Temperature (PCT), Maximum Local Oxidation (including pre-transient oxidation), Core-Wide Oxidation and Core Coolable Geometry, showing that they meet the acceptance criteria of 10 CFR 50.46(b), Paragraphs (1) to (4). Section 3.4 of Reference 1.14 states that there are no limitations on the methodology used in the SBLOCA analysis.

Parameter	**Result**	**Criterion**
Peak Cladding Temperature	1,832 °F	≤ 2,200 °F
Maximum Local Oxidation	4.89 %	≤ 17 %
Core-Wide Oxidation	0.017 %	≤ 1 %
Core Coolable Geometry	Maintained	Maintained

3.1.3. Category 3: Transient-Specific Evaluations

Category 3 includes FSAR transients which apply to HNP, are not explicitly bounded by other FSAR transients, and have not yet accounted for the proposed decrease in the TS minimum RCS flow rate. Where noted, updates to these FSAR analyses are in progress to account for the proposed decrease in the TS minimum RCS flow rate and, in some cases, transition to Duke Energy methods. Category 3 also includes the SBLOCA analysis for HTP fuel. The following sub-sections describe the evaluations for Category 3. The evaluations conclude that the proposed decrease in the TS minimum RCS flow rate will have a negligible effect on the respective analysis results.

3.1.3.1. FSAR Section 15.1.2 – Feedwater System Malfunctions that Result in an Increase in Feedwater Flow

The analysis of record assumes a total RCS flow rate of 293,540 gpm and does not utilize the SCD methodology. The analysis therefore does not account for the proposed reduction in the TS minimum RCS flow rate from 293,540 gpm to 290,000 gpm. The evaluation is described below.

The analysis of record demonstrates that there is significant margin to the DNB and centerline fuel melt (CFM) limits. The transient is considered non-limiting, and the DNB and CFM acceptance criteria are not evaluated on a cycle-specific basis. The proposed reduction in the TS minimum RCS flow rate from 293,540 gpm to 290,000 gpm will result in no or negligible effect on the DNB and CFM results. Reanalysis with Duke Energy methods is in progress and will account for the proposed reduction in the TS minimum RCS flow rate.

3.1.3.2. FSAR Section 15.1.5 – Steam System Piping Failure

This event is analyzed at hot full power (HFP) and hot zero power (HZP) conditions. The HFP condition is analyzed with and without a loss of offsite power. The HZP analysis of record was performed at an RCS flow rate of 290,000 gpm using the Duke Energy non-SCD methodology. The reduction in the TS minimum RCS flow rate has therefore been analyzed for the HZP condition. The HFP analysis is performed to ensure the HZP analysis is limiting (Reference 1.10, Section 5.1.4). However, the HFP analysis assumes an RCS flow of 293,540 gpm and therefore does not account for the proposed reduction in the TS minimum RCS flow rate from 293,540 gpm to 290,000 gpm. The evaluation is described below.

The HFP condition is analyzed to ensure adequate core cooling capability by demonstrating that the release of radioactive material does not result in dose consequences exceeding the regulatory limits. Fuel failures for the HFP condition could occur through DNB or CFM. Fuel failures during the event could occur prior to reactor trip (short-term analysis) or during a return to power following reactor trip (long-term analysis).

Short-Term HFP Analysis: Reactor trip occurs within 5 seconds of the break following a safety injection signal on low steam line pressure. Power increases approximately 1% rated thermal power (RTP) before the rod cluster control assemblies (RCCAs) insert and reduce reactor power to decay heat conditions. The Cycle 23 analysis (current cycle) showed that > 20% margin to CFM exists. The reduction in RCS flow will not result in a significant reduction in CFM margin. The minimum DNB ratio (DNBR) is well above the DNBR limit. The short-term analysis is therefore non-limiting and will not become limiting as a result of the proposed RCS flow reduction.

Long-Term HFP Analysis: The post-trip return to power was analyzed, and the power response was shown to be bounded by the HZP analysis of record, which assumes a total RCS flow rate of 290,000 gpm. Since total RCS flow rate will not significantly impact the return to power for the HFP case, the long-term analysis will not become limiting as a result of the proposed reduction in the TS minimum RCS flow rate.

Reanalysis of the HFP analyses with Duke Energy methods is in progress and will account for the proposed reduction in the TS minimum RCS flow rate.

3.1.3.3. FSAR Section 15.2.7 – Loss of Normal Feedwater Flow

The analysis of record for Loss of Normal Feedwater Flow assumes a total RCS flow rate of 293,540 gpm and does not utilize the SCD methodology. The proposed reduction in the TS minimum RCS flow rate from 293,540 gpm to 290,000 gpm has therefore not been analyzed. The evaluation is described below.

The analysis of record does not analyze for DNB, but demonstrates that fluid mass in each steam generator exceeds 10,000 lbm for the duration of the event (FSAR Figure 15.2.7-5) and subcooling margin exceeds 40 °F after reactor trip. A small reduction in the minimum TS RCS flow rate would have a negligible impact on the long-term core cooling capability. Therefore, the proposed reduction in the TS minimum RCS flow rate is judged to have a negligible impact on the transient.

Reanalysis with Duke Energy methods is in progress and will account for the proposed reduction in the TS minimum RCS flow rate.

3.1.3.4. FSAR Section 15.4.2 – Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power

The short-term core cooling analysis of record assumes a total RCS flow rate of 296,380 gpm and utilizes the SCD methodology. The total RCS flow rate assumed in the analysis is therefore consistent with the proposed reduction in the TS minimum RCS flow rate. The analysis results demonstrate that the short-term core cooling acceptance criteria are met. Therefore, the short-term core cooling portion of the analysis has been analyzed at the new condition and has demonstrated acceptable results.

The peak primary pressure analysis of record assumes a total RCS flow rate of 296,380 gpm and does not utilize the SCD methodology. The proposed reduction in the TS minimum RCS flow rate from 293,540 gpm to 290,000 gpm has therefore not been analyzed. The evaluation is described below.

The peak primary pressure analysis of record demonstrates that > 50 psi margin is maintained to the acceptance criteria (FSAR Figure 15.4.2-10) and concludes that the results are bounded by FSAR Section 15.2.3, Turbine Trip. A small reduction in RCS total flow rate (~1.2%) will have a negligible impact on the existing margin. Therefore, the proposed reduction in the TS minimum RCS flow rate is judged to have a negligible impact on the transient. Regardless, the minimum total RCS flow rate for DNB protection in TS 3.2.5.c does not apply to this acceptance criterion.

3.1.3.5. FSAR Section 15.6.5.3 – Small Break LOCA Transient

The SBLOCA analysis for HTP fuel assumed an initial RCS flow rate of 293,540 gpm (FSAR Table 15.6.5-3). The proposed decrease in the TS minimum RCS flow rate from 293,540 gpm to 290,000 gpm has an insignificant impact on the SBLOCA analysis. RCS flow decreases rapidly early in the transient (due to the assumed loss of offsite power coincident with reactor trip), whereas the PCT occurs about 2,000 seconds later. As a result, the SBLOCA analysis for HTP fuel remains applicable at the decreased RCS flow rate.

3.1.4. Other Considerations

HNP TS Figure 2.1-1 is impacted administratively by the proposed change in the TS minimum RCS flow rate. The TS Figure 2.1-1 title is revised to reduce the measured RCS flow rate from “293,540 GPM × (1.0 + C_1)” to “290,000 GPM × (1.0 + C_1)”. The reactor core safety limits in the HNP Cycle 23 COLR (Reference 1.19) have been generated assuming a measured RCS flow rate of 290,000 gpm × (1.0 + 0.022) = 296,380 gpm, where 0.022 (2.2%) is the measurement uncertainty associated with the precision heat balance for RCS flow. The Cycle 24 COLR will assume the same. No further change to the reactor core safety limits is required to support the proposed reduction in the TS minimum RCS flow rate.

3.1.5. Conclusion

The FSAR transients potentially impacted by the proposed reduction in the TS minimum RCS flow rate have been evaluated. The proposed reduction was determined to have no impact on the FSAR transients in Category 1. The proposed reduction was already incorporated in the FSAR transients in Category 2. The proposed reduction was determined to have a negligible impact on the FSAR transients in Category 3.

Table 1 – Evaluations for Category 1: Transients Not Applicable, Bounded, or Insensitive to RCS Flow

Section	Title	Evaluation
3.6	Protection Against Dynamic Effects Associated with the Postulated Rupture of Piping (LOCA only)	Insensitive to RCS flow rate. (1)
15.1.1	Feedwater System Malfunctions that Result in a Decrease in Feedwater Temperature	Bounded by FSAR Section 15.1.3.
15.1.4	Inadvertent Opening of a Steam Generator Relief or Safety Valve	Core Cooling – Bounded by FSAR Sections 15.1.3 and 15.1.5. (2) Primary Pressure – Bounded by FSAR Section 15.2.3. Secondary Pressure – Bounded by FSAR Section 15.2.3.
15.2.1	Steam Pressure Regulator Malfunction or Failure that Results in Decreasing Steam Flow	This event does not apply to HNP.
15.2.2	Loss of External Electrical Load	Bounded by FSAR Section 15.2.3.
15.2.4	Inadvertent Closure of Main Steam Isolation Valves	Bounded by FSAR Section 15.2.3.
15.2.5	Loss of Condenser Vacuum and Other Events Resulting in Turbine Trip	Bounded by FSAR Section 15.2.3.
15.2.6	Loss of Non-Emergency AC Power to the Station Auxiliaries	Short Term Core Cooling – Bounded by FSAR Section 15.3.2. (2) Primary Pressure – Bounded by FSAR Section 15.3.2. Secondary Pressure – Bounded by FSAR Section 15.2.3. Long Term Core Cooling – Bounded by FSAR Section 15.2.7.
15.3.1	Partial Loss of Forced Reactor Coolant Flow	Bounded by FSAR Section 15.3.2. (2)
15.3.4	Reactor Coolant Pump Shaft Break	Bounded by FSAR Section 15.3.3.
15.4.4	Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature	Not credible during power operation; no analysis necessary at zero power operation.
15.4.5	A Malfunction or Failure of the Flow Controller in a BWR Loop that Results in an Increased Reactor Coolant Flow Rate	This event does not apply to HNP.

Table 1 – Evaluations for Category 1: Transients Not Applicable, Bounded, or Insensitive to RCS Flow (Continued)

Section	Title	Evaluation
15.4.6	Chemical and Volume Control System Malfunction that Results in a Decrease in the Boron Concentration in the Reactor Coolant	Insensitive to RCS flow rate. (3)
15.4.9	Spectrum of Rod Drop Accidents in a BWR	This event does not apply to HNP.
15.5.1	Inadvertent Operation of the Emergency Core Cooling System During Power Operation	Core Cooling – Bounded by FSAR Section 15.6.1. Peak Primary Pressure – Bounded by FSAR Section 15.2.3. Peak Secondary Pressure – Bounded by FSAR Section 15.2.3. Overfill – Insensitive to RCS flow rate. (4)
15.5.2	Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory	Bounded by FSAR Sections 15.4.6 and 15.5.1.
15.5.3	A Number of BWR Transients	This event does not apply to HNP.
15.6.2	Break in Instrument Line or Other Line From Reactor Coolant Pressure Boundary That Penetrate Containment	This event does not involve an NSSS transient.
15.6.4	Spectrum of BWR Steam System Piping Failure Outside Containment	This event does not apply to HNP.
15.7.1	Radioactive Waste Gas System Leak or Failure	This event does not involve an NSSS transient.
15.7.2	Liquid Waste System Leak or Failure	This event does not involve an NSSS transient.
15.7.3	Postulated Radioactive Releases Due to Liquid Tank Failure	This event does not involve an NSSS transient.
15.7.4	Design Basis Fuel Handling Accidents	This event does not involve an NSSS transient.
15.7.5	Spent Fuel Cask Drop Accidents	This event does not involve an NSSS transient.

Table 1 – Evaluations for Category 1: Transients Not Applicable, Bounded, or Insensitive to RCS Flow (Continued)

<u>Notes</u>

1. Only auxiliary line breaks are considered under the Leak-Before-Break methodology.

2. The analyses of record for FSAR Sections 15.1.5 (ANS Condition IV) and 15.3.2 (ANS Condition III) did not result in fuel failure for Cycle 23, and are not expected to result in fuel failure for Cycle 24. Therefore, these analyses bound selected ANS Condition II transients, consistent with the evaluation provided in the response to RAIs 14 and 31 for DPC-NE-3009-P-A (References 1.26 and 1.20). If the FSAR Section 15.1.5 or 15.3.2 analyses result in fuel failure for a given cycle, then the FSAR will be updated with analyses of the pertinent ANS Condition II transients, with the proposed decrease in the TS minimum RCS flow rate.

3. According to Section 5.4.6 of Reference 1.10, the number of reactor coolant pumps in operation is used to determine the dilution volume considered in the analysis, but the TS minimum RCS flow rate is not an input to the analysis.

4. See Reference 1.10, Table 5-16.

Table 2 – Evaluations for Category 2: Transients Bounded by Current RCS Flow Assumption

Section	Title	Methodology	Total RCS Flow Assumed (gpm)
6.2.1 (1)	Containment Functional Design	Non-SCD (Other Analyses)	277,800
15.1.3	Excessive Increase in Secondary Steam Flow	SCD	296,380
15.1.5	Steam System Piping Failure	Non-SCD (Core Cooling Analysis)	290,000 (Hot Zero Power Analysis) 293,540 (Hot Full Power Analysis) (2)
15.2.3	Turbine Trip	SCD (Core Cooling Analysis) Non-SCD (Other Analyses)	296,380 (Core Cooling Analysis) 290,000 (Primary Pressure Analysis) 321,300 (Secondary Pressure Analysis)
15.2.8	Feedwater System Pipe Break	SCD (Core Cooling Analysis) Non-SCD (Other Analyses)	296,380 (Core Cooling Analysis) 290,000 (Primary Pressure Analysis) 290,000 (Long-Term Analysis)
15.3.2	Complete Loss of Forced Reactor Coolant Flow	SCD	296,380
15.3.3	Reactor Coolant Pump Shaft Seizure (Locked Rotor)	SCD (Core Cooling Analysis) Non-SCD (Other Analyses)	296,380 (Core Cooling Analysis) 290,000 (Primary Pressure Analysis)
15.4.1	Uncontrolled Rod Cluster Control Assembly Bank Withdrawal from a Subcritical or Low Power Startup Condition	SCD (Core Cooling Analysis) Non-SCD (Other Analyses)	296,380 (Core Cooling Analysis) 321,300 (Primary Pressure Analysis)
15.4.2	Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power	SCD (Core Cooling Analysis) Non-SCD (Other Analyses)	296,380 (Core Cooling Analysis) 296,380 (Primary Pressure Analysis) (2)
15.4.3.1	Dropped Full Length RCCA or RCCA Bank	SCD	296,380

Table 2 – Evaluations for Category 2: Transients Bounded by Current RCS Flow Assumption (Continued)

Section	Title	Methodology	Total RCS Flow Assumed (gpm)
15.4.3.2	Withdrawal of a Single Full Length RCCA	SCD	296,380
15.4.3.3	Statically Misaligned RCCA or Bank	SCD	296,380
15.4.7	Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position	SCD	296,380
15.4.8	Spectrum of Rod Cluster Control Assembly Ejection Accidents	SCD Non-SCD (Core Cooling and Other Analyses)	296,380 (Cases with no flux-based trip) 290,000 (All other cases)
15.6.1	Inadvertent Opening of a Pressurizer Safety or Power Operated Relief Valve	SCD	296,380
15.6.3	Steam Generator Tube Rupture	Non-SCD (Other Analyses)	277,800 (MTO and Offsite Dose Thermal-Hydraulic Input Analyses)
15.6.5.2	Large Break LOCA	Non-SCD (Other Analyses)	290,000 – 310,600
15.6.5.3	Small Break LOCA	Non-SCD (Other Analyses)	290,000
15.8	Anticipated Transients Without Scram	Non-SCD (Other Analyses)	277,800

<u>Notes</u>

1. Includes subsections that comprise all of the potentially impacted Chapter 6 analyses.
2. See Category 3 for a specific evaluation of the FSAR Section 15.1.5 Hot Full Power analysis and the FSAR Section 15.4.2 Peak Primary Pressure analysis.

3.2 RLBLOCA Analysis

Framatome Reports ANP-3767P (proprietary) and ANP-3767NP (non-proprietary), Revision 0 (Reference 1.13), describe the LBLOCA analysis supporting operation with the GAIA fuel design beginning in Cycle 24. The proprietary and non-proprietary analysis reports are provided as Enclosures 4 and 5, respectively.

The LBLOCA analysis assumed full-power operation at a core power level of 2,958 MWt, a heat flux hot channel factor (F_Q) up to 2.62, and a nuclear enthalpy rise hot channel factor ($F_{\Delta H}$) of 1.73, with uncertainty included for each value. A K(z) adjustment factor was used to define an elevation-dependent F_Q limit. The analysis considered typical operating ranges or TS limits for the following: initial pressurizer pressure and water level; accumulator pressure, temperature and water level; RCS temperature; containment pressure and temperature; and refueling water storage tank temperature. The analysis sampled a range of initial RCS mass flow rates from 108.04 Mlbm/hr to 115.72 Mlbm/hr, corresponding to a range of initial RCS volume flow rates from 290,000 gpm to 310,600 gpm. Therefore, the analysis supports the proposed decrease in TS minimum RCS flow rate from 293,540 gpm to 290,000 gpm. The analysis included calculations for both fresh and once-burned GAIA fuel.

The following table compares the 95/95 results for PCT, Maximum Local Oxidation (including pre-transient oxidation) and Core-Wide Oxidation to the acceptance criteria obtained (or adjusted) from 10 CFR 50.46(b), paragraphs (1) to (3). As can be seen from the table, the results meet the acceptance criteria. Table 3-4 of the LBLOCA analysis report (Enclosures 4 and 5) describes the evaluation of the eleven limitations identified in Section 4.0 of the Safety Evaluation on the EMF-2103(P)(A), Revision 3, methodology, all of which were addressed successfully.

Parameter	Result	Criterion
Peak Cladding Temperature	1,820 °F	≤ 2,200 °F
Maximum Local Oxidation	6.79 %	≤ 13 %
Core-Wide Oxidation	0.07 %	≤ 1 %

In lieu of ANP-3011(P), Revision 1, an evaluation was completed for once-burned HTP fuel using the EMF-2103(P)(A), Revision 3, methodology. Differences between the GAIA and HTP fuel designs include (but are not limited to) the HTP fuel having lower pellet density, a slightly smaller pellet diameter, and a slightly larger cladding thickness. A set of S-RELAP5 runs was completed to determine the impact of the fuel design differences on peak cladding temperature. The calculations used the same random seed and second-cycle peaking factor limits as in the GAIA analysis, but with the fuel rod models adjusted to represent the HTP fuel.

In the EMF-2103(P)(A), Revision 3, methodology, the analysis-specific order statistic is used to establish the 95/95 PCT result. Comparing the analogous cases from the once-burned GAIA and once-burned HTP case sets showed a PCT increase of 46 °F for once-burned HTP fuel. Based on this comparison, a reasonable PCT penalty of +50 °F was established for once-burned HTP fuel relative to the limiting PCT of 1,820 °F for GAIA fuel. This penalty will be applied to the once-burned HTP fuel to support the replacement of ANP-3011(P) with EMF-2103(P)(A) in the licensing basis for Cycle 24.

3.3 COLR List

HNP TS 6.9.1.6.2 lists the methodologies approved for use in the design and safety analysis of core reloads, with the HNP COLR identifying the methods and revisions used each cycle. The NRC has approved the following Duke Energy methodologies for use by HNP to perform the respective analyses in-house:

- DPC-NE-2005-P, Revision 5, "Thermal-Hydraulic Statistical Core Design Methodology," by letter dated March 8, 2016 (Reference 1.5).
- DPC-NE-1008-P, Revision 0, "Nuclear Design Methodology Using CASMO-5/SIMULATE-3 for Westinghouse Reactors," by letter dated May 18, 2017 (Reference 1.6).
- DPC-NF-2010, Revision 3, "Nuclear Physics Methodology for Reload Design," by letter dated May 18, 2017 (Reference 1.6).
- DPC-NE-2011-P, Revision 2, "Nuclear Design Methodology Report for Core Operating Limits of Westinghouse Reactors," by letter dated May 18, 2017 (Reference 1.6).
- DPC-NE-3008-P, Revision 0, "Thermal-Hydraulic Models for Transient Analysis," by letter dated April 10, 2018 (Reference 1.7).
- DPC-NE-3009-P, Revision 0, "FSAR / UFSAR Chapter 15 Transient Analysis Methodology," by letter dated April 10, 2018 (Reference 1.7).

Additionally, by letter dated April 29, 2019 (Reference 1.8), the NRC approved the addition of NRC-approved methodology BAW-10231P-A, "COPERNIC Fuel Rod Design Computer Code," to the list of analytical methodologies in HNP TS 6.9.1.6.2. The NRC also approved the revision of the fuel centerline melt safety limit to that used in the COPERNIC code as to allow Duke Energy the ability to self-perform fuel rod mechanical analyses for HNP.

The list of COLR methodologies in HNP TS 6.9.1.6.2 has been reviewed to identify those methodologies that have been rendered obsolete by the NRC-approved Duke Energy methodologies listed above, as well as BAW-10231P-A. The obsolete methodologies, as identified in Section 2.3 of this license amendment request, are no longer planned for use in the design and safety analysis of core reloads. As such, an administrative change is requested to HNP TS 6.9.1.6.2 to remove the methodologies that have become obsolete and consolidate the remaining methodology listings.

The license amendment request also proposes the removal of details related to the cross-referencing of the HNP TS 6.9.1.6.2 COLR methodologies to their respective TS 6.9.1.6.1 COLR parameters. The HNP TS are based upon the format and content of the NUREG-0452, "Standard Technical Specifications for Westinghouse Pressurized Water Reactors," series. However, the NRC allows for selective incorporation of ISTS requirements (i.e., NUREG-1431 for Westinghouse Plants). As discussed in Section 16.0, Revision 3, "Technical Specifications," dated March 2010 (ADAMS Accession No. ML100351425), of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: Light Water Reactor (LWR) Edition," TS change requests for facilities with TS based on previous standard TS should comply with comparable provisions in current ISTS NUREGs to the extent possible or justify deviations from the ISTS. The proposed removal of the cross-reference content in HNP TS 6.9.1.6.2 would be in alignment with the structure provided in NUREG-1431, ISTS 5.6.3, "Core Operating Limits Report," which does not require the cross-referencing of each methodology to each applicable specification and/or parameter.

Furthermore, 10 CFR 50.36(c)(5), "Administrative controls," identifies administrative controls as the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner. The details currently contained in HNP TS 6.9.1.6.2 that cross-reference each methodology to its respective specification and/or parameter are extraneous to the required content of the Administrative Controls section of TS. As such, it is proposed for deletion from the HNP TS.

4.0 Regulatory Evaluation

4.1 Applicable Regulatory Requirements/Criteria

The following NRC requirements and guidance documents are applicable to the proposed changes:

10 CFR 50.36, Technical Specifications

The NRC's regulatory requirements related to the content of the TS are set forth in Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.36, "Technical specifications." This regulation requires that the TS include items in the following five specific categories: (1) safety limits, limiting safety system settings, and limiting control settings, (2) LCOs, (3) SRs, (4) design features, and (5) administrative controls. The regulation does not specify the particular requirements to be included in a plant's TS.

10 CFR 50.36(b) states that the TS will be derived from the analyses and evaluation included in the safety analysis report, and amendments thereto, submitted pursuant to 10 CFR 50.34. The minimum allowable RCS flow rates for HNP located in the TS are supported by existing safety analyses that demonstrate that the unit can meet safety analysis acceptance criteria at the revised flow rates. All applicable safety analyses either are not affected by the proposed changes or have been reanalyzed or evaluated at the proposed new values.

10 CFR 50.36(c)(5) states, "Administrative controls are the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner. Each licensee shall submit any reports to the Commission pursuant to approved technical specifications as specified in § 50.4." The proposed changes continue to meet the requirements of this regulation.

10 CFR 50.46, Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors

10 CFR 50.46 requires that the calculated ECCS performance for reactors with zircaloy or ZIRLO fuel cladding meet certain criteria. Additionally, Appendix K to 10 CFR Part 50, "ECCS Evaluation Models," presumes the use of zircaloy or ZIRLO fuel cladding when doing calculations for energy release, cladding oxidation, and hydrogen generation after a postulated LOCA. By letter dated February 24, 2012 (Reference 1.21), the NRC granted an exemption to HNP from the requirements of 10 CFR Part 50, Section 50.46, and Appendix K to 10 CFR Part 50 to allow use of M5 Fuel Cladding. The RLBLOCA analysis included in Enclosures 4 and 5 satisfies the requirements of 10 CFR 50.46(b), paragraphs (1) through (3), and the SBLOCA

analysis included in Enclosures 6 and 7 satisfies the requirements of 10 CFR 50.46(b), paragraphs (1) through (4).

10 CFR 50 Appendix A, General Design Criteria 10, 15, and 35

10 CFR 50, Appendix A, General Design Criterion 10 (Reactor design) states that the reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

10 CFR 50, Appendix A, General Design Criterion 15 (Reactor coolant system design) states that the reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.

10 CFR 50, Appendix A, General Design Criterion 35 (Emergency core cooling) states that a system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.

Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

The safety analyses and evaluations demonstrate that the analysis acceptance criteria will continue to be met with the proposed changes.

4.2 Precedent

RCS Flow Rate

The NRC previously approved a change to the TS for Catawba Nuclear Station Units 1 and 2 via letter dated June 2, 2016 (Reference 1.24), that decreased the TS minimum required RCS flow rate for each unit.

