Luminant

Steven K. Sewell
Senior Director,
Engineering & Regulatory Affairs

Comanche Peak Nuclear Power Plant (Vistra Operations Company LLC)
P.O. Box 1002
6322 North FM 56
Glen Rose, TX 76043

T 254.897.6113

CP-202200355
TXX-22078
November 21, 2022

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

Ref 10 CFR 50.90
10 CFR 50.91
10 CFR 2.390

Subject: Comanche Peak Nuclear Power Plant (CPNPP)
Docket Nos. 50-445 and 50-446
Application to Revise Technical Specifications to Apply the Westinghouse Full Spectrum Loss of Coolant Accident Evaluation Model. LAR 22-002.

Dear Sir or Madam:

Pursuant to 10 CFR 50.90, Vistra Operations Company LLC ("Vistra OpCo") is submitting a request for an amendment to the Technical Specifications (TS) for Comanche Peak, Units 1 and 2. The request seeks approval to apply the Westinghouse FULL SPECTRUM™ loss-of-coolant accident (FSLOCA™) evaluation model (EM) (FSLOCA EM), to revise TS 2.1.1.2 to reflect the peak fuel centerline melt temperature specified in the Westinghouse performance analysis and design model (PAD5), and to remove the allowance for zircalloy cladding from TS 4.2.1 for Comanche Peak Unit 1 and Unit 2.

The enclosure provides a description and assessment of the proposed changes. Attachment 1 provides the proposed TS changes. Attachment 2 provides the proposed TS Bases pages for information only. Attachment 3 provides a proprietary version of the technical evaluation. Attachment 4 provides a non-proprietary version of the technical evaluation. Attachment 5 provides the Westinghouse Application for Withholding Proprietary Information from Public Disclosure and accompanying Affidavit. As Attachment 3 contains information proprietary to Westinghouse Electric Company LLC ("Westinghouse"), it is respectfully requested that information which is proprietary to Westinghouse be withheld from public disclosure in accordance with 10 CFR 2.390. Attachment 6 provides supplemental information related to the gamma energy redistribution error.

Vistra OpCo requests that the amendment be reviewed as a normal license amendment request. Approval of the proposed amendment is requested within one year of completion of the NRC's acceptance review. Once approved, implementation of this amendment will be prior to Mode 4 entry for Unit 2 Cycle 22 (Fall 2024) and Unit 1 Cycle 25 (Spring 2025). CPNPP Unit 2 Cycle 22 is currently planned to begin in November 2024 and CPNPP Unit 1 Cycle 25 is currently planned to begin in May 2025.

There are no new regulatory commitments made in this submittal.

In accordance with 10 CFR 50.91, a copy of this application, with attachments, is being provided to the designated State of Texas Official.

Should you have any questions, please contact Nic Boehmisch at (254) 897-5064 or nicholas.boehmisch@luminant.com.

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

Executed on November 21, 2022.

Sincerely,

Steven K. Sewell

Enclosure: Description and Assessment

Attachments:
1. Proposed Technical Specification Changes
2. Proposed Technical Specification Bases Changes (information only)
3 Technical Evaluation WPT-18138 (Proprietary)
4. Technical Evaluation WPT-18138 (Non-Proprietary)
5. Application for Withholding Proprietary Information
6. Supplemental information and PCT summary

c (email) -
Scott Morris, Region IV [Scott.Morris@nrc.gov]
Dennis Galvin, NRR [Dennis.Galvin@nrc.gov]
John Ellegood, Senior Resident Inspector, CPNPP [John.Ellegood@nrc.gov]
Neil Day, Resident Inspector, CPNPP [Neil.Day@nrc.gov]

Mr. Robert Free [robert.free@dshs.state.tx.us]
Environmental Monitoring & Emergency Response Manager
Texas Department of State Health Services
Mail Code 1986
P. O. Box 149347
Austin TX, 78714-9347

Description and Assessment

1.0 SUMMARY DESCRIPTION

Vistra Operations Company LLC (Vistra OpCo) requests amendments to Comanche Peak Nuclear Power Plant Units 1 and 2 (CPNPP) Technical Specifications (TS) by revising the referenced Loss-of-Coolant Accident (LOCA) methodology to reflect the adoption of WCAP-16996-P-A, Revision 1, Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (FULL SPECTRUMTM LOCA Methodology), (FSLOCA).

The requested amendments also revise the TS reactor core safety limit (SL) to reflect the peak fuel centerline melt temperature specified in Topical Report (TR) WCAP-17642-P-A, Revision 1, Westinghouse Performance Analysis and Design Model (PAD5), and revise the TS reactor core fuel assemblies design feature by removing the discussion of Zircalloy fuel rods and ZIRLO lead test assemblies. These additional changes are consistent with the implementation of FSLOCA Methodology.

2.0 DETAILED DESCRIPTION

2.1 System Design and Operation

Comanche Peak Nuclear Power Plant Units 1 and 2 (CPNPP) must ensure acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and anticipated operational occurrences. This is accomplished by having a departure from nucleate boiling (DNB) design basis, which corresponds to a 95% probability at a 95% confidence level (the 95/95 DNB criterion) that DNB will not occur and by requiring that fuel centerline temperature stays below the melting temperature, as required by the reactor core Safety Limit (SL) Technical Specification (TS) 2.1.1.

The restrictions of the SL prevent overheating of the fuel and cladding, as well as possible cladding perforation, that would result in the release of fission products to the reactor coolant. Overheating of the fuel is prevented by maintaining the steady state peak linear heat rate (LHR) below the level at which fuel centerline melting occurs. Fuel centerline melting occurs when the local LHR, or power peaking, in a region of the fuel is high enough to cause the fuel centerline temperature to reach the melting point of the fuel. Expansion of the pellet upon centerline melting may cause the pellet to stress the cladding to the point of failure, allowing an uncontrolled release of activity to the reactor coolant. The proper functioning of the Reactor Protection System and steam generator safety valves prevent violation of the reactor core SLs.

The design requirements of existing plant systems, structures, and components (SSCs) are used as inputs to the CPNPP safety analysis, with appropriate technical conservatisms applied. Therefore, the analysis does not directly impact the existing design or configuration or operation of any plant SSCs.

2.2 Current Technical Specification Requirements

TS 2.1.1.2 Requires that in MODES 1 and 2, the peak fuel centerline temperature shall be maintained < 4700°F.

TS 2.2.1 requires that if SL 2.1.1 is violated, restore compliance and be in MODE 3 within 1 hour.

TS 4.2.1 specifies that each fuel assembly shall consist of a matrix of Zircaloy or ZIRLO clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide as fuel material. It allows a limited number of lead test assemblies that have not completed representative testing or that contain Westinghouse ZIRLO fuel rod cladding to be placed in non-limiting core regions.

TS 5.6.5.b lists approved analytical methods. The current approved analytical methods pertaining to the change are numbers 11 through 14, as listed in TS 5.6.5.b.

2.3 Reason for Proposed Change

In Reference 6.3, Vistra OpCo committed to submit to the NRC for review and approval, a license amendment request reflecting: (1) a Full-scope LOCA (FSLOCA) analysis which applies NRC-approved methods, and which will include the effects of fuel pellet thermal conductivity and, (2) a revised 'not-LOCA' analysis. Both the FSLOCA and 'not-LOCA' analysis utilize the updated Westinghouse PAD5 model. This LAR is being submitted to fulfill the commitment specified in Reference 6.3.

Plant-specific safety analysis are performed to ensure compliance with the Safety Limit is maintained. Westinghouse Topical Report WCAP-17642-P-A, Revision 1, "Westinghouse Performance Analysis and Design Model (PAD5)," Reference 6.2, defined the fuel pellet melting limit that is included in the PAD5 methodology based on available fuel pellet material properties. The NRC reviewed and approved the Westinghouse methodology and concluded that the melting limits defined in Reference 6.2 are acceptable. The proposed change will be implemented to maintain consistency between the value in SL 2.1.1.2 and the criteria used when performing confirmatory safety analyses that rely on the NRC approved methodology in Reference 6.2.

The proposed change removes discussion of Zircalloy fuel rods. The FSLOCA evaluation model analysis considered ZIRLO cladding. There is no Zircalloy fuel loaded in CPNPP. Insertion of Zircalloy fuel would require additional analysis and calculations and is not anticipated to occur. This change is consistent with the NRC approval of FSLOCA evaluation model methodology.

The allowance for a limited number of lead test assemblies to be placed in non-limiting core regions based on containing Westinghouse ZIRLO fuel cladding is removed. This allowance was added to TS to allow CPNPP to initially use eight ZIRLO cladding fuel lead test assemblies while transitioning to general usage of ZIRLO cladding fuel by NRC Safety Evaluation dated March 26, 2002, Reference 6.4. In a subsequent NRC Safety Evaluation dated September 4, 2002, Reference 6.5, the use of ZIRLO cladding fuel was allowed in general. All fuel loaded in CPNPP has ZIRLO cladding. As a result, the allowance specifically for ZIRLO cladding lead test assemblies is no longer required. This portion of the proposed change is administrative in nature.

2.4 Description of Proposed Change

TS 2.1.1.2 will be revised to state; “In MODES 1 and 2, the peak fuel centerline temperature shall be maintained less than 5080°F, decreasing by 9°F per 10,000 MWD/MTU of burnup.”

TS 4.2.1 for the Reactor Core Fuel Assemblies Design Features will be revised to state; “The reactor shall contain 193 fuel assemblies. Each assembly shall consist of a matrix of ZIRLO clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in non-limiting core regions.”

The approved analytical methods listings numbered 11 through 14 as listed in TS 5.6.5.b are deleted. A new approved analytical method “WCAP-16996-P-A, Revision 1, "Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (FULL SPECTRUM LOCA Methodology)," November 2016,” Reference 6.1, will be listed as number 11 and numbers 12 through 14 will state “Not used.”

3.0 TECHNICAL EVALUATION

The proposed change updates the listing of approved analytical methods in TS 5.6.5.b. These changes are administrative in nature. The proposed change reflects the adoption of WCAP-16996-P-A, Revision 1, Reference 6.1, demonstrating CPNPP compliance with the ECCS performance criterion of 10 CFR 50.46 subject to the NRC’s specified conditions and limitations. Attachment 3 to this enclosure "Application of Westinghouse Full Spectrum LOCA Evaluation Model to the Comanche Peak Units 1 and 2 Nuclear Plants," provides the technical evaluation for the application of the Westinghouse FSLOCA Evaluation Model (EM) to CPNPP. This evaluation was performed in accordance with the NRC approved FSLOCA EM in Reference 6.1. Supplemental information and updated PCT summary sheets related to the evaluation of the Gamma Energy Redistribution Error are included in Attachment 6.

The principal design tool used by Westinghouse for evaluating fuel rod performance is the Performance Analysis and Design Model (PAD5) code, Reference 6.2. This computer program iteratively calculates the interrelated effects of fuel and cladding deformations including fuel densification, fuel swelling, fuel relocation, fuel rod temperatures, fill and fission gas release, and rod internal pressure as a function of time and linear power. PAD5 evaluates the power history of a fuel rod as a series of steady-state power levels with instantaneous jumps from one power to another. The length of the fuel rod is divided into several axial segments, and each segment is assumed to operate at a constant set of conditions over its length. Fuel densification and swelling, cladding stresses and strains, temperatures, burn up and fission gas releases are calculated separately for each axial segment and the effects are integrated to obtain the

overall fission gas release and resulting internal pressure for each time step. The coolant temperature rise along the fuel rod is calculated based on the flow rate and axial power distribution, and the cladding surface temperature is determined with consideration of corrosion effects and the possibility of local boiling.

Model updates incorporated into the PAD5 code address the fuel and cladding performance models required for high burnup fuel design. Key fuel performance updates to the PAD5 models include fuel thermal conductivity degradation (TCD) with burnup, enhanced high burnup athermal fission gas release (pellet rim effects) and enhanced high burnup fission gas bubble swelling. Cladding creep and growth models are also updated to reflect high burnup cladding performance. In addition to high burnup analysis capability, a key driver for the implementation of the PAD5 models in fuel design is to address regulatory concerns associated with fuel thermal conductivity degradation with burnup.

The PAD5 models are the latest evolutions of the Westinghouse PAD code, Reference 6.2. As part of the Reference 6.2 development, the burnup-dependent term of the fuel melting limits in PAD5 was updated based on journal-published fuel material data. Additional validation performed in Section 2.1 of Appendix A of Reference 6.2 shows that the PAD5 code, in conjunction with the new fuel melt limit, accurately predicts fuel melt based on comparisons to experimental observations. Section 3.7.12 of the NRC Safety Evaluation Report in Reference 6.2 concludes that the fuel melting limits in PAD5 are acceptable.

The peak fuel centerline temperature SL is independent of the PAD5 methodology, as noted in the Safety Evaluation of the Amendment that reflected the SL revision for Turkey Point Nuclear Plant, Units 3 and 4, Reference 6.8. The current licensing basis safety analyses use the existing SL 2.1.1.2 for fuel melt as an acceptance criterion as required by the current methodology. Thus, Vistra OpCo will continue to meet the existing SL when using its current licensing basis safety analyses even with the implementation of the proposed SL. Since the existing SL for peak fuel centerline temperature is more restrictive than the proposed limit, the current licensing basis safety analyses remain conservative with respect to the proposed SL.

A comprehensive description of the PAD5 models, NRC Requests for Additional Information, and the subsequent NRC Safety Evaluation are documented in Reference 6.2. The NRC Safety Evaluation Limitations and Conditions are discussed in Section 3.1 of this amendment request. As described in Section 3.1, the proposed SL will only be applicable for analyses performed with the method described in Reference 6.2.

3.1 Limits of Applicability

The proposed amendment will only be used in applicable safety analyses that are performed with the approved fuel performance methods in Reference 6.2. The Limitations and Conditions from the NRC Safety Evaluation in Reference 6.2 pertinent to this amendment request are detailed below along with details of how each is satisfied.

- The NRC staff limits the applicability of the PAD5 code and methodology to the cladding, fuel, and reactor parameters listed in Section 4.1 of the Safety Evaluation in Reference 6.2.

  Response: Vistra OpCo will apply PAD5 within the limits specified in Section 4.1 of Reference 6.2 for cladding, fuel, and reactor parameters to be used at CPNPP. Because these PAD5 inputs depend on the reload design, these parameters are validated on a cycle-specific basis.

- The application of PAD5 should at no time exceed the fuel melting temperature as calculated by PAD5 due to the lack of properties for molten fuel in PAD5 and other properties such as thermal conductivity and fission gas release.

  Response: Vistra OpCo will limit the CPNPP peak fuel centerline temperature per this amendment request.

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements

10 CFR 50.36 requires that the TS include items in the following specific categories: (1) safety limits, limiting safety systems settings, and limiting control settings; (2) limiting conditions for operation (LCOs); (3) surveillance requirements per 10 CFR 50.36(c)(3); (4) design features; and (5) administrative controls. This change is related to the categories 1, 4, and 5 above since; a change to the peak fuel centerline melt temperature Safety Limit is proposed, a change to the Reactor Core Fuel Assemblies Design Feature is proposed, and a change to the Core Operating Limits Report, Reporting Requirements Administrative Controls is proposed.

10 CFR 50, Appendix A, GDC 10, “Reactor design,” requires that the reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences. The requirements of GDC 10 are met by the revision of TS 2.1.1.2 to be consistent with the limit approved in Reference 6.2.

The FSLOCA EM in WCAP-16996-P-A, Revision 1, Reference 6.1, satisfies the requirements of 10 CFR 50.46(b) paragraphs (1) through (4). Conformance with General Design Criterion 35, "Emergency Core Cooling," is described in more detail in Reference 6.1.

The proposed change meets the current regulatory requirements and does not affect conformance with the General Design Criteria as described in the Comanche Peak Units 1 and 2 UFSAR.

4.2 Precedent

The proposed change allows the use of the FSLOCA EM in WCAP-16996-P-A, Revision 1, Reference 6.1. Several previous similar requests have been approved by the NRC including Diablo Canyon, Surry, North Anna, Watts Barr, and Turkey Point, Reference 6.6.

The proposed change to TS Safety Limit 2.1.1.2 changes the fuel centerline temperature to reflect that specified in WCAP-17642-P-A, Reference 6.2. The NRC has approved changes to the fuel centerline melt temperature based on Reference 6.2 for several plants including Surry, North Anna, and Millstone, Reference 6.7.

4.3 No Significant Hazards Consideration Determination

The Proposed amendment to Technical Specification (TS) 5.6.5.b adds Westinghouse Topical Report WCAP-16996-P-A, Revision 1, to the listing of approved analytical methods in TS 5.6.5.b. and removes obsolete methods. The proposed amendment to TS 2.1.1.2 changes the fuel centerline temperature to reflect that specified in WCAP-17642-P-A, Revision 1. The proposed amendment to TS 4.2.1 removes the use of zircalloy cladding, and the specification of cladding material for lead test fuel assemblies.

Vistra OpCo has evaluated whether a significant hazards consideration is involved with the proposed amendments 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 proposed amendment neither alters plant equipment nor the manner in which equipment is operated and maintained, and thereby cannot increase the probability of an accident. The proposed amendment cannot adversely affect the type or amount of effluent that may be released off-site or increase individual or cumulative occupational exposures resulting from any design basis accident, and thereby cannot increase the consequences of any accident.

