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June 04, 2020

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

Serial No.: 20-176
NRA/YG: R0
Docket No.: 50-395
License No.: NPF-12

DOMINION ENERGY SOUTH CAROLINA (DESC)
VIRGIL C. SUMMER NUCLEAR STATION (VCSNS) UNIT 1
LICENSE AMENDMENT REQUEST
UPDATE OF ANALYTICAL METHOD TO THE CORE OPERATING LIMITS REPORT WITH THE FULL SPECTRUM LOSS OF COOLANT ACCIDENT APPROACH

Pursuant to 10 CFR 50.90, Dominion Energy South Carolina (DESC), acting for itself and as an agent for South Carolina Public Service Authority hereby is submitting a license amendment request (LAR) to revise the Technical Specifications (TS) for Virgil C. Summer Nuclear Station (VCSNS) Unit 1. The proposed LAR requests NRC approval to replace VCSNS TS Section 6.9.1.11, "Core Operating Limits Report," analytical methods Item (c), which currently references statistically-based best estimate large break loss of coolant accident (LBLOCA) and deterministically-based small break loss of coolant accident (SBLOCA) methods. DESC is proposing to replace the currently referenced methods with a state-of-the-art, unified, and approved full spectrum loss of coolant accident analysis (FSLOCA) approach. The proposed change fulfills a South Carolina Electric and Gas (now DESC) commitment to address fuel pellet thermal conductivity degradation (TCD) as described in the NRC Information Notice 2011-21.

A detailed description and supporting information are contained in the attachments to this letter.

DESC has evaluated (Attachment 1) the proposed LAR and determined that it does not involve a significant hazards consideration as defined in 10 CFR 50.92. In addition, the implementation of the LAR will not result in a significant increase in the amount of effluents that may be released offsite or a significant increase in individual or cumulative occupational radiation exposure. Therefore, it is concluded that the proposed LAR is eligible for categorical exclusion from an environmental assessment as set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), neither an environmental impact statement nor environmental assessment is needed in connection with the approval of the proposed LAR.

DESC requests NRC's review and approval of the proposed LAR by June 30, 2021, with a 90-day implementation period.

Should you have any questions, please contact Mr. Yan Gao at (804)-273-2768.

Respectfully,

Mark D. Sartain
Vice President – Nuclear Engineering and Fleet Support

COMMONWEALTH OF VIRGINIA)
)
COUNTY OF HENRICO)

The foregoing document was acknowledged before me, in and for the County and Commonwealth aforesaid, today by Mark D. Sartain, who is Vice President – Nuclear Engineering and Fleet Support of Dominion Energy South Carolina, Inc. He has affirmed before me that he is duly authorized to execute and file the foregoing document in behalf of that Company, and that the statements in the document are true to the best of his knowledge and belief.

Acknowledged before me this 4th day of June, 2020.

My Commission Expires: March 31, 2022

Diane E. Aitken
Notary Public

DIANE E. AITKEN
NOTARY PUBLIC
REG. #7763114
COMMONWEALTH OF VIRGINIA
MY COMMISSION EXPIRES MARCH 31, 2022

Commitments made in this letter: None.

Attachments:

1. Technical Specification Change Discussion
2. Technical Specification Page with Mark-up
3. Technical Specification Page Proposed
4. License Amendment Request Technical Evaluation

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ATTACHMENT 1

TECHNICAL SPECIFICATION CHANGE DISCUSSION

Virgil C. Summer Nuclear Station (VCSNS) Unit 1
Dominion Energy South Carolina, Inc. (DESC)

TECHNICAL SPECIFICATION CHANGE DISCUSSION

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TECHNICAL SPECIFICATION CHANGE DISCUSSION

1.0 SUMMARY DESCRIPTION

The change requests replacement of Virgil C. Summer Nuclear Station (VCSNS) Technical Specification 6.9.1.11, "Core Operating Limits Report," analytical methods Item (c) [7.1]. The replacement reflects a change in methods for analyzing the Loss of Coolant Accident (LOCA) analyses. LOCA analyses demonstrate the acceptable performance of a plant's Emergency Core Cooling System (ECCS) in accordance with Title 10 of the Code of Federal Regulations (CFR) Part 50.46. VCSNS seeks to transition from the current statistically-based Best Estimate Large Break LOCA [7.2] and deterministically-based Small Break LOCA [7.3] [7.4] methods to a state-of-the-art, unified, and approved Full Spectrum LOCA approach [7.5]. The proposed change in analysis methods also fulfills the South Carolina Electric and Gas (now Dominion Energy South Carolina) commitment [7.6] to address fuel pellet thermal conductivity degradation (TCD) as described in Nuclear Regulatory Commission Information Notice 2011-21 [7.7], by replacing the previous PAD3.4 and PAD4.0 fuel thermal performance codes with the updated and approved PAD5 code [7.8] in the LOCA analyses. Additionally, should 50.46(c) in its current form be approved, the transition to the Full Spectrum LOCA approach should help VCSNS to comply with the new regulation.

These attachments include technical and regulatory evaluations, and a proposed mark-up page to the affected Technical Specification.

2.0 DETAILED DESCRIPTION

2.1. System Design and Operation

The change is relevant to the Emergency Core Cooling System (ECCS) insofar as the methods being changed are used to demonstrate its performance to meet the acceptance criteria set forth in 10 CFR 50.46. However, the ECCS structures, systems, and components are not physically altered by the requested change. Similarly, the way the ECCS is operated is also unaltered by the change.

2.2. Current Technical Specification Requirement

Technical Specification 6.9.1.11 lists the NRC analytical methods used to determine the core operating limits that support the VCSNS Core Operating Limits Report (COLR). Analytical methods Item (c) of TS 6.9.1.11 is currently WCAP-12945 [7.2].

2.3. Reason for the Proposed Change

By letter dated June 1, 2017 [7.9], SCE&G (now Dominion Energy South Carolina) committed to the US Nuclear Regulatory Commission (NRC) to submit for review and approval a LOCA analysis that applies NRC-approved methods that include the effects of fuel pellet thermal conductivity degradation (TCD) by June 15, 2020. The Westinghouse FSLOCA evaluation model (EM) considers the effects of TCD, and submittal of this LAR fulfills the commitment.

2.4. Description of Proposed Changes

Technical Specification 6.9.1.11, analytical methods Item (c) currently lists WCAP-12945 [7.2] as a previously approved analytical method. This change proposes replacement of WCAP-12945 with Westinghouse Topical Report WCAP-16996-P-A [7.5].

3.0 TECHNICAL EVALUATION

Attachment 4 to this letter "License Amendment Request Technical Evaluation – Application of Westinghouse Full Spectrum LOCA (FSLOCA) Evaluation Model to VCSNS Unit 1," provides the technical evaluation for the application of the Westinghouse FSLOCA EM to VCSNS. This evaluation was performed in accordance with the NRC-approved FSLOCA EM in Westinghouse Topical Report WCAP-16996-P-A [7.5].

4.0 REGULATORY EVALUATION

4.1. Applicable Regulatory Requirements/Criteria

For the ECCS, it must be demonstrated that there is a high level of probability that the following criteria in 10 CFR 50.46 are met:

(b)(1) The analysis Peak Cladding Temperature (PCT) corresponds to a bounding estimate of the 95th percentile PCT at the 95 percent confidence level. Since the resulting PCT is less than 2,200°F, the analysis with the FSLOCA EM confirms that 10 CFR 50.46 acceptance criterion (b)(1), i.e., "PCT less than 2,200°F," is demonstrated.

The results are shown in Attachment 4, Table 7 for VCSNS.

(b)(2) The analysis Maximum Local Oxidation (MLO) corresponds to a bounding estimate of the 95th percentile MLO at the 95 percent confidence level. Since the resulting MLO is less than 17 percent when converting the time-at-temperature to an equivalent cladding reacted using the Baker-Just correlation and adding the pre-transient corrosion, the analysis confirms that 10 CFR 50.46 acceptance criterion (b)(2), i.e., "MLO of the cladding less than 17 percent," is demonstrated.

The results are shown in Attachment 4, Table 7 for VCSNS.

(b)(3) The analysis Core-Wide Oxidation (CWO) corresponds to a bounding estimate of the 95th percentile CWO at the 95 percent confidence level. Since the resulting CWO is less than 1 percent, the analysis confirms that 10 CFR 50.46 acceptance criterion (b)(3), i.e., "CWO less than 1 percent," is demonstrated.

The results are shown in Attachment 4, Table 7 for VCSNS.

(b)(4) 10 CFR 50.46 acceptance criterion (b)(4) requires that the calculated changes in core geometry are such that the core remains in a coolable geometry.

This criterion is met by demonstrating compliance with criteria (b)(1), (b)(2), and (b)(3), and by assuring that fuel assembly grid deformation due to combined LOCA and seismic loads is specifically addressed. Criteria (b)(1), (b)(2), and (b)(3) have been met for VCSNS as shown in Attachment 4, Table 7.

Section 32.1 of the NRC-approved FSLOCA EM [7.5] discusses that the effects of LOCA and seismic loads on the core geometry do not need to be considered unless fuel assembly grid deformation extends beyond the core periphery (i.e., deformation in a fuel assembly with no sides adjacent to the core baffle plates). Inboard grid deformation due to combined LOCA and seismic loads is not calculated to occur for VCSNS.

Note that the FSLOCA EM does not address 10 CFR 50.46 (b)(5), "Long-term cooling." Long-term cooling is dependent on the demonstration of continued delivery of cooling water to the core. The FSLOCA EM [7.5] does not alter any actions that are currently in place to maintain long-term cooling.

Based on the analysis results for Region I and Region II presented in Attachment 4, Table 7, it is concluded that VCSNS would continue to comply with the criteria in 10 CFR 50.46 with Westinghouse Topical Report WCAP-16996-P-A [7.5] on the list of approved methodologies for determining core operating limits.

4.2. Precedents

The proposed change to Technical Specification 6.9.1.11 replaces WCAP-12945-P-A [7.2] with Westinghouse Topical Report WCAP-16996-P-A [7.5] on the list of approved methodologies for determining core operating limits at VCSNS. Several previous similar requests have been made to include the FSLOCA methodology for Diablo Canyon [7.10], North Anna [7.11], Surry [7.12], and Watts Bar [7.13]. The Diablo Canyon request was recently approved [7.14].

4.3. No Significant Hazards Consideration

The proposed change to Technical Specification (TS) 6.9.1.11 replaces Westinghouse Topical Report WCAP-12945-P-A [7.2] with WCAP-16996-P-A [7.5] on the list of approved methodologies for use in determining core operating limits.

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

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

   Response: No.

   The proposed change to TS 6.9.1.11 permits the use of an NRC-approved methodology for analysis of the LOCA to determine if VCSNS meets the applicable design and safety analysis acceptance criteria. The proposed change to the list of NRC-approved methodologies in TS 6.9.1.11 has no direct impact upon plant operation or configuration. VCSNS abides by the limitations and conditions of the approved method and application of the method demonstrates compliance with 10 CFR 50.46(b)(1-4) ECCS performance acceptance criteria. The proposed method change does not alter the probability of accident initiation or the mitigation of its consequences.

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

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

   Response: No.

   The proposed change replaces a referenced analysis method for a previously evaluated accident. The change does not involve any credible new failure mechanisms, malfunctions, or accident initiators not previously considered. The proposed change does not result in any physical changes to the plant and does not result in any changes to the way the plant is being operated.

   Therefore, the proposed change does not create the possibility of a new or different kind of accident or malfunction from those previously evaluated within the UFSAR.

3. Does the proposed change involve a significant reduction in a margin of safety?

   Response: No.

   No design basis safety limits are exceeded or altered by this change. Approved methods will be used to ensure that the plant continues to meet applicable design criteria and safety analysis acceptance criteria.

   Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above information, DESC concludes that the proposed change does not involve a 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.

5.0 ENVIRONMENTAL CONSIDERATIONS

DESC has reviewed the proposed license amendment for environmental considerations in accordance with 10 CFR 51.22. The proposed license amendment does 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 from an environmental assessment as set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6.0 CONCLUSION

The request for replacement of analytical methods Item (c) in VCSNS Technical Specification Section 6.9.1.11 "Core Operating Limits Report," [7.1] reflects a change in method for analyzing the LOCA scenarios. Using the approved LOCA analysis method described in WCAP-16996-P-A [7.5], and showing compliance with its applicable Limitations and Conditions, VCSNS has demonstrated the acceptable performance of the plant ECCS in accordance with the applicable requirements of 10 CFR 50.46. Additionally, the proposed change in analysis method fulfills DESC's commitment to address fuel pellet TCD as described in Nuclear Regulatory Commission Information Notice 2011-21 [7.7].

Consistent with precedent set by other utilities, VCSNS shows that adoption of the new method does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c). Accordingly, a finding of "no significant hazards consideration" is justified. Similarly, DESC has determined the proposed amendment meets the eligibility criterion for categorical exclusion from an environmental assessment, and thus pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

Mark-up and proposed Technical Specification pages reflecting the method replacement are included in Attachment 2 and 3, respectively.

7.0 REFERENCES

7.1 V. C. Summer Technical Specifications, through Amendment 217.

7.2 WCAP-12945-P-A, Volume 1 (Revision 2) through Volumes 2 through 5 (Revision 1), Code Qualification Document for Best Estimate LOCA Analysis. March 1998. (PROPRIETARY)

7.3 WCAP-10054-P-A, Westinghouse Small Break LOCA ECCS Evaluation Model Using the NOTRUMP Code. August 1985. (PROPRIETARY)

7.4 WCAP-10079-P-A, NOTRUMP, A Nodal Transient Small Break and General Network Code. August 1985. (PROPRIETARY)

7.5 WCAP-16996-P-A, Revision 1, Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (FULL SPECTRUM LOCA Methodology), November 2016. (PROPRIETARY)

7.6 SCE&G Letter RC-12-0104, Virgil C. Summer Nuclear Station (VCSNS), Unit 1, Docket No. 50-395, Operating License No. NPF-12, ECCS Evaluation Model Revisions 30-Day Report. October 16, 2012 (ADAMS Accession Number ML12293A071).

7.7 NRC Information Notice 2011-21, Realistic Emergency Core Cooling System Evaluation Model Effects Resulting from Nuclear Fuel Thermal Conductivity Degradation. December 13, 2011. (ADAMS Accession Number ML113430785).

7.8 WCAP-17462-P-A Revision 1, Westinghouse Performance Analysis and Design Model (PAD5). November 2017. (PROPRIETARY)

7.9 SCE&G Letter RC-17-0067, Virgil C. Summer Nuclear Station (VCSNS), Unit 1, Docket No. 50-395, Operating License No. NPF-12, Regulatory Commitment Change Related to Best Estimate Large Break Loss of Coolant Accident Analysis that Explicitly Accounts for Thermal Conductivity Degradation. June 1, 2017. (ADAMS Accession Number ML17153A219)

7.10 Letter from Paula Gerfen (Pacific Gas and Electric Company) to USNRC (DCL No. 18-100), License Amendment Request 18-02, License Amendment Request to Revise Technical Specification 5.6.5b, "Core Operating Limits Report (COLR) for Full Spectrum Loss-of-Coolant Accident Methodology." December 26, 2018 (ADAMS Accession Number ML19003A196).

7.11 Letter from Mark Sartain (Dominion Energy Virginia) to USNRC, Virginia Electric and Power Company, North Anna Power Station Units 1 and 2, Proposed License Amendment Request, Addition of Analytical Methodology to the Core Operating Limits Report for a Full Spectrum Loss of Coolant Accident (FSLOCA). October 30, 2019. (ADAMS Accession Number ML19309D197).

7.12 Letter from Mark Sartain (Dominion Energy Virginia) to USNRC, Virginia Electric and Power Company, Surry Power Station Units 1 and 2, Proposed License Amendment Request, Addition of Analytical Methodology to the Core Operating Limits Report for a Large Break Loss of Coolant Accident (LBLOCA). October 30, 2019. (ADAMS Accession Number ML19309D196).

7.13 Letter from James T. Polickoski (Tennessee Valley Authority) to USNRC, Watts Bar Nuclear Plant, Units 1 and 2, Facility Operating License Nos. NFP-90 and NFP-96, NRC Docket Nos. 50-390 and 50-391, Application to Implement FULL SPECTRUM LOCA (FSLOCA) Methodology for Loss of Coolant Accident (LOCA) Analysis and New LOCA-specific Tritium Producing Burnable Absorber Rod Stress Analysis Methodology 9WBN-TS-19-04). January 17, 2020. (ADAMS Accession Number ML20017A338).

7.14 Letter from USNRC to James Welsch (Pacific Gas and Electric Company), Diablo Canyon Nuclear Plant Units 1 and 2 – Issuance of Amendment Nos. 234 and 236 to Revise Technical Specification 5.6.5b, "Core Operating Limits Report (COLR)," for Full Spectrum Loss of Coolant Accident Methodology (EPID L-2018-LLA-0730), January 9, 2020. (ADAMS Accession Number ML19316A109).

ATTACHMENT 2

TECHNICAL SPECIFICATION PAGE WITH MARK-UP

Virgil C. Summer Nuclear Station (VCSNS) Unit 1
Dominion Energy South Carolina, Inc. (DESC)

CORE OPERATING LIMITS REPORT (Continued)

c. WCAP-12945-P-A, Volume 1 (Revision 2) through Volumes 2 through 5 (Revision 1) "Code Qualification Document for Best Estimate LOCA Analysis," March 1998 (Westinghouse Proprietary).

Liparulo, N. (W) to NRC Document Control Desk, NSD-NRC-96-4746, "Re-Analysis Work Plans Using Final Best Estimate Methodology" dated 6/13/1996.

(Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor.)

d. WCAP-12472-P-A, "BEACON CORE MONITORING AND OPERATIONS SUPPORT SYSTEM," August 1994, (W Proprietary).

WCAP-12472-P-A, Addendum 1-A, "BEACON CORE MONITORING AND

REPLACEMENT:

WCAP-16996-P-A Revision 1, "Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (FULL SPECTRUM LOCA Methodology)," November 2016, (Westinghouse Proprietary)

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e. WCAP-13749-P-A, "Safety Evaluation Supporting the Conditional Exemption of the Most Negative EOL Moderator Temperature Coefficient Measurement," March 1997, (Westinghouse Proprietary).

(Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient.)

f. WCAP-12610-P-A, "VANTAGE + Fuel Assembly Reference Core Report," April 1995 (W Proprietary). WCAP-12610-P-A & CENPD-404-P-A, Addendum 1-A, "Optimized ZIRLO™," July 2006 (W Proprietary).

(Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor.)

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.

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

ATTACHMENT 3

TECHNICAL SPECIFICATION PAGE PROPOSED

Virgil C. Summer Nuclear Station (VCSNS) Unit 1
Dominion Energy South Carolina, Inc. (DESC)

CORE OPERATING LIMITS REPORT (Continued)

c. WCAP-16996-P-A Revision 1, "Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (FULL SPECTRUM LOCA Methodology)," November 2016, (Westinghouse Proprietary).

d. WCAP-12472-P-A, "BEACON CORE MONITORING AND OPERATIONS SUPPORT SYSTEM," August 1994, (W Proprietary).

WCAP-12472-P-A, Addendum 1-A, "BEACON CORE MONITORING AND OPERATIONS SUPPORT SYSTEM," January 2000, (W Proprietary)

(Methodology for Specifications 3.2.2 - Heat Flux Hot Channel Factor, 3.2.3- RCS Flow Rate and Nuclear Enthalpy Rise Hot Channel Factor, and 3.2.4 - Quadrant Power Tilt Ratio.)

e. WCAP-13749-P-A, "Safety Evaluation Supporting the Conditional Exemption of the Most Negative EOL Moderator Temperature Coefficient Measurement," March 1997, (Westinghouse Proprietary).

(Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient.)

f. WCAP-12610-P-A, "VANTAGE + Fuel Assembly Reference Core Report," April 1995 (W Proprietary). WCAP-12610-P-A & CENPD-404-P-A, Addendum 1-A, "Optimized ZIRLO™," July 2006 (W Proprietary).

(Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor.)

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.

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

ATTACHMENT 4

LICENSE AMENDMENT REQUEST TECHNICAL EVALUATION

Virgil C. Summer Nuclear Station (VCSNS) Unit 1
Dominion Energy South Carolina, Inc. (DESC)

LICENSE AMENDMENT REQUEST TECHNICAL EVALUATION

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LICENSE AMENDMENT REQUEST TECHNICAL EVALUATION

Application of Westinghouse Full Spectrum LOCA (FSLOCA) Evaluation Model to VCSNS Unit 1

1.0 INTRODUCTION

An analysis with the FULL SPECTRUM™ loss-of-coolant accident (FSLOCA™) evaluation model (EM) has been completed for VCSNS. This LAR requests approval to apply the Westinghouse FSLOCA EM.