LOCA Methodology References

The NRC previously approved a change to the Millstone Power Station, Unit No. 2 (Millstone), TS to add the evaluation model EMF-2103(P)(A), Revision 3, “Realistic Large Break LOCA Methodology for Pressurized Water Reactors,” to the list of analytical methods used to establish core operating limits per letter dated January 24, 2017 (Reference 1.22). Millstone had proposed the change as a result of reanalyzing the LBLOCA with EMF-2103(P)(A), Revision 3 methodology.

COLR Reference List

The NRC previously approved changes to HNP TS 6.9.1.6.2 and H. B. Robinson Steam Electric Plant, Unit No. 2 (RNP) TS 5.6.5, "Core Operating Limits Report," for the removal of analytical methods no longer required to be listed, as per letters dated March 30, 2012, and December 29, 2011 (References 1.23 and 1.25), respectively. The license amendments also revised HNP and RNP TS to permit the use of M5 advanced alloy for fuel rod cladding and fuel assembly structural components in future operating cycles, but that is outside the scope of this license amendment request.

4.3 No Significant Hazards Consideration Determination Analysis

Pursuant to 10 CFR 50.90, Duke Energy Progress, LLC (Duke Energy), hereby requests a revision to the Technical Specifications (TS) for the Shearon Harris Nuclear Power Plant, Unit 1 (HNP). Duke Energy is proposing changes to TS 3/4.2.5, "DNB Parameters," and TS 6.9.1.6, "Core Operating Limits Report," in support of analysis development for HNP Cycle 24 and the introduction of reload batches of Framatome, Inc. (Framatome) GAIA fuel assemblies. HNP TS 3/4.2.5 would be revised to reflect a lower minimum Reactor Coolant System (RCS) flow rate, whereas TS 6.9.1.6.2 would reflect the incorporation of the Framatome topical report EMF-2103(P)(A), Revision 3, "Realistic Large Break LOCA {Loss-of-Coolant Accident} Methodology for Pressurized Water Reactors." HNP TS 6.9.1.6.2 will also be revised to reflect the removal of analytical methods no longer applicable for the determination of HNP core operating limits. As part of this license amendment request, Duke Energy is providing an updated HNP Small Break LOCA (SBLOCA) analysis reflecting the proposed lower minimum RCS flow rate and featuring Framatome GAIA fuel.

Duke Energy has evaluated whether a significant hazards consideration is involved with the proposed amendment(s) by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1) Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The reduction in HNP RCS minimum flow rate from 293,540 gpm to 290,000 gpm will not change the probability of actuation of any Engineered Safeguard Feature or any other device. The consequences of previously analyzed accidents have been found to be insignificantly different when the reduced flow rate is assumed. The proposed change will not result in the modification of any system interface that would increase the likelihood of an accident since these events are independent of the proposed change. The proposed amendment will not change, degrade, or prevent actions, or alter any assumptions previously made in evaluating the radiological consequences of an accident described in the Final Safety Analysis Report (FSAR).

The proposed change to HNP TS 6.9.1.6.2 permits the use of the NRC-approved generic Realistic Large Break LOCA (RLBLOCA) methodology developed by Framatome to analyze the HNP LBLOCA to ensure that the plant continues to meet the Emergency Core Cooling System (ECCS) performance acceptance criteria provided in 10 CFR 50.46. The RLBLOCA analysis demonstrates that HNP continues to satisfy the 10 CFR 50.46 performance acceptance criteria. The proposed change does not involve physical changes to any plant structure, system, or component. As such, the probability of occurrence for a previously analyzed accident is not significantly increased. Additionally, the proposed change does not affect the performance of any equipment used to mitigate the consequences of an analyzed accident. As a result, no analysis assumptions are violated, and there are no adverse effects on the factors that contribute to offsite or onsite dose as the result of an accident. The proposed change does not affect setpoints that initiate protective or mitigative actions. The proposed change ensures that plant structures, systems, or components are maintained consistent with the safety analysis and licensing bases. As such, the proposed change does not result in a significant increase in the consequences of a previously analyzed event.

The license amendment request also proposes the deletion of previously-approved analytical methods that are no longer planned for use by HNP. This change is administrative in nature as it removes methodologies that have become obsolete over time and will no longer be used for design and safety analysis of core reloads at HNP.

Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2) Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The proposed changes do not involve any physical alteration of plant systems, structures, or components. No new or different equipment is being installed. No installed equipment is being operated in a different manner. There is no alteration to the setpoints that initiate protective or mitigative actions. As a result, no new failure modes are being introduced. There are no changes in the methods governing normal plant operation, nor are the methods utilized to respond to plant transients altered.

The deletion of obsolete analytical methods is administrative in nature as it removes methodologies that will no longer be used for design and safety analysis of core reloads at HNP.

Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3) Does the proposed amendment involve a significant reduction in a margin of safety?

Response: No.

Margin of safety is related to the confidence in the ability of the fission product barriers to perform their design functions during and following an accident. These barriers include the fuel cladding, the RCS, and the containment system. The analyses and evaluations associated with the decrease in HNP RCS minimum flow rate confirmed that the applicable acceptance criteria continue to be met. Additionally, the proposed changes do not impact the condition or performance of structures, systems, setpoints, and components relied upon for accident mitigation. Approved methodologies will continue to be used to ensure that the plant continues to meet applicable design criteria and safety analysis acceptance criteria. The proposed changes have no effect on the ability of the plant to mitigate design basis accidents and ensure the consequences of the existing analyzed accidents remain bounding. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

Based on the above, Duke Energy concludes that the proposed changes present no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and accordingly, a finding of "no significant hazards consideration" is justified.

4.4 Conclusions

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

5.0 Environmental Consideration

Pursuant to 10 CFR 50.22(b), an evaluation of this license amendment request has been performed to determine whether or not it meets the criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) of the regulations.

Implementation of the proposed amendments will have no adverse impact upon HNP and will not contribute to any additional quantity of effluent being available for adverse environmental impact or personal exposure.

As such, it has been determined that the proposed changes do not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure.

Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment needs be prepared in connection with the proposed amendment.

6.0 References

1.1. AREVA NP, Inc., ANP-3011(P), Revision 1, "Harris Nuclear Plant Unit 1 Realistic Large Break LOCA Analysis," August 2011.

1.2. Framatome ANP Richland Inc., EMF-2103(P)(A), Revision 0, "Realistic Large Break LOCA Methodology for Pressurized Water Reactors," April 2003.

1.3. Saba, F., U.S. Nuclear Regulatory Commission, letter to C. Burton, Progress Energy Carolinas, Inc., "Shearon Harris Nuclear Power Plant, Unit 1 – Issuance of

Amendment RE: The Revision to Technical Specification Core Operating Limits Report References for Realistic Large Break Loss-of-Coolant-Accident Analysis (TAC No. ME6999)," May 30, 2012. {ADAMS Accession Number ML12076A103}

1.4. Barillas, M., U.S. Nuclear Regulatory Commission, letter to T. Hamilton, Duke Energy Progress, LLC, "Shearon Harris Nuclear Power Plant, Unit 1 – Issuance of Amendment: (1) Adopting TSTF-339, 'Relocate TS Parameters to the COLR'; (2) Adopting TSTF-5, 'Delete Safety Limit Violation Notification Requirements'; and (3) Removing a Plant Procedure Referenced in Technical Specifications as it Pertains to the Core Operating Limits Report (CAC No. MF8894; EPID L-2016-LLA-0023)," November 6, 2017. {ADAMS Accession Number ML17250A202}

1.5. Barillas, M., U.S. Nuclear Regulatory Commission, letter to J. Frisco, Duke Energy Corporation, "Shearon Harris Nuclear Power Plant, Unit 1 and H. B. Robinson Steam Electric Plant, Unit No. 2 – Issuance of Amendments Revising Technical Specifications for Methodology Report DPC-NE-2005-P, Revision 5, 'Thermal-Hydraulic Statistical Core Design Methodology' (CAC Nos. MF5872 and MF5873)," March 8, 2016. {ADAMS Accession Number ML16049A630}

1.6. Barillas, M., U.S. Nuclear Regulatory Commission, letter to K. Henderson, Duke Energy Corporation, "Shearon Harris Nuclear Power Plant, Unit 1 and H. B. Robinson Steam Electric Plant, Unit No. 2 – Issuance of Amendments Revising Technical Specifications for Methodology Reports DPC-NE-1008-P Revision 0, 'Nuclear Design Methodology Using CASMO-5/SIMULATE-3 for Westinghouse Reactors,' DPC-NF-2010 Revision 3, 'Nuclear Physics Methodology for Reload Design,' and DPC-NE-2011-P Revision 2, 'Nuclear Design Methodology Report for Core Operating Limits of Westinghouse Reactors' (CAC Nos. MF6648/MF6649 and MF7693/MF7694)," May 18, 2017. {ADAMS Accession Number ML17102A923}

1.7. Galvin, D., U.S. Nuclear Regulatory Commission, letter to S. Capps, Duke Energy Corporation, "Shearon Harris Nuclear Power Plant, Unit 1 and H. B. Robinson Steam Electric Plant, Unit No. 2 – Issuance of Amendments Revising Technical Specifications for Methodology Reports DPC-NE-3008-P, Revision 0, 'Thermal-Hydraulic Models for Transient Analysis,' and DPC-NE-3009-P, Revision 0, 'FSAR / UFSAR Chapter 15 Transient Analysis Methodology' (CAC Nos. MF8439 and MF8440; EPID L-2016-LLA-0012)," April 10, 2018. {ADAMS Accession Number ML18060A401}

1.8. Galvin, D., U.S. Nuclear Regulatory Commission, letter to J. Donahue, Duke Energy Corporation, "Shearon Harris Nuclear Power Plant, Unit 1 and H. B. Robinson Steam Electric Plant, Unit No. 2 – Issuance of Amendments Revising Technical Specifications to Support Self-Performance of Core Reload Design and Safety Analyses (EPID L-2017-LLA-0356)," April 29, 2019. {ADAMS Accession Number ML18288A139}

1.9. Hood, T., U.S. Nuclear Regulatory Commission, letter to T. Hamilton, Duke Energy Progress, LLC, "Shearon Harris Nuclear Power Plant, Unit 1 – Correction to Amendment No. 171 RE: Revision of Technical Specifications to Support Self-

Performance of Core Reload Design and Safety Analysis (EPID L-2017-LLA-0356)," October 18, 2019. {ADAMS Accession Number ML19290F980}

1.10. Duke Energy Progress, Inc., DPC-NE-3009-P-A, Revision 0, "FSAR / UFSAR Chapter 15 Transient Analysis Methodology," April 2018.

1.11. AREVA Inc., EMF-2103P-A, Revision 3, "Realistic Large Break LOCA Methodology for Pressurized Water Reactors," June 2016.

1.12. U.S. Nuclear Regulatory Commission, "Final Safety Evaluation for AREVA NP INC Topical Report EMF-2103(P), Revision 3, 'Realistic Large Break LOCA Methodology for Pressurized Water Reactors' (TAC No. MF2904)," June 17, 2016. {ADAMS Accession No. ML16172A334}

1.13. Framatome Inc., ANP-3767P (Proprietary) and ANP-3767NP (Non-Proprietary), Revision 0, "Harris Nuclear Plant Unit 1 Realistic Large Break LOCA Analysis with GAIA Fuel Design," July 2019. {Included as Enclosures 4 and 5}

1.14. Framatome Inc., ANP-3766P (Proprietary) and ANP-3766NP (Non-Proprietary), Revision 0, "Harris Nuclear Plant Unit 1 Small Break LOCA Analysis with GAIA Fuel Design," July 2019. {Included as Enclosures 6 and 7}

1.15. Framatome ANP Richland Inc., EMF-2328(P)(A), Revision 0, "PWR Small Break LOCA Evaluation Model, S-RELAP5 Based," March 2001.

1.16. AREVA Inc., EMF-2328(P)(A), Revision 0, Supplement 1(P)(A), Revision 0, "PWR Small Break LOCA Evaluation Model, S-RELAP5 Based," December 2016.

1.17. Duke Energy Progress, Inc., Methodology Report DPC-NE-3008-P-A, Revision 0, "Thermal-Hydraulic Models for Transient Analysis," April 2018.

1.18. Duke Energy Corporation, DPC-NE-2005-P-A, Revision 5, "Thermal-Hydraulic Statistical Core Design Methodology," March 2016.

1.19. Hamilton, T., Duke Energy Progress, LLC, letter to U.S. Nuclear Regulatory Commission, "Cycle 23 Core Operating Limits Report, Revisions 1 and 2," November 21, 2019. {ADAMS Accession Numbers ML19326A831 and ML19326A834}

1.20. Donahue, J., Duke Energy Corporation, letter to U.S. Nuclear Regulatory Commission, "Response to Request for Additional Information (RAI) Regarding Application to Revise Technical Specifications for Methodology Report DPC-NE-3009, Revision 0," October 9, 2017. {ADAMS Accession No. ML17282A023}

1.21. Billoch Colon, A, U.S. Nuclear Regulatory Commission, letter to C. Burton, Progress Energy Carolinas, Inc., "Shearon Harris Nuclear Power Plant, Unit 1 – Exemption Regarding Use of M5™ Alloy Fuel Rod Cladding (TAC No. ME5410)," February 24, 2012. {ADAMS Accession No. ML12025A161}

1.22. Guzman, R., U.S. Nuclear Regulatory Commission, letter to D. Stoddard, Dominion Nuclear Connecticut, Inc., "Millstone Power Station, Unit No. 2 – Issuance of Amendment RE: Realistic Large Break Loss-of-Coolant Accident Analysis (CAC No. MF7761)," January 24, 2017. {ADAMS Accession Number ML17025A218}

1.23. Billoch Colon, A, U.S. Nuclear Regulatory Commission, letter to C. Burton, Progress Energy Carolinas, Inc., "Shearon Harris Nuclear Power Plant, Unit 1 – Issuance of Amendment Re: The Use of AREVA's M5™ Advanced Alloy in Fuel Cladding and Fuel Assembly Components (TAC No. ME5409)," March 30, 2012. {ADAMS Accession No. ML12058A133}

1.24. Whited, J., U.S. Nuclear Regulatory Commission, letter to K. Henderson, Duke Energy Carolinas, LLC, "Catawba Nuclear Station, Units 1 and 2 – Issuance of Amendments Regarding Changes to Technical Specification 3.4.1, 'RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits' (CAC Nos. MF6355 and MF6356)," June 2, 2016. {ADAMS Accession Number ML16124A694}

1.25. Gleaves, W., U.S. Nuclear Regulatory Commission, letter to R. Duncan II, Carolina Power and Light Company, "H. B. Robinson Steam Electric Plant, Unit No. 2 – Issuance of an Amendment on Technical Specifications Related to Use of AREVA's M5 Advanced Alloy in Fuel Cladding and Fuel Assembly Components (TAC No. ME4911)," December 29, 2011. {ADAMS Accession Number ML11342A165}

1.26. Donahue, J., Duke Energy Corporation, letter to U.S. Nuclear Regulatory Commission, "Response to Request for Additional Information (RAI) Regarding Application to Revise Technical Specifications for Methodology Report DPC-NE-3009, Revision 0 (Part 2)," October 30, 2017. {ADAMS Accession No. ML17303B205}

ENCLOSURE 2

PROPOSED HNP TECHNICAL SPECIFICATION CHANGES

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1
DOCKET NO. 50-400
RENEWED LICENSE NUMBER NPF-63

8 PAGES PLUS THE COVER

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

BASES

FIGURE 2.1-1

REACTOR CORE SAFETY LIMITS - THREE LOOPS IN OPERATION

WITH MEASURED RCS FLOW > [~~293,540 GPM~~ X (1.0 + C_1)]

290,000 GPM

This figure is deleted from Technical Specifications and relocated to the COLR.

3/4.2.5 DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

3.2.5 The following DNB-related parameters shall be maintained within the following limits:

a. Reactor Coolant System T_{avg} ≤ the limit specified in the COLR, and

b. Pressurizer Pressure ≥ the limit specified in the COLR*, and

c. RCS total flow rate ≥ ~~293,540 gpm~~ and greater than or equal to the limit specified in the COLR.

290,000 gpm

APPLICABILITY: MODE 1.

ACTION:

With any of the above parameters not within its specified limit, restore the parameter to within its limit within 2 hours or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.2.5.1 Each of the parameters shown in Specification 3.2.5 shall be verified to be within its limit at the frequency specified in the Surveillance Frequency Control Program.

4.2.5.2 Verify, by precision heat balance, that RCS total flow rate is within its limit at the frequency specified in the Surveillance Frequency Control Program.**

---

* This limit is not applicable during either a THERMAL POWER Ramp in excess of ±5% RATED THERMAL POWER per minute or a THERMAL POWER step change in excess of ±10% RATED THERMAL POWER.

** Required to be performed within 24 hours after ≥ 95% RATED THERMAL POWER.

6.9.1.6 CORE OPERATING LIMITS REPORT

6.9.1.6.1 Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT (COLR) prior to each reload cycle, or prior to any remaining portion of a reload cycle, for the following:

a. SHUTDOWN MARGIN limits for Specification 3/4.1.1.1 and 3/4.1.1.2.

b. Moderator Temperature Coefficient Positive and Negative Limits and 300 ppm surveillance limit for Specification 3/4.1.1.3.

c. Shutdown Bank Insertion Limits for Specification 3/4.1.3.5.

d. Control Bank Insertion Limits for Specification 3/4.1.3.6.

e. Axial Flux Difference Limits for Specification 3/4.2.1.

f. Heat Flux Hot Channel Factor $F_Q(X,Y,Z)$ Limits for Specification 3/4.2.2.

g. Enthalpy Rise Hot Channel Factor $F_{\Delta H}(X,Y)$ Limits for Specification 3/4.2.3.

h. Boron Concentration for Specification 3/4.9.1.

i. Reactor Core Safety Limits Figure for Specification 2.1.1.

j. Overtemperature ΔT and Overpower ΔT setpoint parameters and time constant values for Specification 2.2.1.

k. Reactor Coolant System pressure, temperature, and flow Departure from Nucleate Boiling (DNB) limits for Specification 3/4.2.5.

l. Shutdown and Operating Boric Acid Tank and Refueling Water Storage Tank boron concentration limits for Specification 3/4.1.2.5 and 3/4.1.2.6.

m. ECCS Accumulators and Refueling Water Storage Tank boron concentration limits for Specification 3/4.5.1 and 3/4.5.4.

6.9.1.6.2 The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC at the time the reload analyses are performed, and the approved revision number shall be identified in the COLR.

~~a. XN-75-27(P)(A), "Exxon Nuclear Neutronics Design Methods for Pressurized Water Reactors," approved version as specified in the COLR.~~

~~(Methodology for Specification 3.1.1.2 - SHUTDOWN MARGIN - MODES 3, 4 and 5, 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Bank Insertion Limits, 3.1.3.6 - Control Bank Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor, and 3.9.1 - Boron Concentration).~~

~~b. ANF-89-151(P)(A), "ANF-RELAP Methodology for Pressurized Water Reactors: Analysis of Non-LOCA Chapter 15 Events," approved version as specified in the COLR.~~

~~(Methodology for Specification 2.2.1 – Reactor Trip System Instrumentation Setpoints, 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Bank Insertion Limits, 3.1.3.6 - Control Bank Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor, and 3.2.5 – DNB Parameters).~~

6.9.1.6 CORE OPERATING LIMITS REPORT (Continued)

~~c.~~ ~~XN-NF-82-21(P)(A), "Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations," approved version as specified in the COLR.~~

~~(Methodology for Specification 2.1.1 - Reactor Core Safety Limits, 2.2.1 - Reactor Trip System Instrumentation Setpoints, 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor, and 3.2.5 - DNB Parameters).~~

a ~~d.~~ XN-75-32(P)(A), "Computational Procedure for Evaluating Fuel Rod Bowing," approved version as specified in the COLR.

~~(Methodology for Specification 2.1.1 - Reactor Core Safety Limits, 2.2.1 - Reactor Trip System Instrumentation Setpoints, 3.2.2 - Heat Flux Hot Channel Factor, 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor, and 3.2.5 - DNB Parameters).~~

~~e.~~ ~~EMF-84-093(P)(A), "Steam Line Break Methodology for PWRs," approved version as specified in the COLR.~~

~~(Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Bank Insertion Limits, 3.1.3.6 - Control Bank Insertion Limits, 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor, and 3.2.5 - DNB Parameters).~~

b ~~f.~~ ~~ANP-3011(P), "Harris Nuclear Plant Unit 1 Realistic Large Break LOCA Analysis," Revision 1, as approved by NRC Safety Evaluation dated May 30, 2012.~~

~~(Methodology for Specification 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor).~~

~~g.~~ ~~XN-NF-78-44(NP)(A), "A Generic Analysis of the Control Rod Ejection Transient for Pressurized Water Reactors," approved version as specified in the COLR.~~

~~(Methodology for Specification 3.1.3.5 - Shutdown Bank Insertion Limits, 3.1.3.6 - Control Bank Insertion Limits, and 3.2.2 - Heat Flux Hot Channel Factor).~~

Replace text with "EMF-2103(P)(A), "Realistic Large Break LOCA Methodology for Pressurized Water Reactors," approved version as specified in COLR."

6.9.1.6 CORE OPERATING LIMITS REPORT (Continued)

~~h.~~ ~~ANF-88-054(P)(A), "PDC-3: Advanced Nuclear Fuels Corporation Power Distribution Control for Pressurized Water Reactors and Application of PDC-3 to H. B. Robinson Unit 2," approved version as specified in the COLR.~~

~~(Methodology for Specification 3.2.1 - Axial Flux Difference, and 3.2.2 - Heat Flux Hot Channel Factor).~~

~~i.~~ ~~EMF-92-081(P)(A), "Statistical Setpoint/Transient Methodology for Westinghouse Type Reactors," approved version as specified in the COLR.~~

~~(Methodology for Specification 2.1.1 – Reactor Core Safety Limits, 2.2.1 – Reactor Trip System Instrumentation Setpoints, 3.1.1.3 – Moderator Temperature Coefficient, 3.1.3.5 – Shutdown Bank Insertion Limits, 3.1.3.6 – Control Bank Insertion Limits, 3.2.1 – Axial Flux Difference, 3.2.2 – Heat Flux Hot Channel Factor, 3.2.3 – Nuclear Enthalpy Rise Hot Channel Factor, and 3.2.5 – DNB Parameters).~~

c ~~j.~~ EMF-92-153(P)(A), "HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel," approved version as specified in the COLR.

~~(Methodology for Specification 2.1.1 – Reactor Core Safety Limits, 2.2.1 – Reactor Trip System Instrumentation Setpoints, 3.2.3 – Nuclear Enthalpy Rise Hot Channel Factor, and 3.2.5 – DNB Parameters).~~

d ~~k.~~ BAW-10240(P)(A), “Incorporation of M5 Properties in Framatome ANP Approved Methods.”

~~(Methodology for Specification 2.1.1 – Reactor Core Safety Limits, 2.2.1 – Reactor Trip System Instrumentation Setpoints, 3.1.1.2 – SHUTDOWN MARGIN - MODES 3, 4 and 5, 3.1.1.3 – Moderator Temperature Coefficient, 3.1.3.5 – Shutdown Bank Insertion Limits, 3.1.3.6 – Control Bank Insertion Limits, 3.2.1 – Axial Flux Difference, 3.2.2 – Heat Flux Hot Channel Factor, 3.2.3 – Nuclear Enthalpy Rise Hot Channel Factor, 3.2.5 – DNB Parameters, and 3.9.1 – Boron Concentration).~~

~~l.~~ ~~EMF-96-029(P)(A), "Reactor Analysis Systems for PWRs," approved version as specified in the COLR.~~

~~(Methodology for Specification 3.1.1.2 - SHUTDOWN MARGIN - MODES 3, 4 and 5, 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Bank Insertion Limits, 3.1.3.6 - Control Bank Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor, and 3.9.1 - Boron Concentration).~~

e ~~m.~~ EMF-2328(P)(A) PWR Small Break LOCA Evaluation Model, S-RELAP5 Based, approved version as specified in the COLR.

~~(Methodology for Specification 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor).~~

~~n.~~ ~~EMF-2310(P)(A), “SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors”, approved version as specified in the COLR.~~

6.9.1.6 CORE OPERATING LIMITS REPORT (Continued)

~~(Methodology for Specification 2.2.1 – Reactor Trip System Instrumentation Setpoints, 3.1.1.3 – Moderator Temperature Coefficient, 3.1.3.5 – Shutdown Bank Insertion Limits, 3.1.3.6 – Control Bank Insertion Limits, 3.2.1 – Axial Flux Difference, 3.2.2 – Heat Flux Hot Channel Factor, 3.2.3 – Nuclear Enthalpy Rise Hot Channel Factor, and 3.2.5 – DNB Parameters).~~

f ~~o.~~ Mechanical Design Methodologies

~~XN-NF-81-58(P)(A), "RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model," approved version as specified in the COLR.~~

~~ANF-81-58(P)(A), "RODEX2 Fuel Rod Thermal Mechanical Response Evaluation Model," approved version as specified in the COLR.~~

~~XN-NF-82-06(P)(A), "Qualification of Exxon Nuclear Fuel for Extended Burnup," approved version as specified in the COLR.~~

~~ANF-88-133(P)(A), "Qualification of Advanced Nuclear Fuels' PWR Design Methodology for Rod Burnups of 62 GWd/MTU," approved version as specified in the COLR.~~

~~XN-NF-85-92(P)(A), "Exxon Nuclear Uranium Dioxide/Gadolinia Irradiation Examination and Thermal Conductivity Results," approved version as specified in the COLR.~~

~~EMF-92-116(P)(A), "Generic Mechanical Design Criteria for PWR Fuel Designs," approved version as specified in the COLR.~~

BAW-10231P-A, “COPERNIC Fuel Rod Design Computer Code,” approved version as specified in the COLR.