Therefore, the proposed amendment does 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 amendment neither installs new nor modifies existing plant equipment and thereby cannot introduce new equipment failure modes. The

proposed amendment does not alter safety analysis assumptions, or create new accident initiators or precursors, and thereby cannot introduce a new or different type of accident.

Therefore, the proposed amendment does 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.

The revised Safety Limit 2.1.1.2 has been calculated based on the NRC approved methods, which ensure that the plant operates in compliance with applicable regulatory criteria. The proposed amendment has been generically reviewed by the NRC and found to be appropriately conservative with respect to the fuel material properties in Topical Report WCAP-17642-P-A, Revision 1. The Limitations and Conditions from the NRC Safety Evaluation of the Topical Report pertinent to this amendment request have been satisfied for CPNPP 1 and 2.

The proposed amendment does not modify any, limiting safety system settings, or safety analysis assumptions. The proposed amendment does not modify equipment credited in safety analyses, and thereby cannot affect the integrity of any radiological barrier.

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

Based on the above evaluations, Vistra OpCo concludes that the propose amendment presents 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

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 CONSIDERATIONS

Vistra OpCo has determined that the proposed amendment would change requirements with respect to the installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amount of effluent that may be released offsite, or (iii) a significant increase in the

individual or cumulative occupational radiation exposure. Accordingly, the proposed change meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), an environmental assessment of the proposed change is not required.

6.0 REFERENCES

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

6.2 "Westinghouse Performance Analysis and Design Model (PAD5)," WCAP-17642-P-A, Revision 1, November 2017.

6.3 Letter from T. McCool (Vistra), to the NRC, dated August 14, 2019, "ECCS Reanalysis Schedule," CP-201900369, [ML19248B765]

6.4 Letter from NRC to (Luminant), March 26, 2002, "COMANCHE PEAK STEAM ELECTRIC STATION (CPSES), UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS RE: USE OF ZIRLOTM LEAD TEST ASSEMBLIES (TAC NOS. MB3446 AND MB3447)," [ML020660756]

6.5 Letter from NRC to C. Terry (Luminant), September 4, 2002, "COMANCHE PEAK STEAM ELECTRIC STATION (CPSES), UNITS 1 AND 2 -ISSUANCE OF AMENDMENTS RE: USE OF ZIRLOTM FUEL (TAC NOS. MB3101, MB3102, MB4740, AND MB4741)," [ML022180303]

6.6 Letter from NRC to B. Coffey (Florida Power & Light Company), May 24, 2022, "TURKEY POINT NUCLEAR GENERATING UNIT NOS. 3 AND 4 – ISSUANCE OF AMENDMENT NOS. 296 AND 289 REGARDING IMPLEMENTATION OF FULL SPECTRUM LOSS-OF-COOLANT ACCIDENT (FSLOCA) METHODOLOGY (EPID L-2021-LLA-0070)," [ML22028A066]

6.7 Letter from NRC to D. Stoddard (Dominion), January 7, 2022, "MILLSTONE POWER STATION, UNIT NO. 3 ISSUANCE OF AMENDMENT NO. 281 RE: REVISED REACTOR CORE SAFETY LIMIT TO REFLECT TOPICAL REPORT WCAP-17642-P-A, REVISION 1 (EPID L-2020-LLA-0266)," [ML21326A099]

6.8 Letter from NRC to M. Nazar (Florida Power & Light Company), August 15, 2019, "Turkey Point Generating Unit Nos. 3 and 4 - Issuance of Amendment Nos. 288 and 282 Regarding Revised Reactor Core Safety Limit to Reflect Topical Report WCAP- 17642-P-A, Revision 1 (EPID L-2018-LLA-0120)," [ML19031C891]

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 In MODES 1 and 2, the departure from nucleate boiling ratio (DNBR) shall be maintained ≥ the 95/95 DNB criterion for the DNB correlation(s) specified in Section 5.6.5.

2.1.1.2 In MODES 1 and 2, the peak fuel centerline temperature shall be maintained ~~< 4700°F~~.

2.1.2 RCS Pressure SL

In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be maintained ≤ 2735 psig.

2.2 SL Violations

2.2.1 If SL 2.1.1 is violated, restore compliance and be in MODE 3 within 1 hour.

2.2.2 If SL 2.1.2 is violated:

2.2.2.1 In MODE 1 or 2, restore compliance and be in MODE 3 within 1 hour.

2.2.2.2 In MODE 3, 4, or 5, restore compliance within 5 minutes.

less than 5080°F, decreasing by 9°F per 10,000 MWD/MTU of burnup.

4.0 DESIGN FEATURES

4.1 Site Location

The site area is approximately 7,700 acres located in Somervell County in North Central Texas. Squaw Creek Reservoir (SCR), established for station cooling, extends into Hood County. The site is situated along Squaw Creek, a tributary of the Paluxy River, which is a tributary of the Brazos River. The site is over 30 miles southwest of the nearest portion of Fort Worth and approximately 4.5 miles north-northwest of Glen Rose, the nearest community.

4.2 Reactor Core

4.2.1 Fuel Assemblies

The reactor shall contain 193 fuel assemblies. Each assembly shall consist of a matrix of ~~Zircaloy or~~ ZIRLO~~™~~ clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing ~~or that contain Westinghouse ZIRLO™ fuel rod cladding~~ may be placed in non-limiting core regions.

4.2.2 Control Rod Assemblies

The reactor core shall contain 53 control rod assemblies. The control material shall be silver-indium-cadmium as approved by the NRC.

5.6 Reporting Requirements

5.6.5 Core Operating Limits Report (COLR) (continued)

~~11. WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," August 1985.~~

~~12. WCAP-10054-P-A, Addendum 2, Revision 1, "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation Model," July 1997.~~

12. Not used.

~~13. WCAP-10079-P-A, "NOTRUMP, A Nodal Transient Small Break and General Network Code," August 1985.~~

13. Not used.

~~14. WCAP-16009-P-A, "Realistic Large-Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)," January 2005.~~

14. Not used.

15. WCAP-12472-P-A, "BEACON Core Monitoring and Operations Support System," August 1994.

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

a. RCS pressure and temperature limits for heat up, cooldown, low temperature operation, criticality, and hydrostatic testing, and PORV lift settings as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:

1. Specification 3.4.3, "RCS Pressure and Temperature (P/T) Limits," and

2. Specification 3.4.12, "Low Temperature Overpressure Protection (LTOP) System."

b. The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

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

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.1 Heat Flux Hot Channel Factor ($F_Q(Z)$) (RAOC-W(Z) Methodology)

BASES

BACKGROUND

The purpose of the limits on the values of $F_Q(Z)$ is to limit the local (i.e., pellet) peak power density. The value of $F_Q(Z)$ varies along the axial height (Z) of the core.

$F_Q(Z)$ is defined as the maximum local fuel rod linear power density divided by the average fuel rod linear power density, assuming nominal fuel pellet and fuel rod dimensions. Therefore, $F_Q(Z)$ is a measure of the peak fuel pellet power within the reactor core.

During power operation, the global power distribution is limited by LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," and LCO 3.2.4, "QUADRANT TILT POWER RATIO (QPTR)," which are directly and continuously measured process variables. These LCOs, along with LCO 3.1.7, "Control Bank Insertion Limits," maintain the core limits on power distributions on a continuous basis.

$F_Q(Z)$ varies with fuel loading patterns, control bank insertion, fuel burnup, and changes in axial power distribution.

$F_Q(Z)$ is measured periodically using the incore detector system or an OPERABLE PDMS. These measurements are generally taken with the core at or near equilibrium conditions.

Using the measured three dimensional power distributions, it is possible to derive a measured value for $F_Q(Z)$. However, because this value represents an equilibrium condition, it does not include the variations in the value of $F_Q(Z)$ that are present during non-equilibrium situations, such as load following. To account for these possible variations, the steady state value of $F_Q(Z)$ is adjusted by an elevation dependent factor, W(Z), that accounts for calculated worse case transient conditions.

Core monitoring and control under non-steady state conditions are accomplished by operating the core within the limits of the appropriate LCOs, including the limits on AFD, QPTR, and control rod insertion.

APPLICABLE SAFETY ANALYSES

This LCO's principal effect is to preclude core power distributions that could lead to violation of the following fuel design criterion:

a. During a ~~large break~~ loss of coolant accident (LOCA), the ~~peak cladding temperature must not exceed 2200°F~~ (Ref. 1); and

(continued)

10 CFR 50.46 acceptance criteria must be met

BASES

APPLICABLE SAFETY ANALYSES (continued)

b. During an ejected rod accident, the energy deposition to the fuel must not exceed 280 cal/gm, and

c. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn.

Limits on $F_Q(Z)$ ensure that the value of the initial total peaking factor assumed in the accident analyses remains valid. ~~Other criteria must also be met (e.g., maximum cladding oxidation, maximum hydrogen generation, coolable geometry, and long term cooling). However, the LOCA peak cladding temperature is typically most limiting.~~

$F_Q(Z)$ limits assumed in the LOCA analysis are typically limiting relative to (i.e., lower than) the $F_Q(Z)$ limit assumed in safety analyses for other postulated accidents. Therefore, this LCO provides conservative limits for other postulated accidents.

$F_Q(Z)$ satisfies Criterion 2 of the 10 CFR 50.36(c)(2)(ii).

LCO

The Heat Flux Hot Channel Factor, $F_Q(Z)$, shall be limited by the following relationships:

$$F_Q(Z) \le \frac{F_Q^C}{P} K(Z) \qquad \text{for } P > 0.5$$

$$F_Q(Z) \le \frac{F_Q^C}{0.5} K(Z) \qquad \text{for } P \le 0.5$$

where:

F_Q^C is the $F_Q(Z)$ limit at RTP provided in the COLR,

$K(Z)$ is the normalized $F_Q(Z)$ as a function of core height provided in the COLR, and

P = THERMAL POWER/RTP

The actual values of F_Q^C and K(Z) are given in the COLR.

(continued)

ensure that the 10 CFR 50.46 acceptance criteria are met

BASES

LCO (continued)

The $F_Q(Z)$ limits define limiting values for core power peaking that ~~precludes peak cladding temperatures above 2200°F~~ during either a large or small break LOCA.

This LCO requires operation within the bounds assumed in the safety analyses. Calculations are performed in the core design process to confirm that the core can be controlled in such a manner during operation that it can stay within the LOCA $F_Q(Z)$ limits. If $F_Q^C(Z)$ cannot be maintained within the LCO limits, a reduction of the core power is required and if $F_Q^W(Z)$ cannot be maintained within the LCO limits, reduction of the AFD limits is required. Note that sufficient reduction of the AFD limits will also result in a reduction of the core power.

Violating the LCO limits of $F_Q(Z)$ may produce unacceptable consequences if a design basis event occurs while $F_Q(Z)$ is outside its specified limits.

APPLICABILITY

The $F_Q(Z)$ limits must be maintained in MODE 1 to prevent core power distributions from exceeding the limits assumed in the safety analyses. Applicability in other MODES is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the reactor coolant to require a limit on the distribution of core power.

ACTIONS

A.1

Reducing THERMAL POWER by ≥ 1% RTP for each 1% by which $F_Q^C(Z)$ exceeds its limit, maintains an acceptable absolute power density. $F_Q^C(Z)$ is $F_Q^M(Z)$ multiplied by factors that account for manufacturing tolerances and measurement uncertainties. $F_Q^M(Z)$ is the measured value of $F_Q(Z)$. The Completion Time of 15 minutes provides an acceptable time to reduce power in an orderly manner and without allowing the plant to remain in an unacceptable condition for an extended period of time. The maximum allowable power level initially determined by Required Action A.1 may be affected by subsequent determinations of $F_Q^C(Z)$ and would require power reductions within 15 minutes of the $F_Q^C(Z)$ determination, if necessary to comply with the decreased maximum allowable power level. Decreases in $F_Q^C(Z)$ would allow increasing the maximum allowable power level and increasing power up to this revised limit.

(continued)

BASES

BACKGROUND (continued)

no overheating of the fuel that results in possible cladding perforation with the release of fission products to the reactor coolant.

APPLICABLE SAFETY ANALYSES

Limits on $F^N_{\Delta H}$ preclude core power distributions that exceed the following fuel design limits:

a. For ANS Condition II events, there must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hottest fuel rod in the core does not experience a DNB condition;

the 10 CFR 50.46 acceptance criteria must be met (Ref. 3)

b. During a ~~large break~~ loss of coolant accident (LOCA), ~~peak cladding temperature (PCT) must not exceed 2200°F~~;

c. During an ejected rod accident, the average fuel pellet enthalpy at the hot spot must not exceed 280 cal/gm (Ref. 1); and

d. Fuel design limits required by GDC 26 (Ref. 2) for the condition when control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn.

The limits on $F^N_{\Delta H}$ ensure that the DNB design basis is met for normal operation, operational transients, and any transients arising from events of moderate frequency. The DNB design basis is met by limiting the minimum DNBR to the 95/95 DNB criterion applicable to a specific DNBR correlation. This value provides a high degree of assurance that the hottest fuel rod in the core does not experience a DNB condition.

The allowable $F^N_{\Delta H}$ limit increases with decreasing power level. This functionality in $F^N_{\Delta H}$ is included in the analyses that provide the Reactor Core Safety Limits (SLs) of SL 2.1.1. Therefore, any DNB events in which the calculation of the core limits is modeled implicitly use this variable value of $F^N_{\Delta H}$ in the analyses. Likewise, all transients that may be DNB limited are assumed to begin with an initial $F^N_{\Delta H}$ as a function of power level defined by the COLR limit equation.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

~~The LOCA safety analysis also uses $F^N_{\Delta H}$ as an input parameter. The Nuclear Heat Flux Hot Channel Factor ($F_Q(Z)$) and the axial peaking factors are inserted directly into the LOCA safety analyses that verify the acceptability of the resulting peak cladding temperature (Ref. 3).~~

The Nuclear Enthalpy Rise Hot Channel Factor ($F_{\Delta H}{}^N$), the Nuclear Heat Flux Hot Channel Factor ($F_Q(Z)$) and the axial peaking factors are supported by the LOCA safety analyses that verify compliance with the 10 CFR 50.46 acceptance criteria (Ref. 3).

The fuel is protected in part by compliance with Technical Specifications which ensure that the initial conditions assumed in the safety and accident analyses remain valid. The following LCOs ensure this: LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," LCO 3.1.7, "Control Bank Insertion Limits," LCO 3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)," and LCO 3.2.1, "Heat Flux Hot Channel Factor ($F_Q(Z)$)."

$F^N_{\Delta H}$ and $F_Q(Z)$ are measured periodically using the movable incore detector system or an OPERABLE PDMS. Measurements are generally taken with the core at, or near, equilibrium conditions. Core monitoring and control under transient conditions (Condition 1 events) are accomplished by operating the core within the limits of the LCOs on AFD, QPTR, and Bank Insertion Limits.

$F^N_{\Delta H}$ satisfies Criterion 2 of 10CFR50.36(c)(2)(ii).

LCO

$F^N_{\Delta H}$ shall be maintained within the limits of the relationship provided in the COLR.

The $F^N_{\Delta H}$ limit is representative of the coolant flow channel with the maximum enthalpy rise. This channel has the least heat removal capability and thus the highest probability for a DNB condition.

The limiting value of $F^N_{\Delta H}$ described by the equation contained in the COLR, is the design radial peaking factor used in the unit safety analyses.

A power multiplication factor in this equation includes an additional allowance for higher radial peaking factors from reduced thermal feedback and greater control rod insertion at low power levels. The limiting value of $F^N_{\Delta H}$ is allowed to increase by a cycle-dependent factor ($PF_{\Delta H}$, as specified

(continued)

BASES

LCO (continued)

in the COLR) for a 1% RTP reduction in THERMAL POWER.

If the power distribution measurements are performed at a power level less than 100% RTP, then the $F^N_{\Delta H}$ values that would result from measurements if the core was at 100% RTP should be inferred from the available information. A comparison of these inferred values with $F_{\Delta H}^{RTP}$ assures compliance with the LCO at all power levels.

APPLICABILITY

The $F^N_{\Delta H}$ limits must be maintained in MODE 1 to preclude core power distributions from exceeding the fuel design limits for DNBR and ~~PCT.~~ the 10 CFR 50.46 acceptance criteria (Ref. 3). Applicability in other modes is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the coolant to require a limit on the distribution of core power.

ACTIONS

A.1.1

With $F^N_{\Delta H}$ exceeding its limit, the unit is allowed 4 hours to restore $F^N_{\Delta H}$ to within its limits. This restoration may, for example, involve realigning any misaligned rods or reducing power enough to bring $F^N_{\Delta H}$ within its power dependent limit. When the $F^N_{\Delta H}$ limit is exceeded, the DNBR limit is not likely violated in steady state operation, because events that could significantly perturb the $F^N_{\Delta H}$ value (e.g., static control rod misalignment) are considered in the safety analyses. However, the DNBR limit may be violated if a DNB limiting event occurs. Thus, the allowed Completion Time of 4 hours provides an acceptable time to restore $F^N_{\Delta H}$ to within its limits without allowing the plant to remain in an unacceptable condition for an extended period of time. The restoration of the peaking factor to within its limits by power reduction or control rod movement does not restore compliance with the LCO. Thus, this condition can not be exited until a valid surveillance demonstrates compliance with the LCO.