The FSLOCA EM [1] was developed to address the full spectrum of loss-of-coolant accidents (LOCAs) which result from a postulated break in the reactor coolant system (RCS) of a pressurized water reactor (PWR). The break sizes considered in the Westinghouse FSLOCA EM include any break size in which break flow is beyond the capacity of the normal charging pumps, up to and including a double ended guillotine (DEG) rupture of an RCS cold leg with a break flow area equal to two times the pipe area, including what traditionally are defined as Small and Large Break LOCAs.

The break size spectrum is divided into two regions. Region I encompasses breaks that are typically defined as Small Break LOCAs (SBLOCAs). Region II includes break sizes that are typically defined as Large Break LOCAs (LBLOCAs).

The FSLOCA EM explicitly considers the effects of fuel pellet thermal conductivity degradation (TCD) and other burnup-related effects by calibrating to fuel rod performance data input generated by the PAD5 code [2], which explicitly models TCD and is benchmarked to high burnup data. The fuel pellet thermal conductivity model in the WCOBRA/TRAC-TF2 code [1] used for system and fuel response predictions explicitly accounts for fuel pellet thermal conductivity degradation.

Three of the Title 10 of the Code of Federal Regulations (CFR) 50.46 criteria (peak cladding temperature (PCT), maximum local oxidation (MLO), and core-wide oxidation (CWO)) are considered directly in the FSLOCA EM. A high probability statement is developed for the PCT, MLO, and CWO to demonstrate compliance with 10 CFR 50.46 acceptance criteria (b)(1), (b)(2), and (b)(3) via statistical methods. The MLO is defined as the sum of pre-transient corrosion and transient oxidation consistent with the position in Information Notice 98-29 [3]. Compliance with demonstrating the maintenance of coolable geometry, and how long-term core cooling is addressed, are discussed in Section 4.1.

The FSLOCA EM has been generically approved by the Nuclear Regulatory Commission (NRC) for Westinghouse 3-loop and 4-loop plants with cold leg Emergency Core Cooling System (ECCS) injection [1]. Since VCSNS is a Westinghouse designed 3-loop plant with cold leg ECCS injection, the approved method is applicable, with the exceptions identified under Limitation and Condition Number 2 in Section 2.3.

Both VCSNS and its analysis vendor (Westinghouse) have interface processes which identify plant configuration changes potentially impacting safety analyses. These interface processes, along with Westinghouse internal processes for assessing EM changes and errors, are used to identify the need for LOCA analysis impact assessments.

The following methods detail the elements of the VCSNS FSLOCA analyses. The major plant parameter and analysis assumptions are provided in Tables 1 through 6.

2.0 METHOD OF ANALYSIS

2.1. Full Spectrum LOCA Evaluation Model Development

In 1988, the NRC Staff amended the requirements of 10 CFR 50.46 and 10 CFR 50 Appendix K, "ECCS Evaluation Models," to permit the use of a realistic EM to analyze the performance of the ECCS during a hypothetical LOCA. Under the amended rules, best-estimate thermal-hydraulic models may be used in place of models with Appendix K features. After the rule change, Westinghouse developed and received approval for a best-estimate LBLOCA EM [4]. The EM is referred to as the Code Qualification Document (CQD) and was developed following Regulatory Guide (RG) 1.157 [5].

When the FSLOCA EM was being developed, the NRC issued RG 1.203 [6] which expands on the principles of RG 1.157, while providing a more systematic approach to the development and assessment process of a PWR accident and safety analysis EM. Therefore, the development of the FSLOCA EM followed the Evaluation Model Development and Assessment Process (EMDAP), which is documented in RG 1.203. While RG 1.203 expands upon RG 1.157, there are certain aspects of RG 1.157 which are more detailed than RG 1.203; therefore, both RGs were used for the development of the FSLOCA EM.

2.2. WCOBRA/TRAC-TF2 Computer Code

The FSLOCA EM [1] uses the WCOBRA/TRAC-TF2 code to analyze the system thermal hydraulic response for the full spectrum of break sizes. WCOBRA/TRAC-TF2 was created by combining a 1D module (TRAC-P) with a 3D module (based on Westinghouse modified COBRA-TF). The 1D and 3D modules include an explicit non-condensable gas transport equation. The use of TRAC-P allows for the extension of a two-fluid, six-equation formulation of the two-phase flow to the 1D loop components. This new code is WCOBRA/TRAC-TF2, where "TF2" is an identifier that reflects the use of a three-field (TF) formulation of the 3D module derived by COBRA-TF and a two-fluid (TF) formulation of the 1D module based on TRAC-P.

This best-estimate computer code contains the following features:

1. Ability to model transient three-dimensional flows in different geometries inside the reactor vessel
2. Ability to model thermal and mechanical non-equilibrium between phases
3. Ability to mechanistically represent interfacial heat, mass, and momentum transfer in different flow regimes
4. Ability to represent important reactor components such as fuel rods, steam generators (SGs), reactor coolant pumps (RCPs), etc.

A detailed assessment of the computer code WCOBRA/TRAC-TF2 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 for a LOCA. Modeling of a LOCA introduces additional uncertainties which are identified and quantified in the plant-specific analysis. The reactor vessel and loop noding scheme used in the FSLOCA EM is consistent with the noding scheme used for the experiment simulations that form the validation basis for the physical models in the code. Such noding choices have been justified by assessing the model against large- and full-scale experiments.

2.3. Compliance with FSLOCA EM Limitations and Conditions

The NRC's SER for the FSLOCA EM [1] contains 15 Limitations and Conditions. A summary of each and how it is met is provided below.

Limitation and Condition Number 1

Summary

The FSLOCA EM is not approved to demonstrate compliance with 10 CFR 50.46 acceptance criterion (b)(5) related to long-term cooling.

Compliance

The analysis for VCSNS with the FSLOCA EM is only being used to demonstrate compliance with 10 CFR 50.46 (b)(1) through (b)(4).

Limitation and Condition Number 2

Summary

The FSLOCA EM is approved for the analysis of Westinghouse-designed 3-loop and 4-loop PWRs with cold-side injection. Analyses should be executed consistent with the approved method, or any deviations from the approved method should be described and justified.

Compliance

VCSNS is a Westinghouse-designed 3-loop PWR with cold-side injection, so it is within the NRC approved methodology. The analysis utilized the NRC-approved FSLOCA methodology, as supplemented by corrections described below, pursuant to 10 CFR 50.46 error reporting requirements.

1. After completion of the analysis for VCSNS, two errors were discovered in the WCOBRA/TRAC-TF2 code that can occur under certain conditions. These errors were found to have a negligible impact on analysis results with the FSLOCA EM as described in LTR-NRC-19-6 [7].

2. The power increase in the hot rod and hot assembly due to energy redistribution was calculated incorrectly. The treatment for the uncertainty in the gamma energy redistribution is discussed on pages 29-75 and 29-76 of [1], and the equation for the assumed increase in hot rod and assembly relative power is presented on page 29-76. The error resulted in a 0% to 5% deficiency in the modeled hot rod and hot assembly rod-linear heat rates on a run-specific basis, depending on the as-sampled value for the multiplier uncertainty. The effect of the error correction was evaluated against the application of the FSLOCA EM to VCSNS.

   The error correction has only a limited impact on the power modeled for a single assembly in the core. As such, the error correction has a negligible impact on the system thermal-hydraulic response during the postulated LOCA.

   For the Region I analysis, the primary impact of the error correction is on the rate of cladding heatup above the two-phase mixture level in the core during the boiloff phase. The PCT impact was assessed using run-specific PCT versus linear heat rate relationships and the run-specific hot rod and hot assembly linear heat rate increase that would result from the error correction. Using this approach, the correction of the error was estimated to increase the Region I analysis PCT by 12°F, leading to a result of 1,108°F for the Region I analysis.

   For the Region II analysis, parametric PWR sensitivity studies, derived from a subset of uncertainty analysis simulations covering various design features and fuel arrays, were examined to determine the sensitivity of the analysis results to the error correction. The PCT impact from the error correction was found to be different for the transient phases (i.e., blowdown versus reflood) based on the PWR sensitivity studies. The correction of the error is estimated to increase the Region II analysis PCT by 31°F, leading to an analysis result of 1,879°F for the Region II analysis.

The analysis results, including the error correction, continue to demonstrate compliance with the 10 CFR 50.46 acceptance criteria.

Limitation and Condition Number 3

Summary

For Region II, the containment pressure calculation will be executed in a manner consistent with the approved methodology (i.e., the COCO or LOTIC2 model will be based on appropriate plant-specific design parameters and conditions, and engineered safety features which can reduce pressure are modeled). This includes utilizing a plant-specific initial containment temperature, and only taking credit for containment coatings which are qualified and outside of the break zone-of-influence.

Compliance

The containment pressure calculation was performed consistent with the NRC-approved methodology. Appropriate design parameters and conditions were modeled, as were the engineered safety features which can reduce containment pressure. A minimum initial temperature associated with normal full-power operating conditions was modeled, and no coatings were credited on any of the containment structures.

Limitation and Condition Number 4

Summary

The decay heat uncertainty multiplier will be sampled consistent with the NRC-approved methodology for the FSLOCA EM. The analysis simulations for the FSLOCA EM will not be executed for longer than 10,000 seconds following reactor trip unless the decay heat model is appropriately justified. The sampled values of the decay heat uncertainty multiplier for the cases which produced the Region I and Region II analysis results will be provided in the analysis submittal in units of sigma and absolute units.

Compliance

Consistent with the NRC-approved methodology, the decay heat uncertainty multiplier was consistent with the NRC-approved methodology for the FSLOCA EM. The analysis simulations were all executed for no longer than 10,000 seconds following reactor trip. The sampled values of the decay heat uncertainty multiplier for the cases which produced the Region I and Region II analysis results have been provided in units of sigma and approximate absolute units in Table 10.

Limitation and Condition Number 5

Summary

The maximum assembly and rod length-average burnup must remain below the limits contained in the NRC-approved methodology for the FSLOCA EM.

Compliance

The maximum analyzed assembly and rod length-average burnup were less than or equal to the limits contained in the NRC-approved methodology for the FSLOCA EM.

Limitation and Condition Number 6

Summary

The fuel performance data for analyses with the FSLOCA EM should be based on the PAD5 code (at present), which includes the effect of thermal conductivity degradation. The nominal fuel pellet average temperatures and rod internal pressures should be the maximum values, and the generation of all the PAD5 fuel performance data should adhere to the NRC-approved PAD5 methodology.

Compliance

PAD5 fuel performance data were utilized in the analysis with the FSLOCA EM. The analyzed fuel pellet average temperatures bound the maximum values calculated in accordance with Section 7.5.1 of [2], and the analyzed rod internal pressures were calculated in accordance with Section 7.5.2 of [2].

Limitation and Condition Number 7

Summary

The YDRAG uncertainty parameter must be set to the required value for the Region I analysis, including the determination of the limiting break size.

Compliance

Consistent with the NRC-approved methodology, the YDRAG uncertainty parameter was set to the required value for all Region I cases in the determination of the limiting break size as well as the uncertainty analysis.

Limitation and Condition Number 8

Summary

The KCOSI uncertainty parameter should be set to the required value, and the HS_SLUG uncertainty parameter should be set to the required value for the Region I analysis, including the determination of the limiting break size.

Compliance

Consistent with the NRC-approved methodology, the KCOSI and HS_SLUG uncertainty parameters were set to their required values for all cases in the determination of the limiting break size, as well as the uncertainty analysis.

Limitation and Condition Number 9

Summary

For PWR designs which are not Westinghouse 3-loop PWRs, a sensitivity study will be executed to confirm that certain biases Region I breaks produce conservative results for the plant design being analyzed. This sensitivity study should be executed once, and then referenced in all applications to that particular plant class.

Compliance

VCSNS is a Westinghouse-designed 3-loop PWR, so this Limitation and Condition is not applicable.

Limitation and Condition Number 10

Summary

For PWR designs which are not Westinghouse 3-loop PWRs, a sensitivity study will be executed to demonstrate that the applied break size boundary for Region I analyses serves the intended goal.

Additionally, the minimum sampled break area for the analysis of Region II should be 1 ft^2.

Compliance

VCSNS is a Westinghouse-designed 3-loop PWR, so this part of the Limitation and Condition is not applicable.

The minimum sampled break area for the Region II analysis was 1 ft^2.

Limitation and Condition Number 11

Summary

There are various aspects of this Limitation and Condition, which are summarized below:

1. Certain information regarding the Region I and Region II analyses must be declared and documented prior to performing the uncertainty analysis and will not be changed throughout the remainder of the analysis once they have been declared and documented.

2. If the analysis inputs are changed after they have been declared and documented, for the intended purpose of demonstrating compliance with the applicable acceptance criteria, then the changes and associated rationale for the changes will be provided in the analysis submittal. Additionally, the preliminary values for PCT, MLO, and CWO which caused the input changes will be provided. These preliminary values are not subject to Appendix B verification, and archival of the supporting information for these preliminary values is not required.
3. Plant operating ranges which are sampled within the uncertainty analysis will be provided in the analysis submittal for both regions.

Compliance

This Limitation and Condition was met as follows:

1. The information specified in the NRC-approved methodology for the FSLOCA EM was declared and documented prior to analysis execution and was not changed after it was declared and documented.
2. The analysis inputs were not changed once they were declared and documented.
3. The plant operating ranges which were sampled within the uncertainty analyses are provided in Table 1.

<u>Limitation and Condition Number 12</u>

Summary

The plant-specific dynamic pressure loss from the steam generator secondary-side to the main steam safety valves must be adequately accounted for in analysis with the FSLOCA EM.

Compliance

A bounding plant-specific dynamic pressure loss from the steam generator secondary-side to the main steam safety valves (MSSVs) was modeled.

<u>Limitation and Condition Number 13</u>

Summary

In plant-specific models for analysis with the FSLOCA EM, specific modeling considerations for the upper head spray nozzles should be followed as required by the NRC-approved methodology.

Compliance

The specific modeling requirements for the upper head spray nozzles were met.

Limitation and Condition Number 14

Summary

For analyses with the FSLOCA EM to demonstrate compliance against the current 10 CFR 50.46 oxidation criterion, the transient time-at-temperature will be converted to an equivalent cladding reacted (ECR) using either the Baker-Just or the Cathcart-Pawel correlation. In either case, the pre-transient corrosion will be summed with the LOCA transient oxidation. If the Cathcart-Pawel correlation is used to calculate the LOCA transient ECR, then the result shall be compared to a 13 percent limit. If the Baker-Just correlation is used to calculate the LOCA transient ECR, then the result shall be compared to a 17 percent limit.

Compliance

For the VCSNS analysis, the Baker-Just correlation was used to convert the LOCA transient time-at-temperature to an ECR. The resulting LOCA transient ECR was then summed with the pre-existing corrosion for comparison against the 10 CFR 50.46 local oxidation acceptance criterion of 17%.

Limitation and Condition Number 15

Summary

The Region II analysis will be executed twice; once assuming loss-of-offsite power (LOOP) and once assuming offsite power available (OPA). The results from both analysis executions should be shown to comply with the 10 CFR 50.46 acceptance criteria.

The minimum sample size for the Region II analysis meets the requirements of the FSLOCA EM.

Compliance

The Region II uncertainty analysis was performed twice; once assuming a LOOP and once assuming OPA. The results from both analyses meet the 10 CFR 50.46 acceptance criteria (see Section 4.2).

The sample size used in the Region II analyses exceeded the minimum required sample size.

3.0 REGION I ANALYSIS

3.1. Description of Representative Transient

The small break LOCA transient can be divided into time periods in which specific phenomena are occurring, as discussed below.

Blowdown

The rapid depressurization of the RCS coincides with subcooled liquid flow through the break. Following the reactor trip on the low pressurizer pressure setpoint, the pressurizer drains, and safety injection is initiated on the low pressurizer pressure SI setpoint. After reaching this setpoint and applying the safety injection delays, high pressure safety injection flow begins. Phase separation begins in the upper head and upper plenum near the end of this period until the entire RCS eventually reaches saturation, ending the rapid depressurization slightly above the steam generator secondary side pressure near the modeled MSSV setpoint.

Natural Circulation

This quasi-equilibrium phase persists while the RCS pressure remains slightly above the secondary side pressure. The system drains from the top down, and while significant mass is continually lost through the break, the vapor generated in the core is trapped in the higher elevations of the RCS because of the seal formed by liquid in the crossover leg (cold leg pump suction) loop seals. Throughout this period, the core remains covered by a two-phase mixture and the fuel cladding temperatures remain at the saturation temperature.

Loop Seal Clearance

As the system drains through the break, the liquid levels in the downhill side of the pump suction (crossover leg) decrease to the lower elevations of the piping, allowing the vapor trapped during the natural circulation phase to also vent to the break. This is the loop seal clearing process. The break flow and the flow through the RCS loops become primarily vapor. Relief of a static head imbalance allows for a quick but temporary recovery of liquid levels in the core region.

Boil-Off

With a vapor vent path established after the loop seal clearance, the RCS depressurizes at a rate controlled by the critical flow, which continues to be a primarily high-quality mixture of water and steam. The RCS pressure may remain high enough such that safety injection flow cannot make up for the primary system fluid inventory lost through the break in time, potentially leading to core uncovery and a fuel rod cladding temperature heat-up.

Core Recovery

The RCS pressure continues to decrease, and once it reaches that of the accumulator gas pressure, the introduction of additional ECCS water from the accumulators replenishes the reactor vessel inventory and recovers the core mixture level. The accident analysis is terminated when the break flow is matched, then exceeded, by the injected flow.

3.2. Analysis Results

The VCSNS Region I analysis was performed in accordance with the NRC-approved methodology in Reference 1, with exceptions identified under Limitation and Condition Number 2 in Section 2.3. The transient that produced the analysis PCT result is a cold leg break with a break diameter of 2.6-inches. The most limiting ECCS single failure of one ECCS train is assumed in the analysis as identified in Table 1. Control rod drop is modeled for breaks less than 1 square foot assuming a 2.0-second signal delay time and a 4.0-second rod drop time. RCP trip is modeled coincident with reactor trip on the low pressurizer pressure setpoint for LOOP transients. When the low pressurizer pressure SI setpoint is reached, there is a delay to account for emergency diesel generator start-up, filling headers, etc., after which safety injection is initiated into the reactor coolant system.

The results of the Region I uncertainty analysis are summarized in Table 7. The sampled decay heat uncertainty multipliers for the Region I analysis cases are provided in Table 10.

Table 8 contains a sequence of events for the transient that produced the Region I analysis PCT result. Figures 1 through 13 illustrate the calculated key transient response parameters for this transient.

4.0 REGION II ANALYSIS

4.1. Description of Representative Transient

A large-break LOCA transient can be divided into phases in which specific phenomena are occurring. A convenient way to divide the transient is in terms of the various heat-up and cooldown phases that the fuel assemblies undergo. For each of these phases, specific phenomena and heat transfer regimes are important, as discussed below.

Blowdown – Critical Heat Flux (CHF) Phase

In this phase, the break flow is subcooled, the discharge rate of coolant from the break is high, the core flow reverses, the fuel rods go through departure from nucleate boiling (DNB), and the cladding rapidly heats up and the reactor is shut down due to core voiding.

The regions of the RCS with the highest initial temperatures (upper core, upper plenum, and hot legs) begin to flash during this period. This phase is terminated when the water in the lower plenum and downcomer begins to flash. The mixture level swells and a saturated mixture is pushed into the core by the intact loop RCPs, still rotating in single-phase liquid. As the fluid in the broken cold leg reaches saturation conditions, the discharge flow rate at the break decreases significantly.

Blowdown – Upward Core Flow Phase

Heat transfer is increased as the two-phase mixture is pushed into the core. The break discharge rate is reduced because the fluid becomes saturated at the break. This phase ends as the lower plenum mass is depleted, the fluid in the loops become two-phase, and the RCP head degrades with increasing voiding.

Blowdown – Downward Core Flow Phase

The break flow begins to dominate and pulls flow down through the core as RCP head decreases, while liquid and entrained liquid flows also provide core cooling. Heat transfer in this period may be enhanced by liquid flow from the upper head. Once the system has depressurized to less than the accumulator cover pressure, the accumulators begin to inject cold water into the cold legs.