~~(Methodologies for Specification 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor).~~

g ~~p.~~ DPC-NE-2005-P-A, “Thermal-Hydraulic Statistical Core Design Methodology,” approved version as specified in the COLR.

~~(Methodology for Specification 3.2.3 – Nuclear Enthalpy Rise Hot Channel Factor)~~

h ~~q.~~ DPC-NE-1008-P-A, “Nuclear Design Methodology Using CASMO-5/SIMULATE-3 for Westinghouse Reactors,” as approved by NRC Safety Evaluation dated May 18, 2017.

~~(Methodology for Specification 3.1.1.2 – SHUTDOWN MARGIN – MODES 3, 4, and 5, 3.1.1.3 – Moderator Temperature Coefficient, 3.1.3.5 – Shutdown Bank Insertion Limits, 3.1.3.6 – Control Bank Insertion Limits, 3.2.1 – Axial Flux Difference, 3.2.2 – Heat Flux Hot Channel Factor, 3.2.3 – Nuclear Enthalpy Rise Hot Channel Factor, and 3.9.1 – Boron Concentration).~~

i ~~r.~~ DPC-NF-2010-A, “Nuclear Physics Methodology for Reload Design,” as approved by NRC Safety Evaluation dated May 18, 2017.

~~(Methodology for Specifications 3.1.1.1 – SHUTDOWN MARGIN – MODES 1 and 2, 3.1.1.2 – SHUTDOWN MARGIN – MODES 3, 4, and 5, 3.1.1.3 – Moderator Temperature Coefficient, 3.1.2.5 – Borated Water Source – Shutdown, 3.1.2.6 – Borated Water Sources – Operating, 3.1.3.5 – Shutdown Bank Insertion Limits, 3.1.3.6 – Control Bank Insertion Limits, 3.5.1 – ECCS Accumulators – Cold Leg Injection, 3.5.4 – ECCS Refueling Water Storage Tank, and 3.9.1 – Boron Concentration).~~

6.9.1.6 CORE OPERATING LIMITS REPORT (Continued)

j ~~s.~~ DPC-NE-2011-P-A, "Nuclear Design Methodology Report for Core Operating Limits of Westinghouse Reactors" as approved by NRC Safety Evaluation dated May 18, 2017.

~~(Methodology for Specification 3.1.3.5 – Shutdown Bank Insertion Limits, 3.1.3.6 – Control Bank Insertion Limits, 3.2.1 – Axial Flux Difference, 3.2.2 – Heat Flux Hot Channel Factor, and 3.2.3 – Nuclear Enthalpy Rise Hot Channel Factor).~~

k ~~t.~~ DPC-NE-3008-P-A, "Thermal-Hydraulic Models for Transient Analysis," as approved by NRC Safety Evaluation dated April 10, 2018.

~~(Methodology for Specification 3.1.1.3 – Moderator Temperature Coefficient, 3.1.3.5 – Shutdown Bank Insertion Limits, 3.1.3.6 – Control Bank Insertion Limits, 3.2.1 – Axial Flux Difference, 3.2.2 – Heat Flux Hot Channel Factor, and 3.2.3 – Nuclear Enthalpy Rise Hot Channel Factor).~~

l ~~u.~~ DPC-NE-3009-P-A, "FSAR / UFSAR Chapter 15 Transient Analysis Methodology," as approved by NRC Safety Evaluation dated April 10, 2018.

~~(Methodology for Specification 3.1.1.3 – Moderator Temperature Coefficient, 3.1.3.5 – Shutdown Bank Insertion Limits, 3.1.3.6 – Control Bank Insertion Limits, 3.2.1 – Axial Flux Difference, 3.2.2 – Heat Flux Hot Channel Factor, and 3.2.3 – Nuclear Enthalpy Rise Hot Channel Factor).~~

6.9.1.6.3 The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

6.9.1.6.4 The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements, shall be provided, upon issuance for each reload cycle, to the NRC Document Control Desk, with copies to the Regional Administrator and Resident Inspector.

6.9.1.7 STEAM GENERATOR TUBE INSPECTION REPORT

A report shall be submitted within 180 days after the initial entry into HOT SHUTDOWN following completion of an inspection performed in accordance with Specification 6.8.4.l. The report shall include:

a. The scope of inspections performed on each SG,
b. Degradation mechanisms found,
c. Nondestructive examination techniques utilized for each degradation mechanism,
d. Location, orientation (if linear), and measured sizes (if available) of service induced indications,
e. Number of tubes plugged during the inspection outage for each degradation mechanism,
f. The number and percentage of tubes plugged to date, and the effective plugging percentage in each steam generator, and
g. The results of condition monitoring, including the results of tube pulls and in-situ testing.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the NRC in accordance with 10 CFR 50.4 within the time period specified for each report.

6.10 DELETED

(PAGE 6-25 DELETED By Amendment No.92)

ENCLOSURE 3

**AFFIDAVITS FOR WITHHOLDING OF PROPRIETARY INFORMATION
(FRAMATOME, INC.)**

**SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1
DOCKET NO. 50-400
RENEWED LICENSE NUMBER NPF-63**

6 PAGES PLUS THE COVER

AFFIDAVIT

COMMONWEALTH OF VIRGINIA)
) ss.
CITY OF LYNCHBURG)

1. My name is Gayle Elliott. I am Deputy Director, Licensing & Regulatory Affairs, for Framatome Inc. (Framatome) and as such I am authorized to execute this Affidavit.

2. I am familiar with the criteria applied by Framatome to determine whether certain Framatome information is proprietary. I am familiar with the policies established by Framatome to ensure the proper application of these criteria.

3. I am familiar with the Framatome information contained in Licensing Report ANP-3767P, Revision 0, entitled "Harris Nuclear Plant Unit 1 Realistic Large Break LOCA Analysis with GAIA Fuel Design," dated July 2019 and referred to herein as "Document." Information contained in this Document has been classified by Framatome as proprietary in accordance with the policies established by Framatome for the control and protection of proprietary and confidential information.

4. This Document contains information of a proprietary and confidential nature and is of the type customarily held in confidence by Framatome and not made available to the public. Based on my experience, I am aware that other companies regard information of the kind contained in this Document as proprietary and confidential.

5. This Document has been made available to the U.S. Nuclear Regulatory Commission in confidence with the request that the information contained in this Document be withheld from public disclosure. The request for withholding of proprietary information is

made in accordance with 10 CFR 2.390. The information for which withholding from disclosure is requested qualifies under 10 CFR 2.390(a)(4) "Trade secrets and commercial or financial information."

6. The following criteria are customarily applied by Framatome to determine whether information should be classified as proprietary:

(a) The information reveals details of Framatome's research and development plans and programs or their results.

(b) Use of the information by a competitor would permit the competitor to significantly reduce its expenditures, in time or resources, to design, produce, or market a similar product or service.

(c) The information includes test data or analytical techniques concerning a process, methodology, or component, the application of which results in a competitive advantage for Framatome.

(d) The information reveals certain distinguishing aspects of a process, methodology, or component, the exclusive use of which provides a competitive advantage for Framatome in product optimization or marketability.

(e) The information is vital to a competitive advantage held by Framatome, would be helpful to competitors to Framatome, and would likely cause substantial harm to the competitive position of Framatome.

The information in this Document is considered proprietary for the reasons set forth in paragraphs 6(d) and 6(e) above.

7. In accordance with Framatome's policies governing the protection and control of information, proprietary information contained in this Document has been made available,

on a limited basis, to others outside Framatome only as required and under suitable agreement providing for nondisclosure and limited use of the information.

8. Framatome policy requires that proprietary information be kept in a secured file or area and distributed on a need-to-know basis.

9. The foregoing statements are true and correct to the best of my knowledge, information, and belief.

Commonwealth of Virginia

City of Lynchburg

SUBSCRIBED before me this 30th day of July, 2019.

Heidi Hamilton Elder

Heidi Elder
NOTARY PUBLIC, COMMONWEALTH OF VIRGINIA
MY COMMISSION EXPIRES: 12/31/2022
Reg. # 7777873

HEIDI HAMILTON ELDER
COMMONWEALTH
REGISTRATION NO.
7777873
MY COMM. EXPIRES
12/31/2022
OF VIRGINIA
NOTARY PUBLIC

A F F I D A V I T

1. My name is Gayle Elliott. I am Deputy Director, Licensing & Regulatory Affairs, for Framatome Inc. (Framatome) and as such I am authorized to execute this Affidavit.

2. I am familiar with the criteria applied by Framatome to determine whether certain Framatome information is proprietary. I am familiar with the policies established by Framatome to ensure the proper application of these criteria.

3. I am familiar with the Framatome information contained in Licensing Report ANP-3766P, Revision 0, entitled "Harris Nuclear Plant Unit 1 Small Break LOCA Analysis with GAIA Fuel Design," dated July 2019 and referred to herein as "Document." Information contained in this Document has been classified by Framatome as proprietary in accordance with the policies established by Framatome for the control and protection of proprietary and confidential information.

4. This Document contains information of a proprietary and confidential nature and is of the type customarily held in confidence by Framatome and not made available to the public. Based on my experience, I am aware that other companies regard information of the kind contained in this Document as proprietary and confidential.

5. This Document has been made available to the U.S. Nuclear Regulatory Commission in confidence with the request that the information contained in this Document be withheld from public disclosure. The request for withholding of proprietary information is made in accordance with 10 CFR 2.390. The information for which withholding from disclosure is requested qualifies under 10 CFR 2.390(a)(4) "Trade secrets and commercial or financial information."

6. The following criteria are customarily applied by Framatome to determine whether information should be classified as proprietary:

(a) The information reveals details of Framatome's research and development plans and programs or their results.

(b) Use of the information by a competitor would permit the competitor to significantly reduce its expenditures, in time or resources, to design, produce, or market a similar product or service.

(c) The information includes test data or analytical techniques concerning a process, methodology, or component, the application of which results in a competitive advantage for Framatome.

(d) The information reveals certain distinguishing aspects of a process, methodology, or component, the exclusive use of which provides a competitive advantage for Framatome in product optimization or marketability.

(e) The information is vital to a competitive advantage held by Framatome, would be helpful to competitors to Framatome, and would likely cause substantial harm to the competitive position of Framatome.

The information in this Document is considered proprietary for the reasons set forth in paragraphs 6(d) and 6(e) above.

7. In accordance with Framatome's policies governing the protection and control of information, proprietary information contained in this Document has been made available, on a limited basis, to others outside Framatome only as required and under suitable agreement providing for nondisclosure and limited use of the information.

8. Framatome policy requires that proprietary information be kept in a secured file or area and distributed on a need-to-know basis.

9. The foregoing statements are true and correct to the best of my knowledge, information, and belief.

Gayle Elliott

Commonwealth of Virginia)
) ss.
City of Lynchburg)

SUBSCRIBED before me this 30th day of July, 2019.

Steven Bullock
NOTARY PUBLIC, COMMONWEALTH OF VIRGINIA

STEVEN HUNTER BULLOCK
Notary Public
Commonwealth of Virginia
Reg. # 7714983
My Commission Expires June 30, 2021

ENCLOSURE 5

ANP-3767NP, REVISION 0, "HARRIS NUCLEAR PLANT UNIT 1 REALISTIC LARGE BREAK LOCA ANALYSIS WITH GAIA FUEL DESIGN" (NON-PROPRIETARY)

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1
DOCKET NO. 50-400
RENEWED LICENSE NUMBER NPF-63

100 PAGES PLUS THE COVER

framatome

Harris Nuclear Plant Unit 1 Realistic Large Break LOCA Analysis with GAIA Fuel Design

ANP-3767NP
Revision 0

Licensing Report

July 2019

Nature of Changes

Item	Section(s) or Page(s)	Description and Justification
1	All	Initial Issue

Contents

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Nomenclature

Acronym	Definition
AO	Axial Offset
BOCR	Beginning of Core Recovery
CCFL	Counter Current Flow Limiting
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_Q	Total Peaking Factor
$F_{\Delta H}$	Nuclear Enthalpy Rise Factor / Radial Peaking Factor
FSRR	Fuel Swell Rupture and Relocation
GDC	General Design Criteria
HHSI	High Head Safety Injection
HNP	Harris Nuclear Plant
LBLOCA	Large Break Loss of Coolant Accident
LHGR	Linear Heat Generation Rate
LHSI	Low Head Safety Injection
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
PCT	Peak Clad Temperature
PIRT	Phenomena Identification and Ranking Table
PWR	Pressurized Water Reactor
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
SE	Safety Evaluation
SIAS	Safety Injection Actuation Signal

Acronym	Definition
SG	Steam Generator
SRM	Swelling and Rupture Model
UTL	Upper Tolerance Limit(s)

ABSTRACT

This report describes and provides results from the RLBLOCA analysis for the Harris Nuclear Plant (HNP) Unit 1 GAIA fuel transition. The plant is a PWR Westinghouse 3-loop design with an analyzed thermal power of 2958 MWt (including measurement uncertainty) and dry atmospheric containment. The loops contain three RCPs, three U-tube steam generators and a pressurizer.

The analysis supports operation for Cycle 24 and beyond with Framatome's GAIA W17 fuel design using standard UO_2 fuel with 2%, 4%, 6%, and 8% Gd_2O_3 and $M5_{\mathbf{Framatome}}$ cladding. The analysis performed is the HNP-specific implementation of Framatome's NRC-approved EM RLBLOCA methodology (Reference 1), with exceptions noted. The analysis results confirm that the 10 CFR 50.46(b) paragraph (1) through (3) acceptance criteria (Reference 2) are met and serve as the basis for operation of HNP Unit 1 with Framatome GAIA W17 Fuel Design.

1.0 INTRODUCTION

This report summarizes the RLBLOCA analysis for Harris Nuclear Plant (HNP). The purpose of the RLBLOCA analysis is to support the fuel transition for HNP with the Framatome GAIA W17 Fuel Design. This analysis was performed in accordance with the NRC-approved S-RELAP5 methodology described in Reference 1 with the noted exceptions.

The plant is a Westinghouse 3-loop design with a rated thermal power of 2948 MWt and dry atmospheric containment. The loops contain three RCPs, three U-tube steam generators and a pressurizer. The analysis supports operation for Cycle 24 and beyond with Framatome's GAIA W17 fuel design using standard UO_2 fuel with 2, 4, 6, and 8 weight percent Gd_2O_3 and $M5_{\mathbf{Framatome}}$ cladding.

The analysis assumes full-power operation at a core power level of 2958 MWt (including measurement uncertainty), a total peaking factor (F_Q) up to a value of 2.62 (includes uncertainty), a radial peaking factor of 1.73 (includes uncertainty), and up to 3% SG tube plugging. A K(z) correction factor is applied to the total peaking factor F_Q to yield an elevation dependent peaking limit. This analysis also addresses typical operational ranges or technical specification limits (whichever is applicable) with regard to pressurizer pressure and level; accumulator pressure, temperature, and level; core inlet temperature; core flow; containment pressure and temperature; and refueling water storage tank temperature. The analysis explicitly analyzes fresh and once-burned fuel assemblies. The parameter specification for this analysis is provided in Table 3-1. The analysis also 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.

The UTL results, providing 95/95 simultaneous coverage for this evaluation, meet the 10 CFR 50.46(b) criteria with a PCT of 1820°F, a maximum local oxidation of 6.79 percent and a total core-wide oxidation of 0.07 percent. The PCT of 1820°F occurred in a once-burned 8% gad rod with an assembly burnup of 26.4 GWd/mtU.

2.0 DESCRIPTION OF ANALYSIS

2.1 *Acceptance Criteria*

The purpose of the analysis is to verify the adequacy of the HNP 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 separately during 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 is imposing 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 for EMF-2103(P) Rev. 3). The limitation is addressed in Table 3-4.

2.2 *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 vessel for the RCS loop. 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 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 1 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 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 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 of the emergency coolant being injected. 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.

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 expands 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.

2.3 *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 LOCA analysis.

The Framatome S-RELAP5 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).

There are two Condition Reports (CR), 2019-840 and 2019-1130, associated with the RLBLOCA calculations which were discovered and written after the HNP RLBLOCA calculations reported herein were performed. The CRs were evaluated against the HNP RLBLOCA analysis and confirmed that the figure of merits reported herein remain applicable and valid as a licensing basis for the HNP RLBLOCA analysis. The two CRs are listed and described below.

[ ]

The methodology (Reference 1) has been reviewed and approved by the NRC for performing LBLOCA analyses. However, some differences from the approved Reference 1 LBLOCA methodology were included in this analysis, as described below.

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 steam generator (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 2.6.

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.

2.4 GDC-35 Limiting Condition Determination

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. [

]

2.5 Overall Statistical Compliance to Criteria

[

]

2.6 Plant Description

The plant analyzed is the Harris Nuclear Plant Unit 1, W-designed PWR, which has three loops, each with a hot leg, a U-tube steam generator, and a cold leg with a RCP. The RCS includes one pressurizer connected to a hot leg. The ECCS includes one HHSI, one LHSI and one accumulator injection path per RCS loop. 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 S-RELAP5 model explicitly describes the RCS, reactor vessel, pressurizer, and ECCS. The ECCS includes an accumulator path and a LHSI/HHSI path per RCS loop.

The HHSI and LHSI feed into separate headers that connect to each cold leg pipe downstream of the RCP discharge. The ECCS pumped injection is modeled as a table of flow versus backpressure. This model also describes the secondary-side steam generator that is instantaneously isolated (closed main steam isolation valve and feedwater trip) at the time of the break. The analysis includes Framatome fuel with $M5_{\mathbf{Framatome}}$ cladding and utilizes the COPERNIC code for fuel calculations within S-RELAP5. The primary and secondary coolant systems for HNP were nodalized consistent with code input guidelines in Appendix A of Reference 1.

The Harris Unit 1 Cycle 24 core will contain co-resident Framatome HTP fuel and Framatome GAIA fuel designs. The two assembly types have different form loss coefficients for the grid spacers and the upper and lower tie plates. Therefore, consistent with EMF-2103, Revision 3 (Reference 1), a mixed core configuration is modeled.

As described in Section 2.3, 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 3-1. Table 3-2 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 3-3.

2.7 *SE Limitations*

The RLBLOCA analysis for HNP 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) are addressed in Table 3-4.

3.0 RLBLOCA ANALYSIS

3.1 *RLBLOCA Results*

[

] For a simultaneous coverage/confidence level of 95/95, the UTL values, [

] are a PCT of 1820°F, a MLO of 6.79 percent, and a CWO of 0.07 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 3-6. The sequence of event times for the demonstration case is provided in Table 3-7. The heat transfer parameter ranges for the demonstration case are provided in Table 3-8.

[

]

The analysis scatter plots for the case set are shown in Figure 3-1 through Figure 3-5. Figure 3-1 shows linear scatter plots of the key parameters sampled for all cases. Parameter labels appear to the left of each individual plot. 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 3-2. 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 3-2 and Figure 3-3 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 3-4 and Figure 3-5, respectively.

Figure 3-2 shows about 50% of the cases have a PCT during the blowdown period with PCTs occurring before 15 seconds. The next two clusters of PCTs occur either during the early reflood period or the late reflood period. Blowdown PCT cases are dominated by rapid RCS depressurization and stored energy content. Early reflood PCT cases are dominated by decay heat removal capacity, which is dependent on the accumulator liquid volume and the accumulator pressure setpoint. In general, plants with high pressure accumulators inject relatively 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 early reflood phase of the transient.

The high PCT cases in the upper part of Figure 3-2 are mainly influenced by the area of the break. This is demonstrated in Figure 3-3 which shows a general increasing trend in PCT with break size. From all sampled parameters, the break size has a dominant effect on PCT because of its high influence in the rate of primary depressurization. The scatter shown in Figure 3-3 reflects the influence of other transient phenomena and sampled parameters with direct impact on PCT such as global peaking and axial power distributions.

Figure 3-4 shows a strong 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 3-5, as higher PCT cases would have higher oxidation throughout the core.

The demonstration case is a split break with PCT occurring in the blowdown phase of the transient at 8.2 seconds. Figure 3-6 through Figure 3-20 show plots of key parameters from the S-RELAP5 calculations for the demonstration case. Reduced time scale versions are also shown for some of the plots such as the system pressures and break flow to better depict the early stages of the transient. The transient progression shown in the plots of key parameters for the demonstration case follows that described in Section 2.2.

Figure 3-21 compares the Beginning of Core Recovery (BOCR) times calculated by S-RELAP5 to the BOCR times predicted using the Counter Current Flow Limiting (CCFL) correlation developed by MPR Associates. Note that Figure 3-21 uses the total break area, while previous plots use break area per side.

3.2 *Conclusions*

This report describes and provides results from the RLBLOCA analysis for the HNP GAIA W17 Fuel Transition. The plant is a PWR Westinghouse 3-loop design with an analyzed thermal power of 2958 MWt (including measurement uncertainty) and dry atmospheric containment. The loops contain three RCPs, three U-tube steam generators and a pressurizer. The base model and the design inputs used are representative of the HNP Unit 1. 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 1820°F, a MLO of 6.79 percent and a CWO of 0.07 percent.

Table 3-1
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) Cladding thickness	0.0225 in.
	d) Pellet outside diameter	0.3225 in.
	e) Initial Pellet density	97.0 percent of theoretical
	f) Active fuel length	144 in.
	g) Gd_2O_3 concentrations	2, 4, 6, 8 w/o
	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	≤ 3 percent
2.0	**Plant Initial Operating Conditions**	
	2.1 Reactor Power	
	a) Analyzed reactor power	2958 MWt (includes measurement uncertainty)
	b) F_Q	≤2.62 (includes measurement uncertainty and axial dependency)
	c) K(z)	1.00 between 0 ft and 4 ft 0.96183 between 4 ft and 12 ft
	d) $F_{\Delta H}$	1.73 (includes measurement uncertainty)
	2.2 Fluid Conditions	
	a) Loop flow	108.04 Mlbm/hr ≤ M ≤ 115.72 Mlbm/hr
	b) RCS average temperature	583.3°F ≤ T ≤ 594.3°F
	c) Upper head temperature	~Tcold Temperature[1]
	d) Pressurizer pressure	2199.7 psia ≤ P ≤ 2299.7 psia
	e) Pressurizer liquid level	49.5 percent ≤ L ≤ 80.5 percent
	f) Accumulator pressure	574.7 psia ≤ P ≤ 704.7 psia
	g) Accumulator liquid volume	989 ft^3 ≤ V ≤ 1035 ft^3
	h) Accumulator temperature	70°F ≤ T ≤ 130°F (coupled with containment temperature)
	i) Accumulator resistance fL/D	As-built piping configuration
	j) Accumulator boron	2400 ppm

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

Table 3-1
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	125°F
	e) HHSI pump delay	29 s (No-LOOP) 29 s (LOOP)
	f) LHSI pump delay	29 s (No-LOOP) 29 s (LOOP)
	g) Initial containment pressure	14.7 psia
	h) Initial containment temperature	70°F ≤ T ≤ 130°F
	i) Containment sprays delay	0 s
	j) Containment spray water temperature	40°F
	k) LHSI Flow	

RCS Pressure (psia)	Intact Loops (gpm per loop)	Broken Loop (gpm)
0.00	924.49	1813.98
25.37	924.49	1813.98
35.43	888.15	1741.90
50.07	833.24	1632.95
70.06	753.65	1475.03
90.03	655.05	1278.47
100.02	600.22	1169.09
105.21	569.63	1107.78
110.07	539.84	1048.26
115.72	503.46	973.90
131.51	356.47	657.00
142.79	108.86	137.47
144.79	36.94	44.67
144.89	0.0	0.0

Table 3-1
RLBLOCA Analysis - Plant Parameter Values and Ranges
(Continued)

Plant Parameter	Parameter Value
l) HHSI Flow	

RCS Pressure (psia)	Intact Loops (gpm per loop)	Broken Loop (gpm)
0.00	121.33	190.67
15.00	121.33	190.67
125.50	118.19	185.72
592.23	105.68	166.06
849.99	97.79	153.67
1100.74	89.47	140.60
1674.69	69.61	109.38
1908.47	56.06	88.10
2129.25	40.64	63.86
2220.16	30.26	47.55
2283.52	12.26	19.26
2283.62	0.0	0.0

Table 3-2
Statistical Distribution Used for Process Parameters

Parameter	Operational Uncertainty Distribution	Parameter Range	Measurement Uncertainty Distribution[1]	Standard Deviation
Pressurizer Pressure (psia)	Uniform	2199.7 - 2299.7	N/A	N/A
Pressurizer Liquid Level (%)	Uniform	49.5 - 80.5	N/A	N/A
Accumulator Liquid Volume (ft^3)	Uniform	989 - 1035	N/A	N/A
Accumulator Pressure (psia)	Uniform	574.7 - 704.7	N/A	N/A
Containment/Accumulator Temperature (°F)	Uniform	70 - 130	N/A	N/A
Containment Volume ($x10^6$ ft^3)	Uniform	2.266 - 2.344	N/A	N/A
Initial Flow Rate (Mlbm/hr)	Uniform	108.04 - 115.72	N/A	N/A
Initial Operating Temperature (Tavg) (°F)	Uniform	583.3 - 594.3	N/A	N/A

[1] For the items marked N/A, the measurement uncertainty is included in the parameter range.