Condition A is modified by a Note that requires that Required Actions A.2 and A.3 must be completed whenever Condition A is entered. Thus, even if this Required Action is completed within the 4 hour time period, Required

(continued)

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.4 QUADRANT POWER TILT RATIO (QPTR)

BASES

BACKGROUND

The QPTR limit ensures that the gross radial power distribution remains consistent with the design values used in the safety analyses. Precise radial power distribution measurements are made during startup testing, after refueling, and periodically during power operation.

The power density at any point in the core must be limited so that the fuel design criteria are maintained. Together, LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," LCO 3.2.4, and LCO 3.1.7, "Control Rod Insertion Limits," provide limits on process variables that characterize and control the three dimensional power distribution of the reactor core. Control of these variables ensures that the core operates within the fuel design criteria and that the power distribution remains within the bounds used in the safety analyses.

APPLICABLE SAFETY ANALYSES

This LCO precludes core power distributions that violate the following fuel design criteria:

10 CFR 50.46 acceptance criteria must be met

a. During a ~~large break~~ loss of coolant accident, the ~~peak cladding temperature must not exceed 2200°F~~ (Ref. 1);

b. During a loss of forced reactor coolant flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 departure from nucleate boiling (DNB) criterion) that the hot fuel rod in the core does not experience a DNB condition;

c. During an ejected rod accident, the average fuel pellet enthalpy at the hot spot must not exceed 280 cal/gm (Ref. 2); and

d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 3).

The LCO limits on the AFD, the QPTR, the Heat Flux Hot Channel ($F_Q(Z)$), the Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$), and control bank insertion are established to preclude core power distributions that exceed the safety analyses limits.

The QPTR limits ensure that $F^N_{\Delta H}$ and $F_Q(Z)$ remain below their limiting

(continued)

BASES

BACKGROUND (continued)

The bistable outputs from the signal processing equipment are sensed by the SSPS equipment and combined into logic matrices that represent combinations indicative of various transients. If a required logic matrix combination is completed, the system will send actuation signals via master and slave relays to those components whose aggregate Function best serves to alleviate the condition and restore the unit to a safe condition. Examples are given in the Applicable Safety Analyses, LCO, and Applicability sections of this Bases.

Each SSPS train has a built in testing device that can automatically test the decision logic matrix functions and the actuation devices while the unit is at power. When any one train is taken out of service for testing, the other train is capable of providing unit monitoring and protection until the testing has been completed. The testing device is semiautomatic to minimize testing time.

The actuation of ESF components is accomplished through master and slave relays. The SSPS energizes the master relays appropriate for the condition of the unit. Each master relay then energizes one or more slave relays, which then cause actuation of the end devices. The master and slave relays are routinely tested to ensure operation. The test of the master relays energizes the relay, which then operates the contacts and applies a low voltage to the associated slave relays. The low voltage is not sufficient to actuate the slave relays but only demonstrates signal path continuity. The SLAVE RELAY TEST actuates the devices if their operation will not interfere with continued unit operation. For the latter case, actual component operation is prevented by the SLAVE RELAY TEST circuit, and slave relay contact operation is verified by a continuity check of the circuit containing the slave relay.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

Each of the analyzed accidents can be detected by one or more ESFAS Functions. One of the ESFAS Functions is the primary actuation signal for that accident. An ESFAS Function may be the primary actuation signal for more than one type of accident. An ESFAS Function may also be a secondary, or backup, actuation signal for one or more other accidents. For example, Pressurizer Pressure-Low is a primary actuation signal for ~~small~~ loss of coolant accidents (LOCAs) and a backup actuation signal for steam line breaks (SLBs) outside containment. Functions such as manual initiation, not specifically credited in the accident safety analysis, are qualitatively credited. These Functions may provide protection for conditions that do not require dynamic transient analysis to demonstrate Function performance. These Functions may also serve as backups to Functions that

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

1. Safety Injection

Safety Injection (SI) provides two primary functions:

compliance with the 10 CFR 50.46 acceptance criteria (Ref. 14)

1. Primary side water addition to ensure maintenance or recovery of reactor vessel water level (e.g., coverage of the active fuel for heat removal, clad integrity, and ~~for limiting peak clad temperature to < 2200°F)~~; and

2. Boration to ensure recovery and maintenance of SDM (k_{eff} < 1.0).

These functions are necessary to mitigate the effects of certain high energy line breaks (HELBs) both inside and outside of containment as described in the FSAR [Ref. 3]. The SI signal is also used to initiate other Functions such as:

- Phase A Isolation;
- Containment Ventilation Isolation;
- Reactor Trip;
- Turbine Trip;
- Feedwater Isolation;
- Start of motor driven auxiliary feedwater (AFW) pumps;
- Enabling semi-automatic switchover of Emergency Core Cooling Systems (ECCS) suction to containment sumps, coincident with RWST low-low level.
- Emergency DG start;
- Start of station service water pumps;
- Start of component cooling water pumps;
- Start of Containment Spray Pumps; and
- Start of essential ventilation systems.

(continued)

BASES (continued)

REFERENCES

1. FSAR, Chapter 6.

2. FSAR, Chapter 7.

3. FSAR, Chapter 15.

4. IEEE-279-1971.

5. 10 CFR 50.49.

6. WCAP-10271-P-A, Supplement 2, Rev. 1, June 1990.

7. Technical Requirements Manual.

8. WCAP-10271-P-A, Supplement 3, September 1990.

9. "Westinghouse Setpoint Methodology for Protection Systems Comanche Peak Unit 1, Revision 1," WCAP-12123, Revision 2, April, 1989.

10. WCAP-13877-P-A, Revision 2, August 2000.

11. “Elimination of Periodic Protection Channel Response Time Tests”, WCAP-14036-P-A, Revision 1, October 6, 1998.

12. “Probabilistic Risk Analysis of the RPS and ESFAS Test Times and Completion Times,” WCAP-14333-P-A, Revision 1, October 1998.

13. “Risk-Informed Assessment of the RTS and ESFAS Surveillance Test Intervals and Reactor Trip Breaker Test and Completion Times,” WCAP-15376-P-A, Revision 1, March 2003.

14. 10 CFR 50.46.

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.1 Accumulators

large break

BASES

BACKGROUND

The functions of the ECCS accumulators are to supply water to the reactor vessel during the blowdown phase of a loss of coolant accident (LOCA), to provide inventory to help accomplish the refill phase that follows thereafter, and to provide Reactor Coolant System (RCS) makeup for a small break LOCA.

The blowdown phase of a large break LOCA is the initial period of the transient during which the RCS departs from equilibrium conditions, and heat from fission product decay, hot internals, and the vessel continues to be transferred to the reactor coolant. The blowdown phase of the transient ends when the RCS pressure falls to a value approaching that of the containment atmosphere.

large break

In the refill phase of a LOCA, which immediately follows the blowdown phase, reactor coolant inventory has vacated the core through steam flashing and ejection out through the break. The core is essentially in adiabatic heatup. The balance of accumulator inventory is then available to help fill voids in the lower plenum and reactor vessel downcomer so as to establish a recovery level at the bottom of the core and ongoing reflood of the core with the addition of safety injection (SI) water.

As the break size decreases, the role of the accumulators continues to decrease until they are not required for terminating the temperature increase during the recovery phase of a small break LOCA.

The accumulators are pressure vessels partially filled with borated water and pressurized with nitrogen gas. The accumulators are passive components, since no operator or control actions are required in order for them to perform their function. Internal accumulator tank pressure is sufficient to discharge the accumulator contents to the RCS, if RCS pressure decreases below the accumulator pressure.

Each accumulator is piped into an RCS cold leg via an accumulator line and is isolated from the RCS by a motor operated isolation valve and two check valves in series.

The motor operated isolation valves are required to be open with power removed in MODE 3 above 1000 psig to satisfy BTP ICSB-18 [Ref. 1] for small break LOCAs. They are required to be open with power removed in MODES 1 and 2 for large break LOCA.

The accumulator size, water volume, and nitrogen cover pressure are selected so that three of the four accumulators are sufficient to partially cover the core before significant clad melting or zirconium water reaction can occur

(continued)

BASES

BACKGROUND (continued)

following a LOCA. The need to ensure that three accumulators are adequate for this function is consistent with the LOCA assumption that the entire contents of one accumulator will be lost via the RCS pipe break during the blowdown phase of the LOCA.

APPLICABLE SAFETY ANALYSES

The accumulators are assumed OPERABLE in both the large and small break LOCA analyses at full power (Ref. 2). These are the Design Basis Accidents (DBAs) that establish the acceptance limits for the accumulators. Reference to the analyses for these DBAs is used to assess changes in the accumulators as they relate to the acceptance limits.

large break

In performing the LOCA calculations, conservative assumptions are made concerning the availability of ECCS flow. In the early stages of a LOCA, with or without a loss of offsite power, the accumulators provide the sole source of makeup water to the RCS. The assumption of loss of offsite power is required by regulations and conservatively imposes a delay wherein the ECCS pumps cannot deliver flow until the emergency diesel generators start, come to rated speed, and go through their timed loading sequence. In cold leg break scenarios, the entire contents of one accumulator are assumed to be lost through the break.

The largest break area considered for a large break LOCA is a double ended guillotine break in the RCS cold leg.

~~The limiting large break LOCA is a double ended guillotine break at the discharge of the reactor coolant pump.~~ During this event, the accumulators discharge to the RCS as soon as RCS pressure decreases to below accumulator pressure.

(for loss of offsite power assumption)

for a large break LOCA

As a conservative estimate, no credit is taken for ECCS pump flow until an effective delay has elapsed. This delay accounts for the diesels starting and the pumps being loaded and delivering full flow. The delay time is conservatively set with an additional 2 seconds to account for SI signal generation. During this time, the accumulators are analyzed as providing the sole source of emergency core cooling. No operator action is assumed ~~during the blowdown stage~~ of a large break LOCA.

in the modeling

intermediate

The small break LOCA analyses

~~The worst case small break LOCA analyses~~ also assume a time delay before pumped flow ~~reaches the core~~. For ~~the larger range of small~~ breaks, the rate of blowdown is such that the increase in fuel clad temperature is terminated primarily by the accumulators, with pumped flow then providing continued cooling. As break size decreases, the accumulators and centrifugal charging pumps both play a part in terminating the rise in clad temperature. As break size continues to decrease, the role of the accumulators continues to decrease until they are not required and the centrifugal charging pumps become solely responsible for terminating the temperature increase.

is assumed to inject into the reactor coolant system

(continued)

There is a high level of probability that the peak cladding temperature does not exceed 2200ºF;

BASES

APPLICABLE SAFETY ANALYSES (continued)

There is a high level of probability that the maximum cladding oxidation does not exceed 0.17

This LCO helps to ensure that the following acceptance criteria established for the ECCS by 10 CFR 50.46 (Ref. 3) will be met following a LOCA:

a. ~~Maximum fuel element cladding temperature is ≤ 2200°F;~~

b. ~~Maximum cladding oxidation is ≤ 0.17~~ times the total cladding thickness before oxidation;

There is a high level of probability that the maximum

c. ~~Maximum~~ hydrogen generation from a zirconium water reaction ~~is ≤ 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; and

does not exceed 0.01

d. Core is maintained in a coolable geometry.

large break LOCA and the recovery phase of a small break LOCA

Since the accumulators discharge during the blowdown phase of a ~~LOCA~~, they do not contribute to the long term cooling requirements of 10 CFR 50.46.

The large and small break LOCA analyses use a range of accumulator water volumes of 6119 gallons to 6597 gallons (which account for uncertainties) per approved methods (Ref. 7). The contained water volume is the same as the deliverable volume for the accumulators, since the accumulators are emptied, once discharged. Both large and small break analyses use a nominal accumulator line water volume from the accumulator to the check valve.

~~For both the large and small break LOCA analyses, a nominal contained accumulator water volume is used. The contained water volume is the same as the deliverable volume for the accumulators, since the accumulators are emptied, once discharged. For small breaks, an increase in water volume may be either a peak clad temperature penalty or benefit depending on the transient characteristics. Depending on the NRC-approved methodology used to analyze large breaks, an increase in water volume may be either a peak clad temperature penalty or benefit, depending on downcomer filling and subsequent spill through the break during the core reflooding portion of the transient. The analysis makes a conservative assumption with respect to ignoring or taking credit for line water volume from the accumulator to the check valve. The safety analysis assumes values of 6119 gallons and 6597 gallons.~~

The minimum boron concentration setpoint is used in the post LOCA boron concentration calculation. The calculation is performed to assure reactor subcriticality in a post LOCA environment. Of particular interest is the large break LOCA, since no credit is taken for control rod assembly insertion. A reduction in the accumulator minimum boron concentration would produce a subsequent reduction in the available containment sump concentration for post LOCA shutdown and an increase in the maximum sump pH. The maximum boron concentration is used in determining the cold leg to hot leg recirculation injection switchover time and minimum sump pH.

(continued)

use a range of accumulator nitrogen cover pressures of 603 psia to 694 psia per approved methods (Ref. 7).

APPLICABLE SAFETY ANALYSES (continued)

The large and small break LOCA analyses ~~are performed at the minimum nitrogen cover pressure (603 psia), since sensitivity analyses have demonstrated that higher nitrogen cover pressure results in a computed peak clad temperature benefit.~~ The maximum nitrogen cover pressure limit ~~(693 psia)~~ prevents accumulator relief valve actuation, and ultimately preserves accumulator integrity.

To allow for instrument inaccuracy, control room indicated values of 623 psig and 644 psig are specified and used in surveillance.

The effects on containment mass and energy releases from the accumulators are accounted for in the appropriate analyses (Refs. 2 and 4).

The accumulators satisfy Criteria 2 and 3 of 10CFR50.36(c)(2)(ii).

LCO

The LCO establishes the minimum conditions required to ensure that the accumulators are available to accomplish their core cooling safety function following a LOCA. Four accumulators are required to ensure that 100% of the contents of three of the accumulators will reach the core during a LOCA. This is consistent with the assumption that the contents of one accumulator spill through the break. If less than three accumulators are injected during the blowdown phase of a large break LOCA, the ECCS acceptance criteria of 10 CFR 50.46 (Ref. 3) could be violated.

For an accumulator to be considered OPERABLE, the isolation valve must be fully open, power removed above a nominal RCS pressure of 1000 psig, and the limits established in the SRs for contained volume, boron concentration, and nitrogen cover pressure must be met.

APPLICABILITY

In MODES 1 and 2, and in MODE 3 with RCS pressure > 1000 psig, the accumulator OPERABILITY requirements are based on full power operation. Although cooling requirements decrease as power decreases, the accumulators are still required to provide core cooling as long as elevated RCS pressures and temperatures exist.

This LCO is only applicable at pressures > 1000 psig. At pressures ≤ 1000 psig, the rate of RCS blowdown is such that the ECCS pumps can provide adequate injection to ensure that ~~peak clad temperature remains below the 10 CFR 50.46 (Ref. 3) limit of 2200°F~~ the 10 CFR 50.46 (Ref. 3) acceptance criteria are met.

(continued)

BASES (continued)

REFERENCES

1. BTP ICSB-18 (Rev. 2, July 1981) "Application of the single failure criterion to manually controlled electrically operated valves.

2. FSAR, Chapter 6.

3. 10 CFR 50.46.

4. FSAR, Chapter 15.

5. WCAP-15049-A, Rev. 1, April 1999.

6. NUREG-1366, December 1992.

7. WCAP-16996-P-A, Rev. 1, November 2016.

BASES

BACKGROUND (background)

The ECCS subsystems are actuated upon receipt of an SI signal. The actuation of safeguard loads is accomplished in a programmed time sequence. If offsite power is available, the safeguard loads start after a one second sequencer delay in the programmed time sequence. If offsite power is not available, the Engineered Safety Feature (ESF) buses shed normal operating loads and are connected to the emergency diesel generators (EDGs). Safeguard loads are then actuated in the programmed time sequence. The time delay associated with diesel starting, sequenced loading, and pump starting determines the time required before pumped flow is available to the core following a LOCA.

Each ECCS pump is provided with normally open miniflow lines for pump protection. The RHR miniflow isolation valves close on flow to the RCS and have a time delay to prevent them from closing until the RHR pumps are up to speed and capable of delivering fluid to the RCS. The SI pump minflow isolation valves are closed manually from the control room prior to transfer from injection to recirculation. The Charging Pump miniflow isolation valves close on receipt of a safety injection signal and alternate minflow isolation valves open.

The active ECCS components, along with the passive accumulators and the RWST covered in LCO 3.5.1, "Accumulators," and LCO 3.5.4, "Refueling Water Storage Tank (RWST)," provide the cooling water necessary to meet GDC 35 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The LCO helps to ensure that the following acceptance criteria for the ECCS, established by 10 CFR 50.46 (Ref. 2), will be met following a LOCA:

There is a high level of probability that the peak cladding temperature does not exceed 2200ºF;

a. ~~Maximum fuel element cladding temperature is ≤ 2200°F;~~

There is a high level of probability that the maximum cladding oxidation does not exceed 0.17

b. ~~Maximum cladding oxidation is ≤ 0.17~~ times the total cladding thickness before oxidation;

There is a high level of probability that the maximum

c. ~~Maximum~~ hydrogen generation from a zirconium water reaction ~~is ≤ 0.01~~ times the hypothetical amount generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react;

does not exceed 0.01

d. Core is maintained in a coolable geometry; and

e. Adequate long term core cooling capability is maintained.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The LCO also limits the potential for a post trip return to power following an MSLB event and ensures that containment temperature limits are met.