During this period, due to steam upflow in the downcomer, a portion of the injected ECCS water is lost to the break or bypasses the core around the downcomer and exits via the break due to the phenomenon of emergency core cooling bypass. After the initial surge of accumulator inventory is lost out of the break, core bypass diminishes, and the remaining accumulator liquid refills the lower portion of the reactor vessel. As the system pressure continues to decrease, the break flow and consequently the downward core flow are reduced. The system pressure approaches the containment pressure at the end of this last period of the blowdown phase.

During this phase, the core begins to heat up as the system equilibrates with containment pressure, and the phase ends when the reactor vessel begins to refill with ECCS water.

Refill Phase

The core continues to heat up as the lower plenum refills with ECCS water. This phase is characterized by a rapid increase in fuel cladding temperature at all elevations due to the lack of liquid and steam flow in the core region. The water completely refills the lower plenum and the refill phase ends. As ECCS water enters the core, the fuel rods in the lower core region begin to quench and liquid entrainment begins, resulting in increased fuel rod heat transfer.

Reflood Phase

During the early reflood phase, the accumulators begin to empty and their nitrogen fill gas discharges into the RCS. The nitrogen surge forces water into the core, causing system re-pressurization and a temporary reduction of pumped ECCS flow. During this time, core cooling may increase due to vapor generation and liquid entrainment, but conversely the early reflood pressure spike results in loss of mass out through the broken cold leg.

The pumped ECCS water aids in the filling of the downcomer throughout the reflood period. As the quench front progresses further into the core, the PCT elevation moves increasingly higher in the fuel assembly.

As the transient progresses, continued injection of pumped ECCS water refloods the core, effectively removes the reactor vessel metal mass stored energy and core decay heat and leads to an increase in the reactor vessel fluid mass. Eventually the core inventory increases enough that liquid entrainment can quench all the fuel assemblies in the core.

4.2. Analysis Results

The Region II analysis was performed in accordance with the NRC-approved methodology [1], with exceptions identified under Limitation and Condition Number 2 in Section 2.3. The analysis was performed assuming both LOOP and OPA, and the results of both the LOOP and OPA analyses are compared to the 10 CFR 50.46 acceptance criteria. The most limiting ECCS single failure of one ECCS train is assumed in the analysis as identified in Table 1. The results of the VCSNS Region II LOOP and OPA uncertainty analyses are summarized in Table 7. The sampled decay heat uncertainty multipliers for the Region II analysis cases are provided in Table 10.

Table 9 contains a sequence of events for the transient that produced the more limiting analysis PCT result relative to the offsite power assumption. Figures 14 through 27 illustrate the key response parameters for this transient.

The containment pressure is calculated for each LOCA transient in the analysis using the COCO code [8][9]. The COCO containment code is integrated into the WCOBRA/TRAC-TF2 thermal-hydraulic code. The transient-specific mass and energy releases calculated by the thermal hydraulic code at the end of each timestep are transferred to COCO. COCO then calculates the containment pressure based on the containment model (the inputs are summarized in Tables 2 and 3) and the mass and energy releases, and transfers the pressure back to the thermal-hydraulic code as a boundary condition at the break, consistent with the methodology [1]. The containment model for COCO calculates a conservatively low containment pressure, including the effects of all the installed pressure-reducing systems and processes such as assuming all trains of containment spray are operable and assuming fan cooler operation. The containment backpressure for the transient that produced the analysis PCT result is provided in Figure 21.

5.0 REFERENCES

1. WCAP-16996-P-A Revision 1, Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (FULL SPECTRUM LOCA Methodology). November 2016. (PROPRIETARY)
2. WCAP-17462-P-A Revision 1, Westinghouse Performance Analysis and Design Model (PAD5). November 2017. (PROPRIETARY)
3. NRC Information Notice 98-29, Predicted Increase in Fuel Rod Cladding Oxidation." August 1998.
4. WCAP-12945-P-A, Volume 1 (Revision 2) through Volumes 2 through 5 (Revision 1), Code Qualification Document for Best Estimate LOCA Analysis. March 1998. (PROPRIETARY)
5. NRC Regulatory Guide 1.157, Best Estimate Calculations of Emergency Core Cooling System Performance. May 1989.
6. NRC Regulatory Guide 1.203, Transient and Accident Analysis Methods. December 2005.
7. Westinghouse Letter, LTR-NRC-19-6, U.S. Nuclear Regulatory Commission 10 CFR 50.46 Annual Notification and Reporting for2018." February 2019. (ADAMS Accession Number ML19042A379)
8. WCAP-8339, Westinghouse Emergency Core Cooling System Evaluation Model – Summary. June 1974. (PROPRIETARY)
9. WCAP-8327, Containment Pressure Analysis Code (COCO). June 1974. (PROPRIETARY)

6.0 TABLES AND FIGURES

Table 1. Plant Operating Range Analyzed and Key Parameters for VCSNS

	Parameter	As-Analyzed Value or Range
1.0	**Core Parameters**	
	a) Core power	≤ 2900 MWt ± 2% Uncertainty
	b) Fuel type	17x17 Vantage+ fuel, **Optimized ZIRLO™** cladding material, Intermediate Flow Mixers (IFMs), Integral Fuel Burnable Absorbers (IFBA) or Non-IFBA
	c) Maximum total core peaking factor (F_Q), including uncertainties	2.50
	d) Maximum hot channel enthalpy rise peaking factor ($F_{\Delta H}$), including uncertainties	1.70
	e) Axial flux difference (AFD) band at 100% power	-12% / +10%
2.0	**Reactor Coolant System Parameters**	
	a) Thermal design flow (TDF)	92,600 gpm/loop
	b) Vessel average temperature (T_{AVG})	572.0 – 5.4°F ≤ T_{AVG} ≤ 587.4 + 6.3°F
	c) Pressurizer pressure (P_{RCS})	2250 – 65 psia ≤ P_{RCS} ≤ 2250 + 54 psia
	d) Reactor coolant pump (RCP) model and power	Model 93A, 7000 hp
3.0	**Containment Parameters**	
	a) Containment modeling	Region I: Constant pressure equal to initial containment pressure Region II: Calculated for each transient using transient-specific mass and energy releases and the information in Tables 2 and 3

Table 1. Plant Operating Range Analyzed and Key Parameters for VCSNS

Parameter		As-Analyzed Value or Range
4.0	**Steam Generator (SG) and Secondary Side Parameters**	
	a) Steam generator tube plugging level	≤ 10%
	b) Main steam safety valve (MSSV) nominal set pressures, uncertainty and accumulation	Table 6
	c) Main feedwater temperature	Nominal (440°F)
	d) Auxiliary feedwater temperature (T_{AFW})	32°F ≤ T_{AFW} ≤ 95°F
	e) Auxiliary feedwater flow rate	133.3 gpm/SG
5.0	**Safety Injection (SI) Parameters**	
	a) Single failure configuration	ECCS: Loss of one train of pumped ECCS Region II containment pressure: All containment spray trains are available
	b) Safety injection temperature (T_{SI})	40°F ≤ T_{SI} ≤ 95°F
	c) Low pressurizer pressure safety injection safety analysis limit	1715 psia
	d) Initiation delay time from low pressurizer pressure SI setpoint to full SI flow	≤ 27 seconds (OPA) or ≤ 37 seconds (LOOP)
	e) Safety injection flow	Minimum flows in Table 4 (Region I) or Table 5 (Region II)
6.0	**Accumulator Parameters**	
	a) Accumulator temperature (T_{ACC})	75°F ≤ T_{ACC} ≤ 120°F
	b) Accumulator water volume (V_{ACC})	994 ft^3 ≤ V_{ACC} ≤ 1034 ft^3
	c) Accumulator pressure (P_{ACC})	585 psia ≤ P_{ACC} ≤ 701 psia
	d) Accumulator boron concentration	≥ 2200 ppm
7.0	**Reactor Protection System Parameters**	
	a) Low pressurizer pressure reactor trip signal processing time	≤ 2 seconds
	b) Low pressurizer pressure reactor trip setpoint	1835 psia

Table 2. Containment Data Used for Region II Calculation of Containment Pressure for VCSNS

Parameter	Value
Maximum containment net free volume	1.91×10^6 ft^3
Minimum initial containment temperature at full power operation	75°F
Refueling water storage tank (RWST) temperature for containment spray (T_{RWST})	40°F ≤ T_{RWST} ≤ 95°F
Minimum RWST temperature for broken loop spilling SI	40°F
Minimum containment outside air / ground temperature	-5°F
Minimum initial containment pressure at normal full power operation	14.6 psia
Minimum containment spray pump initiation delay from containment high pressure signal time	≥ 32 seconds (OPA) or ≥ 38 seconds (LOOP)
Maximum containment spray flow rate from all pumps	6600 gpm
Maximum number of containment fan coolers in operation during LOCA transient	2
Minimum fan cooler initiation delay time	≥ 33 seconds (OPA) or ≥ 40 seconds (LOOP)
Maximum heat removal rate per fan cooler as a function of containment temperature	Table 3
Maximum number of containment venting lines (including purge lines, pressure relief lines or any others) which can be OPEN at onset of transient at full power operation	1
Maximum effective valve diameter of each containment venting line	6.065 inches
Maximum containment pressure setpoint for venting valve closure	3.6 psig
Maximum delay time between reaching containment pressure setpoint and start of venting valve closure	1.6 seconds
Maximum venting valve closure time at normal full power operation	5.4 seconds
SI spilling flows	256.5 lbm/sec

Table 3. Fan Cooler Performance Data Used for Region II Calculation of Containment Pressure for VCSNS

Containment Temperature (°F)	Heat Removal Rate (BTU/hr)	Heat Removal Rate (BTU/sec)
148.5	44,592,834	12,387
170.1	68,015,125	18,893
181.7	82,961,400	23,045
190.8	95,496,000	26,527
200.7	109,945,231	30,540
211.0	125,463,080	34,851
220.2	140,202,609	38,945
227.1	151,025,483	41,952
234.2	162,115,201	45,032
254.0	192,871,878	53,576

Table 4. Safety Injection Flow Used for Region I Calculation for VCSNS

Pressure (psia)	High Head Safety Injection (HHSI) Flow (gpm)
14.7	327.1
114.7	335.8
214.7	331.8
314.7	323.9
414.7	315.9
514.7	307.8
614.7	299.5
714.7	291.2
814.7	282.7
914.7	274.2
1014.7	265.5
1114.7	256.6
1214.7	247.5
1314.7	237.9
1414.7	228.1
1514.7	218.1
1614.7	208.1
1714.7	197.8
1814.7	185.0
1914.7	171.8
2014.7	158.0
2114.7	143.7
2214.7	126.3
2314.7	102.9
2414.7	77.4
2414.71	0.0
2514.7	0.0

Table 5. Safety Injection Flow Used for Region II Calculation for VCSNS

Pressure (psia)	High Head Safety Injection (HHSI) Flow (gpm)	Low Head Safety Injection (LHSI) Flow (gpm)
14.7	327.1	2414.3
34.7	328.4	1993.3
54.7	329.7	1537.0
74.7	331.1	1024.4
94.7	332.4	412.4
94.71	332.4	0.0
114.7	333.7	0.0
214.7	325.5	
314.7	313.6	
414.7	301.4	
514.7	288.9	
614.7	276.2	
714.7	263.2	
814.7	249.8	
914.7	236.1	
1014.7	222.0	
1114.7	207.4	
1214.7	192.3	
1314.7	176.9	
1414.7	160.7	
1514.7	143.6	
1614.7	125.3	
1714.7	106.0	
1814.7	85.3	
1914.7	63.3	
2014.7	37.5	
2114.7	2.6	
2114.71	0.0	
2514.7	0.0	

Table 6. Steam Generator Main Steam Safety Valve Parameters for VCSNS

Stage	Set Pressure (psig)	Uncertainty (%)	Accumulation (%)
1	1176	1	3
2	1190	3	3
3	1205	3	3
4	1220	3	3
5	1235	3	3

Table 7. VCSNS Analysis Results with the FSLOCA EM

Outcome	Region I Value	Region II Value (LOOP)	Region II Value (OPA)
95/95 PCT[1]	1,096 + 12 = 1,108 °F	1,848 + 31 = 1879 °F	1,837 + 31 = 1868 °F
95/95 MLO	8.43%	9.13%	9.06%
95/95 CWO	0.00%	0.36%	0.33%

NOTE:

1. The PCT values presented in the table show the analysis-of-record result, which is the sum of the uncertainty analysis result plus the impact of the energy redistribution error correction. The figures presenting the analysis results correspond to the uncertainty analysis results. The MLO and CWO were confirmed to demonstrate compliance with the 10 CFR 50.46 acceptance criteria with the error correction.

Table 8. VCSNS Sequence of Events for the Region I Analysis PCT Case

Event	Time after Break (sec)
Start of Transient	0.0
Reactor Trip Signal	14.9
Safety Injection Signal	26.8
Safety Injection Begins	63.8
Loop Seal Clearing Occurs	680
Top of Core Uncovered	1,208
Accumulator Injection Begins	1,894
PCT Occurs	1,921
Top of Core Recovered	2,976

Table 9. VCSNS Sequence of Events for the Region II Analysis PCT Case

Event	Time after Break (sec)
Start of Transient	0.0
Fuel Rod Burst Occurs	4.3
Safety Injection Signal	4.6
Accumulator Injection Begins	11.1
End of Blowdown	15.5
Safety Injection Begins	41.6
Accumulator Empty	45.0
PCT Occurs	95.4
All Rods Quenched	266

Table 10. VCSNS Sampled Value of Decay Heat Uncertainty Multiplier, DECAY_HT, for the Region I and Region II Analysis Cases

Region	Case	DECAY_HT (units of σ)	DECAY_HT (absolute units)[1]
Region I	PCT	+1.5102σ	7.38%
	MLO	+0.5534σ	2.82%
	CWO	N/A[2]	N/A[2]
Region II (LOOP)	PCT	+0.3023σ	1.45%
	MLO	+0.0946σ	0.48%
	CWO	+0.6064σ	2.88%
Region II (OPA)	PCT	+0.5520σ	2.62%
	MLO	+0.4110σ	2.09%
	CWO	+0.6064σ	2.88%

Notes

1. Approximate uncertainty in total decay heat power at 1 second after shutdown as defined by the ANSI/ANS-5.1-1979 decay heat standard for ^{235}U, ^{239}Pu, and ^{238}U assuming infinite operation.
2. No decay heat uncertainty value is provided for the Region I CWO case since the analysis result for all runs is 0.00%.

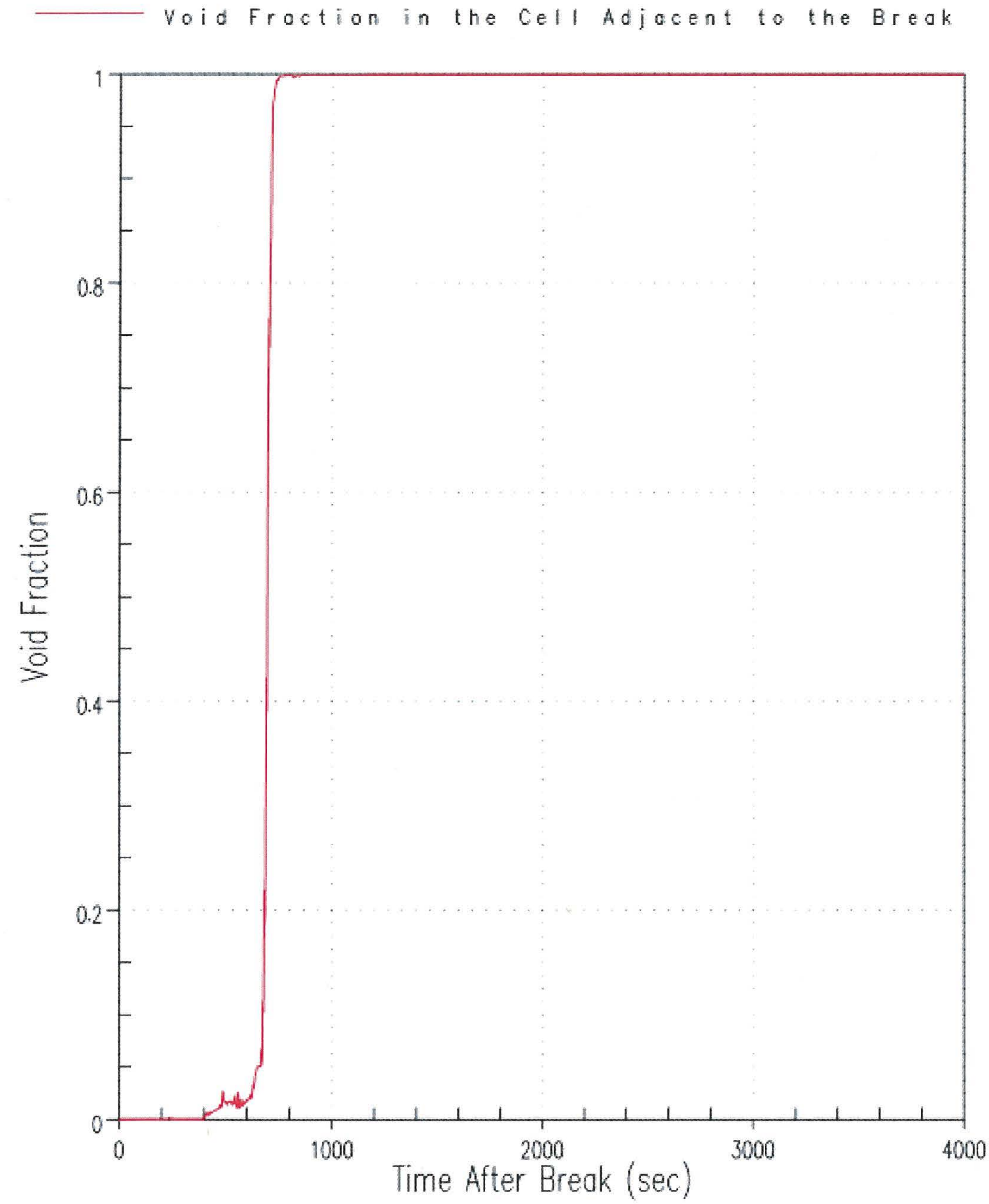

Figure 1 VCSNS Break Flow Void Fraction for the Region I Analysis PCT Case

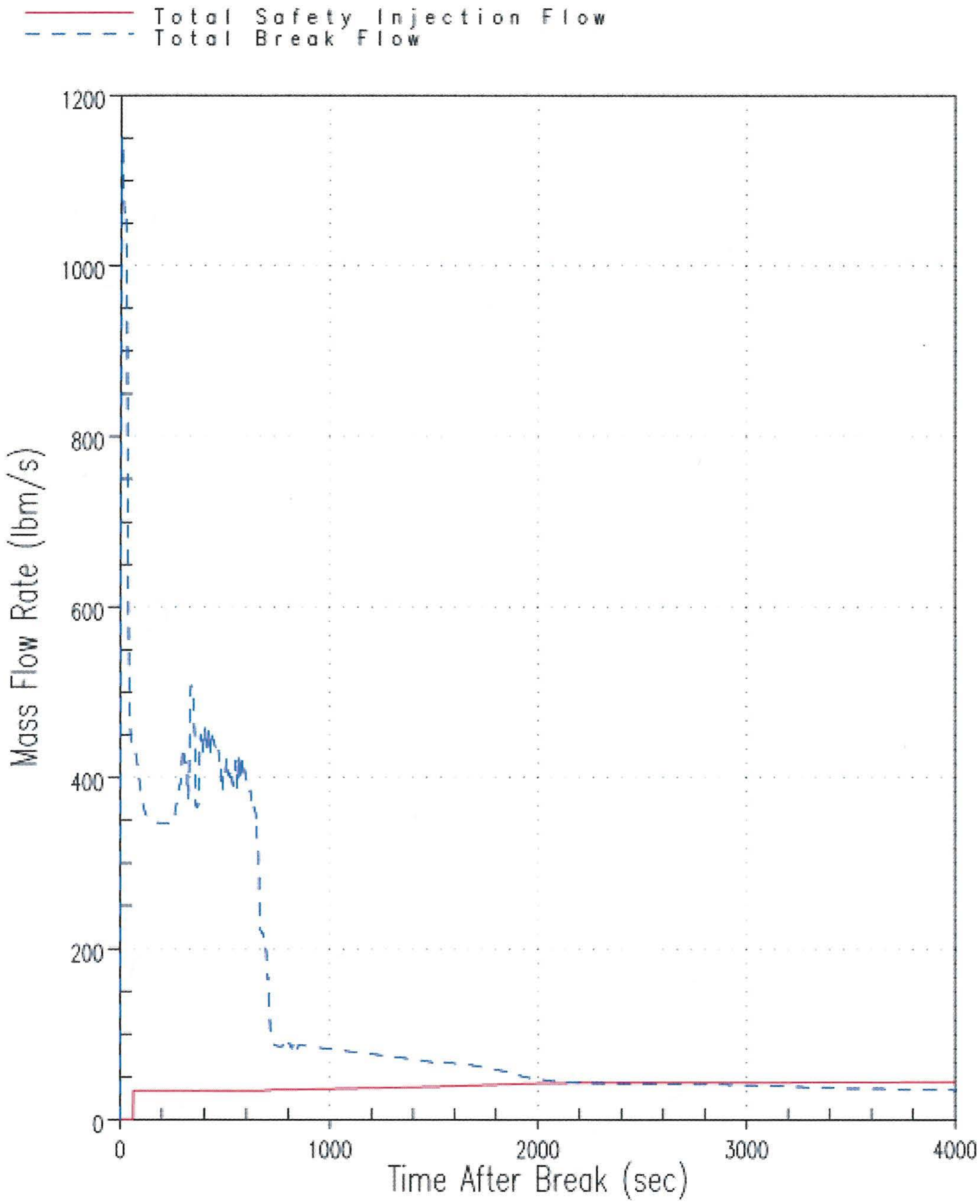

Figure 2 VCSNS Total Safety Injection Flow (not Including Accumulator Flow) and Total Break Flow for the Region I Analysis PCT Case