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

Heat Sink Description	Surface Area, ft^2	Thickness, ft	Material
1. HVAC Ductwork, Galvanized Carbon Steel	44097.0	0.00011	Zinc
		0.00227	Carbon Steel
2. HVAC Ductwork, Painted Carbon Steel	15286.0	0.001	Paint-4
		0.00227	Carbon Steel
3. Grating, Galvanized Carbon Steel	60367.0	0.00011	Zinc
		0.00733	Carbon Steel
4. Painted Carbon Steel, Exposed	134824.0	0.00083	Paint-5
		0.0132	Carbon Steel
5. Painted Carbon Steel (0.5" nominal)	83705.0	0.00083	Paint-5
		0.03389	Carbon Steel
6. Painted Carbon Steel (1" nominal)	28076.0	0.00083	Paint-5
		0.06644	Carbon Steel
7. Painted Carbon Steel (1.5" nominal)	11773.0	0.00083	Paint-5
		0.1084	Carbon Steel
8. Concrete walls with 3/16" liner	6886.0	0.00108	Paint-2
		0.01563	Carbon Steel
		2.29	Concrete
9. Electrical, Galvanized Carbon Steel	14934.0	0.00013	Zinc
		0.0355	Carbon Steel
10. Containment Dome	27056.0	0.00108	Paint-5
		0.04167	Carbon Steel
		2.5	Concrete
11. Containment Cylinder	63686.0	0.00108	Paint-5
		0.03125	Carbon Steel
		4.5	Concrete
12. Concrete Walls and Slabs (combined)	121128.0	0.00225	Paint-3
		2.54	Concrete
13. Hangers, Painted Carbon Steel (above flood level)	90447.0	0.00083	Paint-1
		0.0175	Carbon Steel
14. Equipment, Painted Carbon Steel	7815.0	0.00083	Paint-1
		0.02358	Carbon Steel
15. Equipment, Painted Carbon Steel	13880.0	0.00092	Paint-6
		0.44693	Carbon Steel

Heat Sink Description	Surface Area, ft^2	Thickness, ft	Material
16. Piping, Stainless Steel with Refractory Insulation	25988.0	0.16887	Insulation
		0.02135	Stainless Steel
17. Piping & Equipment, Stainless Steel	3819.0	0.01697	Stainless Steel
18. Painted Carbon Steel below LOCA Flood Level	12201.0	0.00108	Paint-2
		0.0268	Carbon Steel
19. Miscellaneous	2837.6	0.03067	Carbon Steel
20. Sump Strainer and Associated Structural Steel (combined)	5901.0	0.01057	Stainless Steel

Heat Sink Material	Thermal Conductivity, Btu/hr-ft-°F	Volumetric Heat Capacity, Btu/ft^3-°F
Concrete	1	31.9
Carbon Steel	25.9	53.5
Stainless Steel	8.6	54
Zinc	64	40.6
Insulation	0.01	0.00127
Paint-1	0.078	28.8
Paint-2	0.13833	10.55
Paint-3	0.16	14.93
Paint-4	0.2333	38.4
Paint-5	1.379	43.75
Paint-6	0.208	28.8

Table 3-4
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.	Harris is a 3-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 $M5_{\textbf{Framatome}}$ 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 3-5). [ ]

	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.	This is the first statistical calculation for this plant application.

Table 3-5
Compliance with 10 CFR 50.46(b)

UTL for 95/95 Simultaneous Coverage/Confidence		
Parameter	**Value**	**Case Number**
PCT, °F	1820	106
MLO, %	6.79	94
CWO, %	0.07	238

Characteristics of Case Setting the PCT UTL	
PCT, °F	1820
PCT Rod Type	Once-burned 8% gad Rod
Time of PCT, s	8.17
Elevation within Core, ft	1.78
Local Maximum Oxidation, %	2.26
Total Core-Wide Oxidation, %	0.019
PCT Rod Rupture Time, s	25.5
Rod Rupture Elevation within Core, ft	1.51

Table 3-6
Summary of Major Parameters for the Demonstration Case

Parameter	Value
Core Power (MWt)	2958
Time in Cycle (hrs)	288
Break Type	split
Break Size (ft^2/side)	2.6657
[ ]	[ ]
Limiting Rod Assembly Burnup (GWd/mtU)	26.4
Limiting Rod LHGR (kW/ft)	14.66
Limiting Rod Equivalent F_Q	2.45
Limiting Rod Radial Peak, $F_{\Delta H}$	1.73
Limiting Rod Axial Offset	-0.2458

Table 3-7
Calculated Event Times for the Demonstration Case

Event	Time (sec)
Break Opens	0.0
RCP Trip	0.0
SIAS Issued	0.4
PCT Occurred	8.2
Start of Broken Loop Accumulator Injection	14.2
Start of Intact Loop Accumulator Injection (Loop 2 and 3 respectively)	14.7 and 14.7
Beginning of Core Recovery (Beginning of Reflood)	27.8
HHSI Available	29.4
Broken Loop HHSI Delivery Began	29.4
Intact Loop HHSI Delivery Began (Loop 2 and 3 respectively)	29.4 and 29.4
LHSI Available	29.4
Broken Loop LHSI Delivery Began	29.4
Intact Loop LHSI Delivery Began (Loop 2 and 3 respectively)	29.4 and 29.4
Intact Loop Accumulator Emptied (Loop 2 and 3 respectively)	37.6 and 38.7
Broken Loop Accumulator Emptied	39.1
Transient Calculation Terminated	900

Table 3-8
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 (lb/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.

[ ]

4 Not important in pre-CHF heat transfer.

5 Vapor superheat is meaningless during blowdown and system depressurization.

Table 3-9
Fuel Rod Rupture Ranges of Parameters

Parameter Name	Minimum Value	Maximum Value

Figure 3-1
Scatter Plot Key Parameters

Figure 3-1
Scatter Plot Key Parameters (continued)

[ ]

Figure 3-2
PCT versus PCT Time Scatter Plot

Figure 3-3
PCT versus Break Size Scatter Plot

[ ]

Figure 3-4
Maximum Local Oxidation versus PCT Scatter Plot

[ ]

Figure 3-5
Total Core Wide Oxidation versus PCT Scatter Plot

Figure 3-6
Peak Cladding Temperature (Independent of Elevation) for the Demonstration Case

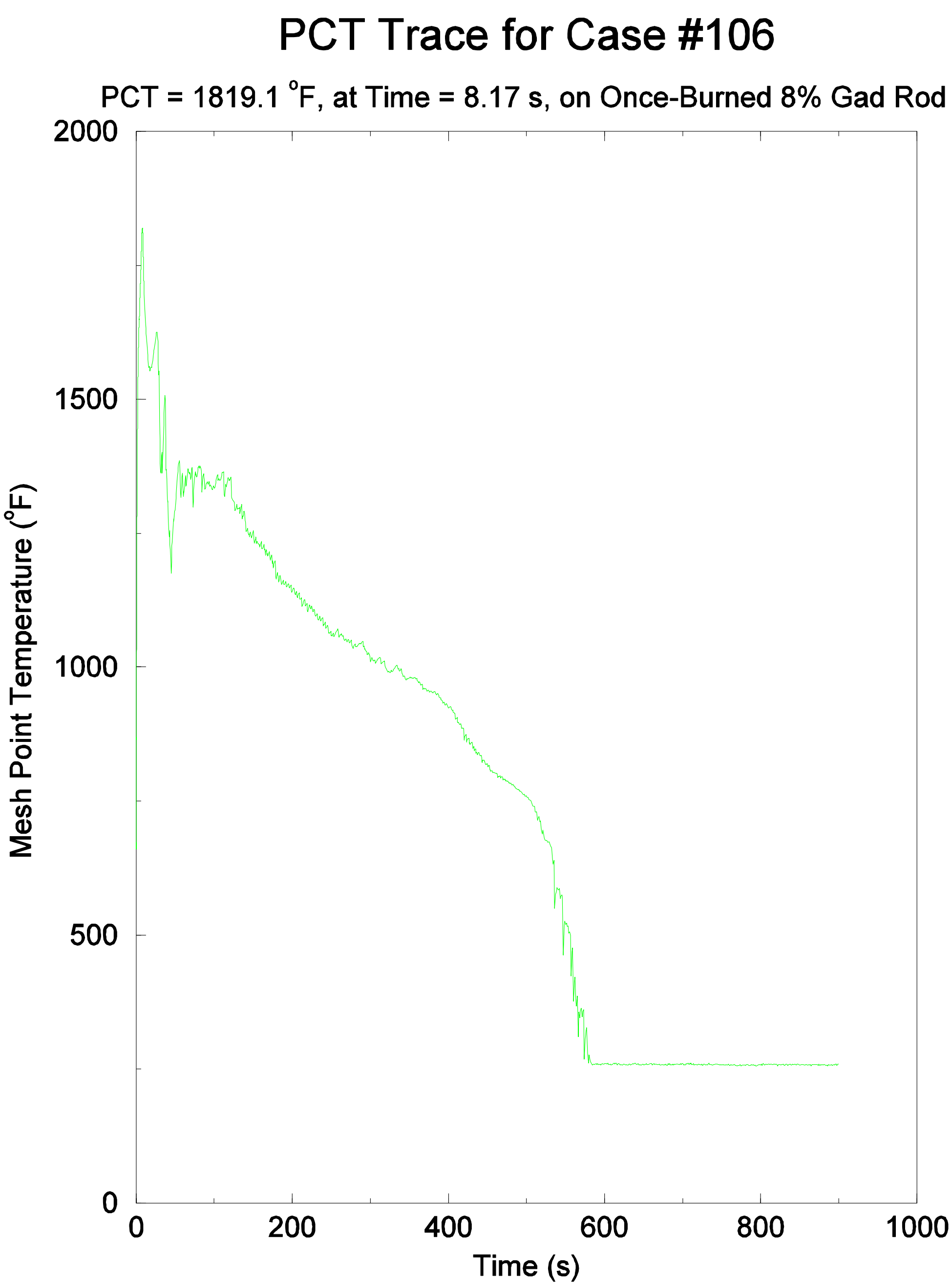

Figure 3-7
Break Flow for the Demonstration Case

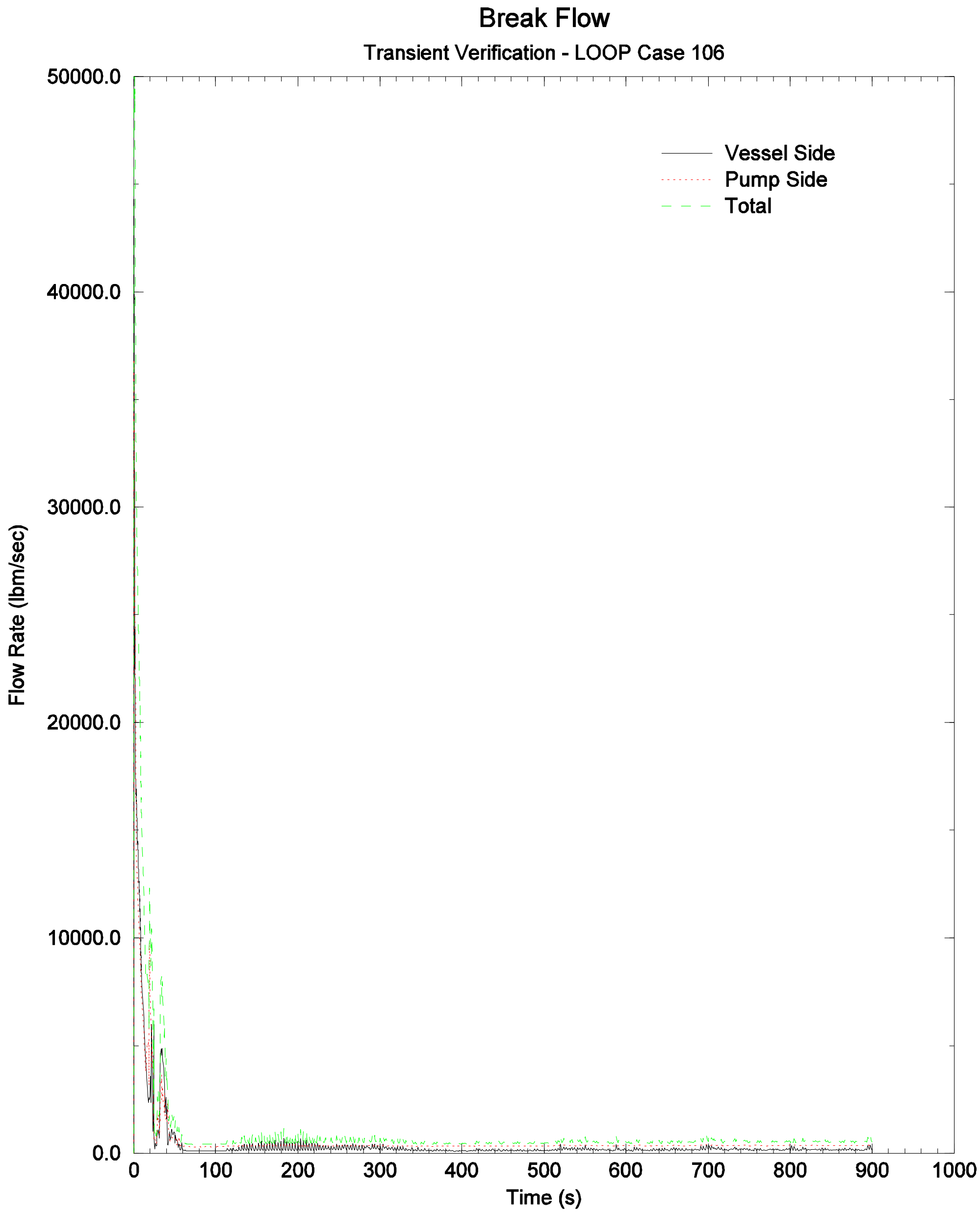

Figure 3-8
Break Flow for the Demonstration Case – Reduced Scale

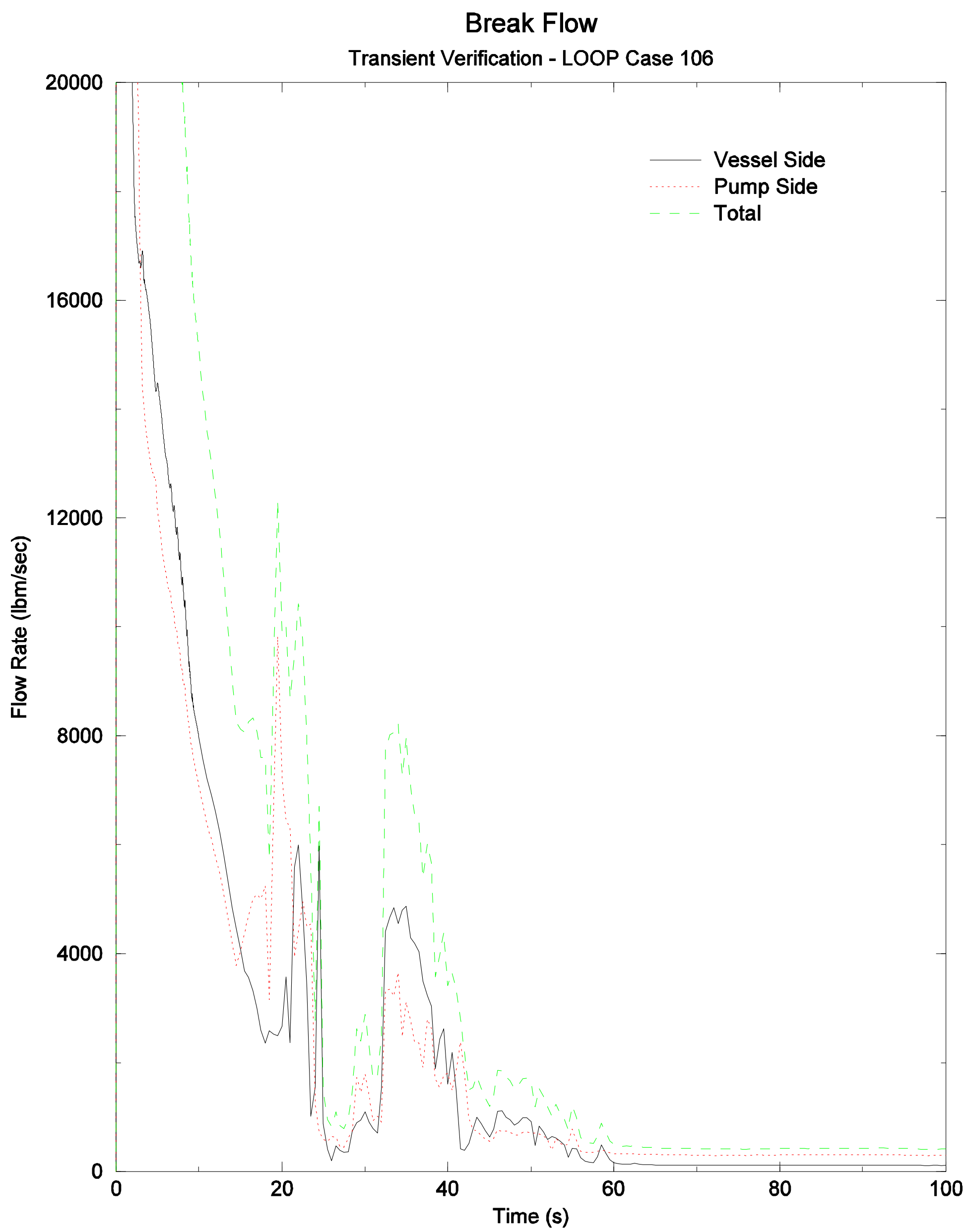

Figure 3-9
Core Inlet Mass Flux for the Demonstration Case

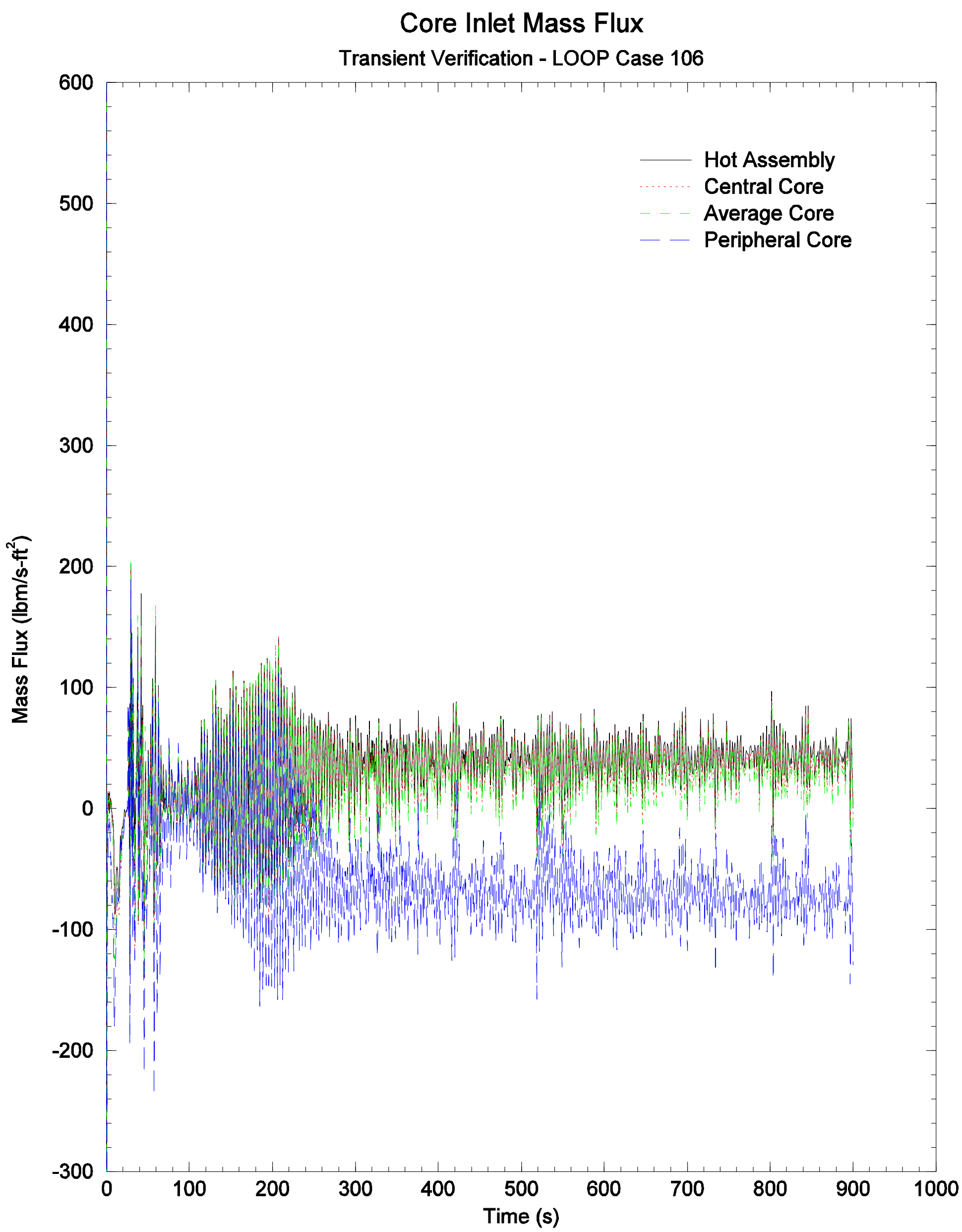

Figure 3-10
Core Outlet Mass Flux for the Demonstration Case

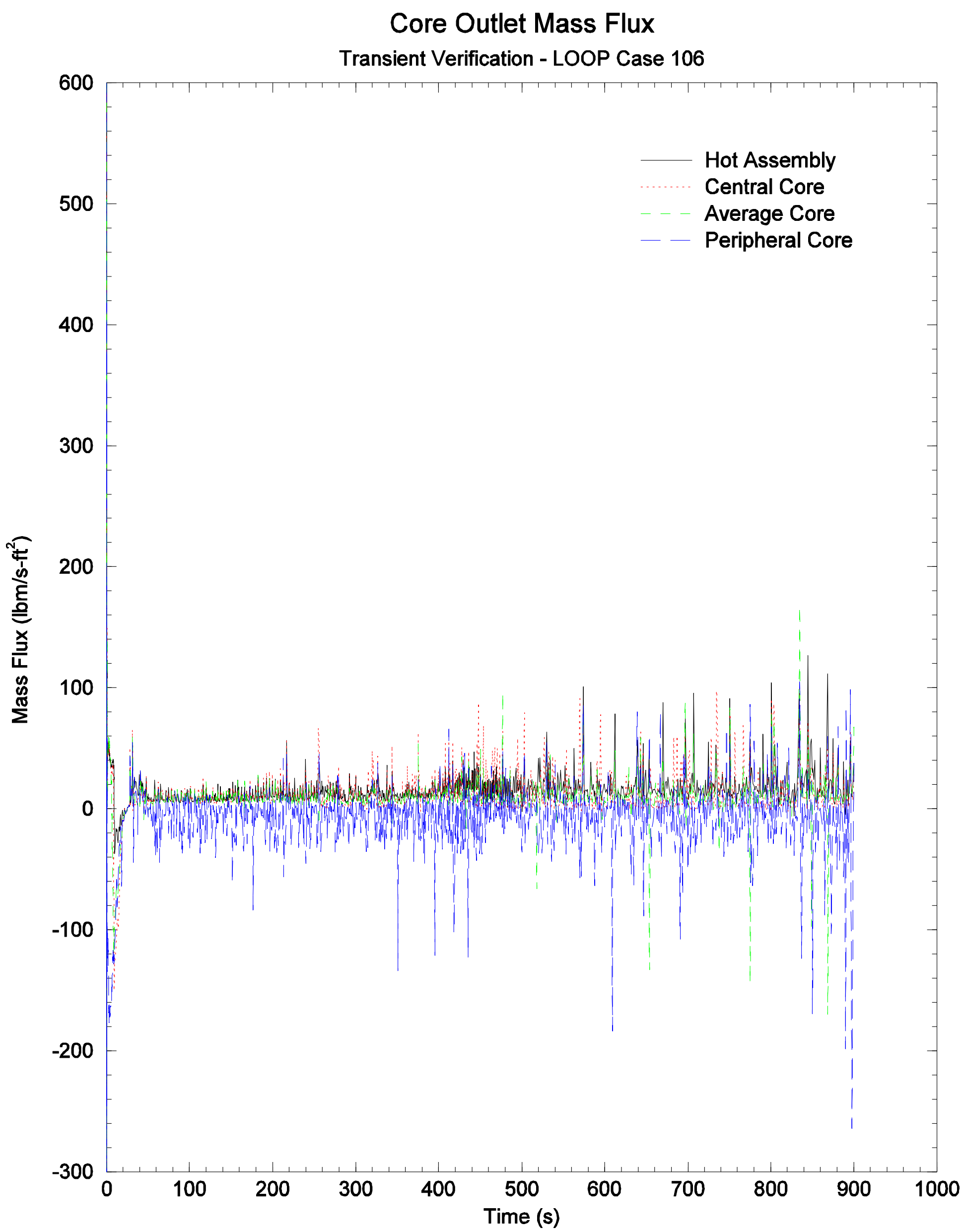

Figure 3-11
Void Fraction at RCS Pumps for the Demonstration Case

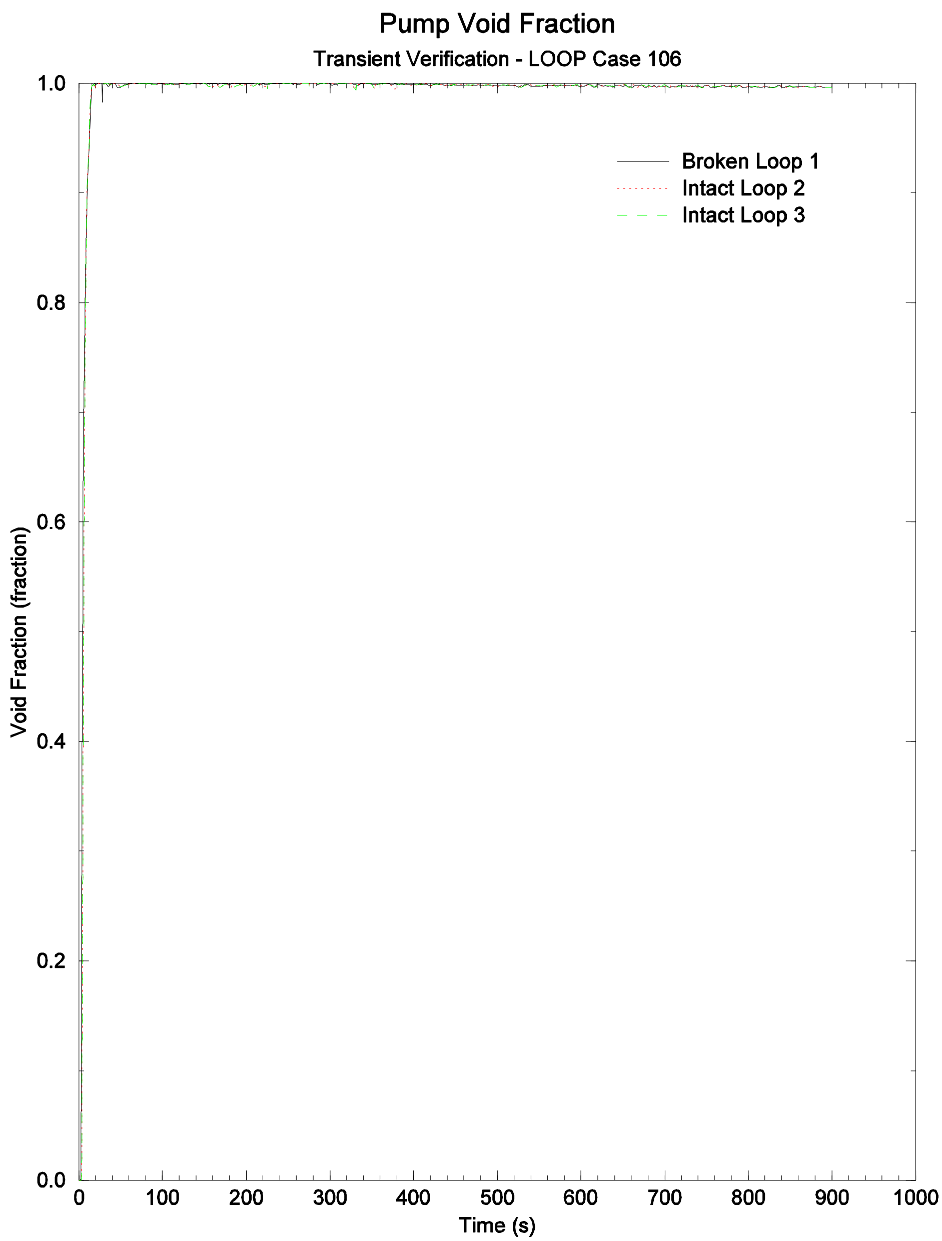

Figure 3-12
ECCS Flows (Includes Accumulator, HHSI and LHSI) for the Demonstration Case