Each ECCS subsystem is taken credit for in a large break LOCA event at full power (Refs. 3 and 4). This event establishes the requirement for runout flow for the ECCS pumps, as well as the maximum response time for their actuation. The centrifugal charging pumps and SI pumps are credited in a small break LOCA event. This event establishes the flow and discharge head at the design point for the centrifugal charging pumps. The SGTR and MSLB events also credit the centrifugal charging pumps. The OPERABILITY requirements for the ECCS are based on the following LOCA analysis assumptions:

a. A large break LOCA event, with or without loss of offsite power and a single failure disabling one ~~RHR pump~~ ECCS train (both EDG trains are assumed to operate due to requirements for modeling full active containment heat removal system operation); and

b. A small break LOCA event, with a loss of offsite power and a single failure disabling one ECCS train.

During the blowdown stage of a LOCA, the RCS depressurizes as primary coolant is ejected through the break into the containment. The nuclear reaction is terminated either by moderator voiding during large breaks or control rod insertion for small breaks. Following depressurization, emergency cooling water is injected into the cold legs, flows into the downcomer, fills the lower plenum, and refloods the core.

The effects on containment mass and energy releases are accounted for in appropriate analyses (Refs. 3 and 4). The LCO ensures that an ECCS train will deliver sufficient water to match boiloff rates soon enough to minimize the consequences of the core being uncovered following a large LOCA. It also ensures that the centrifugal charging and SI pumps will deliver sufficient water and boron during a small LOCA to maintain core subcriticality. For smaller LOCAs, the centrifugal charging pump delivers sufficient fluid to maintain RCS inventory. For a small break LOCA, the steam generators continue to serve as the heat sink, providing part of the required core cooling.

The ECCS trains satisfy Criterion 3 of 10CFR50.36(c)(2)(ii).

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

analysis

concentrations. Although it only has a minor effect, the maximum temperature is used in the feedline break ~~and small break LOCA analyses~~; the minimum temperature is an assumption in both the MSLB and inadvertent ECCS actuation analyses, although the inadvertent ECCS actuation event is typically non-limiting.

The MSLB analysis has considered a delay associated with the interlock between the VCT and RWST isolation valves, and the results show that the departure from nucleate boiling design basis is met. The delay has been established as 27 seconds, with offsite power available, or 37 seconds without offsite power. This response time includes 2 seconds for electronics delay, a 15 second stroke time for the RWST valves, and a 10 second stroke time for the VCT valves.

For a large break LOCA analysis, the minimum contained water volume limit of 473,731 gallons and the lower boron concentration limit of 2400 ppm are used to compute the post LOCA sump boron concentration necessary to assure subcriticality. The large break LOCA is the limiting case since the safety analysis assumes that all control rods are out of the core. The limits on minimum contained water volume and maximum boron concentration of the RWST also ensure a maximum equilibrium sump pH for the solution recirculated within containment after a LOCA which limits corrosion and hydrogen production. The limit on maximum boron concentration is also used to determine a minimum equilibrium sump pH. This minimum pH level minimizes the evolution of iodine and minimizes the effect of chloride stress corrosion on mechanical systems and components.

The upper limit on boron concentration of 2600 ppm is used to determine the maximum allowable time to switch to hot leg recirculation following a LOCA. The purpose of switching from cold leg to hot leg injection is to avoid boron precipitation in the core following the accident.

a range of RWST temperatures of 40ºF to 120ºF is used for the containment spray temperature per approved methods (Ref. 2).

In the ECCS analysis, ~~the containment spray temperature is assumed to be equal to the RWST lower temperature limit of 40°F.~~ If the lower temperature limit is violated, the containment spray further reduces containment pressure, which decreases the rate at which steam can be vented out the break and increases peak clad temperature. ~~The upper temperature limit of 120°F is used in the small break LOCA analysis and containment analysis.~~

The large and small break LOCA analyses use a range of RWST temperatures of 40ºF to 120ºF per approved methods (Ref. 2).

could

large and

Exceeding this temperature ~~will~~ result in a higher peak clad temperature, because there is less heat transfer from the core to the injected water for the small break LOCA ~~and higher containment pressures due to reduced~~

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

~~containment spray cooling capacity~~. For the containment response following an MSLB, the lower limit on boron concentration and the upper limit on RWST water temperature are used to maximize the total energy release to containment.

The RWST satisfies Criteria 2 and 3 of 10CFR50.36(c)(2)(ii).

LCO

The RWST ensures that an adequate supply of borated water is available to cool and depressurize the containment in the event of a Design Basis Accident (DBA), to cool and cover the core in the event of a LOCA, to maintain the reactor subcritical following a DBA, and to ensure adequate level in the containment sump to support ECCS and Containment Spray System pump operation in the recirculation mode.

To be considered OPERABLE, the RWST must meet the water volume, boron concentration, and temperature limits established in the SRs.

APPLICABILITY

In MODES 1, 2, 3, and 4, RWST OPERABILITY requirements are dictated by ECCS and Containment Spray System OPERABILITY requirements. Since both the ECCS and the Containment Spray System must be OPERABLE in MODES 1, 2, 3, and 4, the RWST must also be OPERABLE to support their operation. Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled," and LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled." MODE 6 core cooling requirements are addressed by LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation - High Water Level," and LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation - Low Water Level."

ACTIONS

A.1

With RWST boron concentration or borated water temperature not within limits, they must be returned to within limits within 8 hours. Under these conditions neither the ECCS nor the Containment Spray System can perform its design function. Therefore, prompt action must be taken to restore the tank to OPERABLE condition. The 8 hour limit to restore the RWST temperature or boron concentration to within limits was developed considering the time required to change either the boron concentration or temperature and the fact that the contents of the tank are still available for injection.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.4.2 (continued)

5% measurement uncertainty, is a conservative verification of contained volume. Other means of surveillance which consider measurement uncertainty may also be used.

SR 3.5.4.3

The boron concentration of the RWST should be verified to be within the required limits. This SR ensures that the reactor will remain subcritical following a LOCA. Further, it assures that the resulting sump pH will be maintained in an acceptable range so that boron precipitation in the core will not occur and the effect of chloride and caustic stress corrosion on mechanical systems and components will be minimized. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. FSAR, Chapter 6 and Chapter 15.

2. WCAP-16996-P-A, Rev. 1, November 2016.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.4 Containment Pressure

BASES

BACKGROUND

The containment pressure is limited during normal operation to preserve the initial conditions assumed in the accident analyses for a loss of coolant accident (LOCA) or steam line break (SLB). These limits also prevent the containment pressure from exceeding the containment design negative pressure differential with respect to the outside atmosphere in the event of inadvertent actuation of the Containment Spray System.

Containment pressure is a process variable that is monitored and controlled. The containment pressure limits are derived from the input conditions used in the containment functional analyses and the containment structure external pressure analysis. Should operation occur outside these limits coincident with a Design Basis Accident (DBA), post accident containment pressures could exceed calculated values.

APPLICABLE SAFETY ANALYSES

Containment internal pressure is an initial condition used in the DBA analyses to establish the maximum peak containment internal pressure. The limiting DBAs considered, relative to containment pressure, are the LOCA and SLB, which are analyzed using computer pressure transients.

The containment was designed for an internal pressure load equivalent to 50 psig. The LOCA and SLB are examined under a variety of initial conditions to ensure that the containment design limit is not exceeded. Although only two cases can yield pressure and temperature peaks, there are several cases that are near these peaks; furthermore, the time to the maximum temperature or pressure also varies with the assumed initial conditions. The full spectrum of cases for both LOCA and SLB transients determines the envelopes for which plant equipment is qualified. The containment was also designed for an internal pressure load equivalent to -5 psig. The inadvertent actuation of the Containment Spray System was analyzed to determine the resulting reduction in containment pressure. The initial pressure condition used in this analysis was -0.5 psig. This resulted in a minimum pressure greater than the design load.

For certain aspects of transient accident analyses, maximizing the calculated containment pressure is not conservative. In particular, the cooling effectiveness of the Emergency Core Cooling System during the core reflood phase of a LOCA analysis increases with increasing containment backpressure. ~~Therefore, for the reflood phase~~, the containment

For these calculations

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

backpressure is calculated in a manner designed to conservatively minimize, rather than maximize, the containment pressure response in accordance with ~~10 CFR 50, Appendix K~~ (Ref. 2).

WCAP-16996-P-A

Containment pressure satisfies Criterion 2 of 10CFR50.36(c)(2)(ii).

LCO

Maintaining containment pressure at less than or equal to the LCO upper pressure limit ensures that, in the event of a DBA, the resultant peak containment accident pressure will remain below the containment design pressure. Maintaining containment pressure at greater than or equal to the LCO lower pressure limit ensures that the containment will not exceed the design negative differential pressure following the inadvertent actuation of the Containment Spray System. An instrument uncertainty of ± 0.2 psi is conservatively included in the pressure limits (-0.3 to +1.3 psig) to allow the use of installed instrumentation for pressure measurements.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. Since maintaining containment pressure within limits is essential to ensure initial conditions assumed in the accident analyses are maintained, the LCO is applicable in MODES 1, 2, 3 and 4.

In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, maintaining containment pressure within the limits of the LCO is not required in MODE 5 or 6.

ACTIONS

A.1

When containment pressure is not within the limits of the LCO, it must be restored to within these limits within 8 hours. The Required Action is necessary to return operation to within the bounds of the containment analysis. The 8 hour Completion Time is reasonable to return pressure to normal.

B.1 and B.2

If containment pressure cannot be restored to within limits within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.4.1

Verifying that containment pressure is within limits ensures that unit operation remains within the limits assumed in the containment analysis. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. FSAR, Section 6.2.

2. ~~10 CFR 50, Appendix K.~~ WCAP-16996-P-A, Revision 1.

BASES

APPLICABLE SAFETY ANALYSES (continued)

specific power level of 100% for the LOCA and 70% for the SLB, one containment spray train actuates, and initial (pre-accident) containment conditions of 120°F and 1.5 psig. The analyses also assume a response time delayed initiation to provide conservative peak calculated containment pressure and temperature responses.

For certain aspects of transient accident analyses, maximizing the calculated containment pressure is not conservative. In particular, the effectiveness of the Emergency Core Cooling System during the core reflood phase of a LOCA analysis increases with increasing containment backpressure. For these calculations, the containment backpressure is calculated in a manner designed to conservatively minimize, rather than maximize, the calculated transient containment pressures in accordance with ~~10 CFR 50, Appendix K~~ WCAP-16996-P-A (Ref. 2).

The effect of an inadvertent containment spray actuation has been analyzed. An inadvertent spray actuation results in a -3.79 psig containment pressure and is associated with the sudden cooling effect in the interior of the leak tight containment. Additional discussion is provided in the Bases for LCO 3.6.4.

The modeled Containment Spray System actuation from the containment analysis is based on a response time associated with exceeding the containment "P" signal (High-3) pressure setpoint to achieving full flow through the containment spray nozzles. The Containment Spray System total response time includes diesel generator (DG) startup (for loss of offsite power), sequenced loading of equipment, containment spray pump startup, and spray line filling (Ref. 4).

The Containment Spray System satisfies Criterion 3 of 10CFR50.36(c)(2)(ii).

LCO

During a DBA, a minimum of one containment spray train is required to maintain the containment peak pressure and temperature below the design limits (Ref. 3). The containment spray train is also required to remove iodine from the containment atmosphere and maintain concentrations below those assumed in the safety analysis. To ensure that these requirements are met, two containment spray trains must be OPERABLE. Therefore, in the event of an accident, at least one train operates, assuming the worst case single active failure occurs.

Each Containment Spray System train includes two spray pumps, spray headers, nozzles, valves, piping, instruments, and controls to ensure an

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.6.5 and SR 3.6.6.6 (continued)

for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.6.6.7

Not Used

SR 3.6.6.8

With the containment spray inlet valves closed and the spray header drained of any solution, low pressure air or smoke can be blown through test connections. This SR ensures that each spray nozzle is unobstructed and provides assurance that spray coverage of the containment during an accident is not degraded. Due to the passive design of the nozzle, confirmation of operability following maintenance activities that can result in obstruction of spray nozzle flow is considered adequate to detect obstruction of the nozzles. Confirmation that the spray nozzles are unobstructed may be obtained by utilizing foreign materials exclusion (FME) controls during maintenance, a visual inspection of the affected portions of the system, or by an air or smoke flow test following maintenance involving opening portions of the system downstream of the containment isolation valves or draining of the filled portions of the system inside containment. Maintenance that could result in nozzle blockage is generally a result of a loss of foreign material control or a flow of borated water through a nozzle. Should either of these events occur, a supervisory evaluation will be required to determine whether nozzle blockage is a possible result of the event. For the loss of FME event, an inspection or flush of the affected portions of the system should be adequate to confirm that the spray nozzles are unobstructed since water flow would be required to transport any debris to the spray nozzles. An air flow or smoke test may not be appropriate for a loss of FME event but may be appropriate for the case where borated water inadvertently flows through the nozzles.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 38, GDC 39, GDC 40, GDC 41, GDC 42, and GDC 43.
2. ~~10 CFR 50, Appendix K.~~ WCAP-16996-P-A, Revision 1.
3. FSAR, Section 6.2.1.

(continued)

Application of Westinghouse FULL SPECTRUM LOCA Evaluation Model to the Comanche Peak Units 1 and 2 Nuclear Plants

(81 pages, including cover page)

1. TECHNICAL EVALUATION

1.1 Introduction

Analyses with the **FULL SPECTRUM™** loss-of-coolant accident (**FSLOCA™**) evaluation model (EM) have been completed for Comanche Peak Nuclear Power Plant (CPNPP) Unit 1 and Unit 2. This license amendment request (LAR) for Comanche Peak Unit 1 and Unit 2 requests approval to apply the Westinghouse FSLOCA EM.

The FSLOCA EM (Reference 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 includes 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 (Reference 2), which explicitly models TCD and is benchmarked to high burnup data in Reference 2. The fuel pellet thermal conductivity model in the WCOBRA/TRAC-TF2 code used in the FSLOCA EM explicitly accounts for pellet TCD.

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 that is needed to demonstrate compliance with 10 CFR 50.46 acceptance criteria (b)(1), (b)(2), and (b)(3) (Reference 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 (Reference 4). The coolable geometry acceptance criterion, 10 CFR 50.46 (b)(4), is assured by compliance with acceptance criteria (b)(1), (b)(2), and (b)(3), and demonstrating that fuel assembly grid deformation due to combined seismic and LOCA loads does not extend to the in-board fuel assemblies such that a coolable geometry is maintained.

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 (Reference 1). Since Comanche Peak Unit 1 and Unit 2 are Westinghouse designed 4-loop plants with cold leg ECCS injection, the approved method is applicable. Information required to address Limitations and Conditions 9 and 10 of the NRC's Safety Evaluation Report (SER) for Reference 1 was docketed in Reference 5 in support of application of the FSLOCA EM to Westinghouse 4-loop plants.

This report summarizes the application of the Westinghouse FSLOCA EM to Comanche Peak Unit 1 and Unit 2. The application of the FSLOCA EM to Comanche Peak Unit 1 and Unit 2 is consistent with the NRC-approved methodology (Reference 1), with exceptions identified under Limitation and Condition Number 2 in Section 1.2.3. The application of the FSLOCA EM to Comanche Peak Unit 1 and Unit 2 is consistent with the conditions and limitations as identified in the NRC's SER for Reference 1, and is also applicable for the Comanche Peak Unit 1 and Unit 2 plant design and operating conditions.

Two separate analyses with the FSLOCA EM were performed for Comanche Peak Unit 1 and Unit 2. A single composite vessel model was developed to represent both units. The plant designs and operating parameters were assessed to create the composite model for a conservative application of the FSLOCA EM. Separate loop models were used to account for the difference in the steam generator type.

Both Luminant 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 major plant parameter and analysis assumptions used in the Comanche Peak Unit 1 and Unit 2 analyses with the FSLOCA EM are provided in Tables 1 through 6.

1.2 Method of Analysis

1.2.1 FULL SPECTRUM LOCA Evaluation Model Development

In 1988, the NRC Staff amended the requirements of 10 CFR 50.46 (Reference 3 and Reference 7) and Appendix K, "ECCS Evaluation Models," to permit the use of a realistic EM to analyze the performance of the ECCS during a hypothetical LOCA. Westinghouse's previously approved best-estimate LBLOCA EM is discussed in Reference 8. The EM is referred to as the Automated Statistical Treatment of Uncertainty Method (ASTRUM), and was developed following Regulatory Guide (RG) 1.157 (Reference 9).

When the FSLOCA EM was being developed, the NRC issued RG 1.203 (Reference 10) 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.

1.2.2 WCOBRA/TRAC-TF2 Computer Code

The FSLOCA EM (Reference 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.

1.2.3 Compliance with FSLOCA EM Limitations and Conditions

The NRC's SER for Reference 1 contains 15 limitations and conditions on the NRC-approved FSLOCA EM. A summary of each limitation and condition and how it was 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 the long-term cooling.