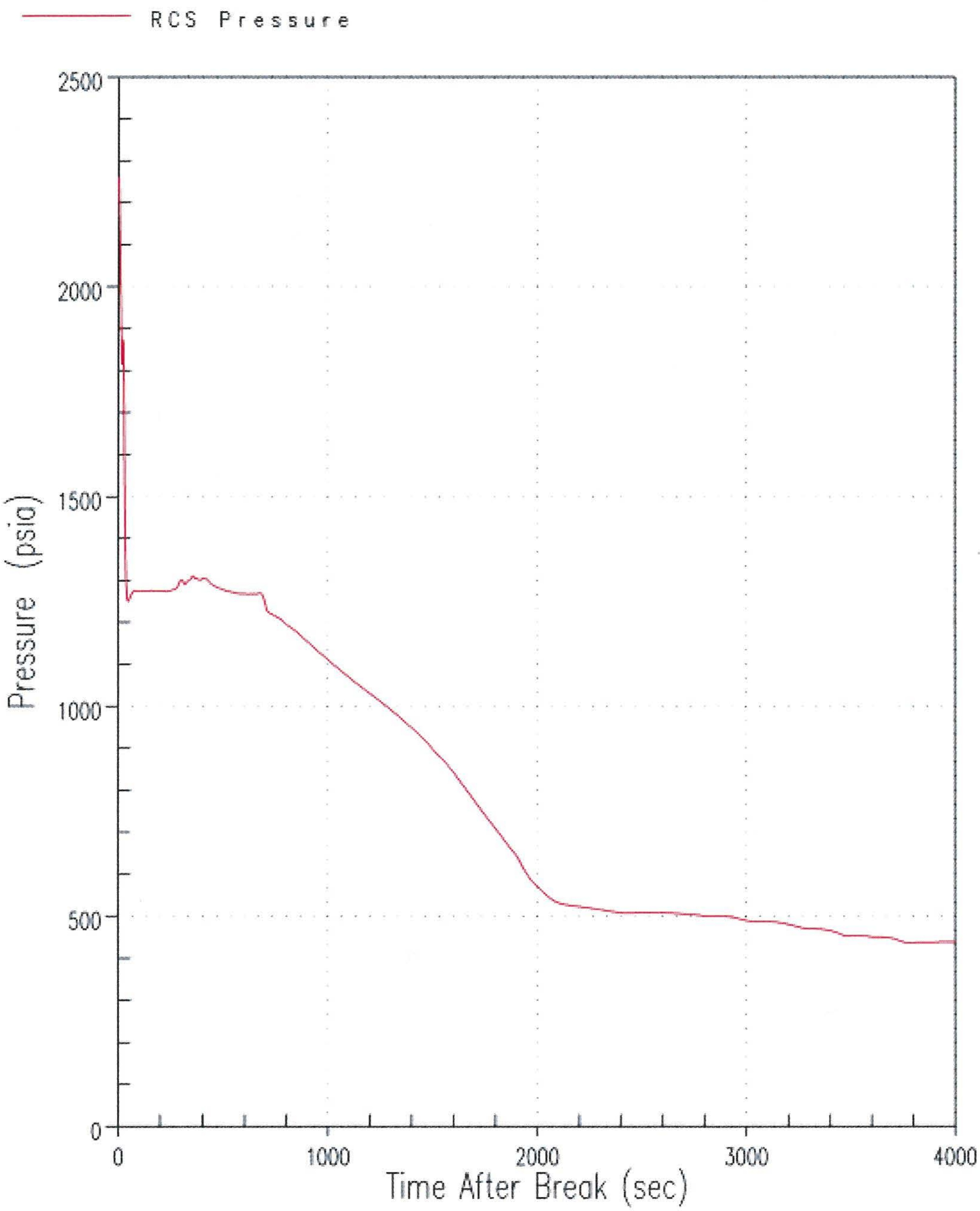

Figure 3 VCSNS RCS Pressure for the Region I Analysis PCT Case

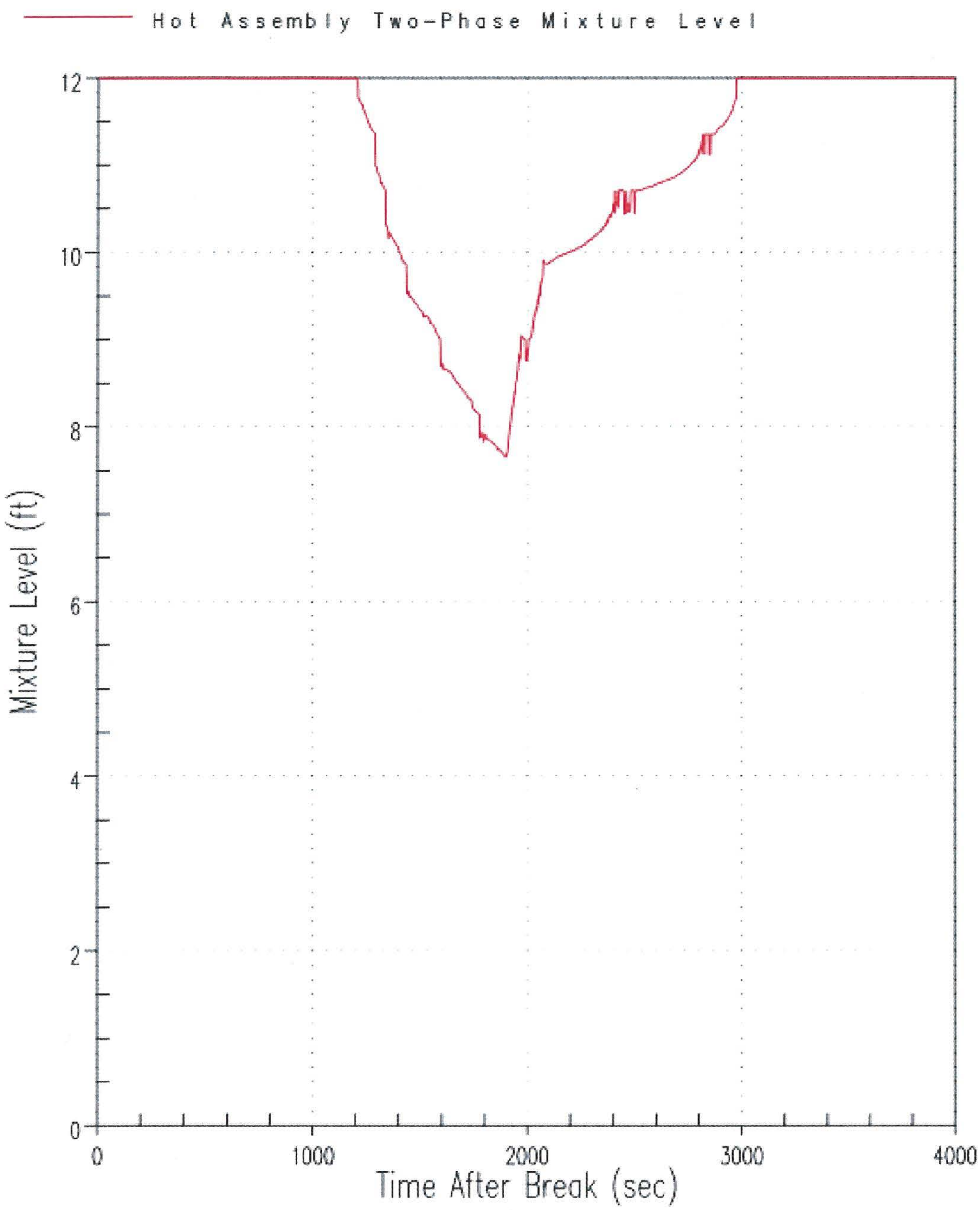

Figure 4 VCSNS Hot Assembly Two-Phase Mixture Level (Relative to Bottom of Active Fuel) for the Region I Analysis PCT Case

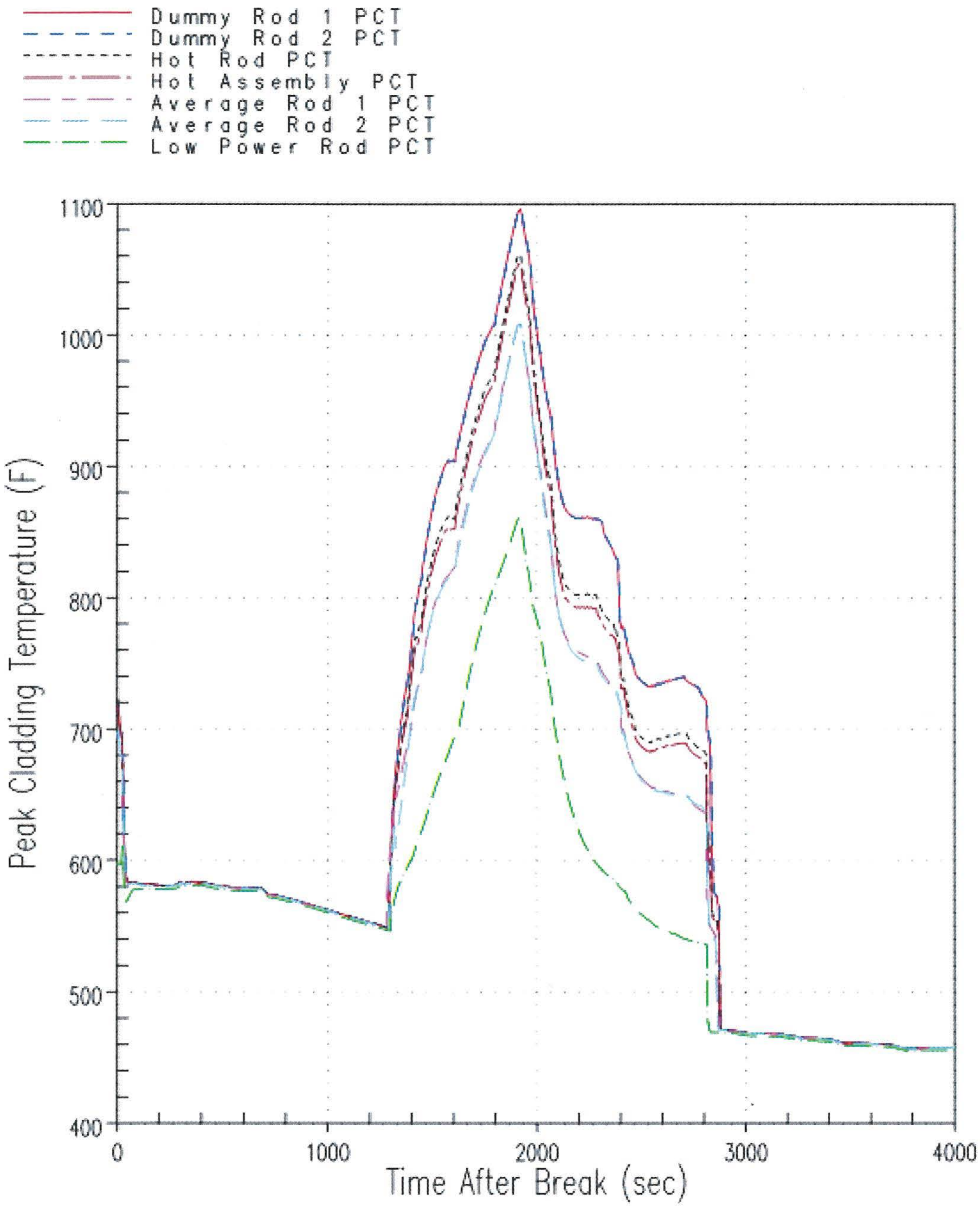

Figure 5 VCSNS Peak Cladding Temperature for all Rods for the Region I Analysis PCT Case

Note: This figure presents the uncertainty analysis results without the PCT penalty for the gamma energy redistribution error correction.

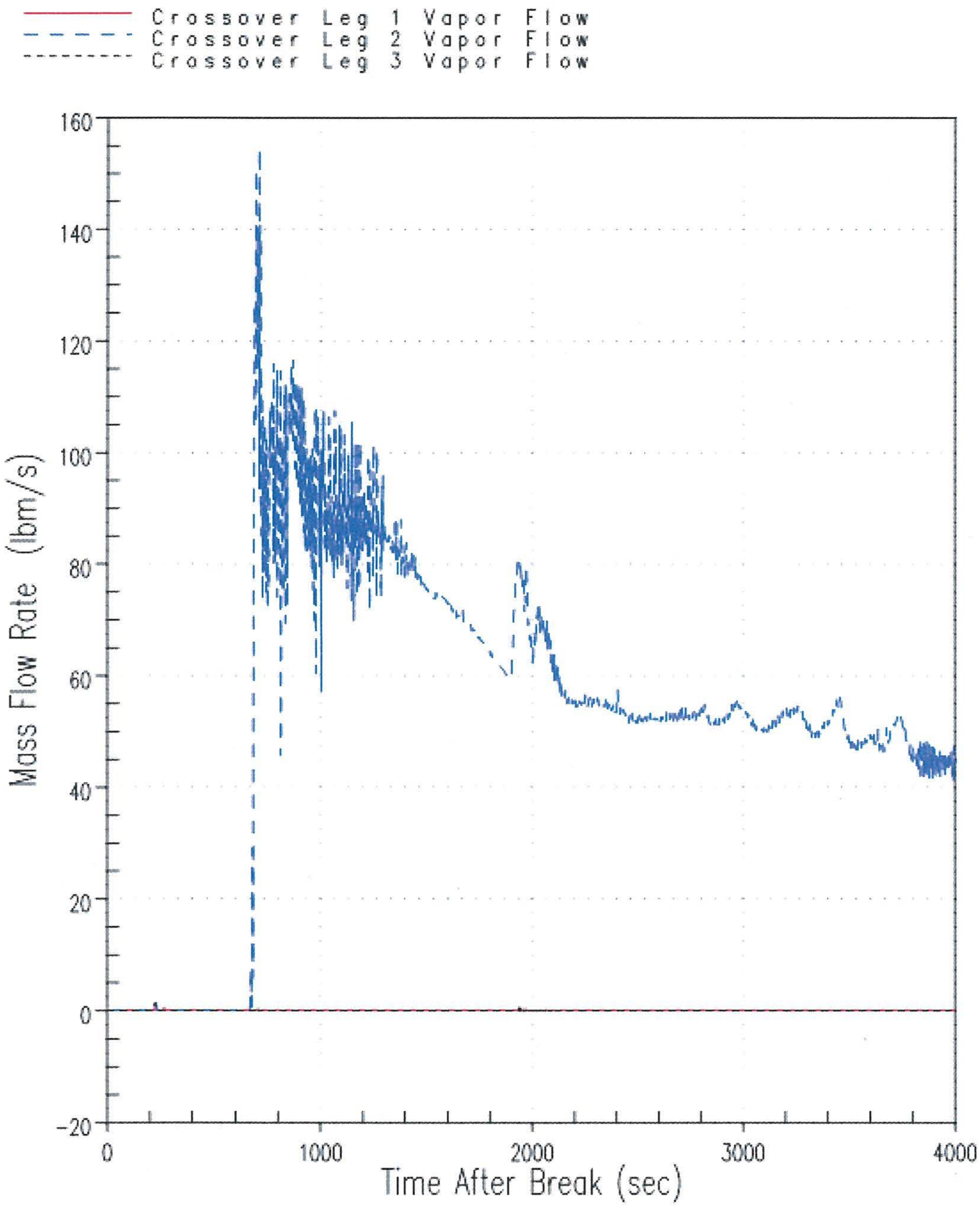

Figure 6 VCSNS Vapor Mass Flow Rate through the Crossover Legs for the Region I Analysis PCT Case

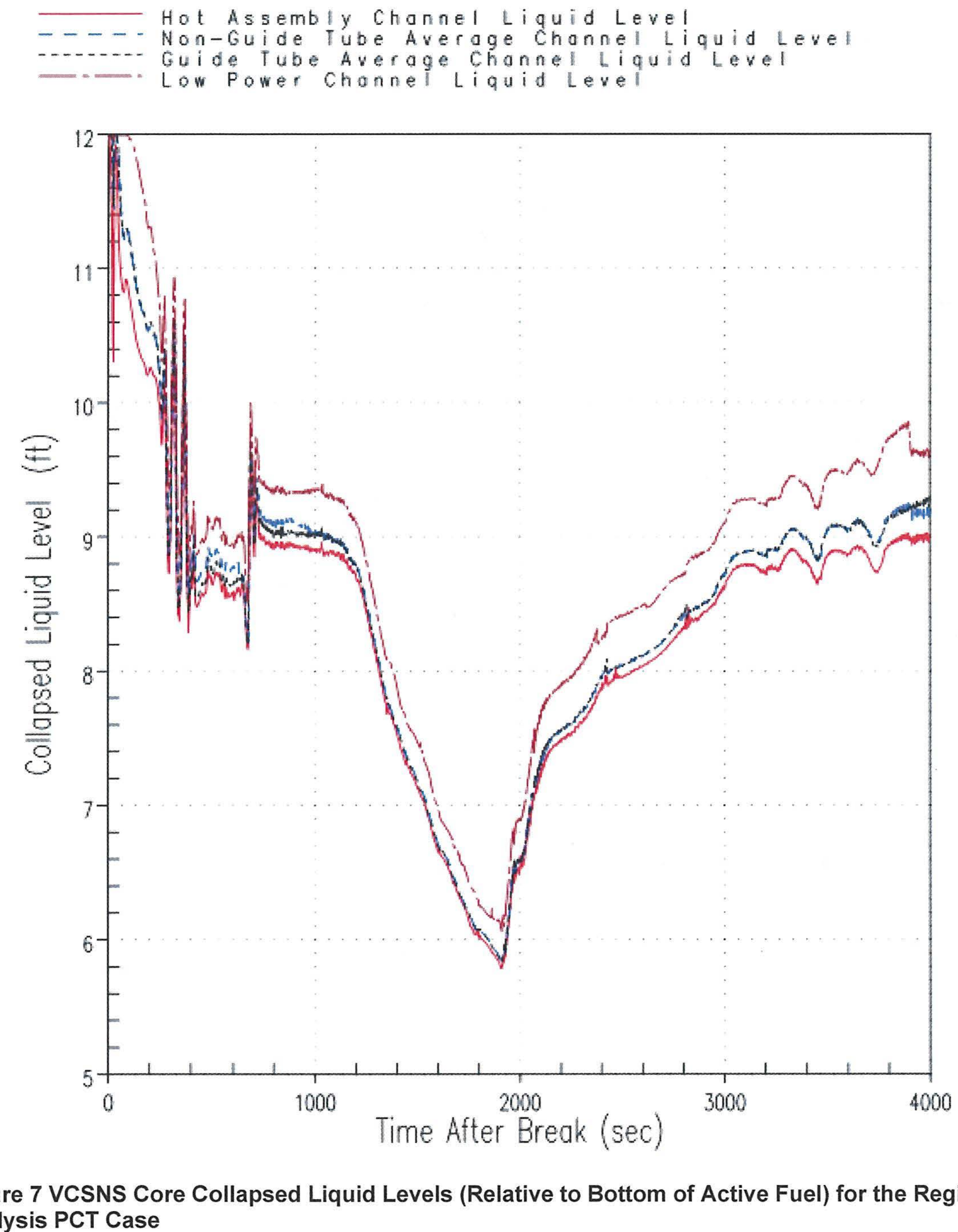

Figure 7 VCSNS Core Collapsed Liquid Levels (Relative to Bottom of Active Fuel) for the Region I Analysis PCT Case