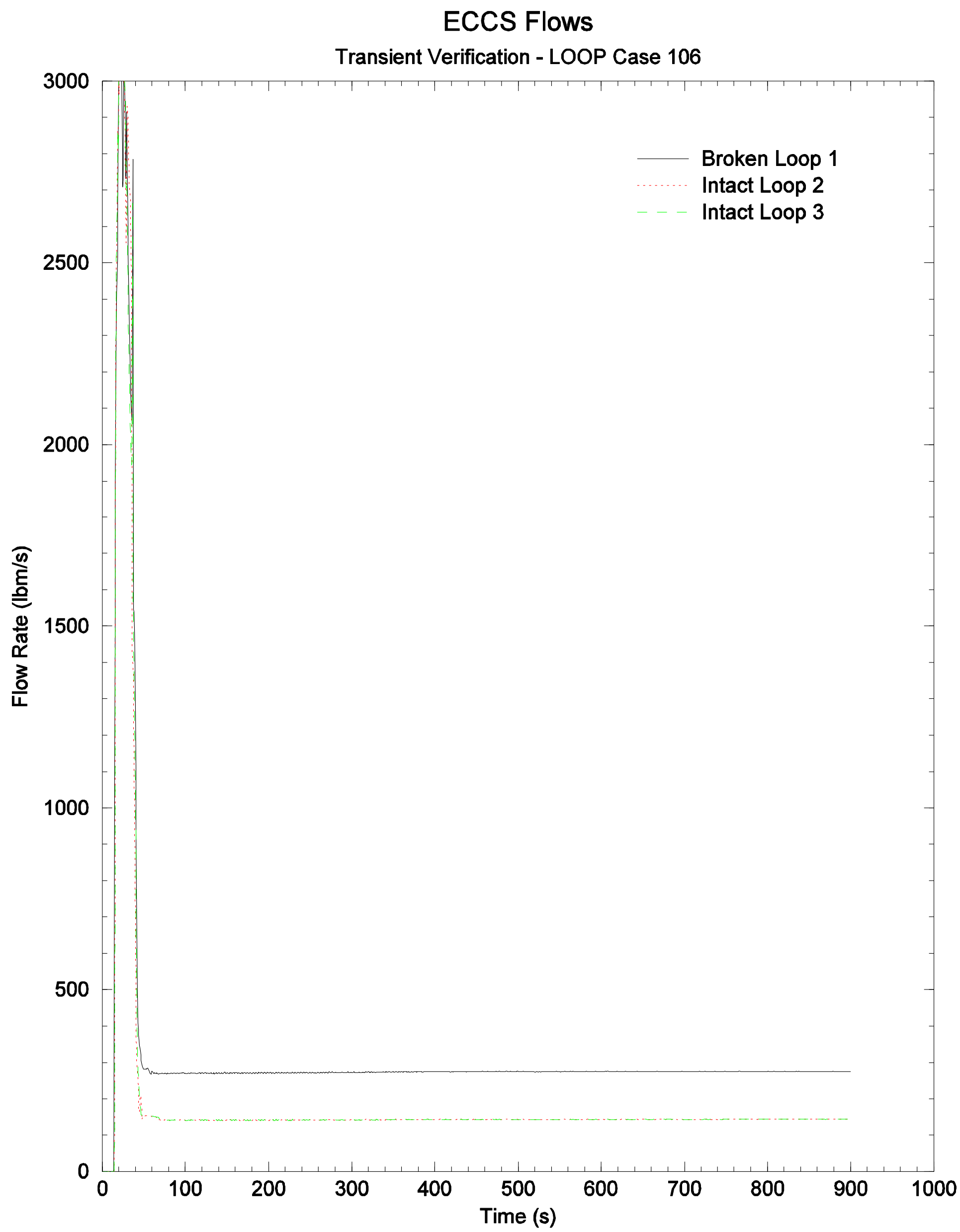

Figure 3-13
Upper Plenum Pressure for the Demonstration Case

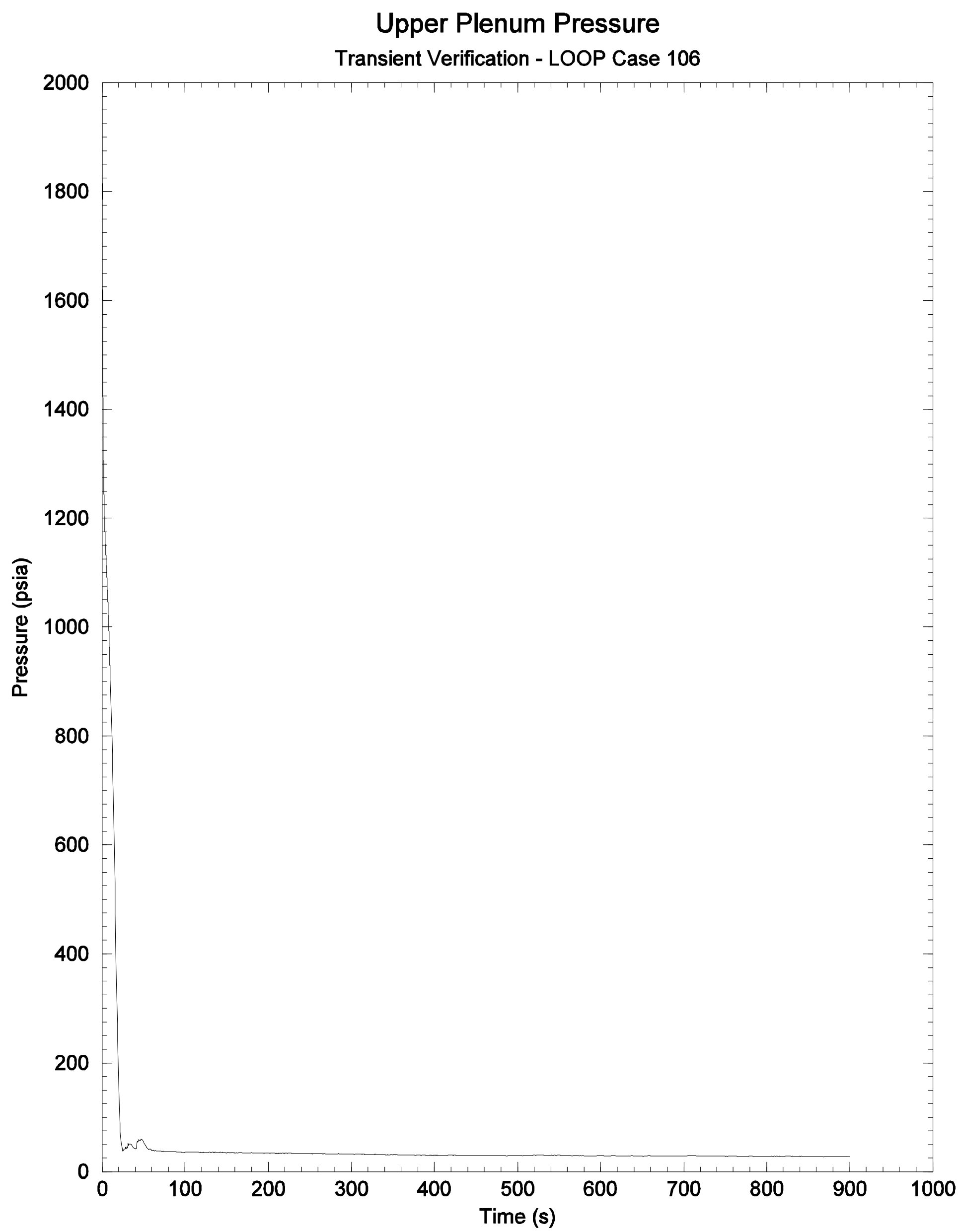

Figure 3-14
Upper Plenum Pressure for the Demonstration Case – Reduced Scale

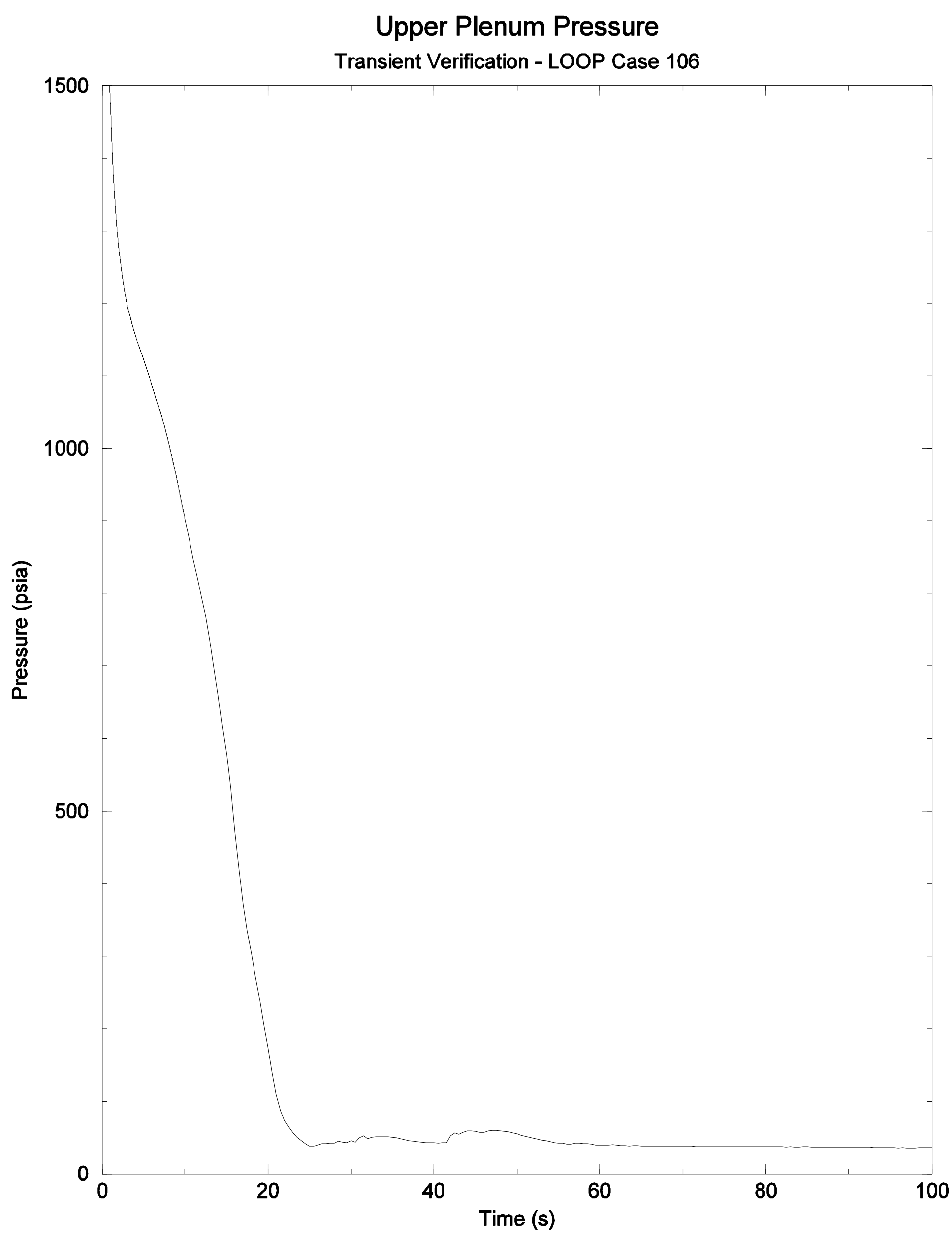

Figure 3-15
Collapsed Liquid Level in the Downcomer for the Demonstration Case

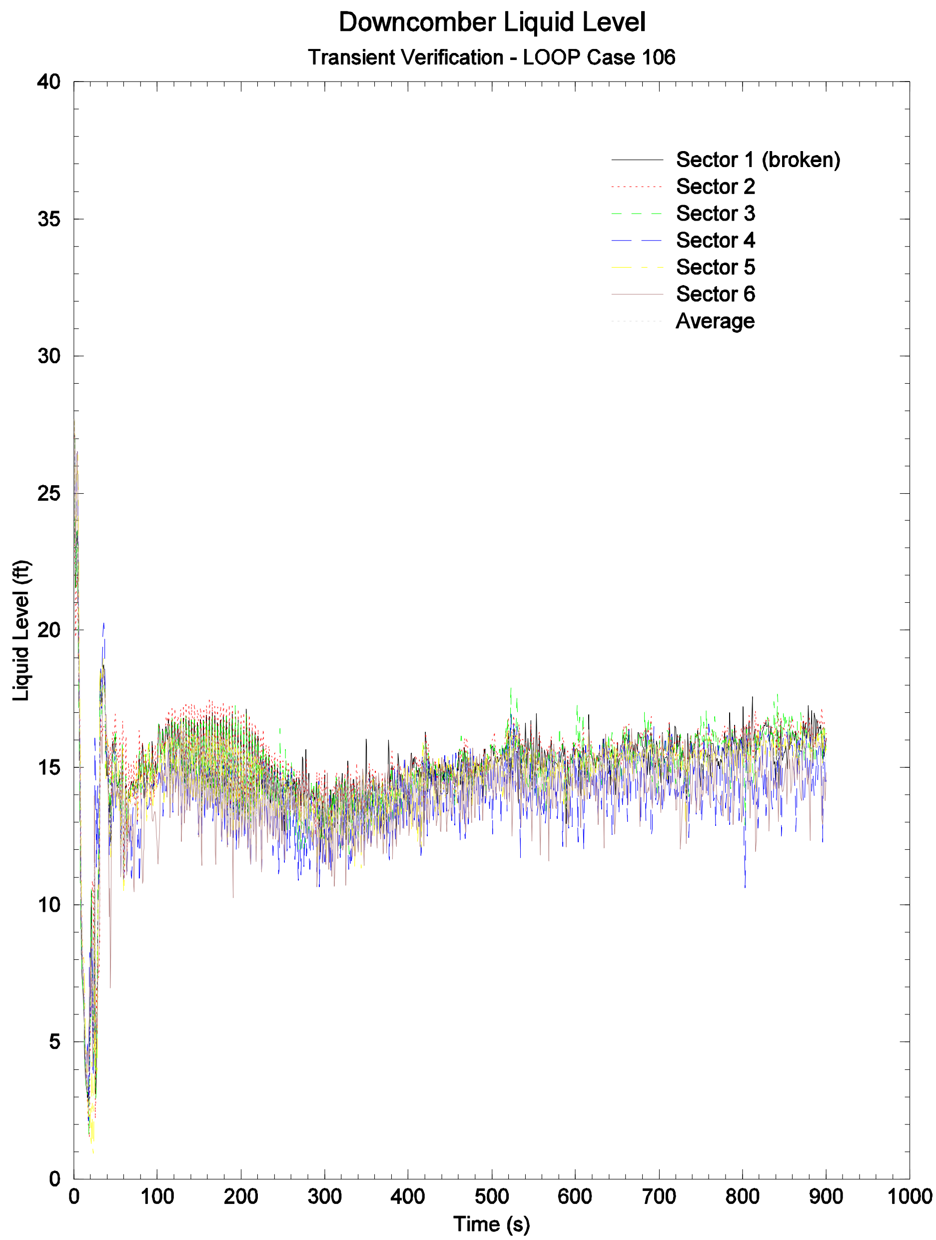

Figure 3-16
Collapsed Liquid Level in the Lower Plenum for the Demonstration Case