Compliance

The analyses for Comanche Peak Unit 1 and Unit 2 with the FSLOCA EM are only being used to demonstrate compliance with 10 CFR 50.46 (b)(1) through (b)(4). The actions that are currently in place to demonstrate compliance with 10 CFR 50.46 acceptance criterion (b)(5) related to the long-term cooling are not impacted by the application of the NRC-approved FSLOCA EM (Reference 1).

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

Comanche Peak Unit 1 and Unit 2 are Westinghouse-designed 4-loop PWRs with cold-side injection, so they are within the NRC-approved methodology. The analyses for Comanche Peak Unit 1 and Unit 2 utilize the NRC-approved FSLOCA methodology, except for the changes which were previously transmitted to the NRC pursuant to 10 CFR 50.46 in LTR-NRC-18-30 (Reference 6). After completion of the analyses for Comanche Peak Unit 1 and Unit 2, two errors were discovered in the WCOBRA/TRAC-TF2 code that can occur under certain conditions. These errors were found to have negligible impact on analysis results with the FSLOCA EM as described in LTR-NRC-19-6 (Reference 13).

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 for the Comanche Peak Unit 1 and Unit 2 analyses was performed consistent with the NRC-approved methodology. Appropriate design parameters and conditions were modeled, as were the engineered safety features which can reduce the containment pressure. A plant-specific 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 [ ][a,c] 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 [ ][a,c] for the Comanche Peak Unit 1 and Unit 2 analyses. 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 10A (Unit 1) and Table 10B (Unit 2).

Limitation and Condition Number 5

Summary

The maximum assembly and rod length-average burnup is limited to [ ][a,c] respectively.

Compliance

The maximum analyzed assembly and rod length-average burnup is less than or equal to [ ][a,c] respectively, for Comanche Peak Unit 1 and Unit 2.

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 is utilized in the Comanche Peak Unit 1 and Unit 2 analyses with the FSLOCA EM. The generation of all the PAD5 fuel performance data adheres to the NRC-approved PAD5 methodology. The analyzed fuel pellet average temperatures bound the maximum values calculated in accordance with Section 7.5.1 of Reference 2, and the analyzed rod internal pressures were calculated in accordance with Section 7.5.2 of Reference 2.

Limitation and Condition Number 7

Summary

The YDRAG uncertainty parameter should be [ ]^{a,c}

Compliance

Consistent with the NRC-approved methodology, the YDRAG uncertainty parameter was [ ]^{a,c} for the Comanche Peak Unit 1 and Unit 2 Region I analyses.

Limitation and Condition Number 8

Summary

The [ ]^{a,c}

Compliance

Consistent with the NRC-approved methodology, the [ ]^{a,c} for the Comanche Peak Unit 1 and Unit 2 Region I analyses.

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 the **[** **]**^{a,c} 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

Comanche Peak Unit 1 and Unit 2 are both Westinghouse-designed 4-loop PWRs. The requested sensitivity study was performed for a 4-loop Westinghouse-designed PWR and is discussed in Reference 5.

Limitation and Condition Number 10

Summary

For PWR designs which are not Westinghouse 3-loop PWRs, a sensitivity study will be executed to: 1) demonstrate that no unexplained behavior occurs in the predicted safety criteria across the region boundary, and 2) ensure that the **[** **]**^{a,c} must cover the equivalent 2 to 4-inch break range using RCS-volume scaling relative to the demonstration plant. This sensitivity study should be executed once, and then referenced in all applications to that particular plant class.

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

Compliance

Comanche Peak Unit 1 and Unit 2 are both Westinghouse-designed 4-loop PWRs. The requested sensitivity study was performed for a 4-loop Westinghouse-designed PWR and is discussed in Reference 5.

The minimum sampled break area for the Comanche Peak Unit 1 and Unit 2 Region II analyses was 1 ft^2.

Limitation and Condition Number 11

Summary

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

1. The **[** **]**^{a,c} the Region I and Region II analysis seeds, and the analysis inputs will be declared and documented prior to performing the Region I and Region II uncertainty analyses. The **[** **]**^{a,c} and the Region I and Region II analysis seeds 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 peak cladding temperature (PCT), maximum local oxidation (MLO), and core-wide oxidation (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.

<u>Compliance</u>

This Limitation and Condition was met for the Comanche Peak Unit 1 and Unit 2 analyses as follows:

1. The **[** **]**^{a,c} the Region I and Region II analysis seeds, and the analysis inputs were declared and documented prior to performing the Region I and Region II uncertainty analyses. The **[** **]**^{a,c} and the Region I and Region II analysis seeds were not changed once they were 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 for Comanche Peak Unit 1 and Unit 2 in Table 1.

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

<u>Summary</u>

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.

<u>Compliance</u>

A conservatively high plant-specific dynamic pressure loss from the steam generator secondary-side to the main steam safety valves (MSSVs) was modeled in the Comanche Peak Unit 1 and Unit 2 analyses.

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

<u>Summary</u>

In plant-specific models for analysis with the FSLOCA EM: 1) the **[** **]**^{a,c} and 2) the **[** **]**^{a,c}

<u>Compliance</u>

The **[** **]**^{a,c} in the analyses for Comanche Peak Unit 1 and Unit 2. The **[** **]**^{a,c} in the analyses.

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 Comanche Peak Unit 1 and Unit 2 analyses, 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 be in compliance with the 10 CFR 50.46 acceptance criteria.

The **[** **]**^{a,c}

Compliance

The Region II uncertainty analyses for Comanche Peak Unit 1 and Unit 2 were performed twice; once assuming a LOOP and once assuming OPA. The results from both analyses that were performed are in compliance with the 10 CFR 50.46 acceptance criteria (see Section 1.5).

The **[**
]^{a,c}

1.3 Region I Analysis

1.3.1 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 upper regions by the liquid remaining in the crossover leg loop seals. Throughout this period, the core remains covered by a two-phase mixture and the fuel cladding temperatures remain at the saturation temperature level.

Loop Seal Clearance

As the system drains, the liquid levels in the downhill side of the pump suction (crossover leg) become depressed all the way to the bottom elevations of the piping, allowing the steam trapped during the natural circulation phase to vent to the break (i.e., a process called loop seal clearance). 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 inner portion of the reactor vessel.

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 remains high enough such that safety injection flow cannot make up for the primary system fluid inventory lost through the break, leading to core uncovery and a fuel rod cladding temperature heatup.

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 transient is considered over as the break flow is compensated by the injected flow.

1.3.2 Analysis Results

The Comanche Peak Unit 1 and Unit 2 Region I analyses were performed in accordance with the NRC-approved methodology in Reference 1, with exceptions identified under Limitation and Condition Number 2 in Section 1.2.3. The transient that produced the analysis PCT result is a cold leg break with a break diameter of 3.7-inches for Unit 1 and 3.6-inches for Unit 2. 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 2.7-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 Comanche Peak Region I uncertainty analyses are summarized in Table 7A (Unit 1) and Table 7B (Unit 2). The sampled decay heat uncertainty multipliers for the Region I analysis cases are provided in Table 10A (Unit 1) and Table 10B (Unit 2).

Table 8A (Unit 1) and Table 8B (Unit 2) contain a sequence of events for the transient that produced the Region I analysis PCT result. Figures 1A through 13A (Unit 1) and Figures 1B through 13B (Unit 2) illustrate the calculated key transient response parameters for this transient.

1.4 Region II Analysis

1.4.1 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 heatup 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 the 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 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.

Blowdown – Downward Core Flow Phase

The break flow begins to dominate and pulls flow down through the core as the RCP head degrades due to increased voiding, 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 bypassed around the downcomer and sent out through the break. 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 approaches 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 nitrogen is discharged into the RCS. The nitrogen surge forces water into the core, which is then evaporated, causing system re-pressurization and a temporary reduction of pumped ECCS flow; this re-pressurization is illustrated by the increase in RCS pressure. 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 is able to quench all the fuel assemblies in the core.

1.4.2 Analysis Results

The Comanche Peak Unit 1 and Unit 2 Region II analyses were performed in accordance with the NRC-approved methodology in Reference 1, with exceptions identified under Limitation and Condition Number 2 in Section 1.2.3. The analysis was performed assuming both LOOP and OPA, and the results of both of 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 Comanche Peak Region II LOOP and OPA uncertainty analyses are summarized in Table 7A (Unit 1) and Table 7B (Unit 2). The sampled decay heat uncertainty multipliers for the Region II analysis cases are provided in Table 10A (Unit 1) and Table 10B (Unit 2).

Table 9A (Unit 1) and Table 9B (Unit 2) contain a sequence of events for the transient that produced the more limiting analysis PCT result relative to the offsite power assumption. Figures 14A through 28A (Unit 1) and Figures 14B through 28B (Unit 2) illustrate the key response parameters for this transient.

The containment pressure is calculated for each LOCA transient in the analysis using the COCO code (References 11 and 12). 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 in Reference 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. The containment backpressure for the transient that produced the analysis PCT result is provided in Figure 22A (Unit 1) and Figure 22B (Unit 2).

1.5 Compliance with 10 CFR 50.46

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 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., "Peak Cladding Temperature does not exceed 2,200°F," is demonstrated.

The results are shown in Table 7A (Unit 1) and Table 7B (Unit 2) for Comanche Peak.

(b)(2) The analysis 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., "Maximum Local Oxidation of the cladding does not exceed 17 percent," is demonstrated.

The results are shown in Table 7A (Unit 1) and Table 7B (Unit 2) for Comanche Peak.

(b)(3) The analysis 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., "Core-Wide Oxidation does not exceed 1 percent," is demonstrated.

The results are shown in Table 7A (Unit 1) and Table 7B (Unit 2) for Comanche Peak.

(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 Comanche Peak as shown in Table 7A (Unit 1) and Table 7B (Unit 2).

It is discussed in Section 32.1 of the NRC-approved FSLOCA EM (Reference 1) 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 Comanche Peak.

(b)(5) 10 CFR 50.46 acceptance criterion (b)(5) requires that long-term core cooling be provided following the successful initial operation of the ECCS.

Long-term cooling is dependent on the demonstration of the continued delivery of cooling water to the core. The actions that are currently in place to maintain long-term cooling are not impacted by the application of the NRC-approved FSLOCA EM (Reference 1).

Based on the analysis results for Region I and Region II presented in Table 7A (Unit 1) and Table 7B (Unit 2) for Comanche Peak, it is concluded that Comanche Peak complies with the criteria in 10 CFR 50.46.

1.6 References

1. "Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (FULL SPECTRUM LOCA Methodology)," WCAP-16996-P-A, Revision 1, November 2016.
2. "Westinghouse Performance Analysis and Design Model (PAD5)," WCAP-17642-P-A, Revision 1, November 2017.
3. "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Cooled Nuclear Power Reactors," 10 CFR 50.46 and Appendix K of 10 CFR 50, Federal Register, Volume 39, Number 3, January 1974.
4. "Information Notice 98-29: Predicted Increase in Fuel Rod Cladding Oxidation," USNRC, August 1998.
5. "'Information to Satisfy the FULL SPECTRUM LOCA (FSLOCA) Evaluation Methodology Plant Type Limitations and Conditions for 4-loop Westinghouse Pressurized Water Reactors (PWRs)' (Proprietary/Non-Proprietary)," LTR-NRC-18-50, July 2018.
6. "U.S. Nuclear Regulatory Commission 10 CFR 50.46 Annual Notification and Reporting for 2017," LTR-NRC-18-30, July 2018.
7. "Emergency Core Cooling Systems: Revisions to Acceptance Criteria," Federal Register, V53, N180, pp. 35996-36005, September 1988.
8. "Realistic Large-Break LOCA Evaluation Methodology Using the Automated Statistical Treatment Of Uncertainty Method (ASTRUM)," WCAP-16009-P-A, January 2005.
9. "Best Estimate Calculations of Emergency Core Cooling System Performance," Regulatory Guide 1.157, USNRC, May 1989.
10. "Transient and Accident Analysis Methods," Regulatory Guide 1.203, USNRC, December 2005.
11. "Westinghouse Emergency Core Cooling System Evaluation Model – Summary," WCAP-8339, June 1974.
12. "Containment Pressure Analysis Code (COCO)," WCAP-8327, June 1974.
13. "U.S. Nuclear Regulatory Commission 10 CFR 50.46 Annual Notification and Reporting for 2018," LTR-NRC-19-6, February 2019.

Table 1. Plant Operating Range Analyzed and Key Parameters for Comanche Peak Unit 1 and Unit 2

Parameter		As-Analyzed Value or Range
1.0	**Core Parameters**	
	a) Core power	≤ 3612 MWt ± 0.6% Uncertainty
	b) Fuel type	Unit 1: 17x17 OFA, Vantage+ with IFM grids, IFBA and WABA, **ZIRLO**® cladding and Robust Protective Grid Unit 2: 17x17 OFA, Vantage+ with IFM grids, IFBA, ZIRLO cladding and Robust Protective Grid (WABA not normally used)
	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	1.60
	e) Axial flux difference (AFD) band at 100% power	+10% / -15%
2.0	**Reactor Coolant System Parameters**	
	a) Thermal design flow (TDF)	95,700 gpm/loop
	b) Vessel average temperature (T_{AVG})	574.2 – 6.0°F ≤ T_{AVG} ≤ 589.2 + 6.0°F
	c) Pressurizer pressure (P_{RCS})	2250 – 30 psia ≤ P_{RCS} ≤ 2250 + 30 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 Comanche Peak Unit 1 and Unit 2

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 (420°F)
	d) Auxiliary feedwater temperature	Nominal (80°F)
	e) Auxiliary feedwater flow rate	430 gpm (total flow to all SGs)
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} ≤ 120°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	≤ 37 seconds (OPA) or 47 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})	88°F ≤ T_{ACC} ≤ 120°F
	b) Accumulator water volume (V_{ACC})	6119 gal ≤ V_{ACC} ≤ 6597 gal
	c) Accumulator pressure (P_{ACC})	588.3 psig ≤ P_{ACC} ≤ 679.3 psig
	d) Accumulator boron concentration	≥ 2300 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	1860 psia

Table 2. Containment Data Used for Region II Calculation of Containment Pressure for Comanche Peak Unit 1 and Unit 2

Parameter	Value
Maximum containment net free volume	3.063×10^6 ft^3
Minimum initial containment temperature at full power operation	88°F
Refueling water storage tank (RWST) temperature for containment spray (T_{RWST})	$40°F \leq T_{RWST} \leq 120°F$
Minimum RWST temperature for broken loop spilling SI	40°F
Minimum containment outside air / ground temperature	4°F
Minimum initial containment pressure at normal full power operation	14.2 psia
Minimum containment spray pump initiation delay from containment high pressure signal time	≥ 22 seconds (OPA) or 32 seconds (LOOP)
Maximum containment spray flow rate from all pumps	15,504 gpm
Maximum number of containment fan coolers in operation during LOCA transient	0
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	3 inches
Maximum containment pressure setpoint for venting valve closure	3.8 psig
Maximum delay time between reaching containment pressure setpoint and start of venting valve closure	2 seconds
Maximum venting valve closure time at normal full power operation	5 seconds
Containment walls / heat sink properties	Table 3
SI spilling flows	390.7 lbm/sec

Table 3. Containment Heat Sink Data Used for Region II Calculation of Containment Pressure for Comanche Peak Unit 1 and Unit 2

Wall	Area (ft^2)	Thickness (ft)	Material
1	29487.0	0.04167 2.458	Steel Concrete
2	84091.0	0.03133 4.467	Steel Concrete
3	3374.0	0.125	Steel
4	198763.5	4.27	Concrete
5	1078360.0	0.00125	Steel
6	93014.86	0.0263	Steel
7	7477.66	0.0398	Steel
8	2520.82	0.04858	Steel
9	753.28	0.0625	Steel
10	2061.96	0.08333	Steel
11	3239.25	0.1592	Steel
12	6819.36	0.2083	Steel
13	963.64	0.25	Steel
14	7012.0	0.5	Steel
15	730.0	0.75	Steel
16	1287.0	1.5	Steel
17	1152.1	0.71	Steel
18	3012.0	6.5 0.02083 12.0	Concrete Steel Concrete
19	8798.0	2.5 0.02083 12.0	Concrete Steel Concrete
20	4086.8	0.01625	Steel

Table 4. Safety Injection Flow Used for Region I Calculation for Comanche Peak Unit 1 and Unit 2

Pressure (psia)	High Head Safety Injection (HHSI) Flow (gpm)	Intermediate Head Safety Injection (IHSI) Flow (gpm)
14.7	253.4	398.1
114.7	246.3	398.1
214.7	239.0	398.1
314.7	231.7	381.5
414.7	224.2	363.5
514.7	216.6	344.7
614.7	209.0	324.9
714.7	201.1	302.9
814.7	193.2	279.8
914.7	184.7	255.5
1014.7	175.9	228.7
1114.7	167.0	197.0
1214.7	157.9	158.8
1314.7	148.5	114.2
1414.7	138.9	39.6
1414.71	138.9	0.0
1514.7	129.0	0.0
1614.7	118.8	
1714.7	107.0	
1814.7	93.7	
1914.7	79.6	
2014.7	62.1	
2114.7	42.8	
2214.7	17.1	
2214.71	0.0	
3000.0	0.0	

Table 5. Safety Injection Flow Used for Region II Calculation for Comanche Peak Unit 1 and Unit 2