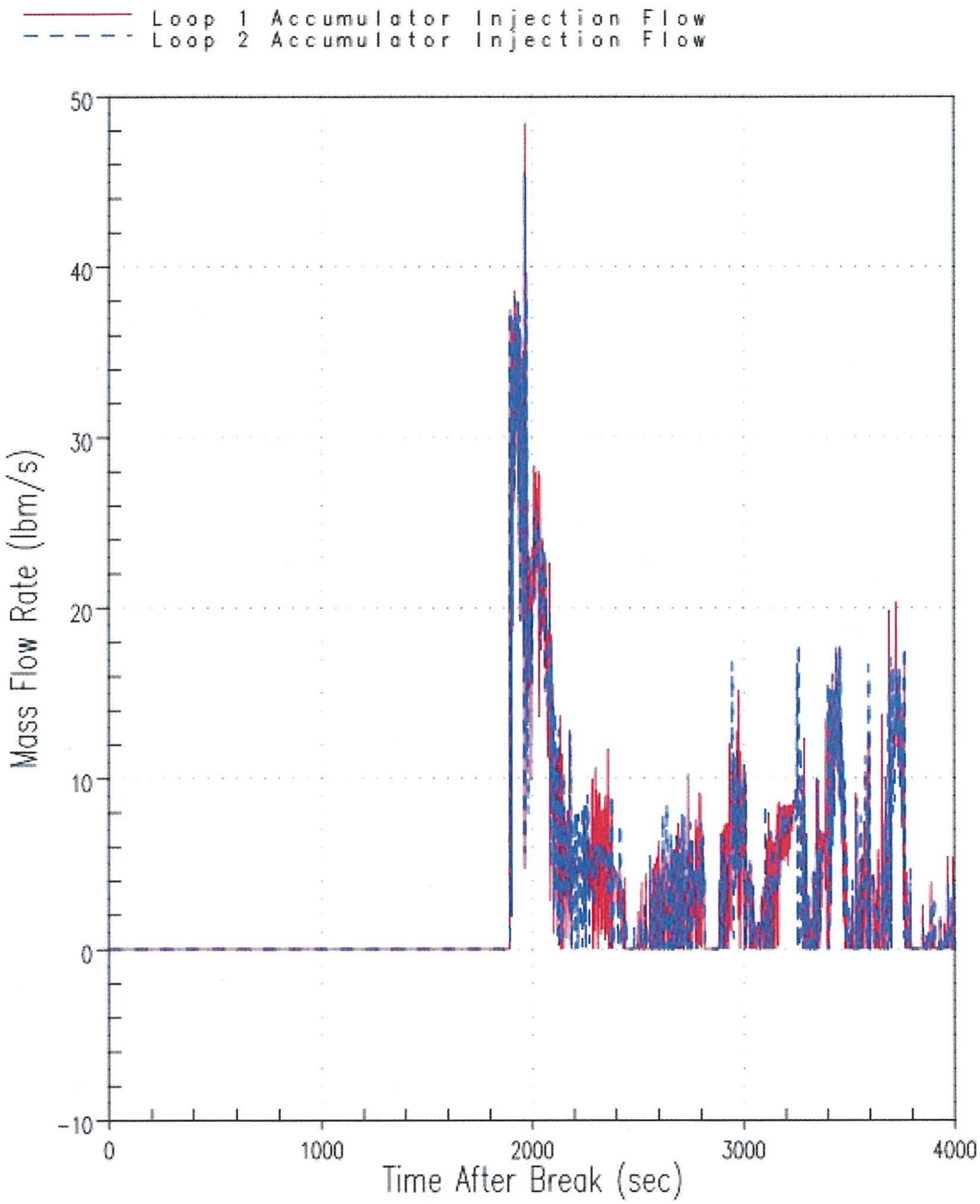

Figure 8 VCSNS Accumulator Injection Flow for the Region I Analysis PCT Case

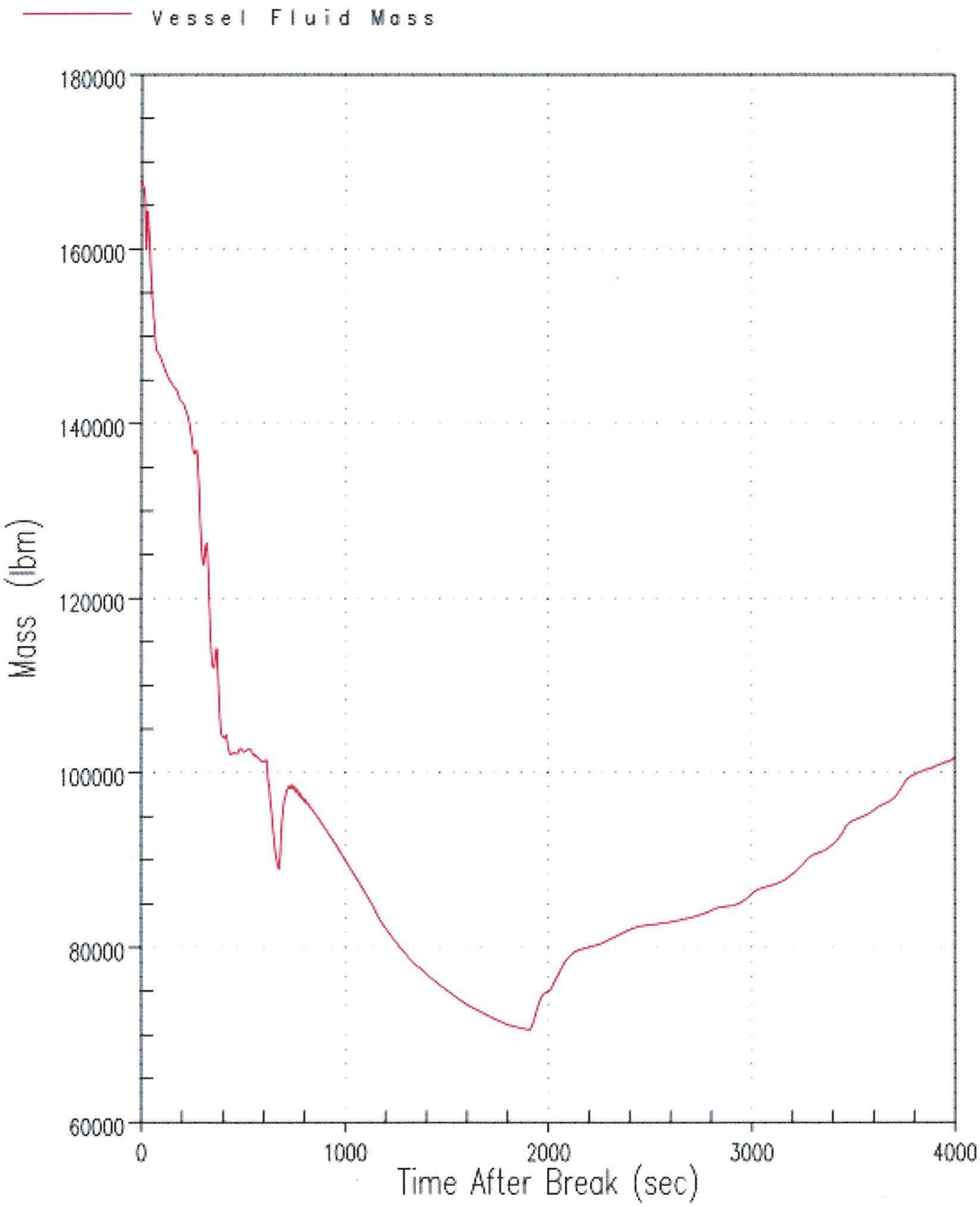

Figure 9 VCSNS Vessel Fluid Mass for the Region I Analysis PCT Case

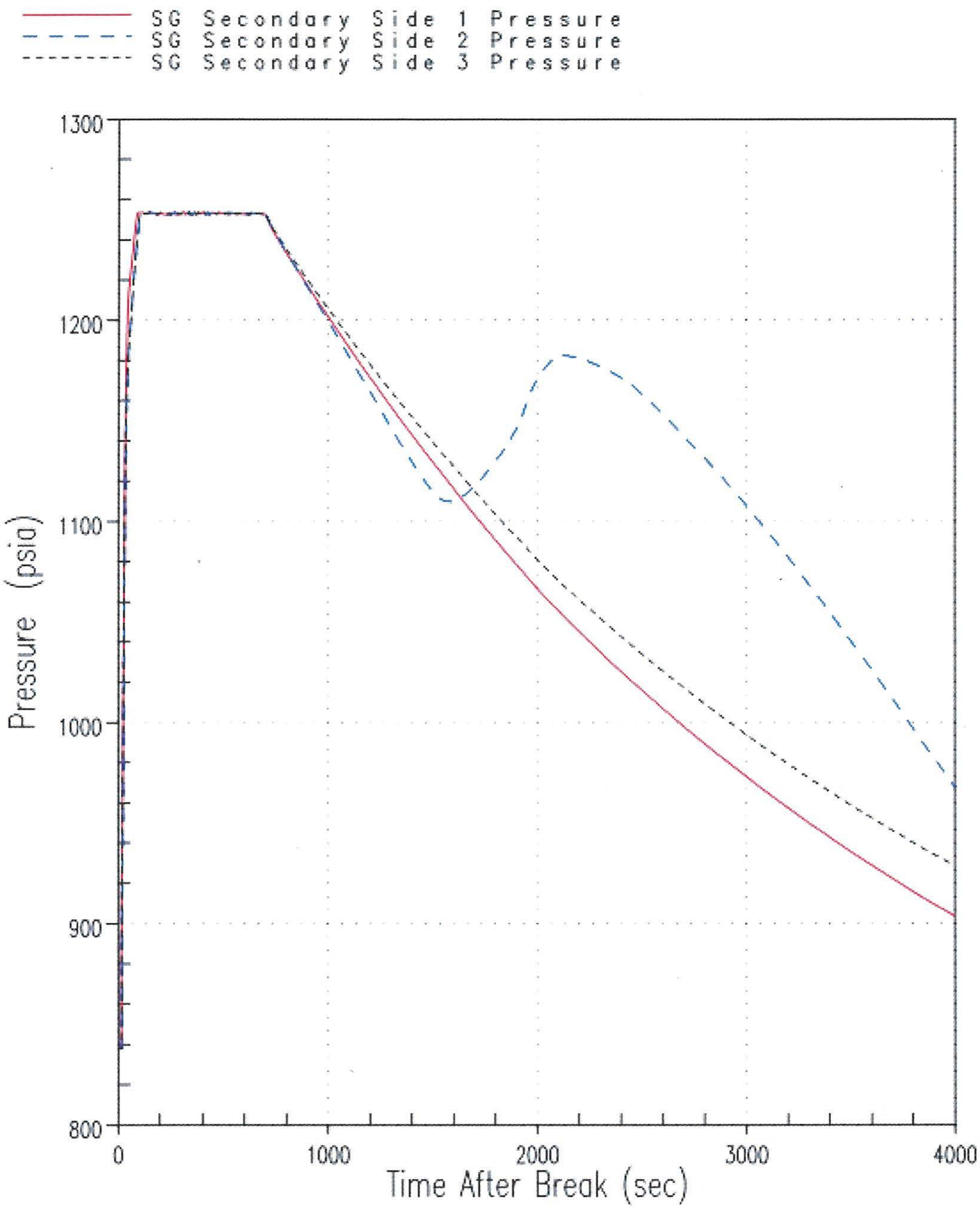

Figure 10 VCSNS Steam Generator Secondary Side Pressure for the Region I Analysis PCT Case

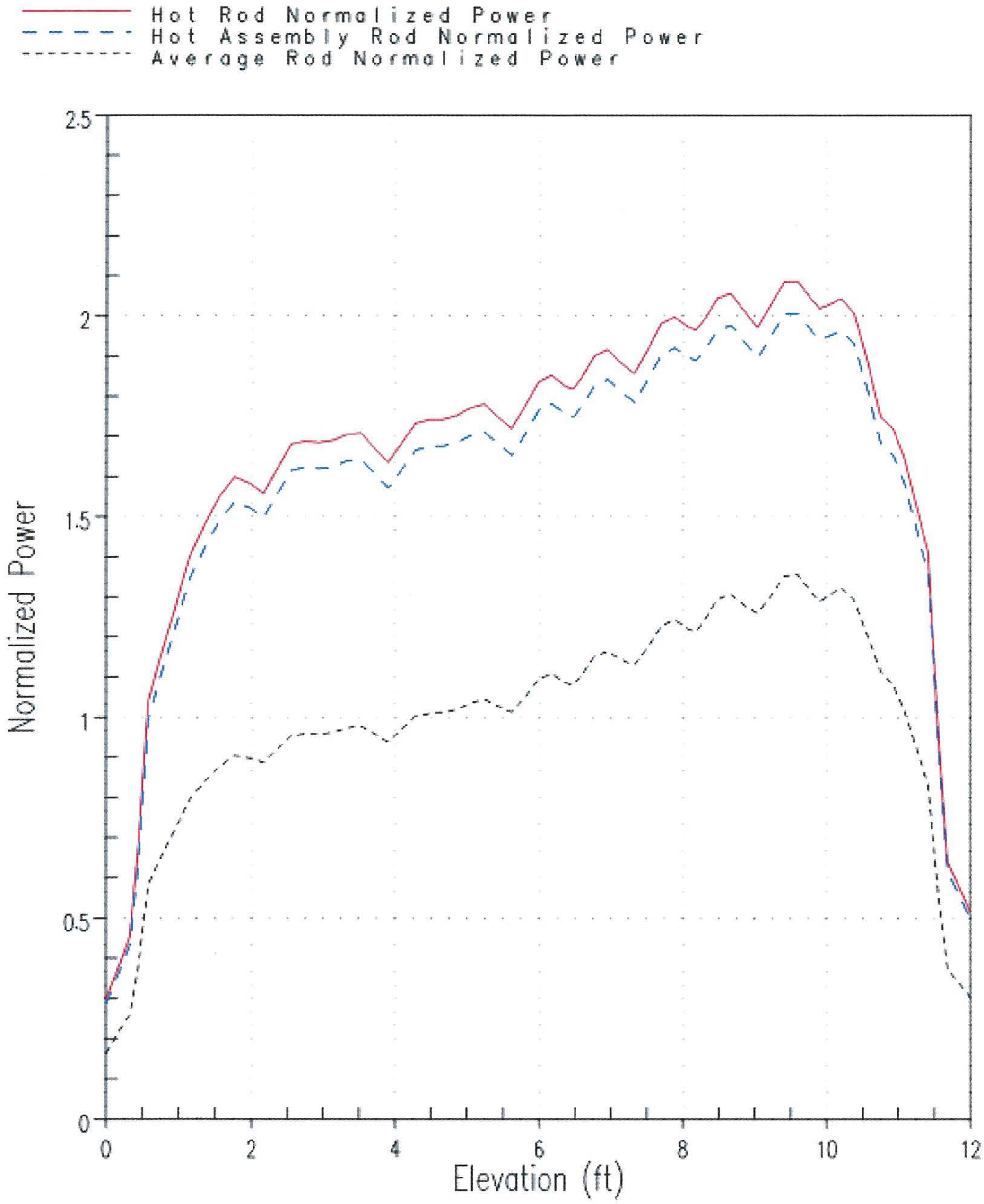

Figure 11 VCSNS Normalized Core Power Shapes for the Region I Analysis PCT Case

Note: The localized power decreases occur at grid elevations.