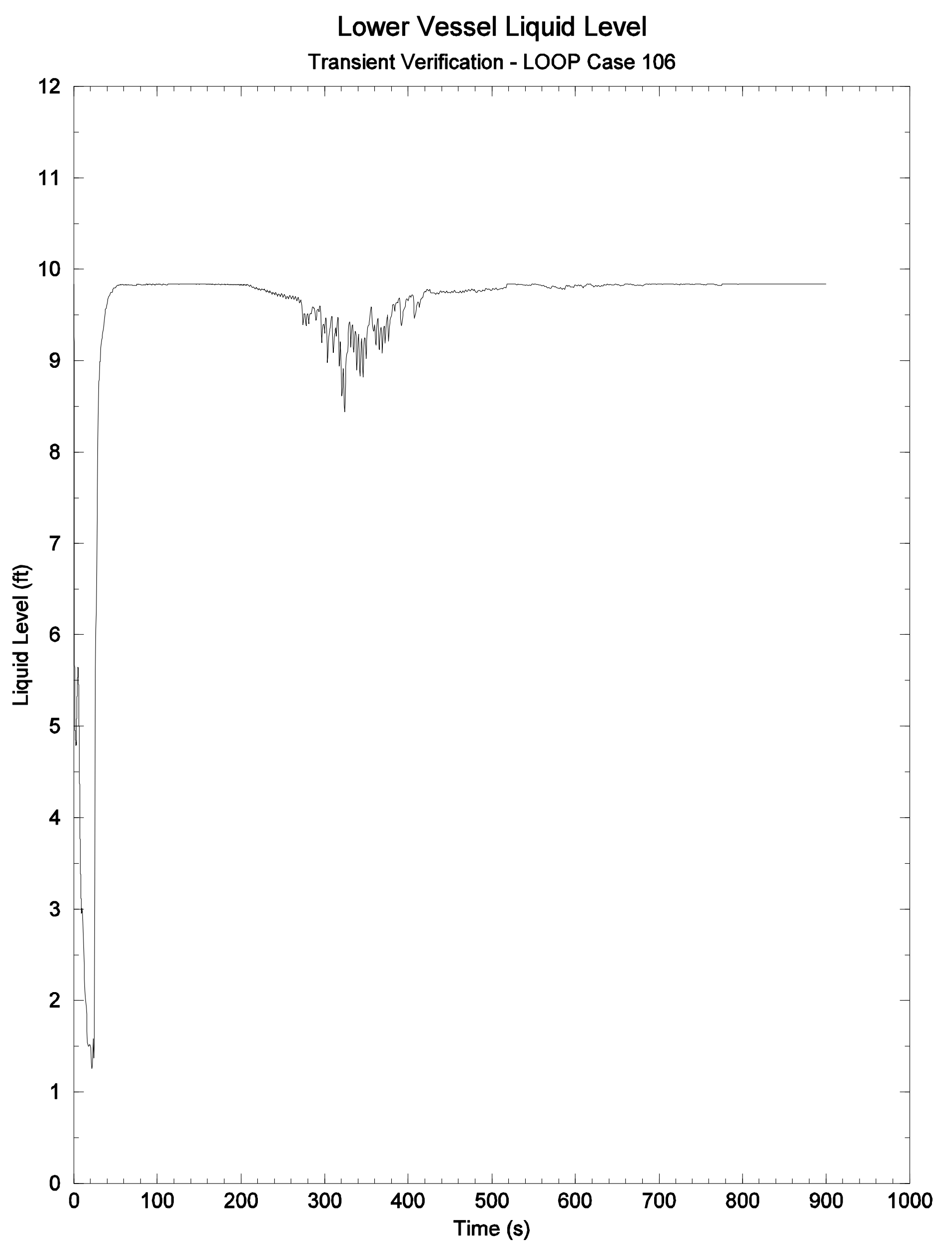

Figure 3-17
Collapsed Liquid Level in the Core for the Demonstration Case

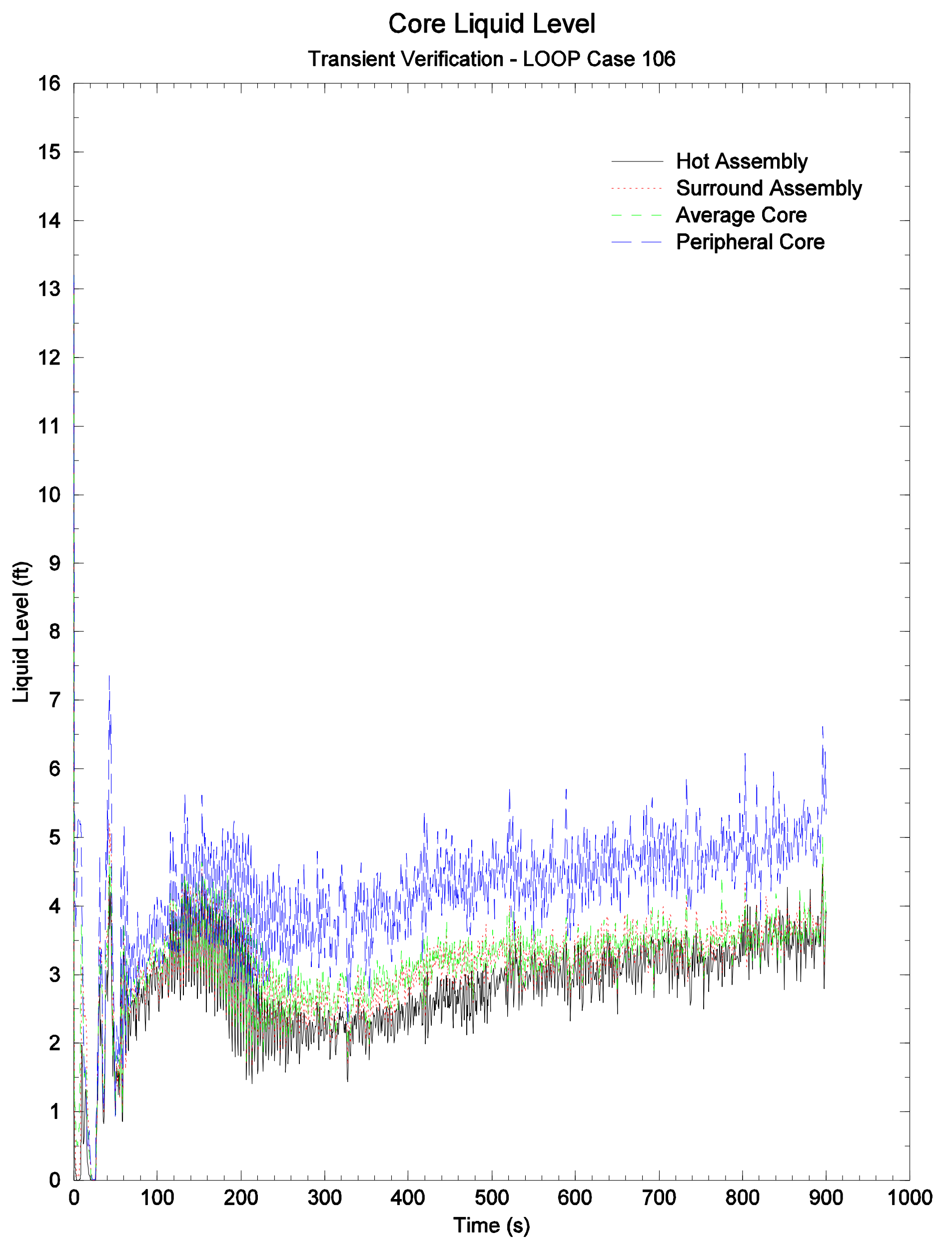

Figure 3-18
Containment and Loop Pressures for the Demonstration Case

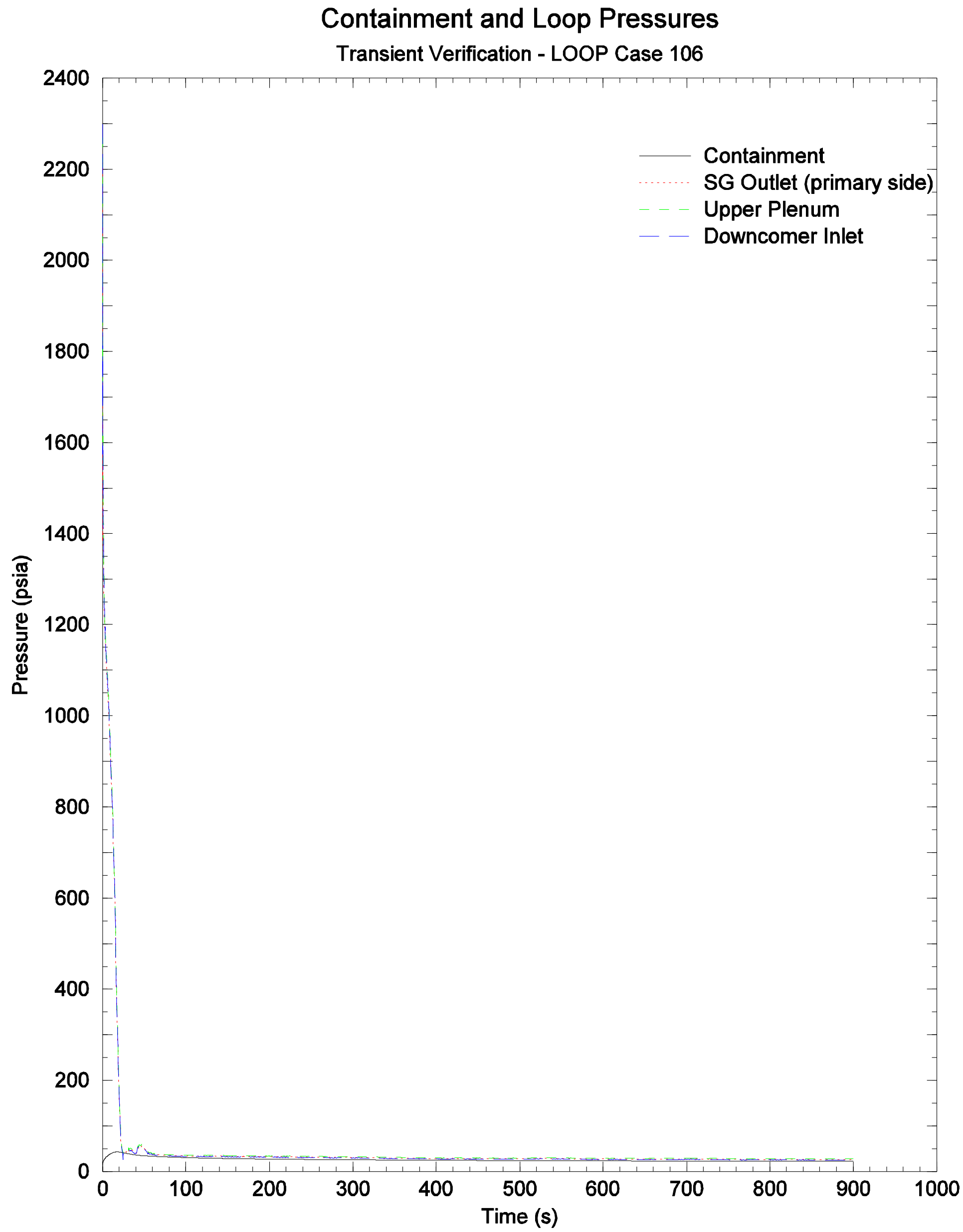

Figure 3-19
Containment and Loop Pressures for the Demonstration Case – Reduced Scale

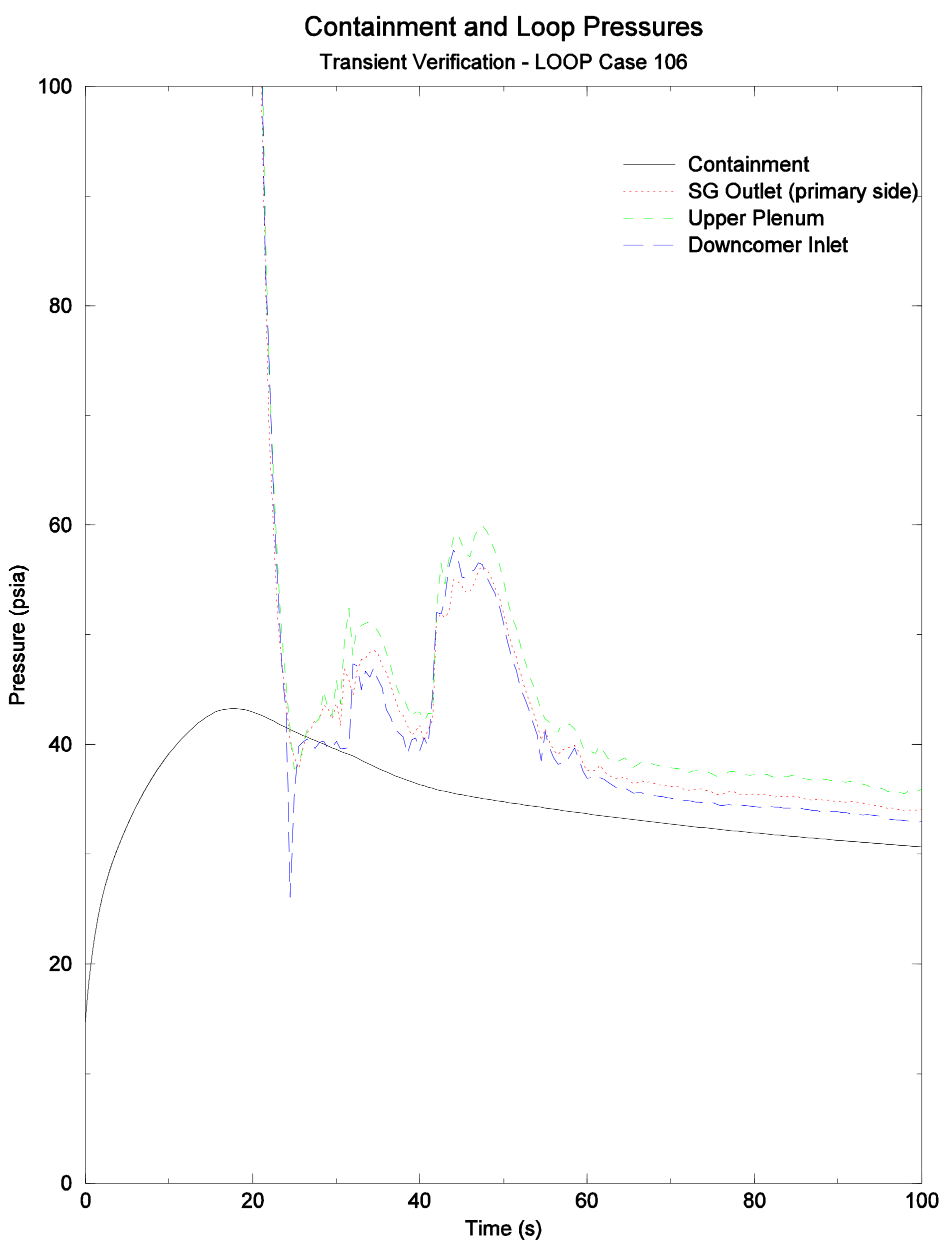

Figure 3-20
Pressure Differences between Upper Plenum and Downcomer for the Demonstration Case

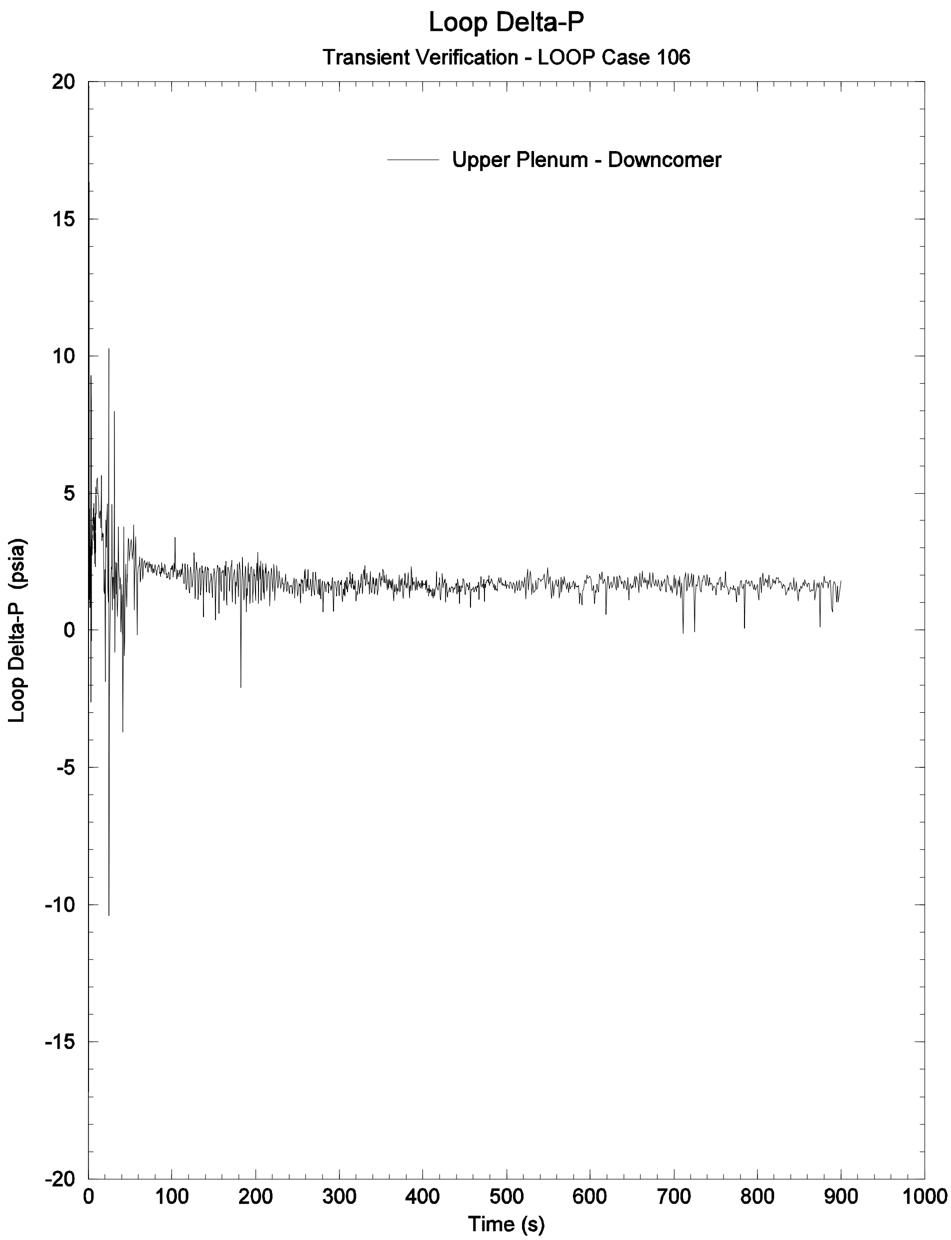

Figure 3-21
Validation of BOCR Time using MPR CCFL Correlation

[

]

Figure 3-22

[ ]

4.0 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. BAW-10231(P)(A) Revision 1, *COPERNIC Fuel Rod Design Computer Code*.

APPENDIX A SUMMARY OF KEY INPUT AND OUTPUT PARAMETERS

The following tables contain the sampled input values for the demonstration case set. 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 2958 MWt with a [ ].

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

[ ]

[ ]

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

[ ]

[

]

[ ]

[

]

[ ]

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

ENCLOSURE 7

ANP-3766NP, REVISION 0, "HARRIS NUCLEAR PLANT UNIT 1 SMALL BREAK LOCA ANALYSIS WITH GAIA FUEL DESIGN" (NON-PROPRIETARY)

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1
DOCKET NO. 50-400
RENEWED LICENSE NUMBER NPF-63

59 PAGES PLUS THE COVER

Harris Nuclear Plant Unit 1 Small Break LOCA Analysis with GAIA Fuel Design

ANP-3766NP
Revision 0

Licensing Report

July 2019

Nature of Changes

Item	Section(s) or Page(s)	Description and Justification
1	All	Initial Issue

Contents

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Nomenclature

Acronym	**Definition**
AFW	Auxiliary Feedwater
BOC	Beginning-of-Cycle
CFR	Code of Federal Regulations
CWO	Core Wide Oxidation
DC	Downcomer
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
EM	Evaluation Model
EOC	End-of-Cycle
$F_{\Delta H}$	Nuclear Enthalpy Rise Factor/Radial Peaking Factor
F_Q	Total Peaking Factor
Framatome	Framatome Inc.
HHSI	High Head Safety Injection
K(z)	Axial-Dependent Peaking Factor
LHSI	Low Head Safety Injection
LOCA	Loss-of-Coolant Accident
LOOP	Loss-of-Offsite Power
LS	Loop Seal
LSC	Loop Seal Clearing
MFW	Main Feedwater
MLO	Maximum Local Oxidation
MSSV	Main Steam Safety Valve
NRC	Nuclear Regulatory Commission
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RHR	Residual Heat Removal
RPS	Reactor Protection System
RT	Reactor Trip
RV	Reactor Vessel
PCT	Peak Cladding Temperature
PWR	Pressurized Water Reactor
PZR	Pressurizer
SBLOCA	Small Break Loss-of-Coolant Accident
SE	Safety Evaluation
SG	Steam Generator
SI	Safety Injection
SIAS	Safety Injection Actuation Signal

Acronym	Definition
SRM	Swelling and Rupture Model
TT	Turbine Trip
W17	Westinghouse 17x17 Fuel Rod Array

1.0 INTRODUCTION

This report summarizes the small break loss-of-coolant accident (SBLOCA) analysis for Shearon Harris Nuclear Plant Unit 1, a three-loop Westinghouse-designed pressurized water reactor (PWR) also referred to as Harris Nuclear Plant Unit 1. The purpose of the SBLOCA analysis is to support the transition to the Framatome Inc. (Framatome) GAIA W17 fuel design with $M5_{Framatome}$ cladding. This analysis was performed in accordance with the Nuclear Regulatory Commission (NRC)-approved S-RELAP5 methodology described in Reference 1 as modified by Reference 2. Reference 3 discusses the incorporation of $M5_{Framatome}$ properties into the SBLOCA methodology.

A complete spectrum of cold leg break sizes was considered, ranging from 1.00 inch diameter to 8.70 inch diameter. In addition, other supporting analyses prescribed by the methodology were performed which consider a delayed reactor coolant pump (RCP) trip, attached piping breaks, and sensitivity to reduced Emergency Core Cooling System (ECCS) fluid temperature.

The analysis supports plant operation at a core power level of 2958 MWt (including measurement uncertainty), a total peaking factor (F_Q) of 2.52 (represents maximum-allowed total peaking of 2.62 with uncertainty and axial-dependent factor (K(z)) applied), a nuclear enthalpy rise factor ($F_{\Delta H}$) of 1.73 (including uncertainty), and 3% steam generator (SG) tube plugging per SG.

2.0 SUMMARY OF RESULTS

An SBLOCA break spectrum analysis was performed for Harris Nuclear Plant Unit 1 using the NRC-approved Framatome SBLOCA method (Reference 1) as modified by Reference 2. The analyses are performed to demonstrate that the following acceptance criteria for ECCS, as stated in 10 CFR 50.46(b)(1-4) (Reference 4), have been met.

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.

4. Calculated changes in core geometry shall be such that the core remains amenable to cooling.

The limiting peak cladding temperature (PCT) is 1832°F for a 7.50 inch diameter cold leg pump discharge break. The same break produced the limiting maximum local oxidation (MLO) and core wide oxidation (CWO) values. The limiting total MLO and limiting CWO values for the spectrum are 4.89% and 0.017%, respectively. The total MLO value includes **[ ]**. The results of the analysis demonstrate the adequacy of the ECCS to support the 10 CFR 50.46(b) (1-4) criteria (Reference 4).

In addition to the cold leg pump discharge break spectrum analysis, three studies were performed to consider a delayed RCP trip, break in an attached pipe and sensitivity to reduced ECCS temperature. The results of the delayed RCP trip study demonstrated that there is at least 5 minutes for operators to trip all three RCPs after the specified trip criteria being met. The attached piping study analyzed a break in both the pumped safety injection (SI) line connection and accumulator line. The ECCS temperature sensitivity study analyzed the sensitivity to ECCS fluid temperatures reduced from those used in the break spectrum analysis. The conclusions of these studies support the applicability of the break spectrum as the licensing basis.

3.0 DESCRIPTION OF ANALYSIS

Section 3.1 of this report provides a brief general description of the postulated SBLOCA event. Section 3.2 describes the analytical models used in the analysis. Section 3.3 presents a description of the Harris Nuclear Plant Unit 1 plant parameters and outlines the system parameters used in the SBLOCA analysis. Section 3.4 describes compliance with the NRC Safety Evaluation (SE) of the methodology.

3.1 *Description of SBLOCA Event*

The postulated SBLOCA is defined as a break in the RCS pressure boundary with an area less than or equal to 10% of the cold leg pipe area. The most limiting break location is in the cold leg pipe on the discharge side of the RCP. This break location results in the largest amount of RCS inventory loss, the largest fraction of ECCS fluid ejected out through the break, and the largest pressure drop between the core exit and the top of the downcomer (DC). This produces the greatest degree of core uncovery, the longest fuel rod heatup time, and consequently, the greatest challenge to the 10 CFR 50.46(b)(1-4) criteria (Reference 4).

The SBLOCA event progression develops in the following distinct phases: (1) subcooled depressurization (also known as blowdown), (2) natural circulation, (3) loop seal clearing, (4) core boil-off, (5) core recovery and long-term cooling. The duration of each of these phases is break size and system dependent.

Following the break, the RCS rapidly depressurizes to the saturation pressure of the hot leg fluid. During the initial depressurization phase, a reactor trip is generated on low pressurizer pressure; the turbine is tripped on the reactor trip. The assumption of a loss-of-offsite-power (LOOP) concurrent with the reactor scram results in RCP trip.

In the second phase of the transient, the RCS transitions to a quasi-equilibrium condition in which the core decay heat, leak flow, SG heat removal, and system hydrostatic head balance combine to control the core inventory. During this period, the RCPs are coasting down and the system drains top down with voids beginning to form at the top of the SG tubes and continuing to form in the reactor vessel (RV) upper head and at the top of the RV upper plenum region. Also, the loop seals remain plugged during this phase, trapping vapor generated by the core in the RCS, resulting in a low quality flow at the break.

The third phase in the transient is characterized by loop seal clearing (LSC). During this phase the loop seal, with liquid trapped in the RCP suction piping can prevent steam from venting via the break. The maximum pressure difference between the RV upper head and DC is reached when the liquid level on the downhill side of the SG is depressed to the elevation of the horizontal loop seal piping. When this point is reached, the loop seal clears, and the trapped steam can be vented to the break. For some of the break sizes, the transient develops slowly, and the core can become temporarily uncovered in this LSC process. Following LSC, the break flow transitions to primarily steam and the core recovers to approximately the cold leg elevation, as pressure imbalances throughout the RCS are relieved.

The fourth phase is characterized as core boil-off. With the loop seal cleared, the venting of steam through the break causes a rapid RCS depressurization below the secondary pressure. As boiling increases in the core, the core mixture level decreases. The core mixture level will reach a minimum, in some cases resulting in deep core uncovery. The transient boil-off period ends when the core liquid level reaches this minimum. At this time, the RCS has depressurized to the point where ECCS flow into the RV matches the rate of boil-off from the core.

The last phase of the transient is characterized as core recovery. The core recovery period extends from the time at which the core mixture level reaches a minimum in the core boil-off phase until all parts of the core are quenched and covered by a low quality mixture. Core recovery is provided by pumped injection and passive accumulator injection when the RCS pressure decreases below the accumulator pressure. Generally, PCT occurs at the beginning of the core recovery phase before the mixture level has increased high enough to provide enhanced cooling to the PCT location on the hot rod.

The SBLOCA transient progression is dependent on the size of the break and is typically broken into three different break size ranges. For break sizes towards the larger end of the break spectrum, significant RCS inventory loss results in more rapid RCS depressurization to the accumulator actuation pressure. Accumulator flow provides sufficient inventory early in the transient to limit the core uncovery and hot rod heatup is typically not significant. For break sizes in the middle of the spectrum, the rate of inventory loss from the RCS is such that the high head safety injection (HHSI) pumps cannot preclude significant core uncovery. The RCS depressurization rate is slow, extending the time required to reach the accumulator injection pressure, if reached at all. This tends to maximize the heatup time of the hot rod which produces the maximum PCT and local cladding oxidation. Break sizes in this range, will either exhibit core recovery with the HHSI pumped injection alone while the RCS pressure remains barely above the accumulator injection setpoint, or exhibit core recovery from accumulator injection. For break sizes at the low end of the spectrum, the RCS pressure does not reach the accumulator injection pressure. However, RCS inventory loss is not significant and typically within the means of HHSI makeup capacity such that core uncovery is minimal if not precluded.

3.2 *Analytical Methods*

The Framatome S-RELAP5 SBLOCA 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. The appropriate conservatisms, as prescribed by Appendix K of 10 CFR 50 (Reference 5), are incorporated. This analysis was performed in accordance with the NRC-approved S-RELAP5 SBLOCA methodology described in Reference 1 as modified by Reference 2.

The two Framatome computer codes used in this analysis are:

1. The RODEX2-2A code was used to determine the burnup dependent initial fuel rod conditions for the system calculations.

2. The S-RELAP5 code was used to predict the thermal-hydraulic response of the primary and secondary sides of the reactor system and the hot rod response. The code version addressed all known Framatome condition reports and modeling issues at the time of analysis.

The SBLOCA methodology (Reference 1, Reference 2) has been reviewed and approved by the NRC to perform SBLOCA analyses. However, several modeling deviations from the approved SBLOCA methodology were included in this analysis, as described below.

Representative system nodalization figures for a Westinghouse three-loop plant are shown in Figure 3-2 through Figure 3-4. As such, minor variations for Harris Nuclear Plant Unit 1 specific details are not shown. For example, the charging system is not simulated in the SBLOCA analysis; therefore, the charging system noding diagram shown in Figure 3-2 is not used.

Figure 3-1

[ ]

[ ]

Figure 3-2
S-RELAP5 SBLOCA Reactor Coolant System Nodalization

Figure 3-3
S-RELAP5 SBLOCA Secondary System Nodalization

Figure 3-4
S-RELAP5 SBLOCA Reactor Vessel Nodalization

[ ]

3.3 *Plant Description and Summary of Analysis Parameters*

Harris Nuclear Plant Unit 1 is a Westinghouse-designed PWR with three loops. Each loop contains a hot leg, a U-tube SG, an RCP, and a cold leg. A pressurizer is connected to the hot leg of one of the loops. The reactor has a core power of 2958 MWt (including measurement uncertainty). The reactor vessel contains a downcomer, upper and lower plenums, and a reactor core containing 157 fuel assemblies. The hot legs connect to the reactor vessel with the vertical U-tube steam generators. Main feedwater (MFW) is injected into the downcomer of each SG. The auxiliary feedwater (AFW) system provides flow to the three SGs when normal feedwater is not available. The ECCS provides injection to each of the three loops via the centrifugal charging/HHSI system, low head safety injection (LHSI)/residual heat removal (RHR) system, and accumulators. For the purpose of this report, the centrifugal charging/HHSI system and LHSI/RHR system are referred to as the HHSI system and LHSI system, respectively.

The RCS was nodalized in the S-RELAP5 model with control volumes interconnected by flow paths or "junctions." The model includes three accumulators, a pressurizer, and three SGs with both primary and secondary sides modeled. All of the loops were modeled explicitly to provide an accurate representation of the plant. A SG tube plugging level of 3% was modeled in each SG. Important system parameters and initial conditions used in the analysis are given in Table 3-1. The heat generation rate in the S-RELAP5 reactor core model was determined from reactor kinetics equations with actinide and decay heating as prescribed by 10 CFR 50 Appendix K (Reference 5).

The break spectrum analysis assumed an LOOP concurrent with reactor scram, which is based on the reactor protection system (RPS) low pressurizer pressure reactor trip plus trip delay. The assumption of LOOP concurrent with reactor scram results in an RCP trip.

The RCPs are tripped at the time of reactor scram, instead of the opening of the break (time zero). This is considered to be conservative, since continued RCP operation will delay LSC. This delay in LSC will result in additional RCS inventory loss since the break flow is mostly liquid until the time of LSC. After LSC, a path for steam venting is established and the break flow transitions from liquid to steam, lowering the break mass flow rate.

The single failure criterion required by 10 CFR 50 Appendix K (Reference 5) was satisfied by assuming the loss of one emergency diesel generator (EDG). As a result, one motor-driven AFW pump, one HHSI pump, and one LHSI pump are assumed available to mitigate the transient. Following the safety injection actuation system (SIAS) activation on low pressurizer pressure, injection from the HHSI and LHSI systems were delayed by 29 seconds.

Table 3-2 and Table 3-3 show the minimum ECCS flow rates with one EDG failure for HHSI and LHSI, respectively. The HHSI system was modeled to deliver flow equally among the three loops. The LHSI system was modeled to deliver higher flow to the loop containing the broken leg [ ]. Although there is a charging system, it is not modeled in the analysis.