Pressure (psia)	High Head Safety Injection (HHSI) Flow (gpm)	Intermediate Head Safety Injection (IHSI) Flow (gpm)	Low Head Safety Injection (LHSI) Flow (gpm)
14.7	253.4	375.0	3236.8
34.7	252.0	371.7	1914.8
54.7	250.6	368.4	1344.9
74.7	249.1	365.0	710.0
74.71	249.1	365.0	0.0
94.7	247.7	361.7	0.0
114.7	246.3	358.4	
214.7	239.0	341.3	
314.7	231.7	323.6	
414.7	224.2	305.1	
514.7	216.6	284.9	
614.7	209.0	263.7	
714.7	201.1	241.2	
814.7	193.2	217.1	
914.7	184.7	191.2	
1014.7	175.9	162.0	
1114.7	167.0	128.6	
1214.7	157.9	81.6	
1314.7	148.5	4.3	
1314.71	148.5	0.0	
1414.7	138.9	0.0	
1514.7	129.0		
1614.7	118.8		
1714.7	107.0		
1814.7	93.7		
1914.7	79.6		
2014.7	62.1		
2114.7	42.8		
2214.7	17.1		
2214.71	0.0		
3000.0	0.0		

Table 6. Steam Generator Main Steam Safety Valve Parameters for Comanche Peak Unit 1 and Unit 2

Stage	Set Pressure (psig)	Uncertainty (%)	Accumulation (psi)
1	1185	3	5
2	1195	3	5
3	1205	3	5
4	1215	3	5
5	1235	3	5

Table 7A. Comanche Peak Unit 1 Analysis Results with the FSLOCA EM

Outcome	Region I Value	Region II Value (LOOP)	Region II Value (OPA)
95/95 PCT	1,017°F	1,546°F	1,546°F
95/95 MLO	8.96%	8.64%	8.64%
95/95 CWO	0.00%	0.02%	0.02%

Table 7B. Comanche Peak Unit 2 Analysis Results with the FSLOCA EM

Outcome	Region I Value	Region II Value (LOOP)	Region II Value (OPA)
95/95 PCT	1,113°F	1,579°F	1,569°F
95/95 MLO	8.70%	8.82%	8.82%
95/95 CWO	0.00%	0.04%	0.04%

Table 8A. Comanche Peak Unit 1 Sequence of Events for Region I Analysis PCT Transient

Event	Time after Break (sec)
Start of Transient	0.0
Reactor Trip Signal	18.0
Safety Injection Signal	30.6
Safety Injection Begins	77.6
Loop Seal Clearing Occurs	640
Top of Core Uncovered	820
Accumulator Injection Begins	1,080
PCT Occurs	1,081
Top of Core Recovered	1,120

Table 8B. Comanche Peak Unit 2 Sequence of Events for Region I Analysis PCT Transient

Event	Time after Break (sec)
Start of Transient	0.0
Reactor Trip Signal	17.7
Safety Injection Signal	29.4
Safety Injection Begins	76.4
Loop Seal Clearing Occurs	620
Top of Core Uncovered	870
Accumulator Injection Begins	1,195
PCT Occurs	1,200
Top of Core Recovered	1,280

Table 9A. Comanche Peak Unit 1 Sequence of Events for Region II Analysis PCT Transient

Event	Time after Break (sec)
Start of Transient	0.0
Fuel Rod Burst Occurs	3.1
PCT Occurs	4.4
Safety Injection Signal	5.0
Accumulator Injection Begins	12
End of Blowdown	25
Accumulator Empty	48
Safety Injection Begins	52
All Rods Quenched	170

Table 9B. Comanche Peak Unit 2 Sequence of Events for Region II Analysis PCT Transient

Event	Time after Break (sec)
Start of Transient	0.0
Safety Injection Signal	5.6
Fuel Rod Burst Occurs	7.9
Accumulator Injection Begins	9.0
End of Blowdown	20
Accumulator Empty	47
Safety Injection Begins	53
PCT Occurs	82
All Rods Quenched	190

Table 10A. Comanche Peak Unit 1 Sampled Value of Decay Heat Uncertainty Multiplier, DECAY_HT, for Region I and Region II Analysis Cases

Region	Case	DECAY_HT (units of σ)	DECAY_HT (absolute units)[1]
Region I	PCT	+0.4889σ	2.37%
	MLO	+1.6486σ	8.41%
	CWO	N/A[2]	N/A[2]
Region II (LOOP)	PCT	+1.3147σ	6.25%
	MLO	+1.0069σ	5.13%
	CWO	+0.7952σ	3.77%
Region II (OPA)	PCT	+1.3147σ	6.25%
	MLO	+1.0069σ	5.13%
	CWO	+1.6986σ	8.08%

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 0.0%.

Table 10B. Comanche Peak Unit 2 Sampled Value of Decay Heat Uncertainty Multiplier, DECAY_HT, for Region I and Region II Analysis Cases

Region	Case	DECAY_HT (units of σ)	DECAY_HT (absolute units)[1]
Region I	PCT	+1.2485σ	6.17%
	MLO	+0.2048σ	1.04%
	CWO	N/A[2]	N/A[2]
Region II (LOOP)	PCT	+1.9393σ	9.19%
	MLO	+1.2676σ	6.47%
	CWO	+0.8144σ	4.11%
Region II (OPA)	PCT	+1.9393σ	9.19%
	MLO	+1.2676σ	6.47%
	CWO	+0.4329σ	2.06%

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 0.0%.

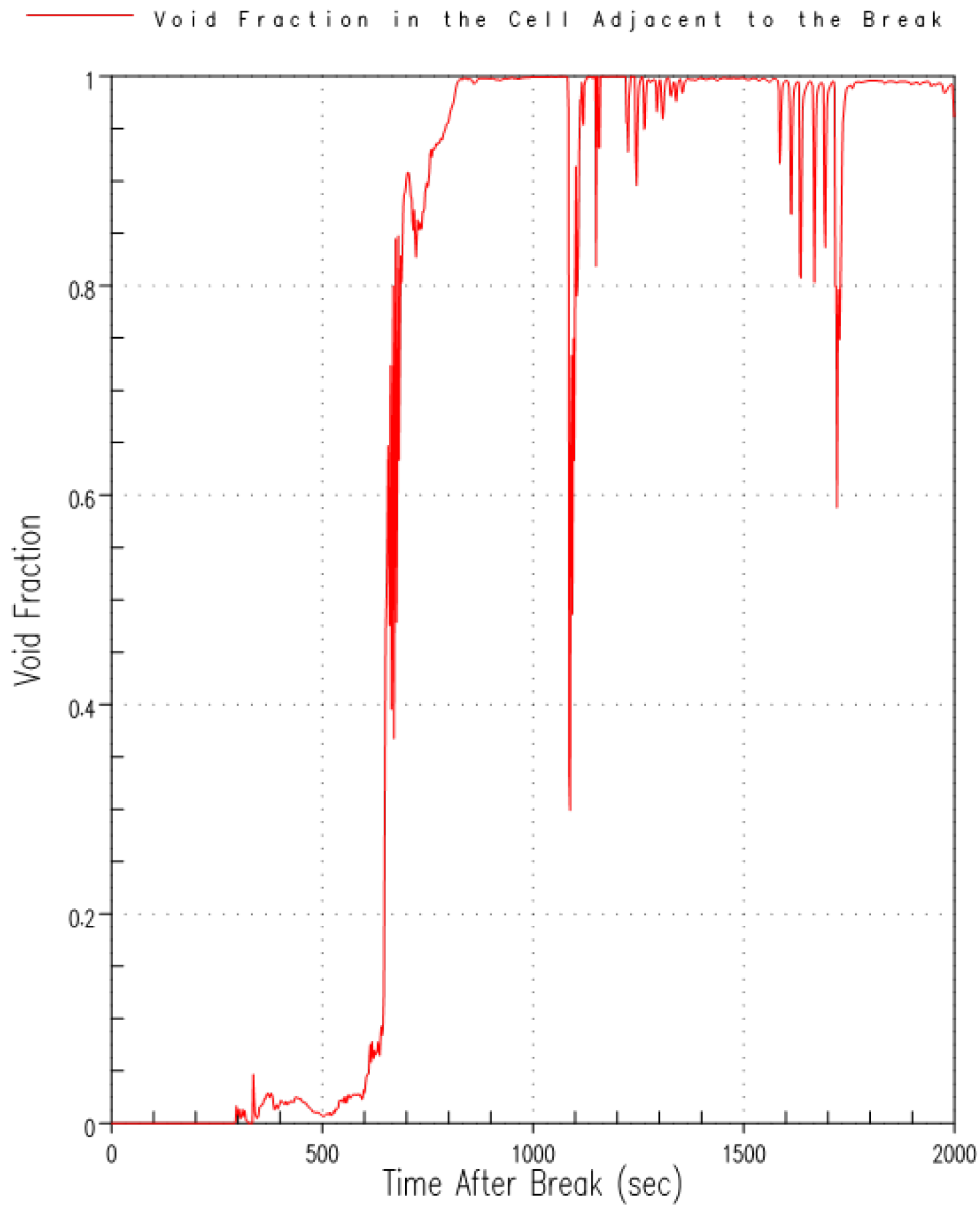

Figure 1A: Comanche Peak Unit 1 Break Flow Void Fraction for the Region I Analysis PCT Case

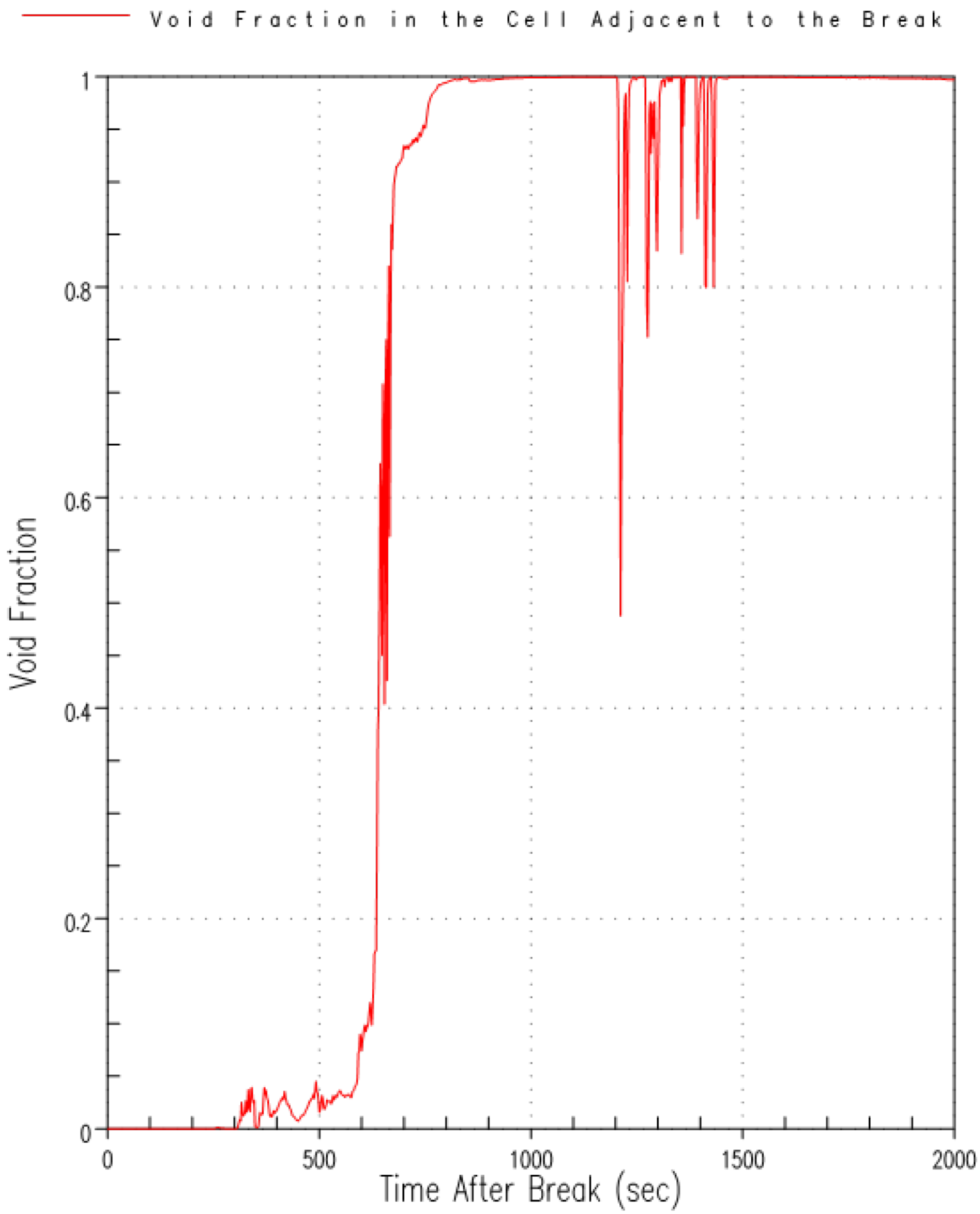

Figure 1B: Comanche Peak Unit 2 Break Flow Void Fraction for the Region I Analysis PCT Case

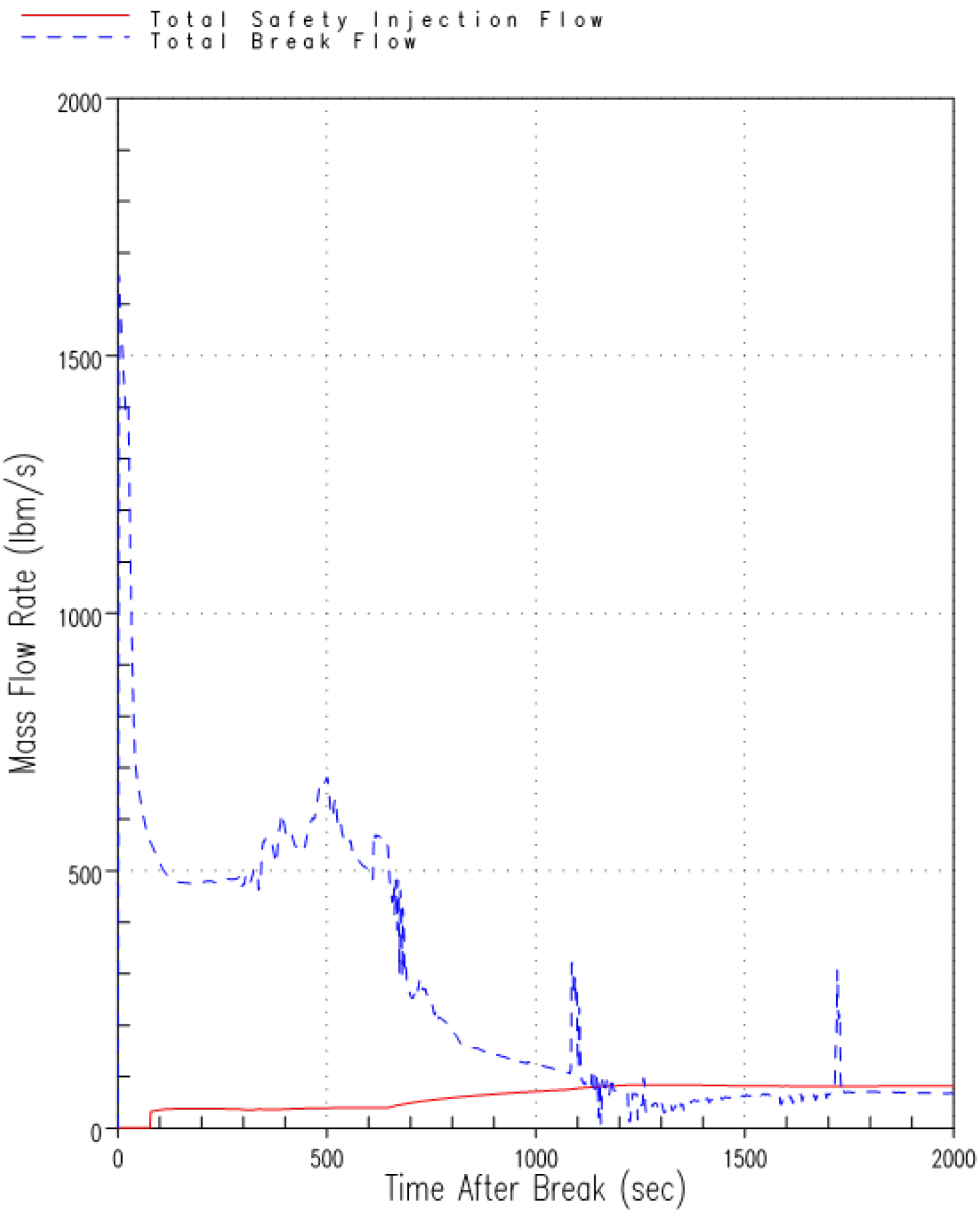

Figure 2A: Comanche Peak Unit 1 Total Safety Injection Flow (not including Accumulator Injection Flow) and Total Break Flow for the Region I Analysis PCT Case