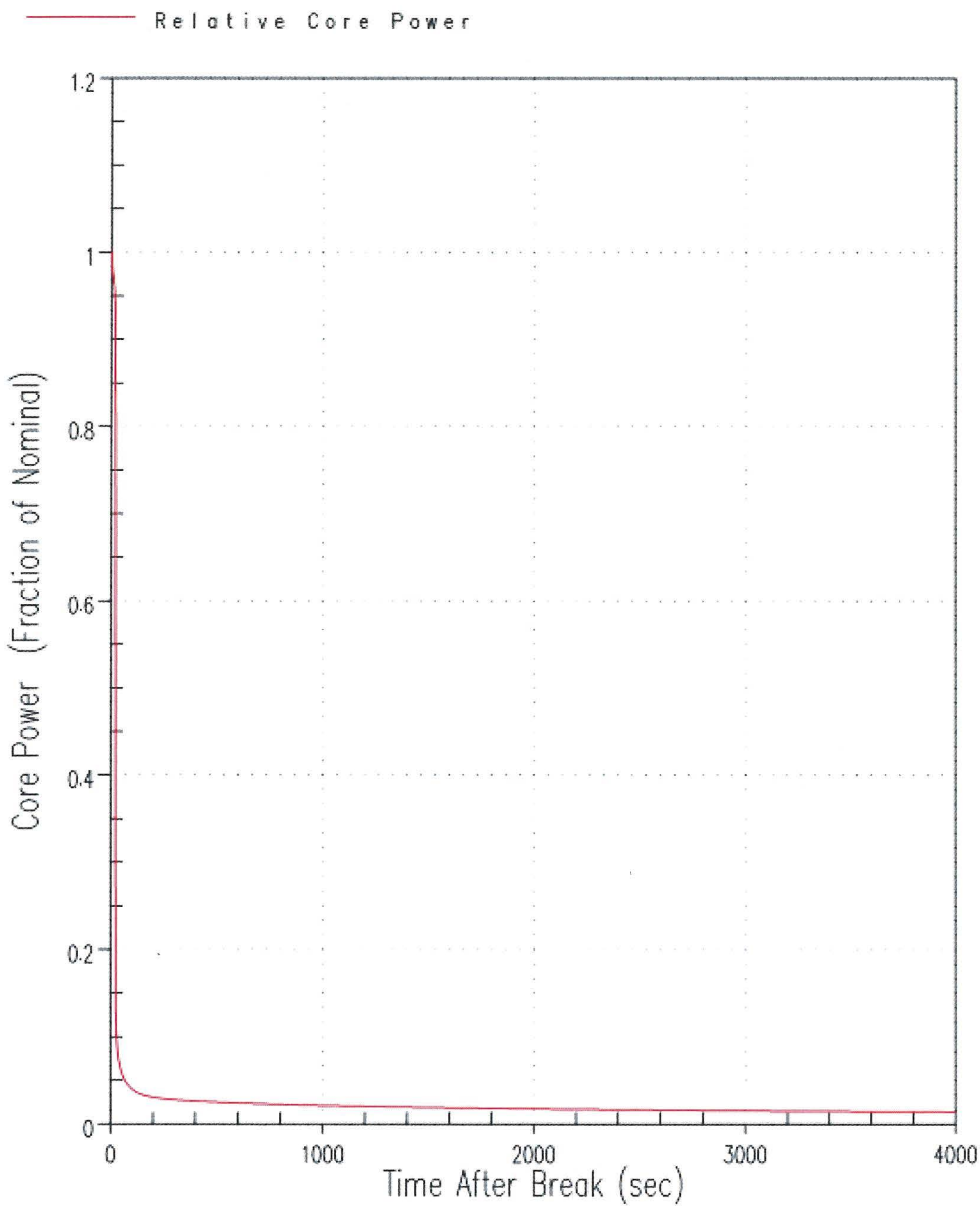

Figure 12 VCSNS Relative Core Power for the Region I Analysis PCT Case

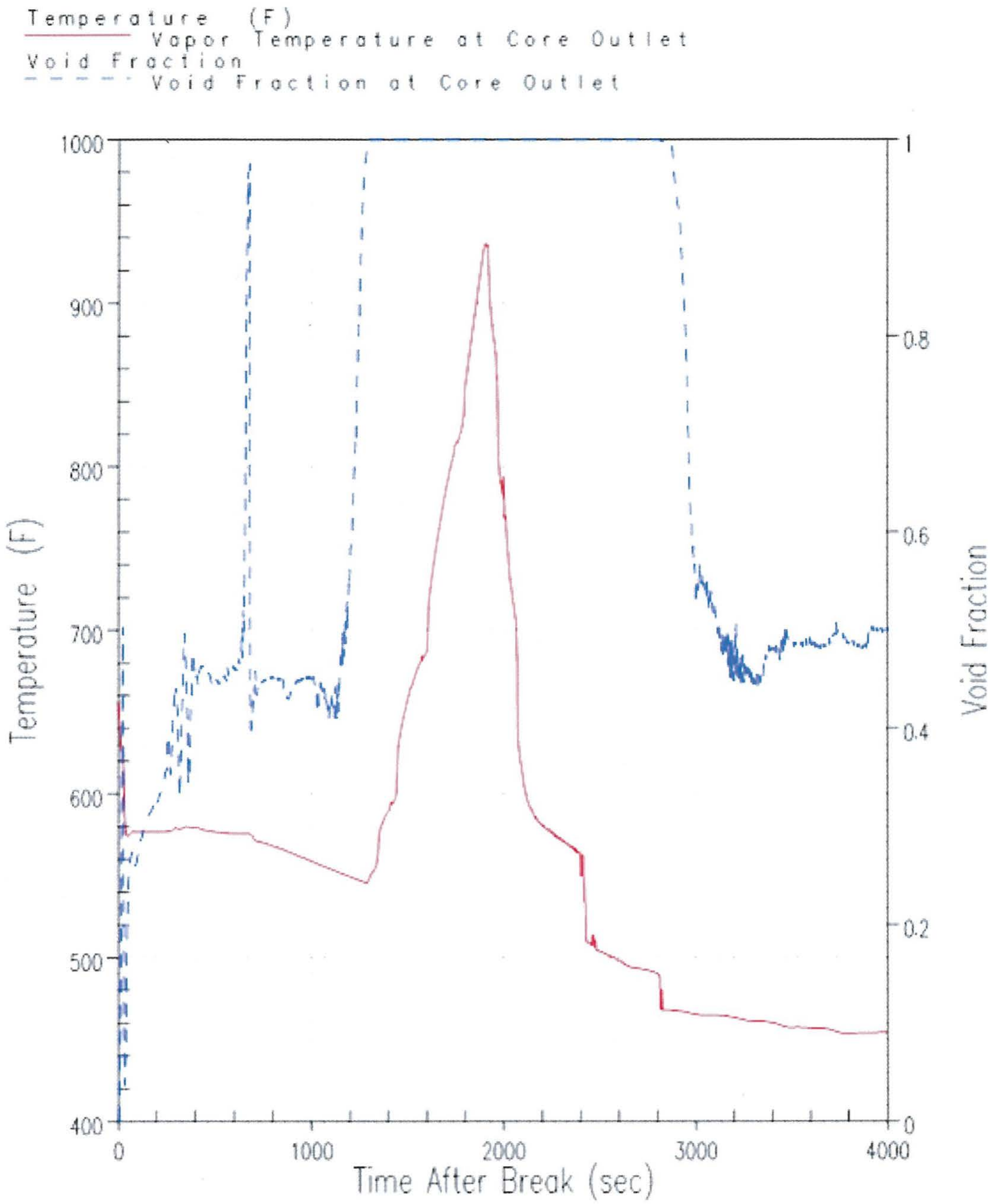

Figure 13 VCSNS Vapor Temperature and Void Fraction at Core Outlet for the Region I Analysis PCT Case

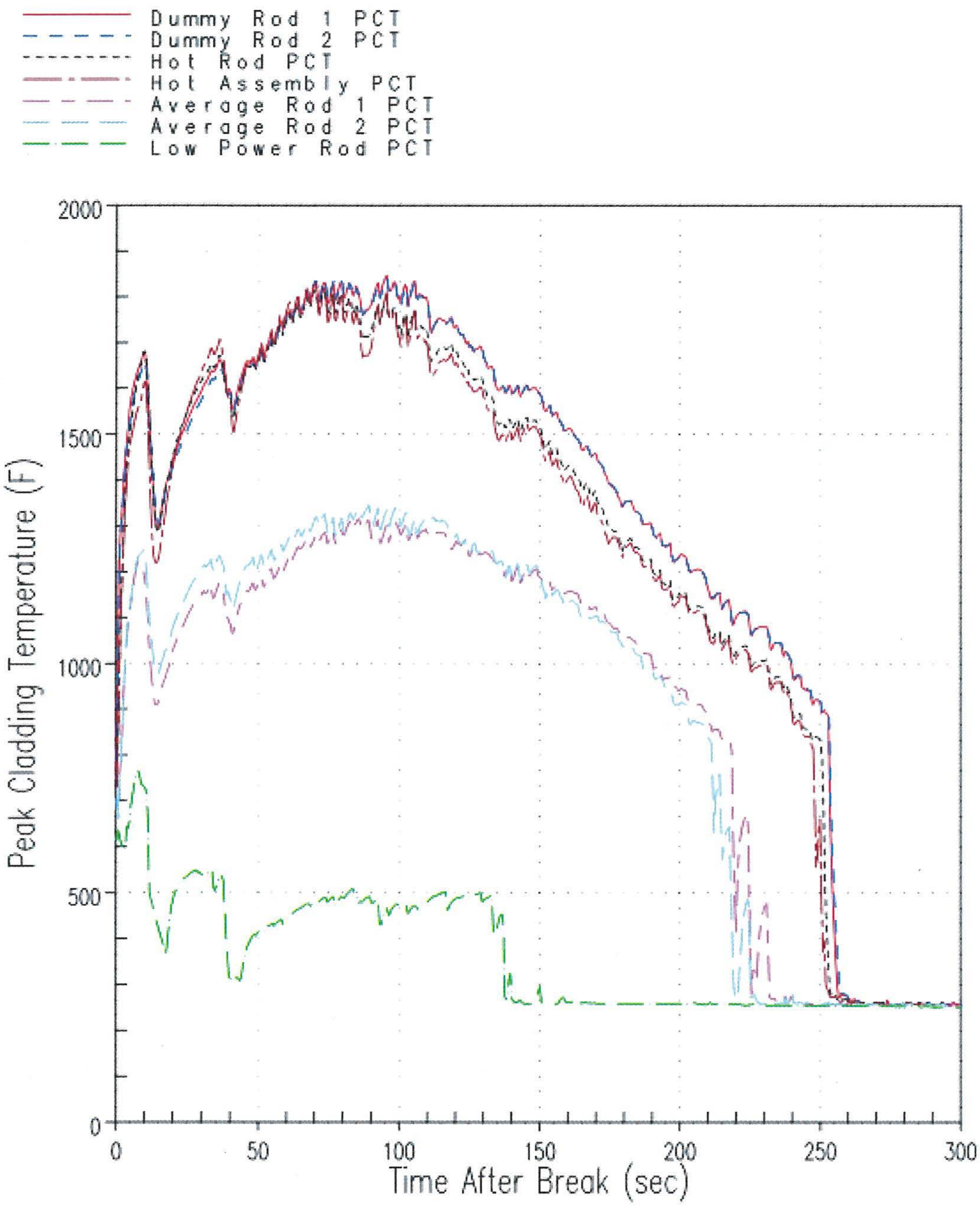

Figure 14 VCSNS Peak Cladding Temperature for all Rods for the Region II Analysis PCT Case

Note: This figure presents the uncertainty analysis results without the PCT penalty for the gamma energy redistribution error correction.

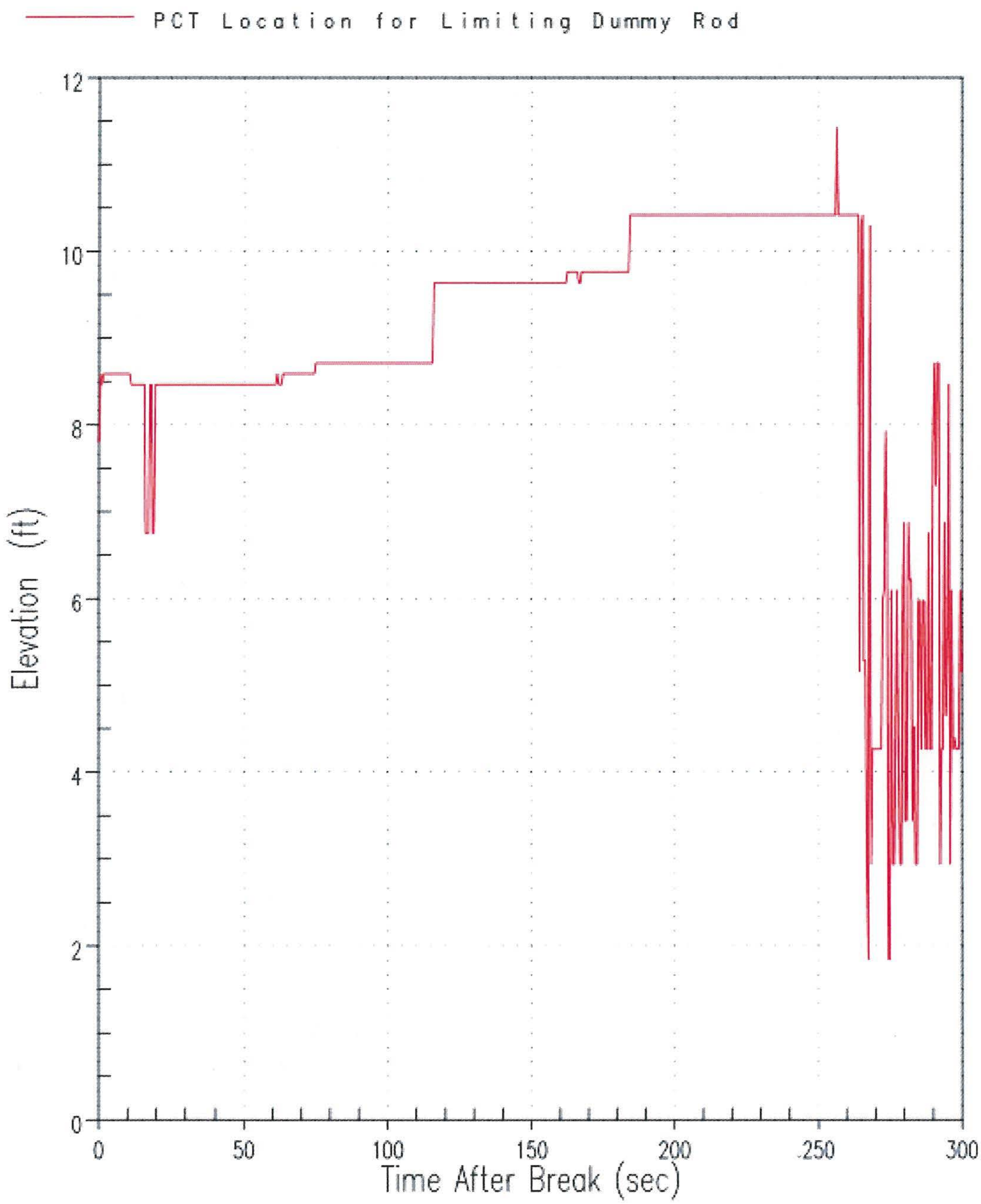

Figure 15 VCSNS Peak Cladding Temperature Elevation (Relative to Bottom of Active Fuel) for the Region II Analysis PCT Case

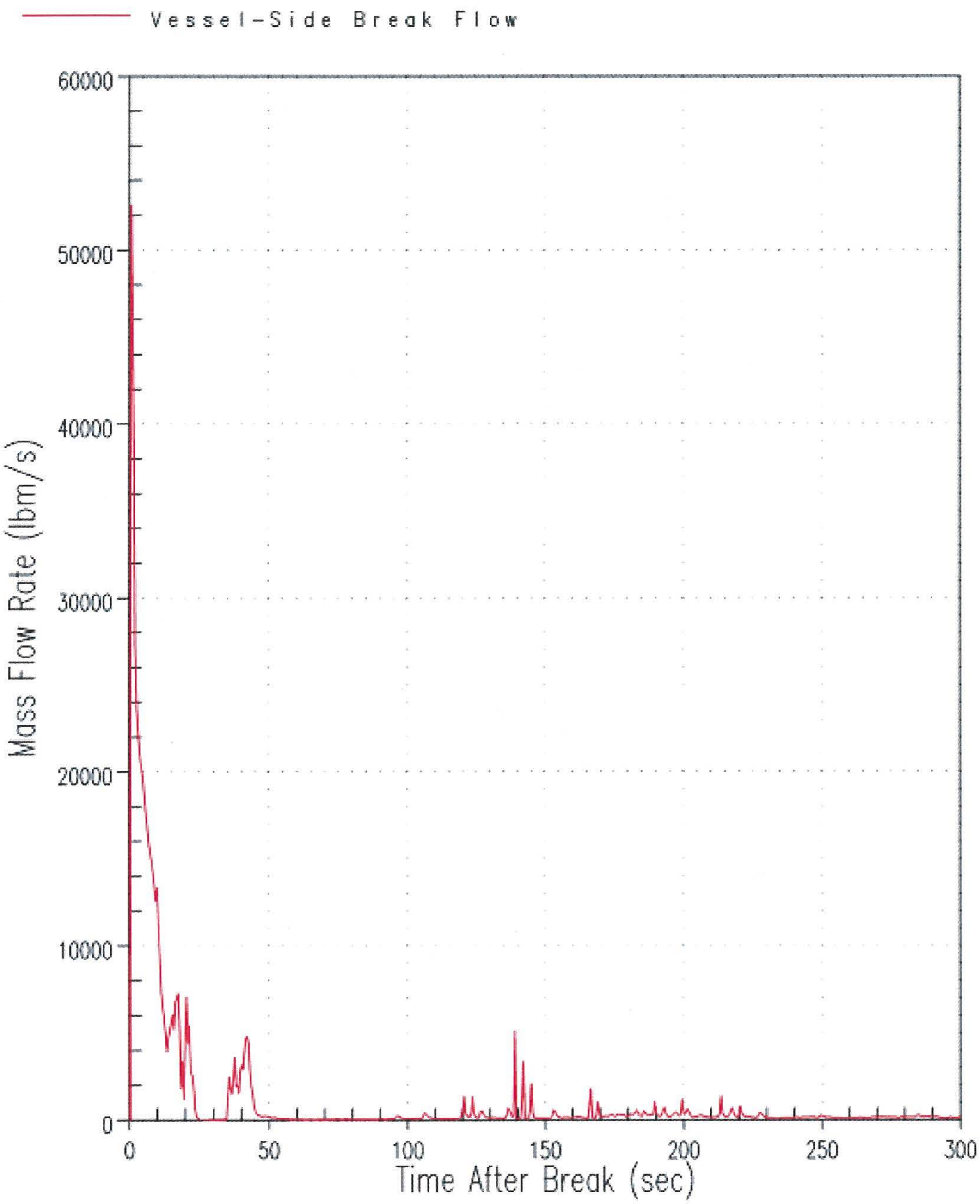

Figure 16a VCSNS Vessel-Side Break Mass Flow Rate for the Region II Analysis PCT Case

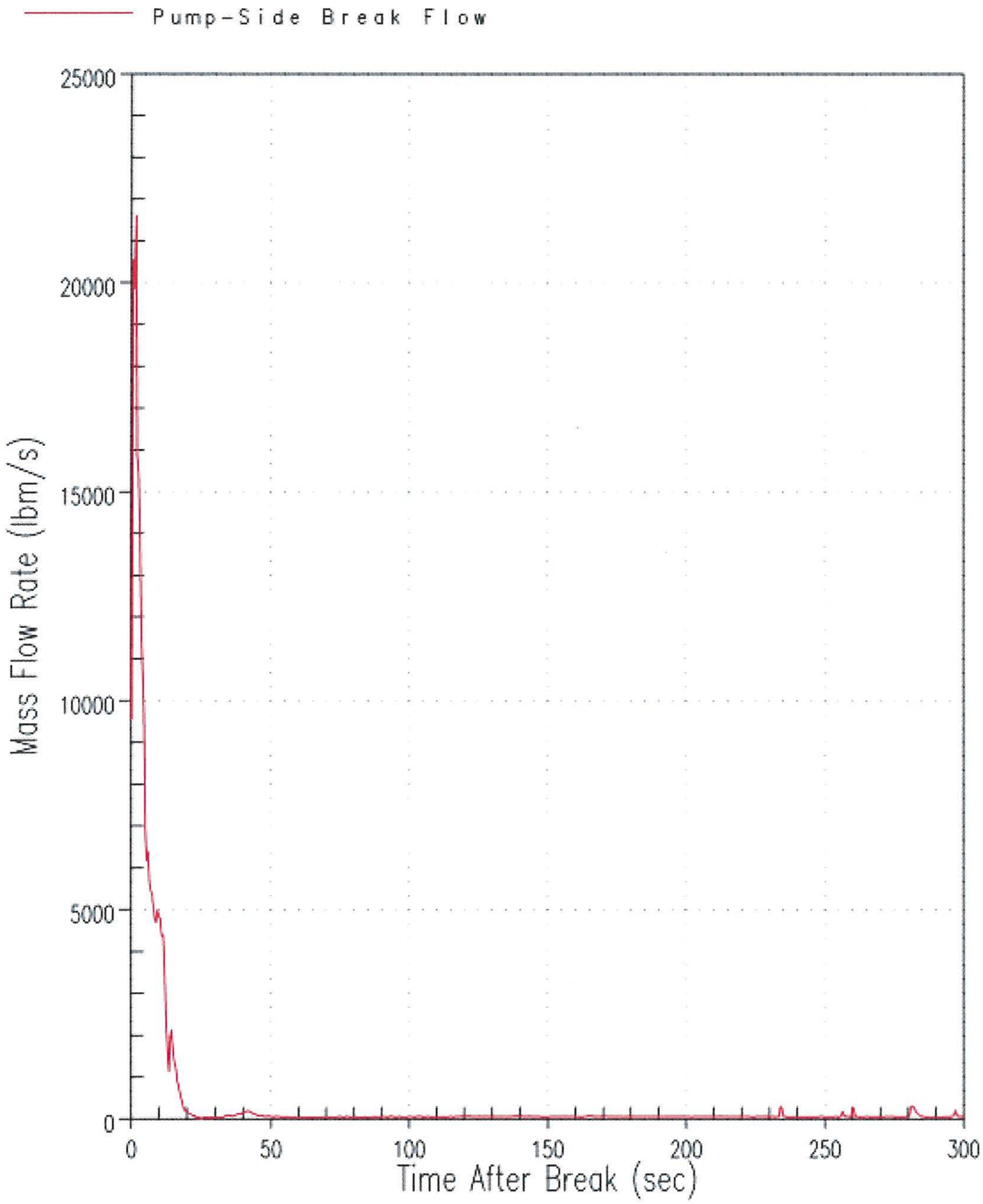

Figure 16b VCSNS Pump-Side Break Mass Flow Rate for the Region II Analysis PCT Case