With one of the two motor-driven AFW pumps assumed unavailable for the single failure criterion, the remaining motor-driven AFW pump delivers flow to the three SGs. The input model included the main steam lines between their respective SGs and the turbine control valve, including the connected main steam safety valve (MSSV) inlet piping. The MSSVs were set to open at their nominal setpoints plus 3% tolerance.

The axial power shapes for this analysis are shown in Figure 3-5. The figure shows the input axial power shape and the axial power shape after being adjusted so that it is consistent with the Technical Specification total peaking and radial peaking factors.

3.4 *SE Compliance*

The NRC-approved supplemented EMF-2328 method (Reference 1 and Reference 2) contains no restrictions. The analysis was performed in accordance with the approved methodology except as indicated in Section 3.2.

Table 3-1
System Parameters and Initial Conditions

Parameter	Value
Reactor Power (MWt)	2958[1]
Axial Power Shape	Figure 3-5
Enthalpy Rise Factor ($F_{\Delta H}$)	1.73[1]
Total Power Peaking Factor (F_Q)	2.62[2]
K(z)	1.00 between 0 and 4 ft 0.96183 between 4 and 12 ft
RCS Flow Rate (gpm)	290,000
Pressurizer Pressure (psia)	2249.7
RCS Operating Temperature (ºF)	588.8
SG Tube Plugging (%)	3
SG Secondary Pressure (psia)	985
MFW Temperature (ºF)	440
Pressurizer Low Pressure for Reactor Trip (psig)	1923[1]
RPS Low Pressurizer Pressure Trip Delay (sec)	2.0
RPS Scram Delay (sec)	0.0
SIAS Low Pressurizer Pressure Activation Setpoint (psia)	1756.7
Accumulator Pressure (psig)	560
Accumulator Fluid Temperature (ºF)	130
Accumulator Water Volume (ft^3)	1012
AFW Temperature (ºF)	80
AFW Flow Rate (gpm)	374
AFW Initiation on Low-Low SG Narrow Range Level Setpoint (% Narrow Range Span)	0
AFW Injection Delay Time (sec)	61.5
RWST Fluid Temperature (ºF)	125
HHSI and LHSI Injection Delay Time on SIAS (sec)	29
MSSV Lift Pressure and Tolerance	Nominal + 3% Tolerance

[1] Includes measurement uncertainty.
[2] Includes measurement uncertainty. Used with K(z) to determine F_Q analysis value.

Table 3-2
High Head Safety Injection Flow Rates

RCS Pressure (psia)	Loop 1 (gpm)	Loop 2 (gpm)	Loop 3 (gpm)
0.00	162.49	162.49	162.49
15.00	162.49	162.49	162.49
398.93	148.61	148.61	148.61
646.81	138.04	138.04	138.04
829.46	130.73	130.73	130.73
1012.11	123.19	123.19	123.19
1142.57	117.27	117.27	117.27
1390.45	105.27	105.27	105.27
1638.33	92.14	92.14	92.14
1755.74	85.16	85.16	85.16
1886.20	77.03	77.03	77.03
2003.62	67.11	67.11	67.11
2134.08	54.04	54.04	54.04
2251.50	38.86	38.86	38.86
2378.23	16.97	16.97	16.97
2378.33	0.0	0.0	0.0

Table 3-3
Low Head Safety Injection Flow Rates

RCS Pressure (psia)	Intact Loops (gpm)	Broken Loop (gpm)
0.00	924.49	1813.98
25.37	924.49	1813.98
35.43	888.15	1741.90
50.07	833.24	1632.95
70.06	753.65	1475.03
90.03	655.05	1278.47
100.02	600.22	1169.09
105.21	569.63	1107.78
110.07	539.84	1048.26
115.72	503.46	973.90
131.51	356.47	657.00
142.79	108.86	137.47
144.79	36.94	44.67
144.89	0.0	0.0

Figure 3-5
Axial Power Distribution Comparison

4.0 ANALYTICAL RESULTS

The analysis results demonstrate the adequacy of the ECCS to support the criteria given in 10 CFR 50.46(b)(1-4) for Harris Nuclear Plant Unit 1 operating with Framatome supplied W17 GAIA fuel design with $M5_{Framatome}$ cladding.

Section 4.1 describes the SBLOCA break spectrum for the cold leg break. Section 4.2 describes the event for the limiting break size. Section 4.3 discusses the delayed RCP trip study. Section 4.4 discusses the attached piping break study. Section 4.5 discusses the ECCS temperature sensitivity study.

4.1 *Break Spectrum Results*

The Harris Nuclear Plant Unit 1 break spectrum analysis for SBLOCA includes breaks of varying diameter up to 10% of the flow area for the cold leg. The spectrum includes a break size range from 1.00 to 8.70 inches in diameter, which is wide enough to establish a PCT trend. Additional break sizes are analyzed with a smaller break interval once the potential limiting break size is determined to confirm the limiting break size. Figure 4-1 shows the calculated PCTs for these breaks. For the break spectrum analysis, RCP trip is assumed to occur on reactor scram.

The results of the cold leg pump discharge SBLOCA break spectrum analysis are presented in Table 4-1. The predicted event times for the break spectrum are provided in Table 4-2. The limiting PCT break size was determined to be 7.50 inches in diameter (0.30680 ft^2), resulting in a PCT of 1832°F. The 7.50 inch break size yielded the highest transient MLO and CWO from the spectrum. The limiting total MLO and limiting CWO values for the spectrum are 4.89% and 0.017%, respectively. The total MLO value of 4.89% includes [ ] .

4.2 *Discussion of Transient for Limiting PCT Break*

The limiting PCT break spectrum case is a 7.50 inch diameter cold leg break. The PCT of this case is 1832°F. The break opens at t=0 seconds and initiates a subcooled depressurization of the RCS. The low pressurizer pressure trip setpoint is reached at 3.55 seconds and at 5.55 seconds the reactor is scrammed, coincident with the RCP, MFW, and turbine trips (Figure 4-2, Figure 4-9, Figure 4-10, and Table 4-2). The pressure in the secondary side begins to rise but does not reach the MSSV setpoints, which remain closed for the duration of the transient (Figure 4-11).

The SIAS is issued at 6.75 seconds. Following the EDG delay and associated valve delays, the HHSI begins to inject at 36 seconds (Figure 4-15). However, HHSI does not provide sufficient inventory to offset the large amounts lost out the break at this time (Figure 4-18). Therefore, the core begins to uncover at 53 seconds, with effective cooling of the majority of the hot assembly lost in a short period of time (Figure 4-19).

All three loop seals clear before time of PCT, with the broken loop clearing first after 91 seconds, followed closely by the other two loops clearing at 92 and 94 seconds (Figure 4-6 and Table 4-2). The clearing of the loop seals produces a temporary increase in core level at approximately 100 seconds (Figure 4-19). However, the mixture level remains near the bottom of the active core during the increase, resulting in continued poor cooling in the upper regions of the core and allowing the clad temperature excursion to proceed (Figure 4-20).

The accumulators inject at 182 seconds (Figure 4-17 and Table 4-2). The minimum RV mass occurs around 200 seconds (Figure 4-8). There is a time delay from the accumulator injection to the mixture level reaching sufficient levels to cool the upper locations in the core. The delay results in a rupture of the hot rod after 195 seconds (Table 4-1). The rupture allows for interior metal-water reaction, thereby increasing the local oxidation at the rupture node.

The cladding temperature excursion is terminated at 207 seconds with a PCT of 1832ºF (Figure 4-20 and Table 4-1). The core is quenched at approximately 240 seconds with accumulator injection ending after approximately 300 seconds. At this point, enough decay heat is being removed and adequate mixture level is sustained by mainly HHSI flow injection (Figure 4-15). LHSI actuates two times, briefly at 207 seconds and then again at around 500 seconds, where injection is sustained for the duration of the transient. However, since sustained LHSI actuation begins well after the time of PCT, the effects of LHSI on the transient mitigation are considered minimal (Figure 4-16).

4.3 *Delayed RCP Trip Study*

The delayed RCP trip study is performed in accordance with the NRC-approved supplement to the EMF-2328 methodology (Reference 2). For plants such as Harris Nuclear Plant Unit 1 that does not have an automatic RCP trip, a delayed RCP trip can potentially result in a more limiting condition than tripping the RCPs at reactor scram. Continued operation of the RCPs can result in more overall inventory loss out the break. It has been postulated that tripping the pumps when the minimum RCS inventory occurs could cause a collapse of voids in the core, thus depressing the core level and provoking a deeper core uncovery, and a potentially higher PCT. Therefore, the methodology prescribes an RCP trip study for both the cold and hot leg breaks consistent with the plant licensing basis and Emergency Operating Procedures.

For Harris Nuclear Plant Unit 1, the condition for which all three RCPs are tripped is based on the RCS pressure with consideration of required operator action times specified in the plant Emergency Operating Procedure. A delayed RCP trip time of 5 minutes following the specified trip criteria being met was analyzed to demonstrate compliance to 10 CFR 50.46(b)(1-4) criteria (Reference 4).

The spectrum of cold and hot leg breaks in this study includes break sizes from 1.00 to 8.70 inches. The results of the delayed RCP trip cases indicate that there is at least 5 minutes for operators to trip all three RCPs after the specified trip criteria being met with considerable margin to the 10 CFR 50.46(b)(1-4) criteria.

4.4 *Attached Piping Break Study*

The ECCS must cope with ruptures of the main RCS piping and breaks in attached piping. To demonstrate this, as prescribed by the NRC-approved supplement to EMF-2328 (Reference 2), an analysis of the ruptures in attached piping that compromise the ability to inject emergency coolant into the RCS is performed. The size of the rupture and the portion of ECCS lost directly to containment are dependent on the plant design.

Harris Nuclear Plant Unit 1 has a separate line for the accumulator and the pumped SI injection connected to each cold leg. The high head and low head system connect to the cold leg through a common line. Both the accumulator and SI line breaks are analyzed, where each break location represents a double-ended guillotine break area. The accumulator line break resulted in a PCT of 1483ºF and a transient MLO of 0.15%. The SI line break resulted in a PCT of 1257ºF and transient MLO of 0.07%. The results are less limiting than those of the break spectrum analysis.

4.5 *ECCS Temperature Sensitivity Study*

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[

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Table 4-1
Summary of SBLOCA Break Spectrum Results

[ ]

3 [ ]
4 [ ]

Table 4-2
Sequence of Events for Break Spectrum (seconds)

[ ]

Table 4-2
Sequence of Events for Break Spectrum (seconds) (cont.)

Table 4-2
Sequence of Events for Break Spectrum (seconds) (cont.)

[5] [ ]

Table 4-2
Sequence of Events for Break Spectrum (seconds) (cont.)

Figure 4-1
Peak Cladding Temperature versus Break Size (SBLOCA Break Spectrum)