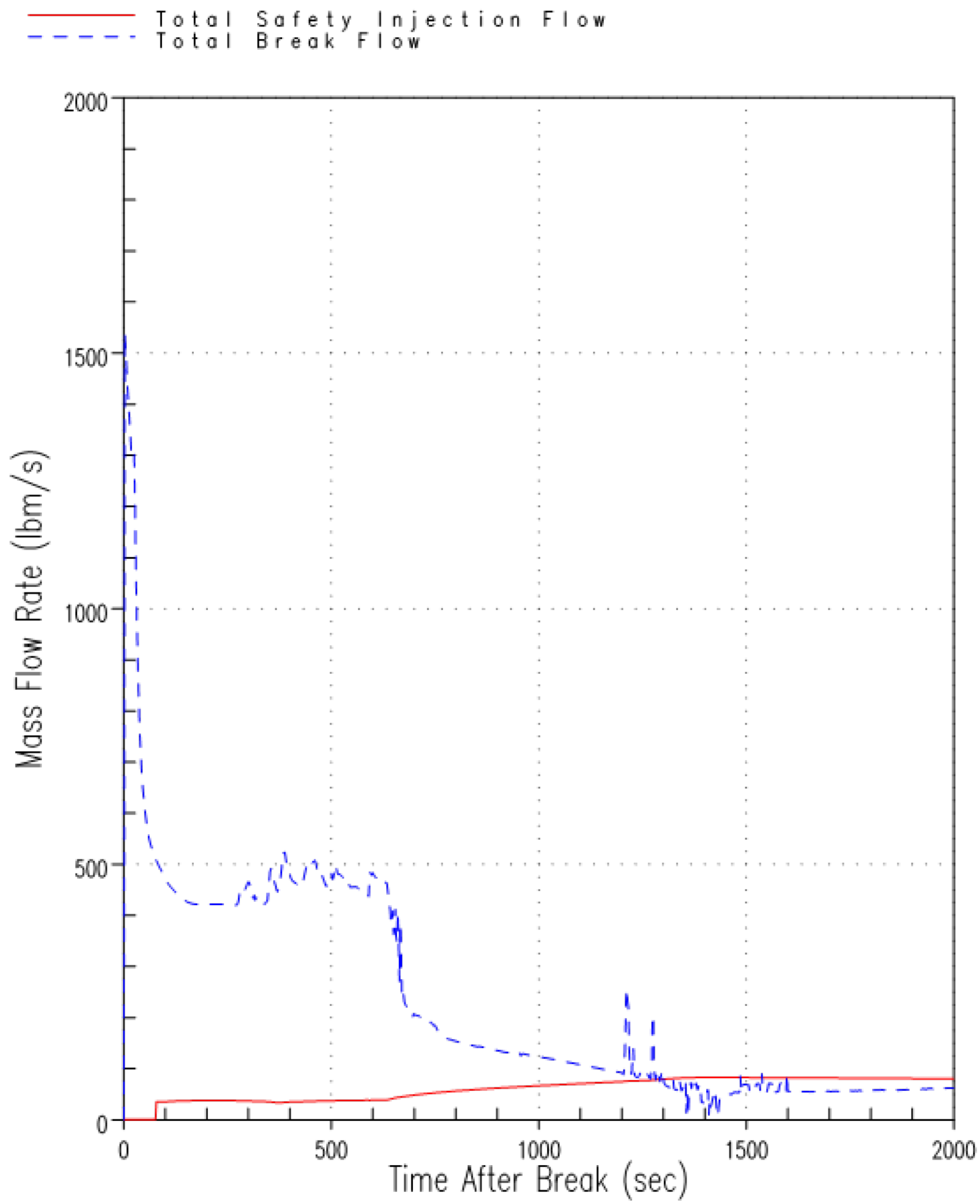

Figure 2B: Comanche Peak Unit 2 Total Safety Injection Flow (not including Accumulator Injection Flow) and Total Break Flow for the Region I Analysis PCT Case

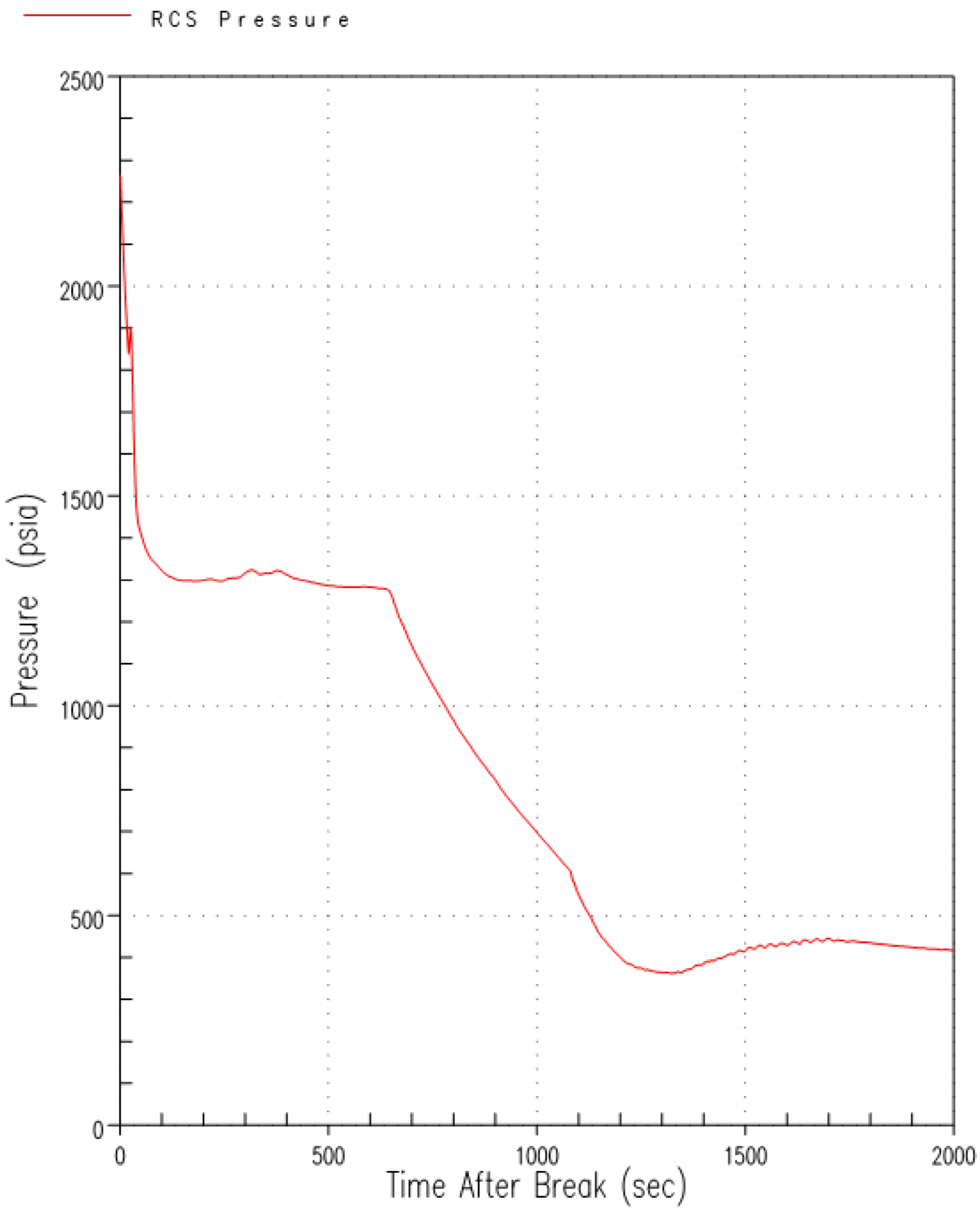

Figure 3A: Comanche Peak Unit 1 RCS Pressure for the Region I Analysis PCT Case

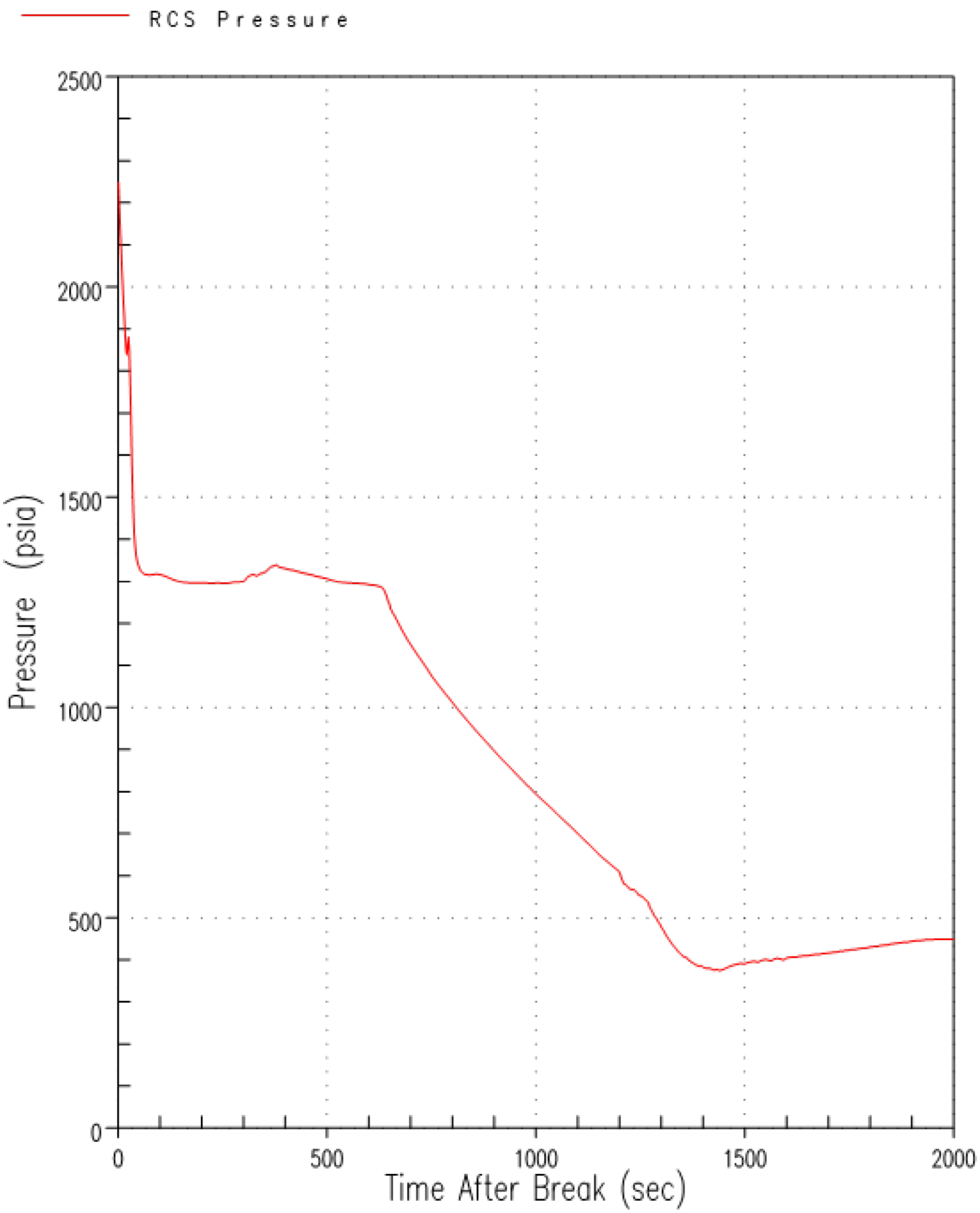

Figure 3B: Comanche Peak Unit 2 RCS Pressure for the Region I Analysis PCT Case

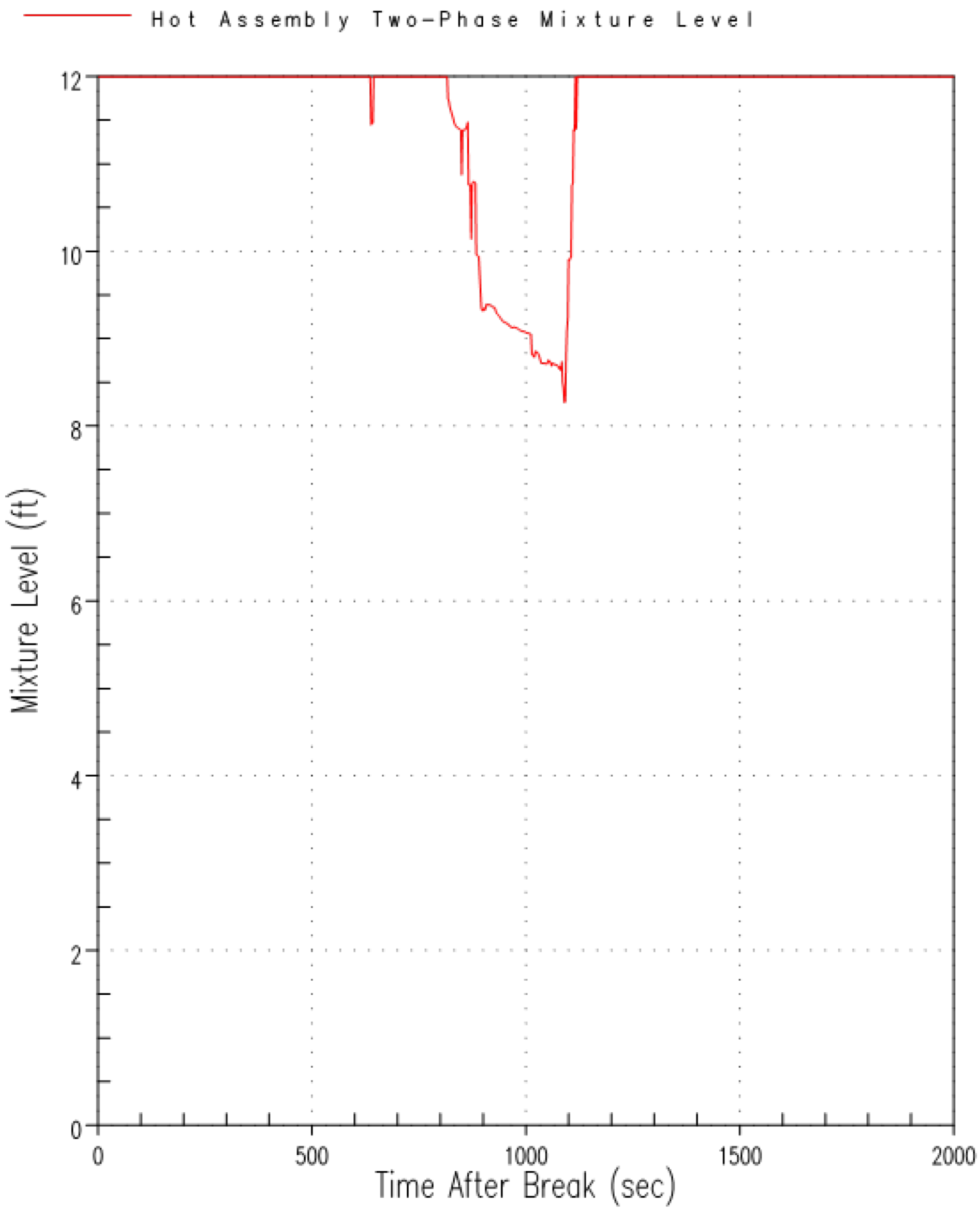

Figure 4A: Comanche Peak Unit 1 Hot Assembly Two-Phase Mixture Level (Relative to Bottom of Active Fuel) for the Region I Analysis PCT Case

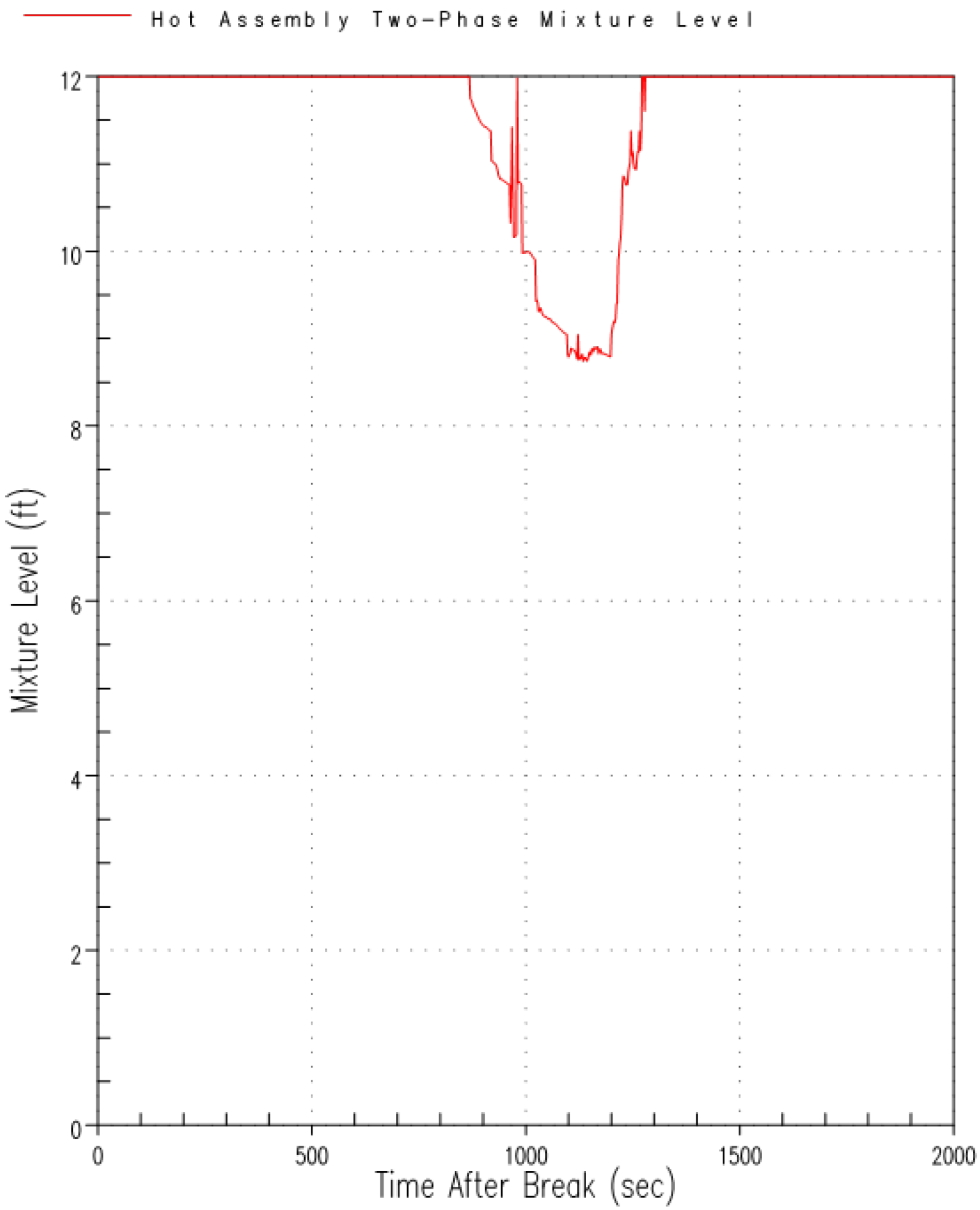

Figure 4B: Comanche Peak Unit 2 Hot Assembly Two-Phase Mixture Level (Relative to Bottom of Active Fuel) for the Region I Analysis PCT Case