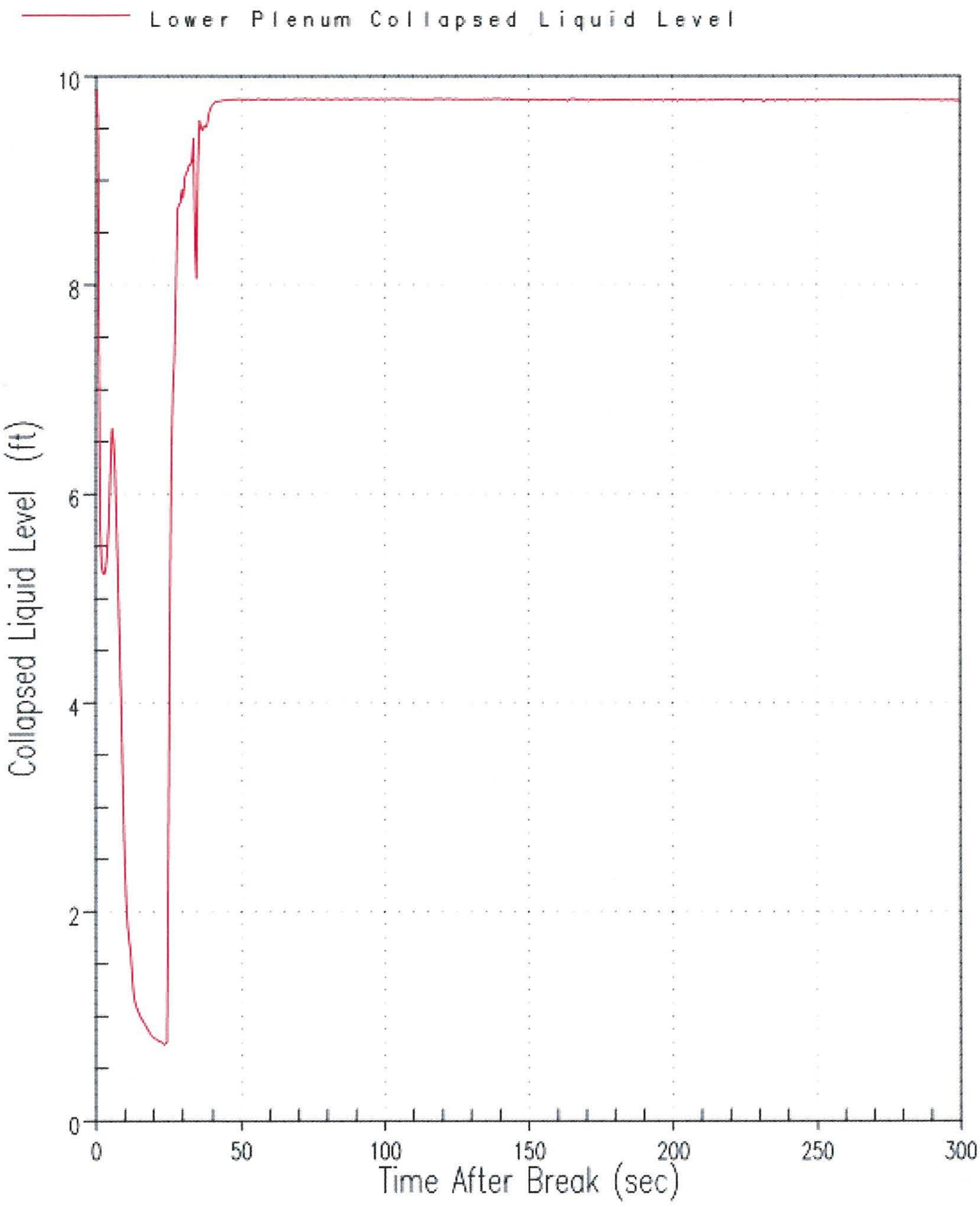

Figure 17 VCSNS Lower Plenum Collapsed Liquid Level (Relative to Inside Bottom of Vessel) for the Region II Analysis PCT Case

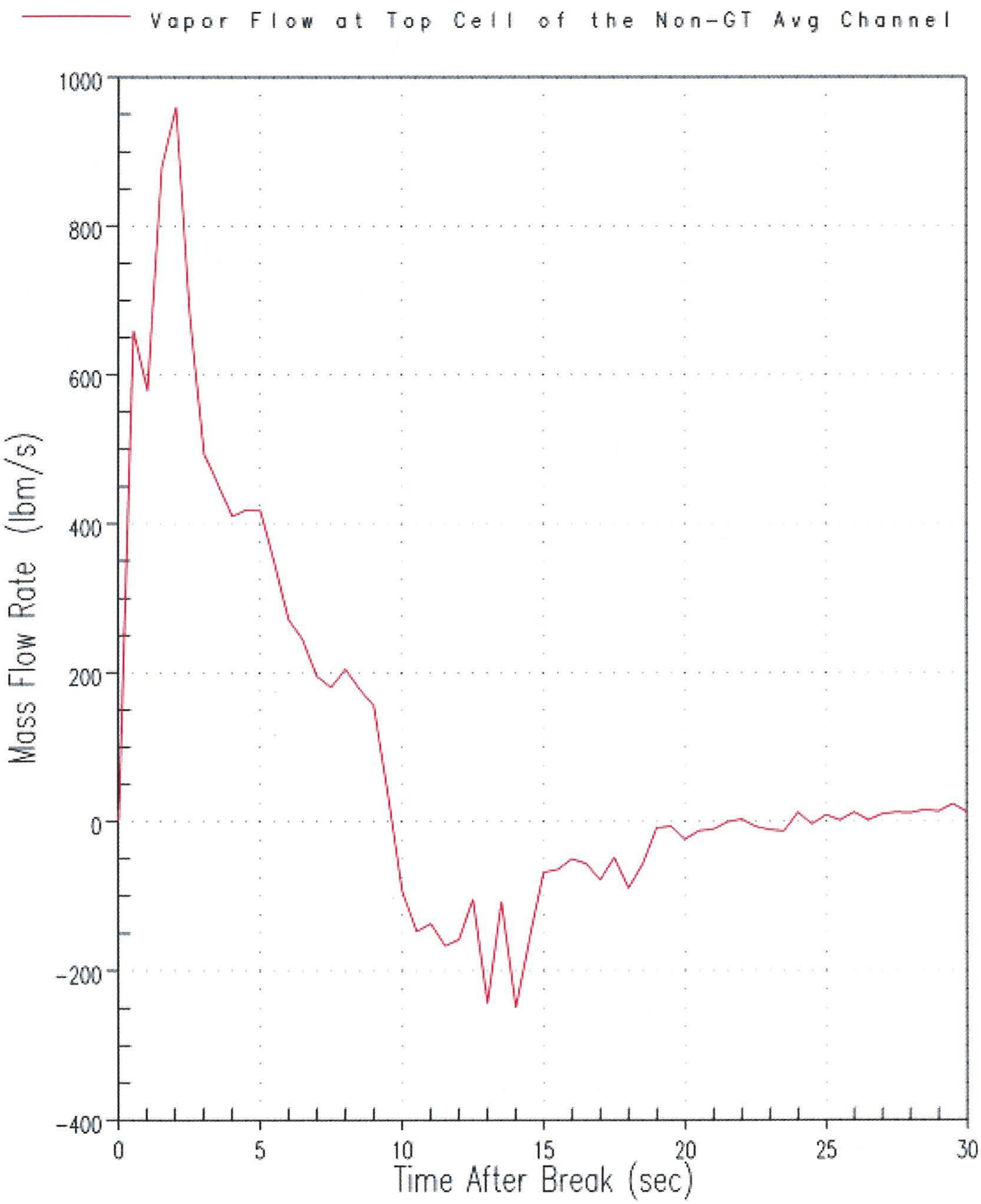

Figure 18 VCSNS Vapor Mass Flow Rate at the Top Cell Face of the Core Average Channel not Under Guide Tubes for the Region II Analysis PCT Case

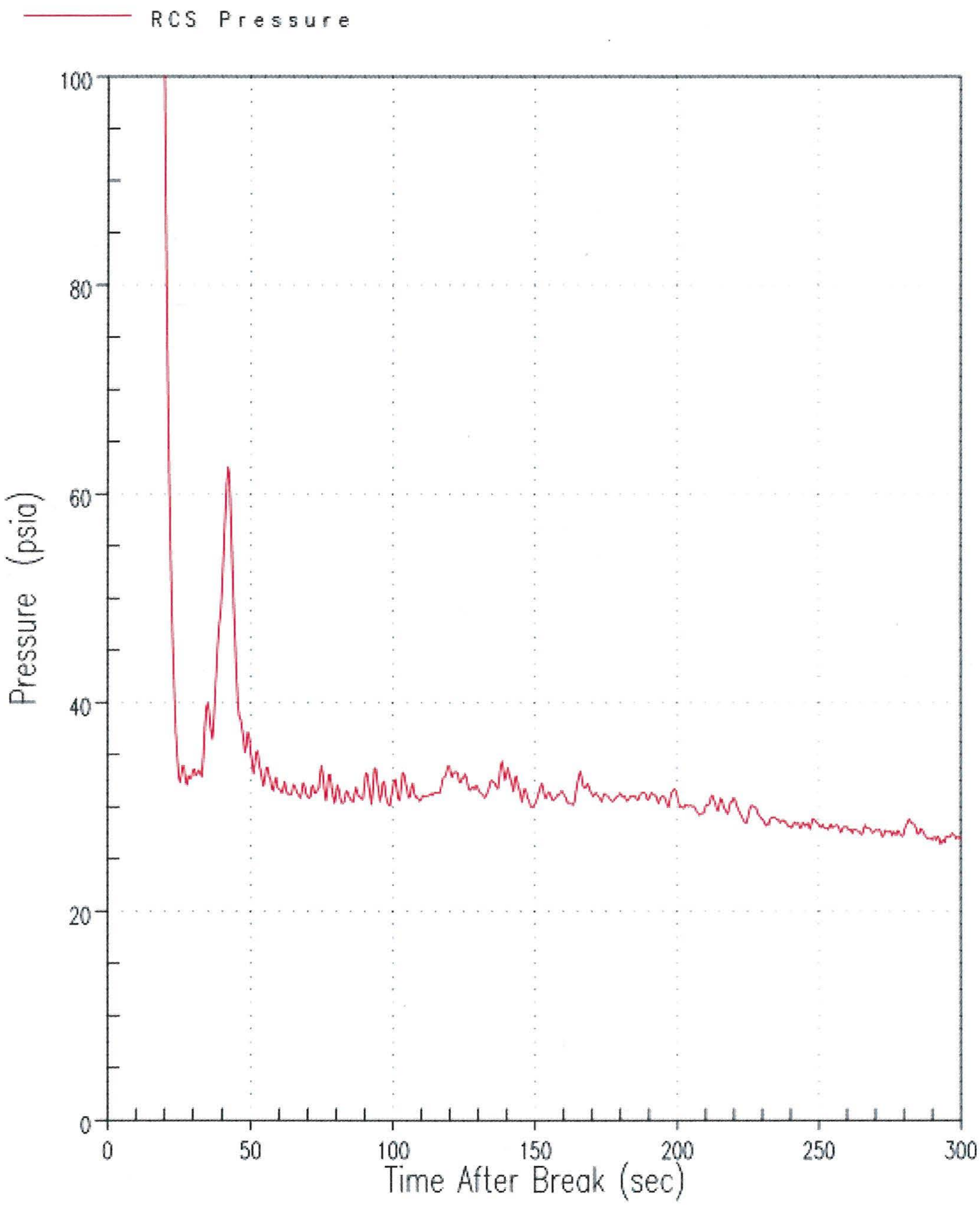

Figure 19 VCSNS RCS Pressure for the Region II Analysis PCT Case

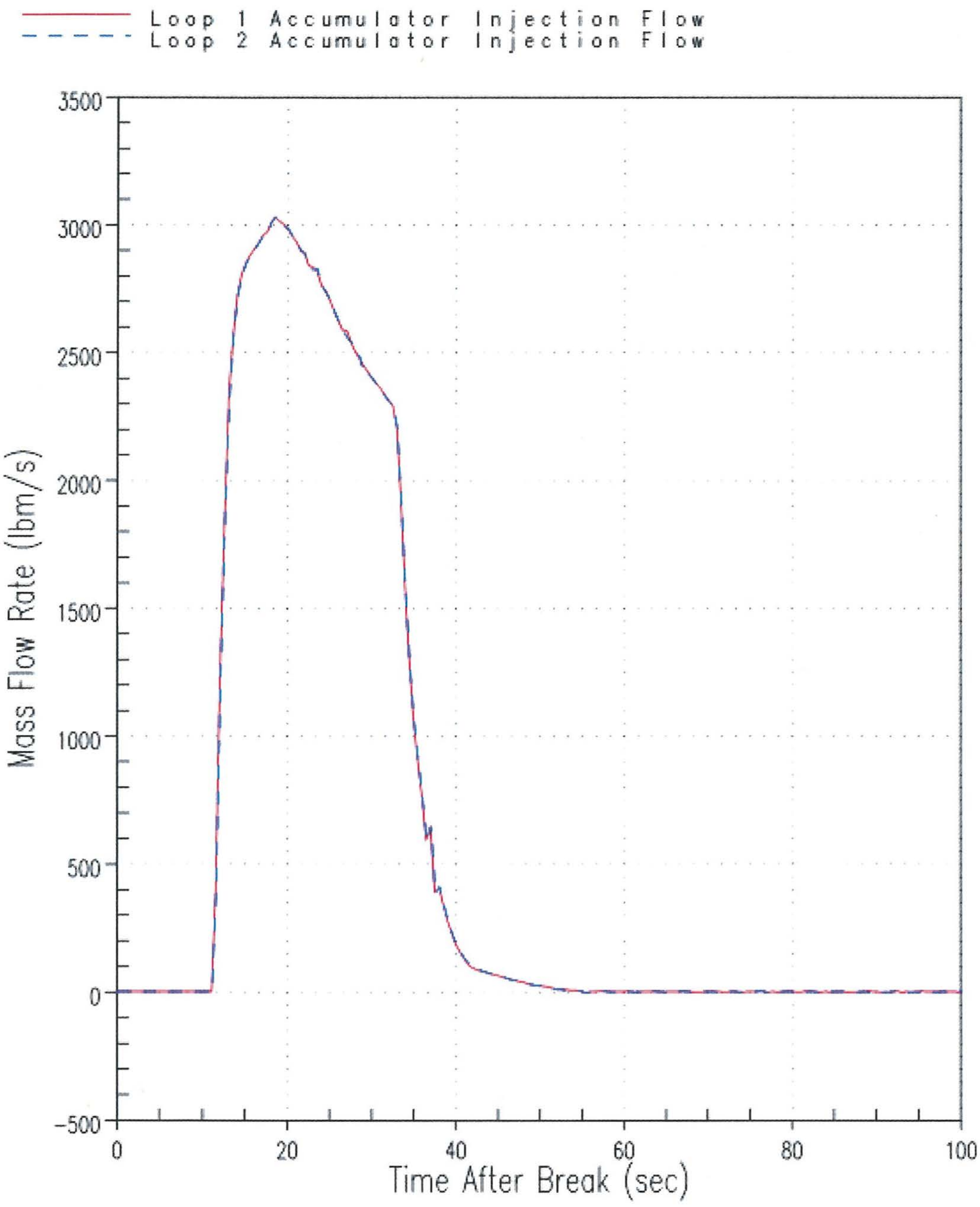

Figure 20 VCSNS Accumulator Injection Flow per Loop for the Region II Analysis PCT Case