Figure 4-2
Reactor Power – 7.50 inch Break

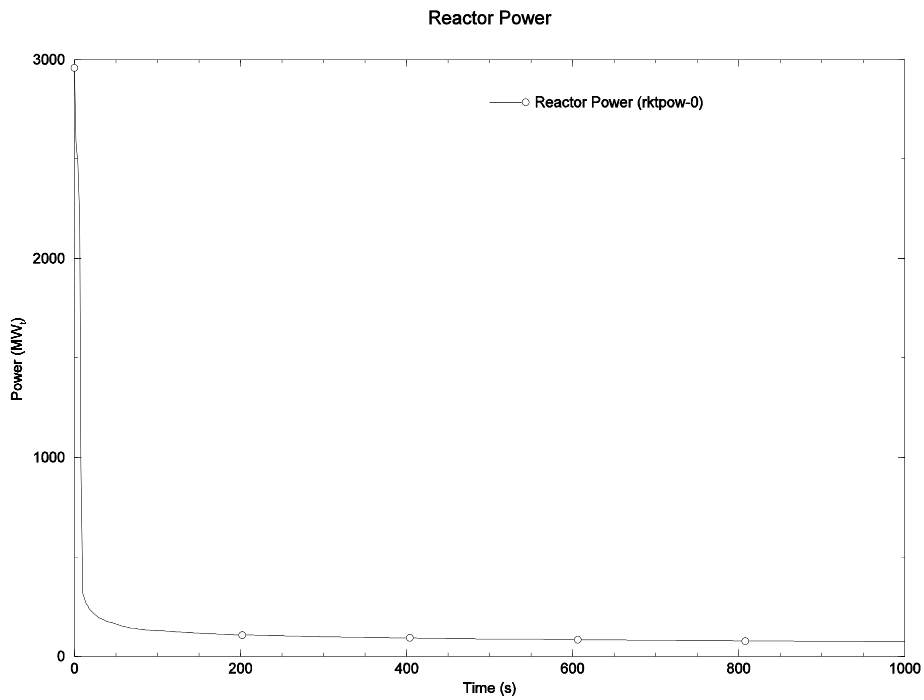


Figure 4-3
Primary and Secondary System Pressures – 7.50 inch Break

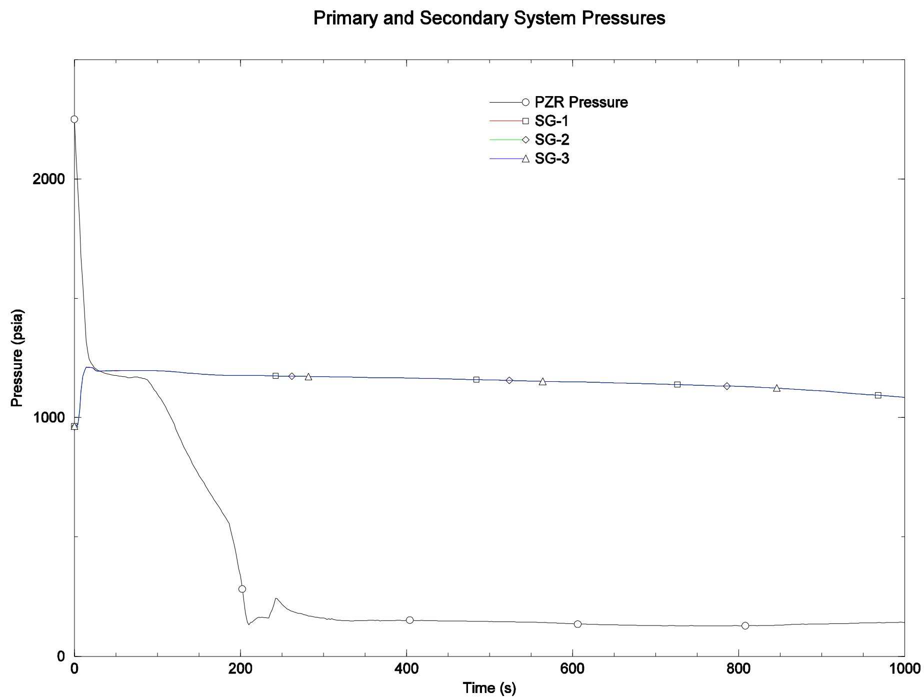


Figure 4-4
Break Mass Flow Rate – 7.50 inch Break

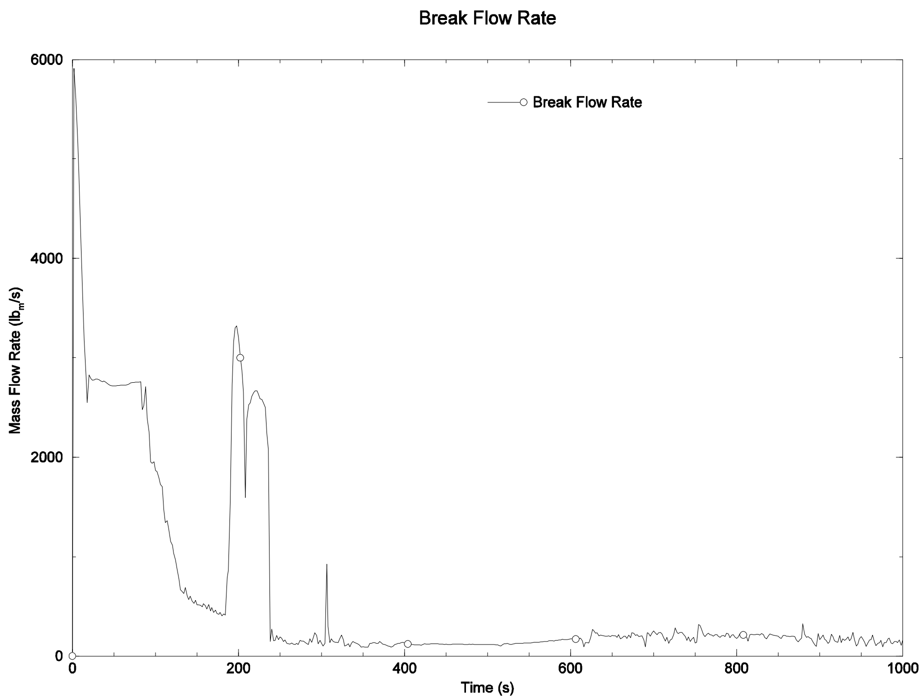


Figure 4-5
Break Vapor Void Fraction– 7.50 inch Break

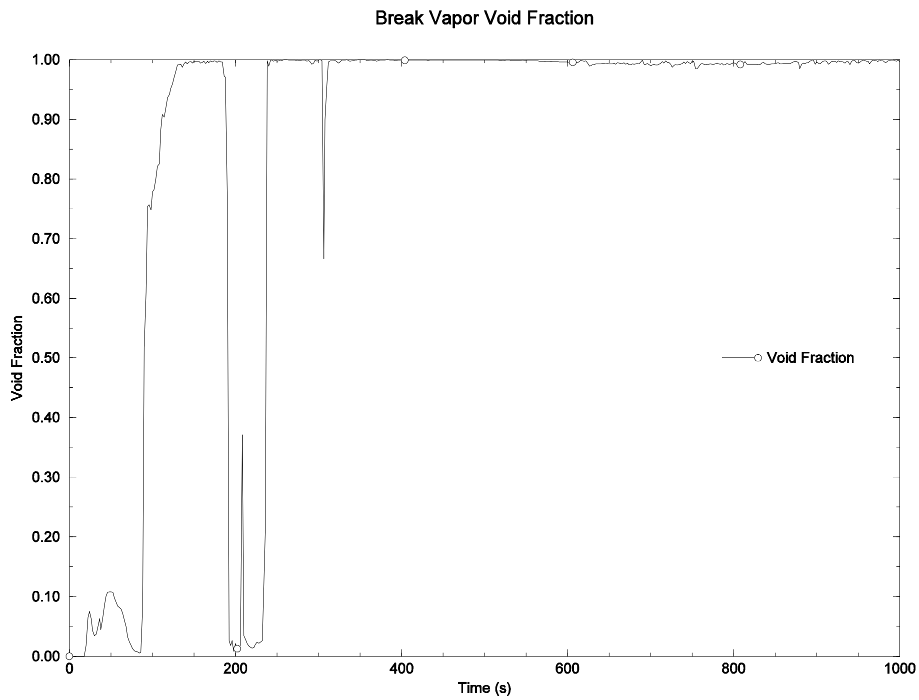


Figure 4-6
Loop Seal Upside Collapsed Levels – 7.50 inch Break

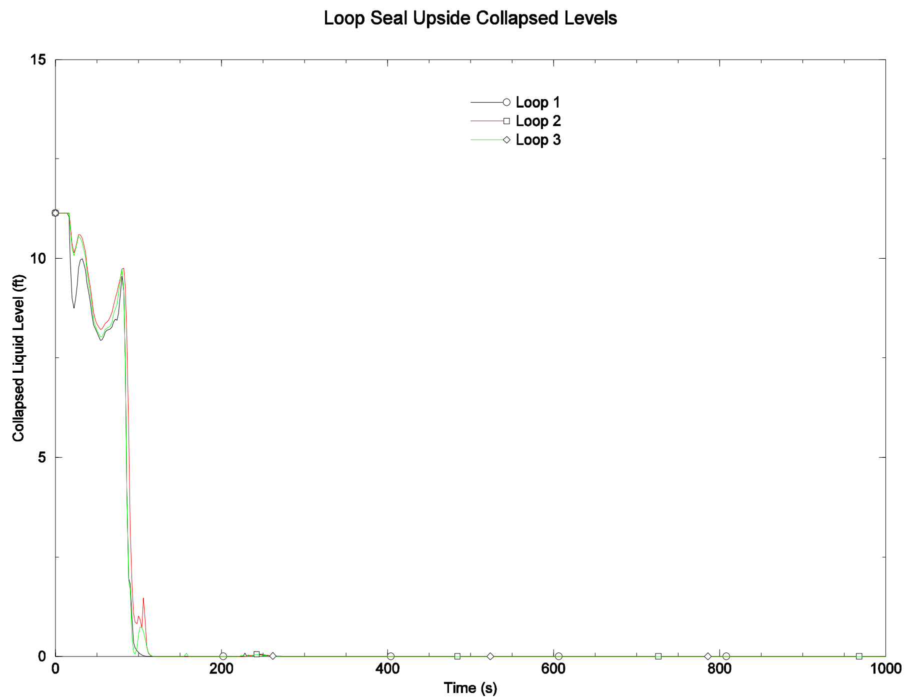


Figure 4-7
Downcomer Collapsed Liquid Level –7.50 inch Break

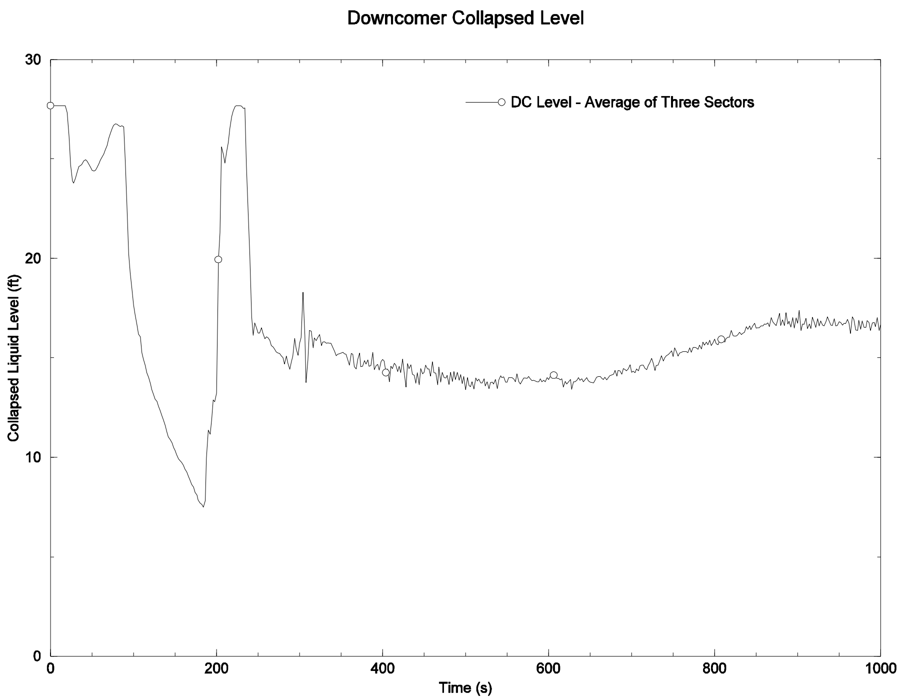


Figure 4-8
Primary System Masses – 7.50 inch Break

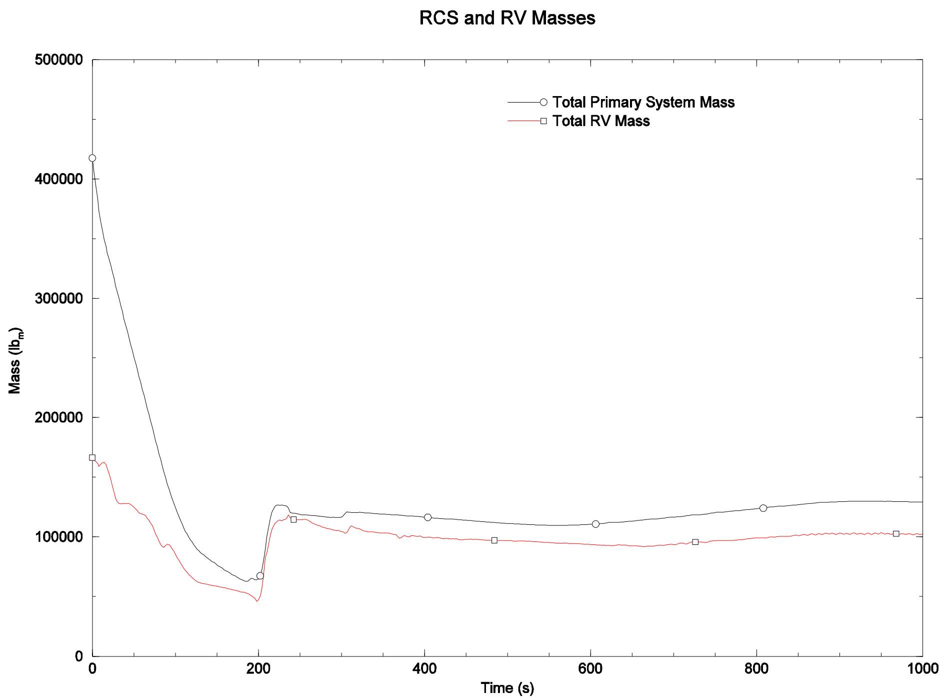


Figure 4-9
RCS Loop Mass Flow Rates – 7.50 inch Break

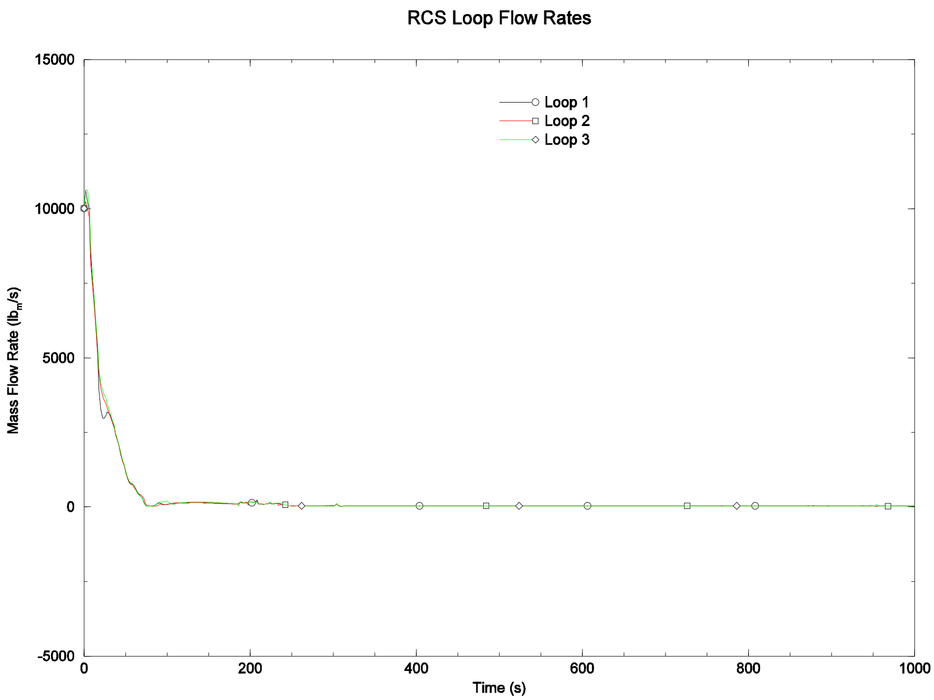


Figure 4-10
Steam Generator Main Feedwater Flow Mass Rates – 7.50 inch Break

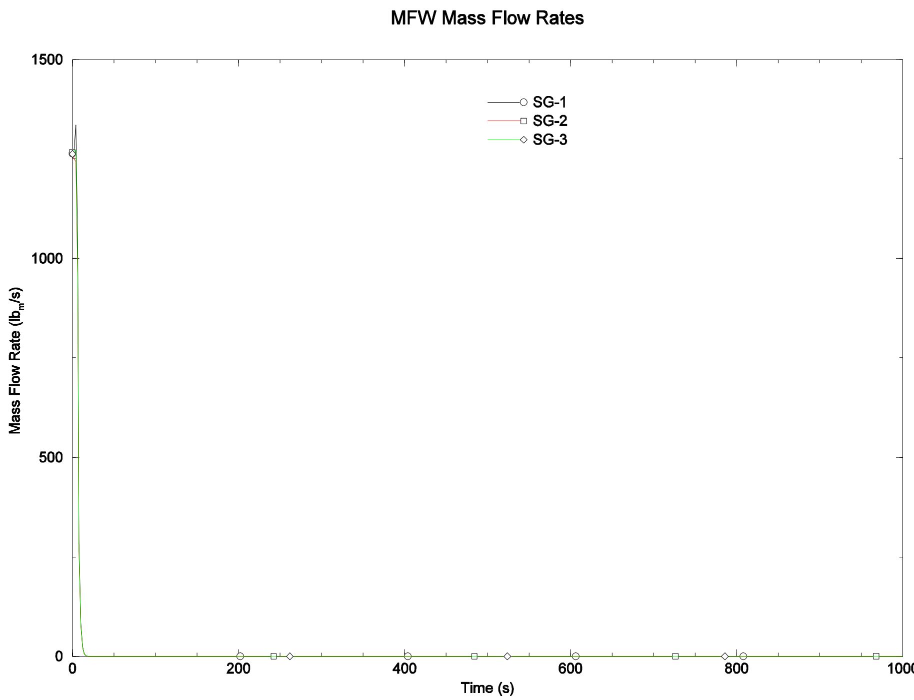


Figure 4-11
Steam Generator MSSV Mass Flow Rates – 7.50 inch Break

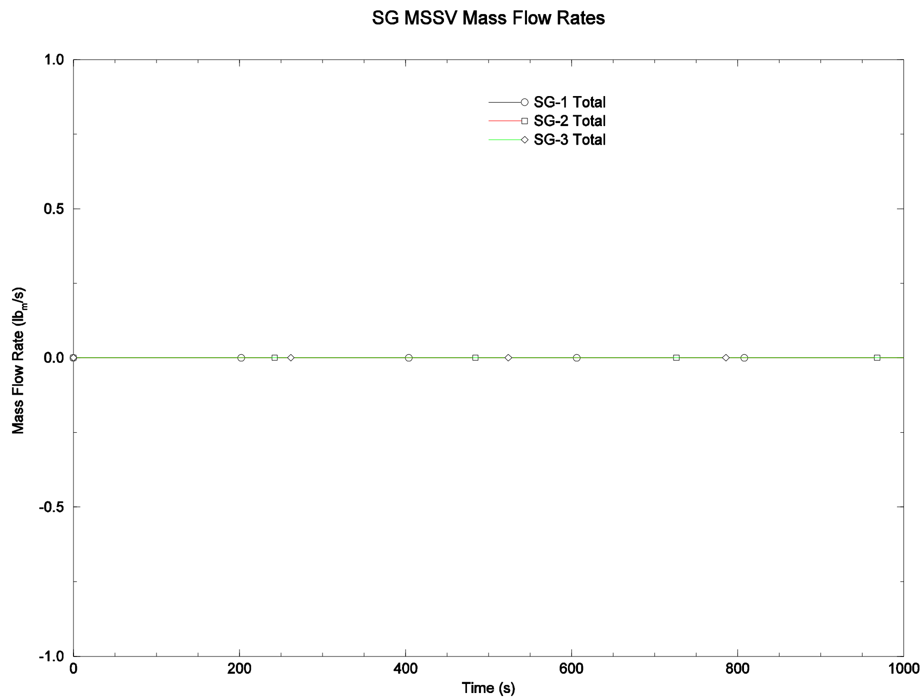


Figure 4-12
Steam Generator Auxiliary Feedwater Flow Rate – 7.50 inch Break

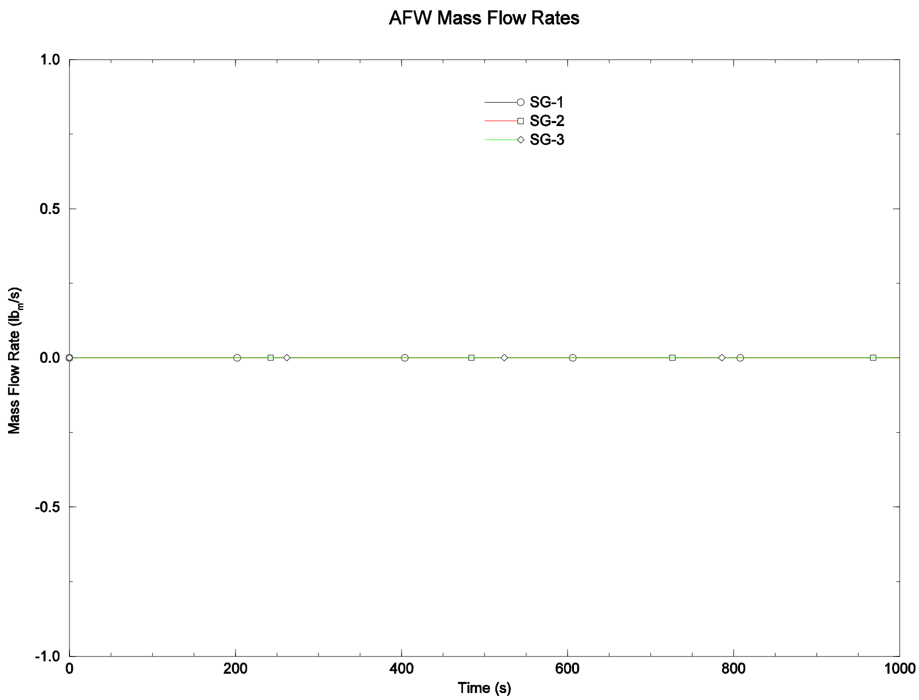


Figure 4-13
Steam Generator Total Mass – 7.50 inch Break

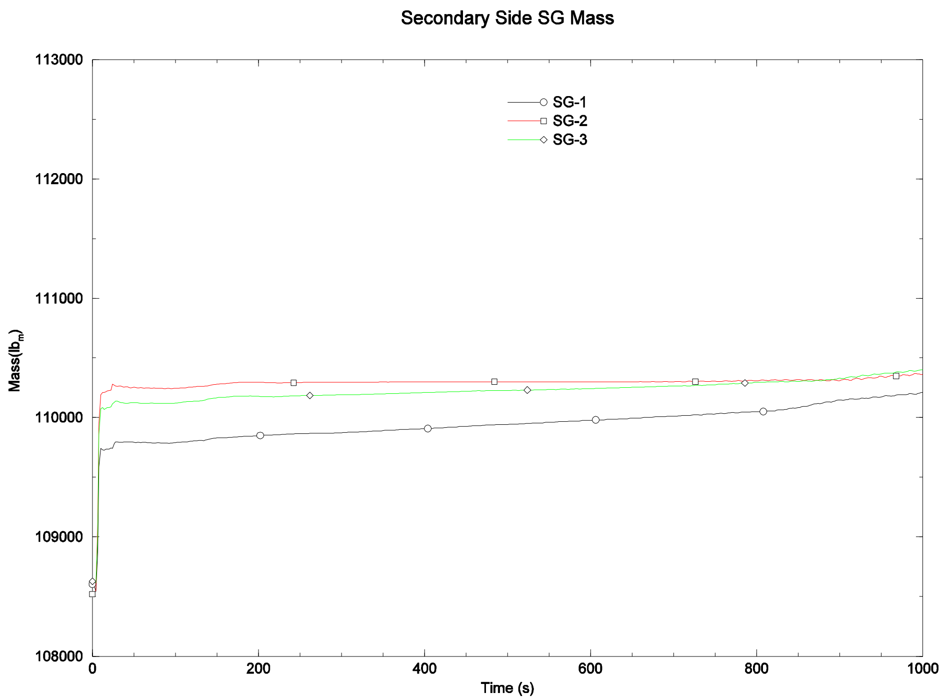


Figure 4-14
Steam Generator Narrow Range Level – 7.50 inch Break

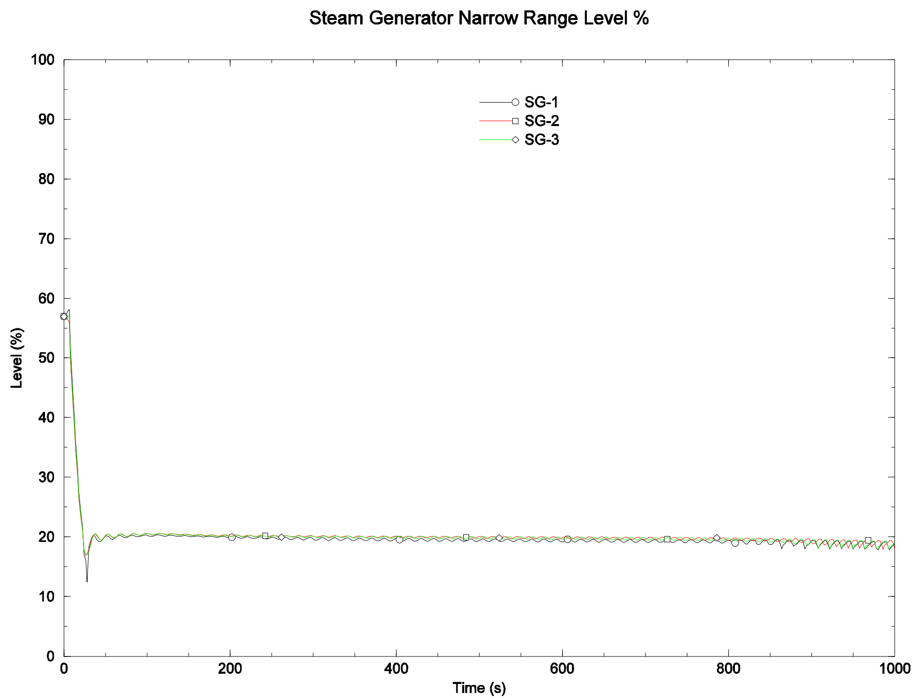


Figure 4-15
High Head Safety Injection Mass Flow Rates– 7.50 inch Break

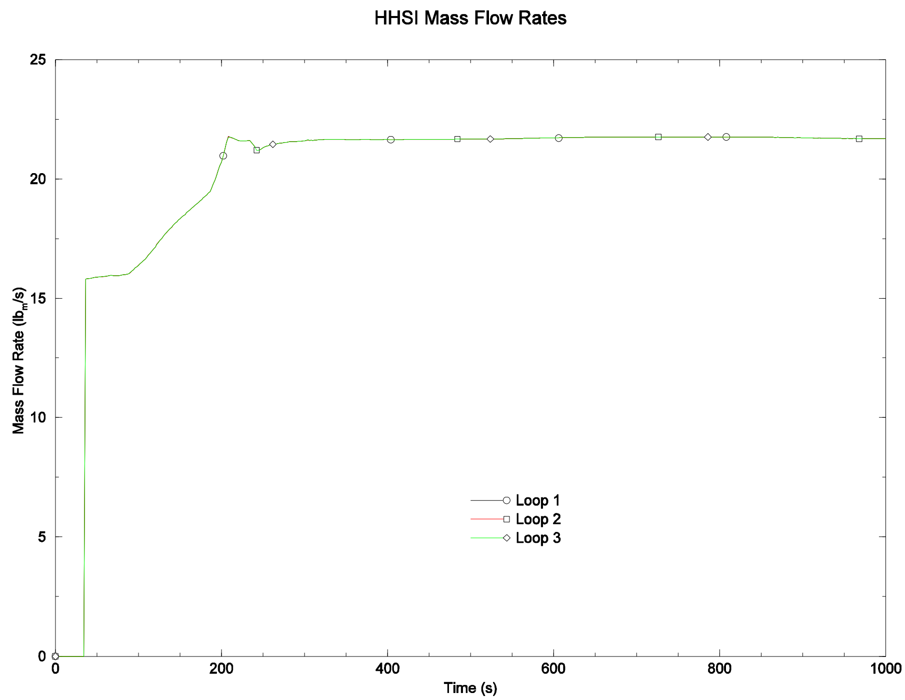


Figure 4-16
Low Head Safety Injection Mass Flow Rates– 7.50 inch Break

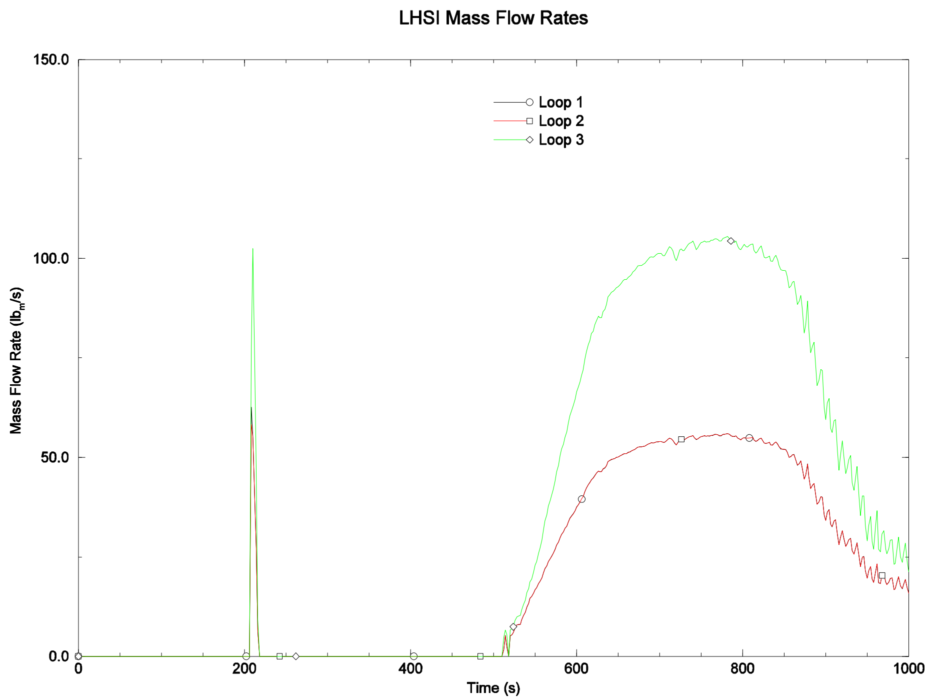


Figure 4-17
Accumulator Mass Flow Rates – 7.50 inch Break

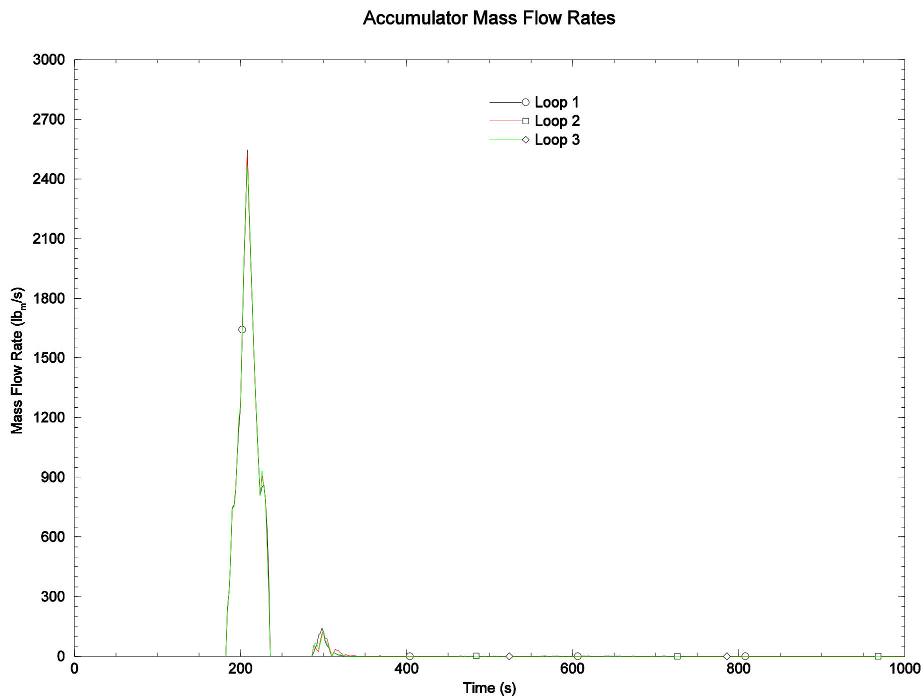


Figure 4-18
Total ECCS and Break Mass Flow Rates – 7.50 inch Break

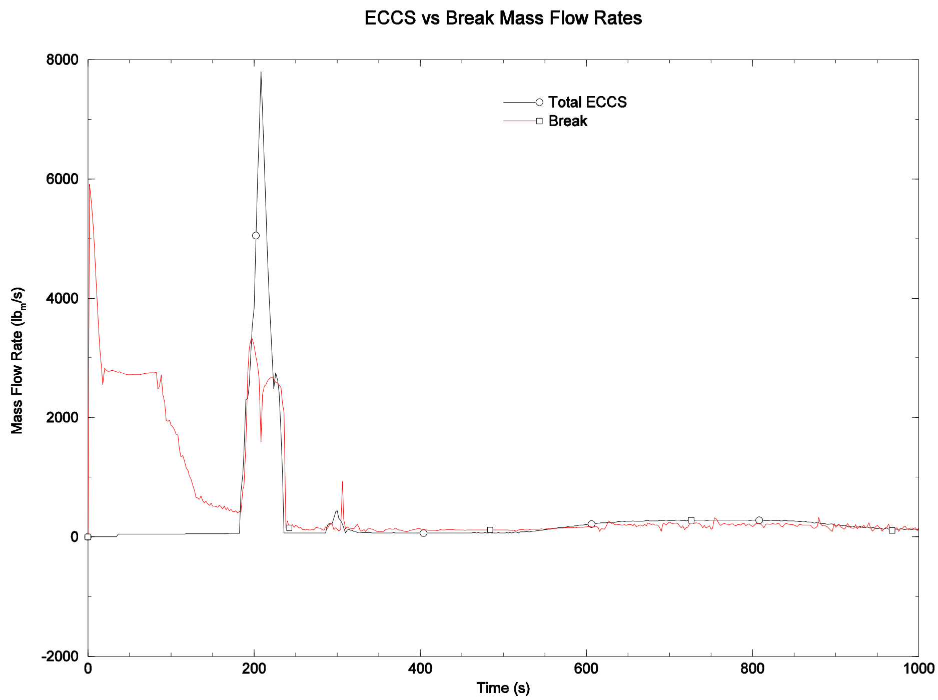


Figure 4-19
Hot Assembly Collapsed Level – 7.50 inch Break

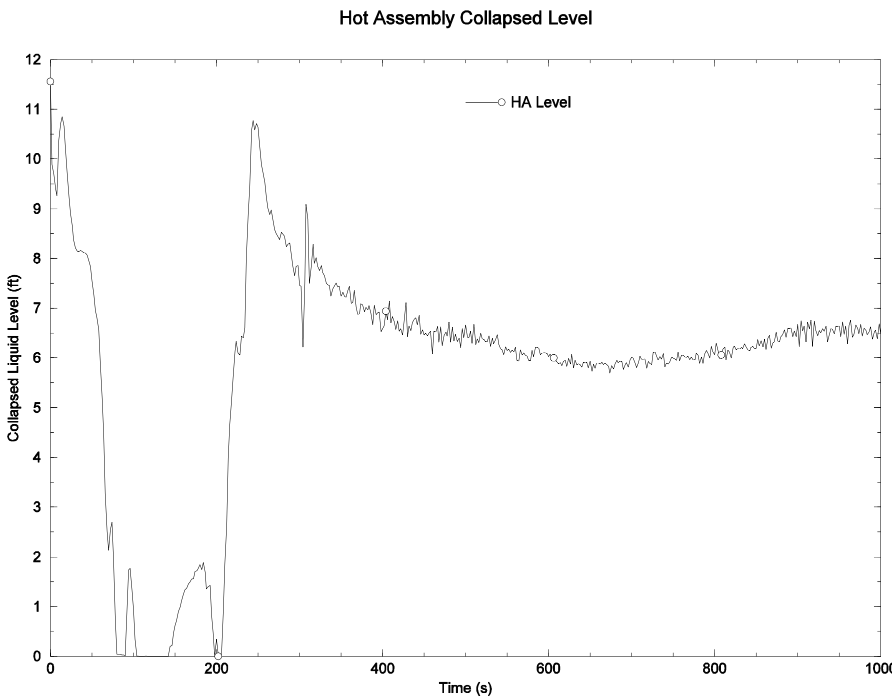
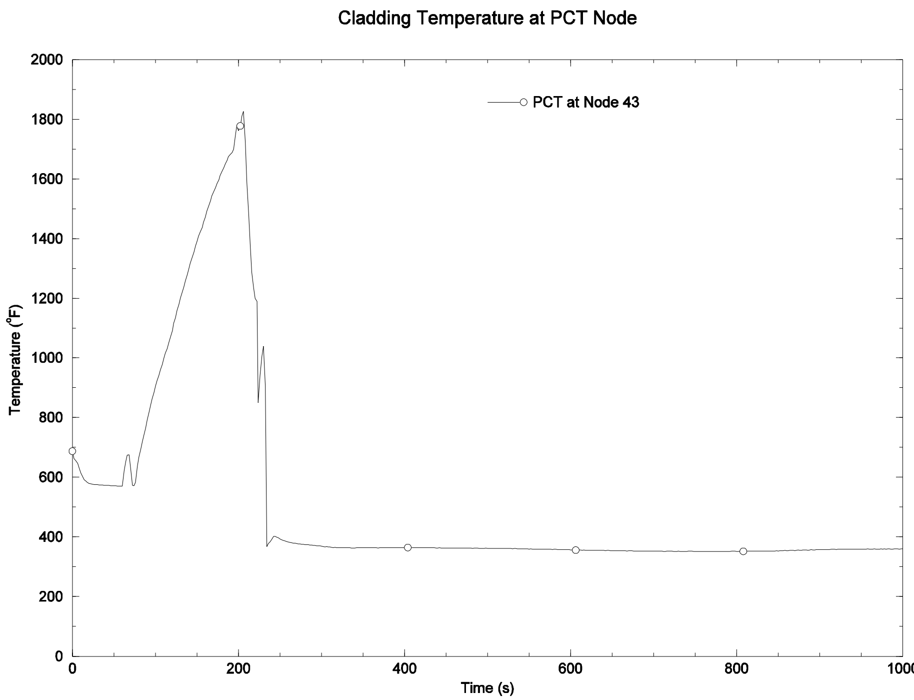


Figure 4-20
Cladding Temperature at PCT Node – 7.50 inch Break



5.0 REFERENCES

1. Framatome Inc. Topical Report EMF-2328(P)(A) Revision 0, *PWR Small Break LOCA Evaluation Model*, S-RELAP5 Based, March 2001.

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3. Framatome Inc. Topical Report BAW-10240(P)(A) Revision 0, *Incorporation of M5™ Properties in Framatome ANP Approved Methods*, May 2004.

4. Code of Federal Regulations, Title 10, Part 50, Section 46, *Acceptance Criteria For Emergency Core Cooling Systems For Light-Water Nuclear Power Reactors*, January 2010.

5. Code of Federal Regulations, Title 10, Part 50, Appendix K, *ECCS Evaluation Models*, June 2000.