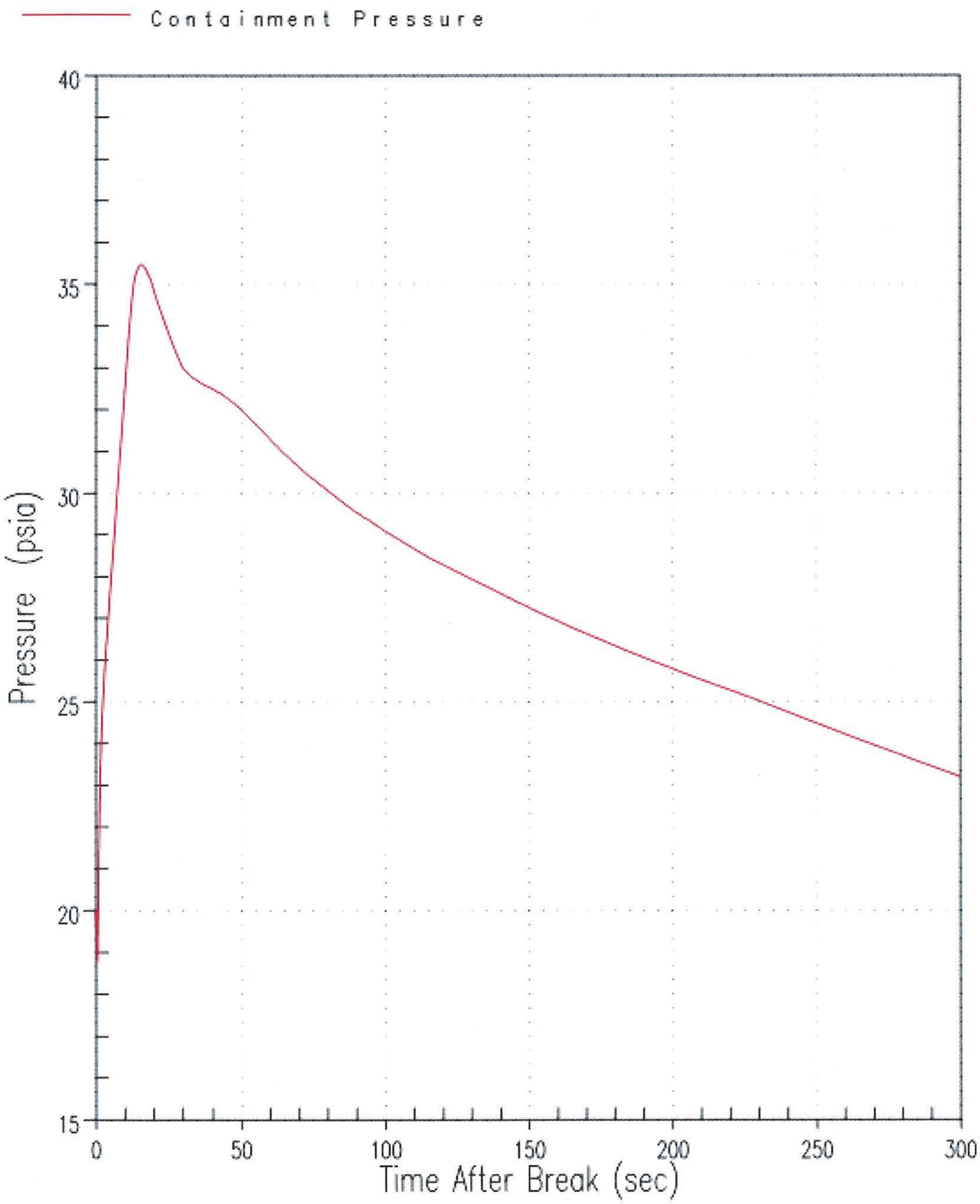

Figure 21 VCSNS Containment Pressure for the Region II Analysis PCT Case

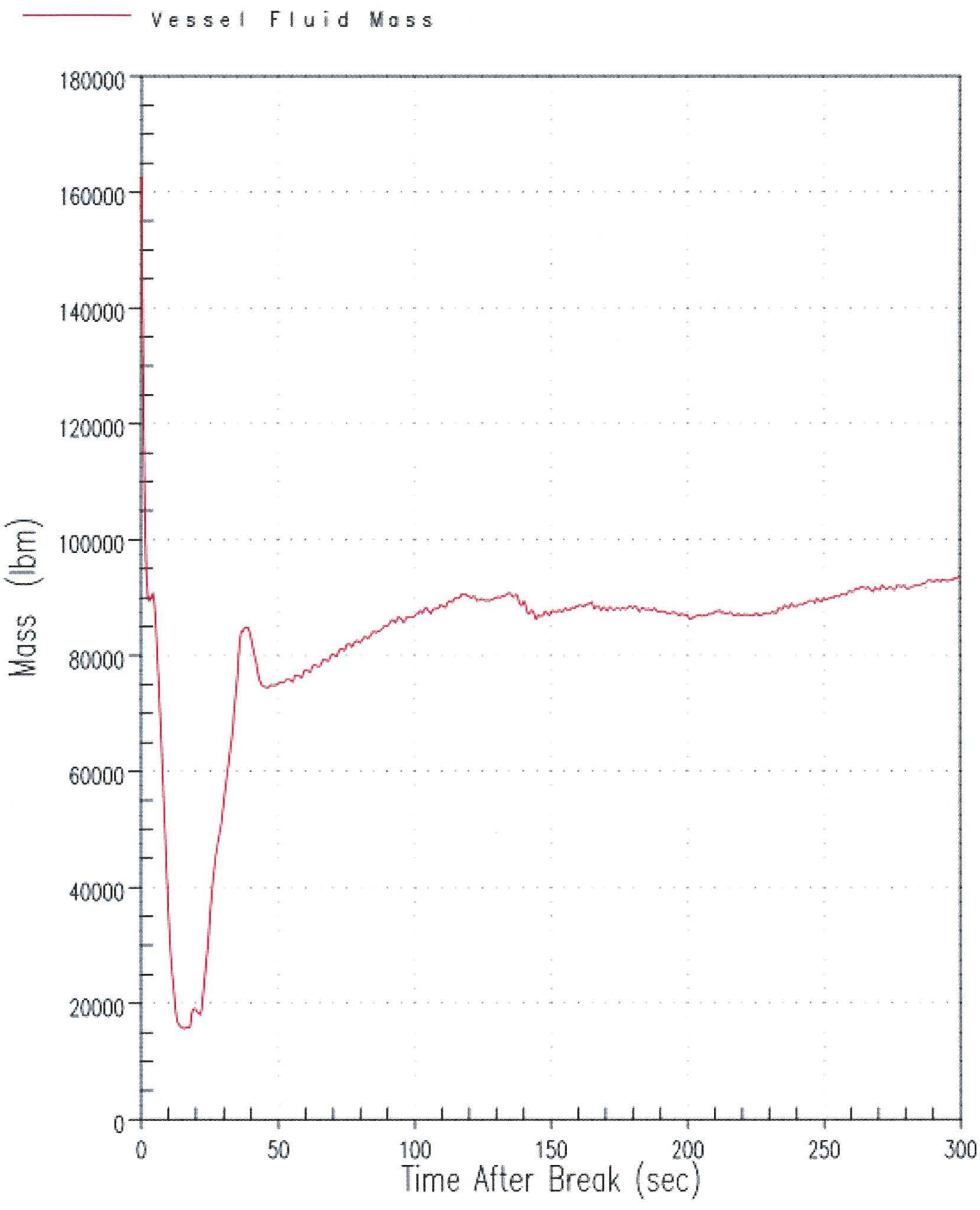

Figure 22 VCSNS Vessel Fluid Mass for the Region II Analysis PCT Case

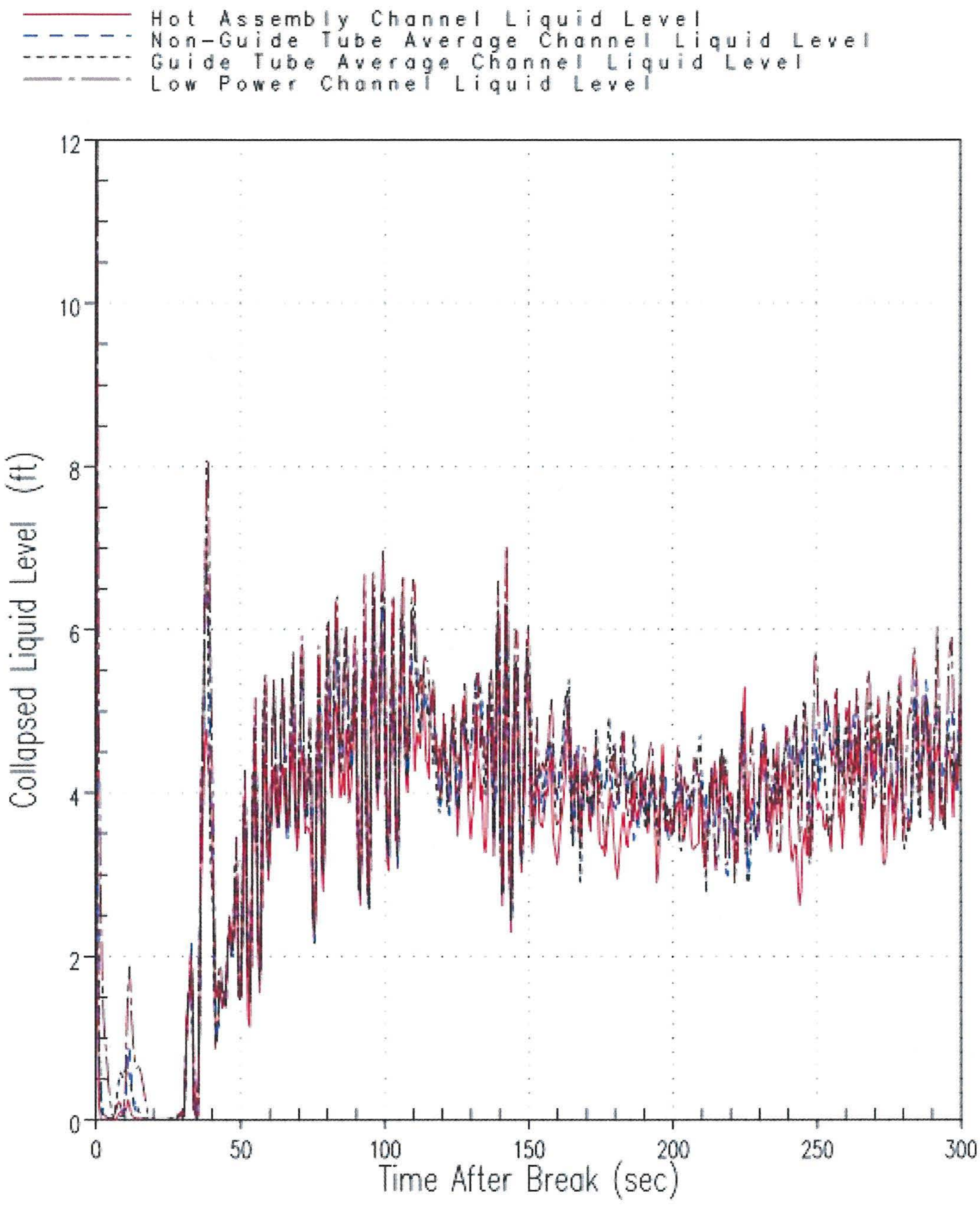

Figure 23 VCSNS Collapsed Liquid Level for Each Core Channel (Relative to Bottom of Active Fuel) for the Region II Analysis PCT Case

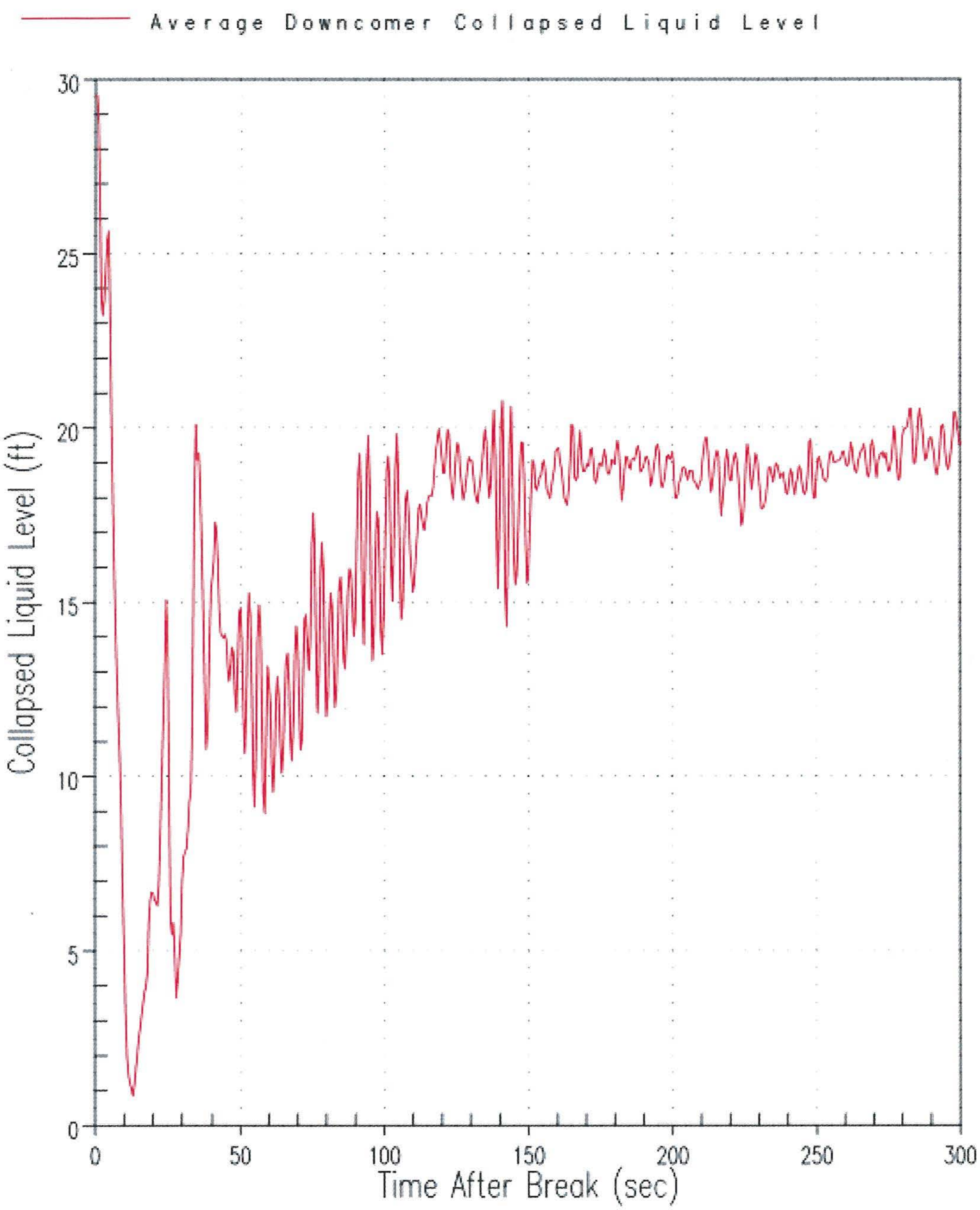

Figure 24 VCSNS Average Downcomer Collapsed Liquid Level (Relative to Bottom of Upper Tie Plate) for the Region II Analysis PCT Case

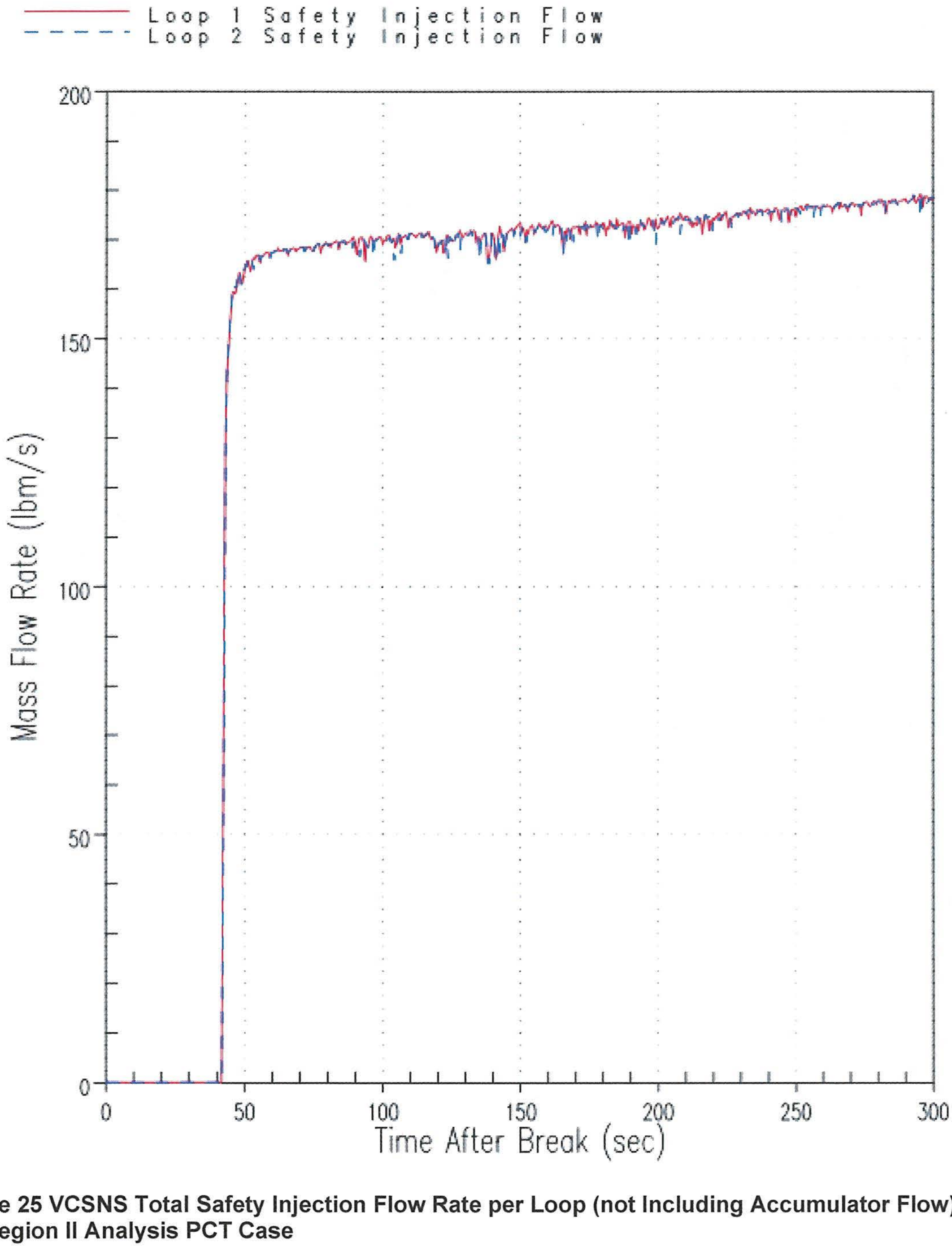

Figure 25 VCSNS Total Safety Injection Flow Rate per Loop (not Including Accumulator Flow) for the Region II Analysis PCT Case

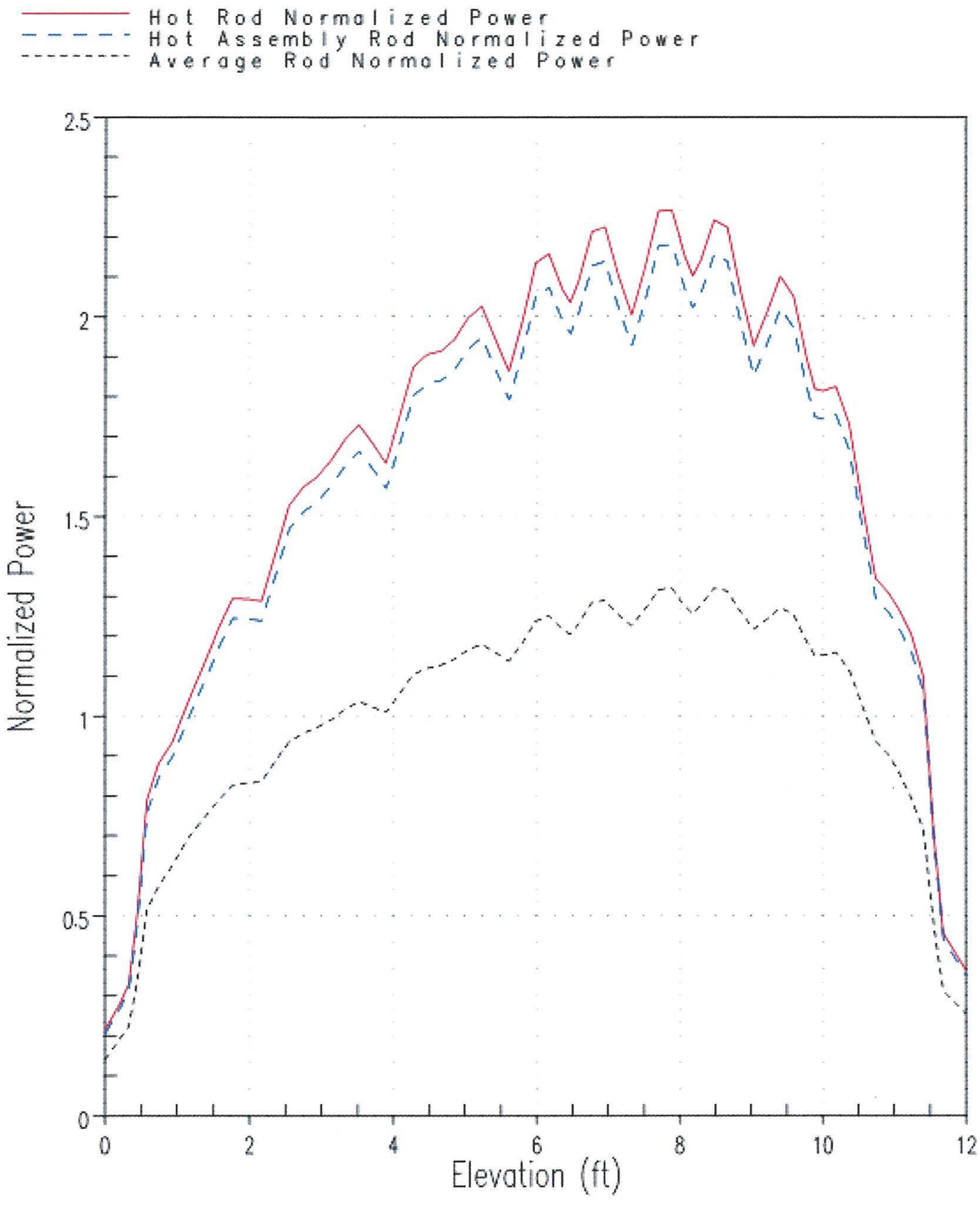

Figure 26 VCSNS Normalized Core Power Shapes for the Region II Analysis PCT Case

Note: The localized power decreases occur at grid elevations

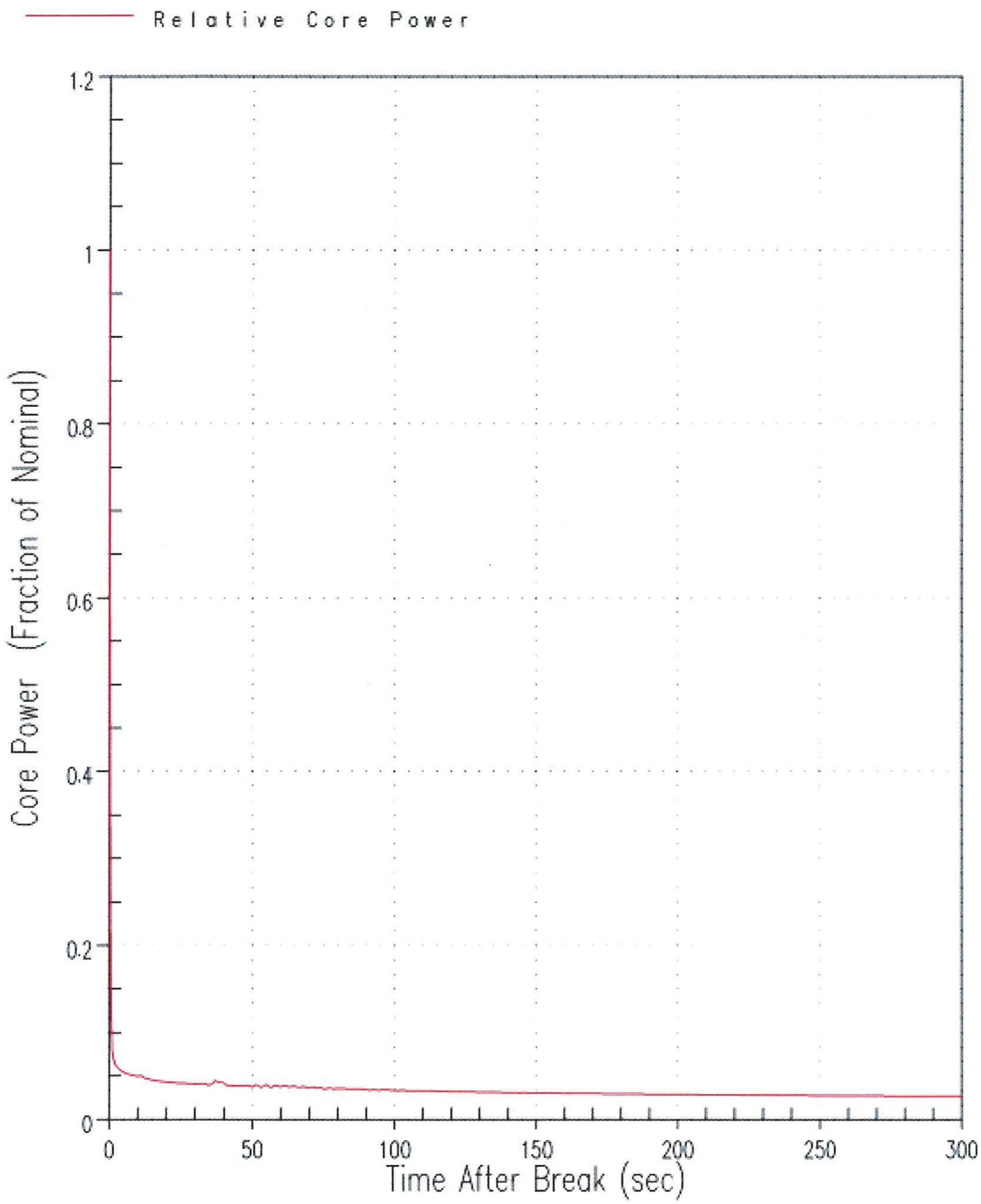

Figure 27 VCSNS Relative Core Power for the Region II Analysis PCT Case