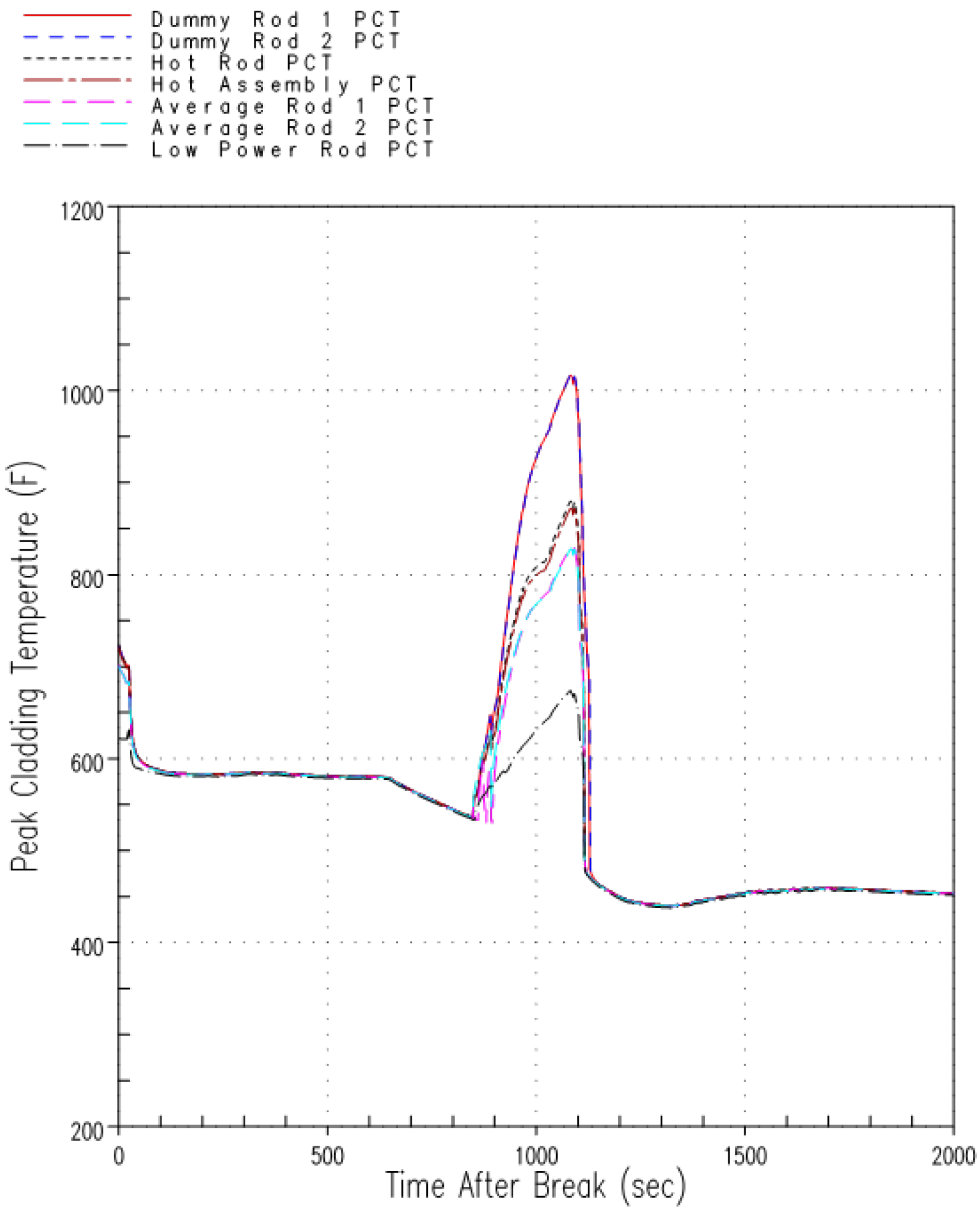

Figure 5A: Comanche Peak Unit 1 Peak Cladding Temperature for all Rods for the Region I Analysis PCT Case

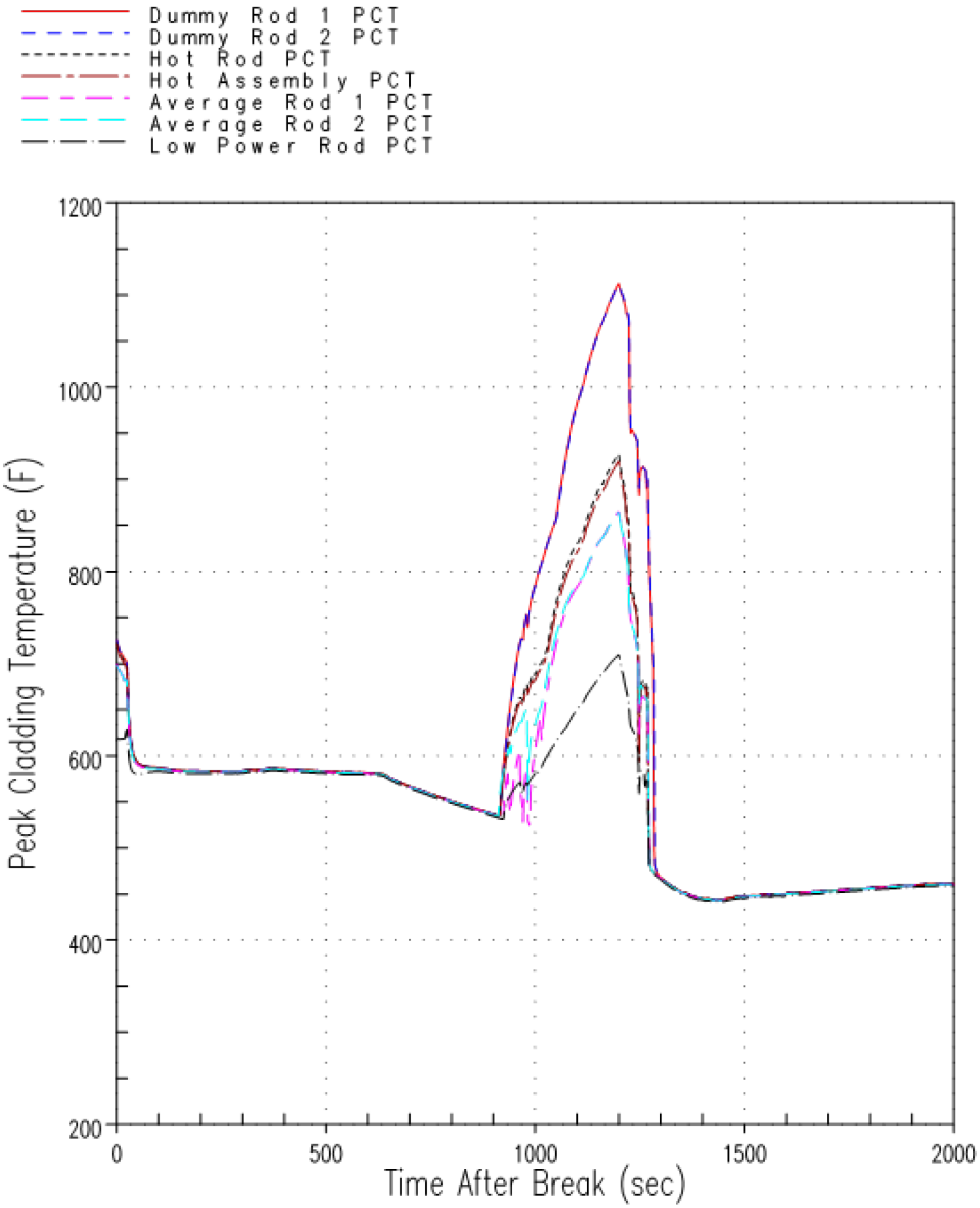

Figure 5B: Comanche Peak Unit 2 Peak Cladding Temperature for all Rods for the Region I Analysis PCT Case

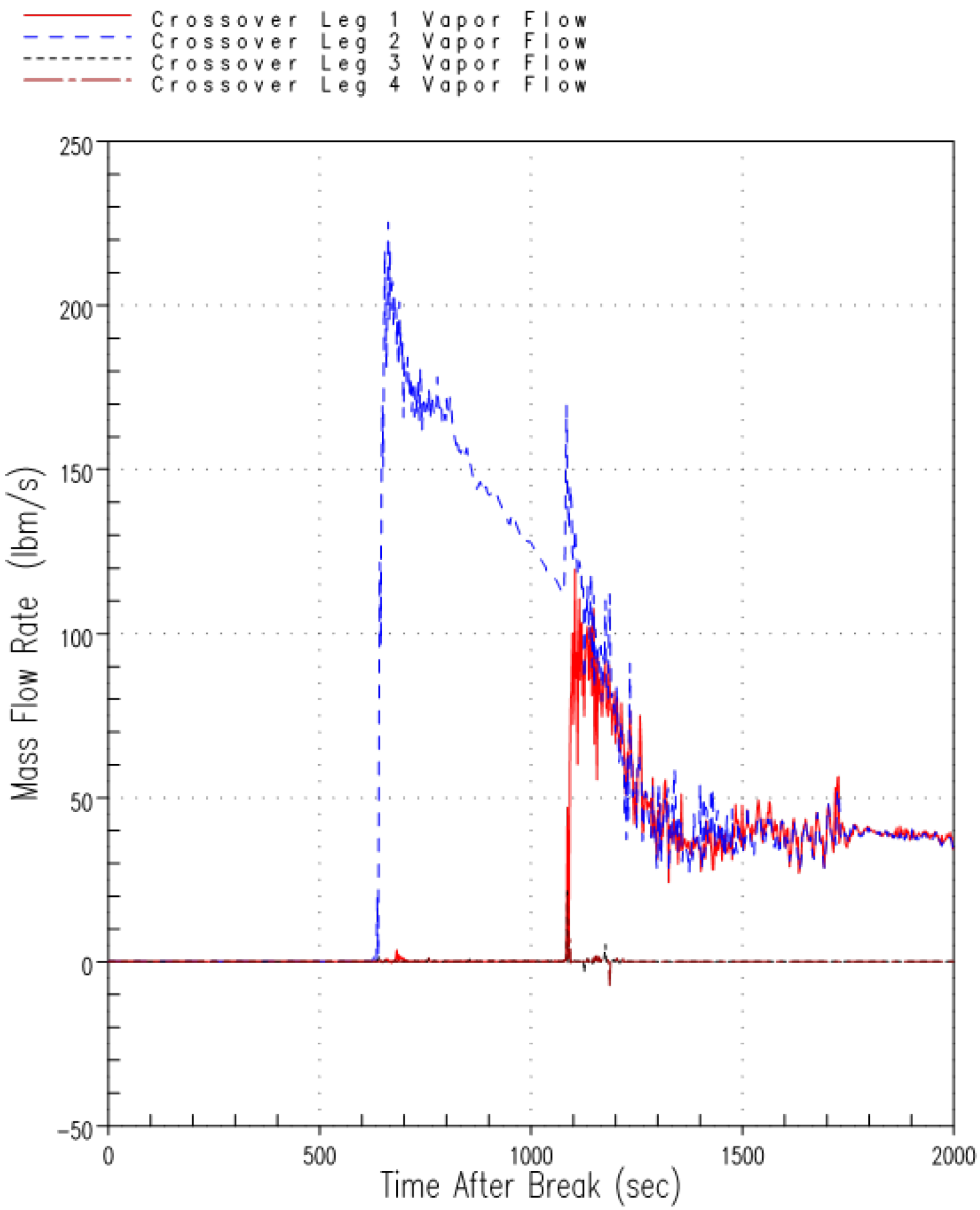

Figure 6A: Comanche Peak Unit 1 Vapor Mass Flow Rate through the Crossover Legs for the Region I Analysis PCT Case

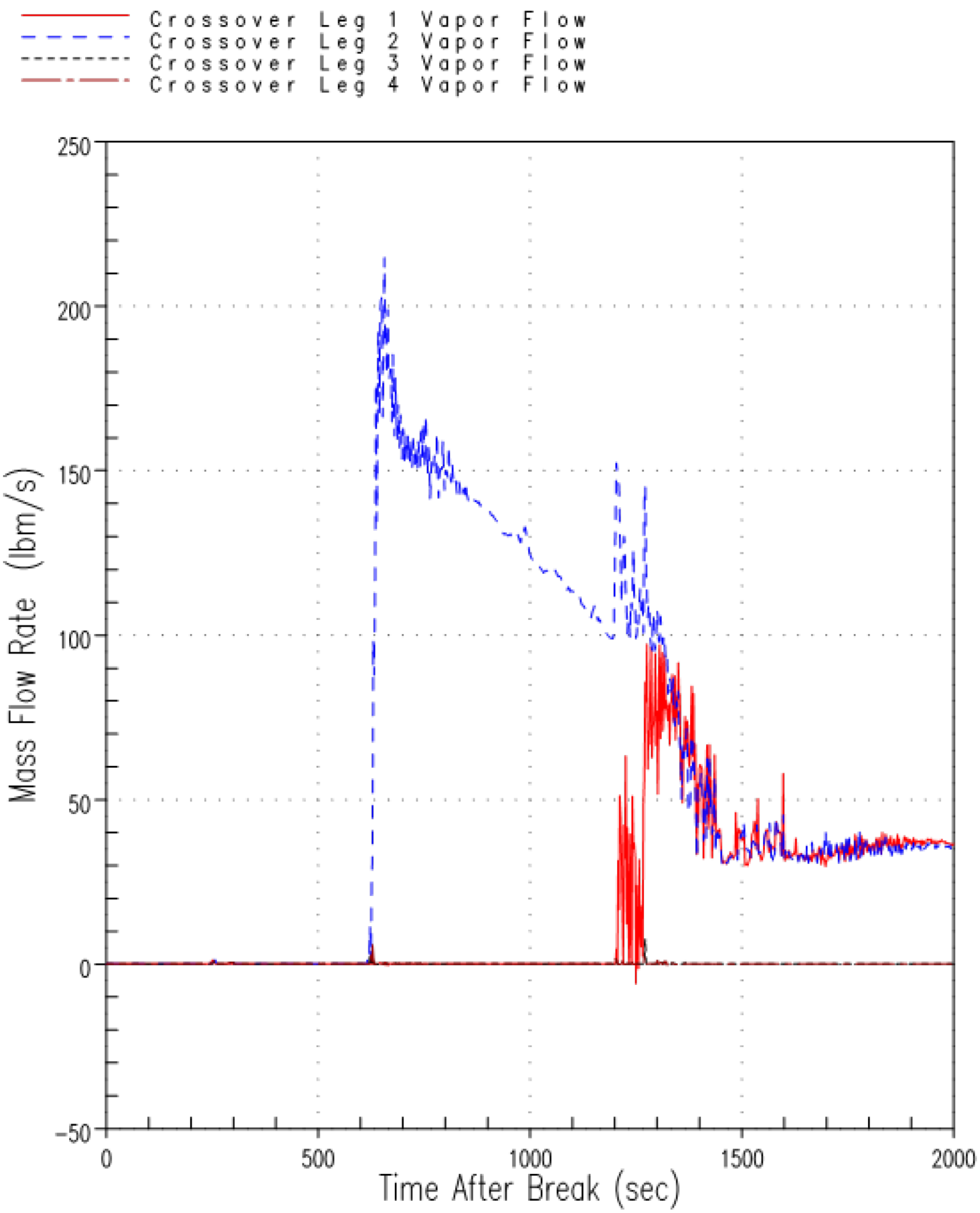

Figure 6B: Comanche Peak Unit 2 Vapor Mass Flow Rate through the Crossover Legs for the Region I Analysis PCT Case

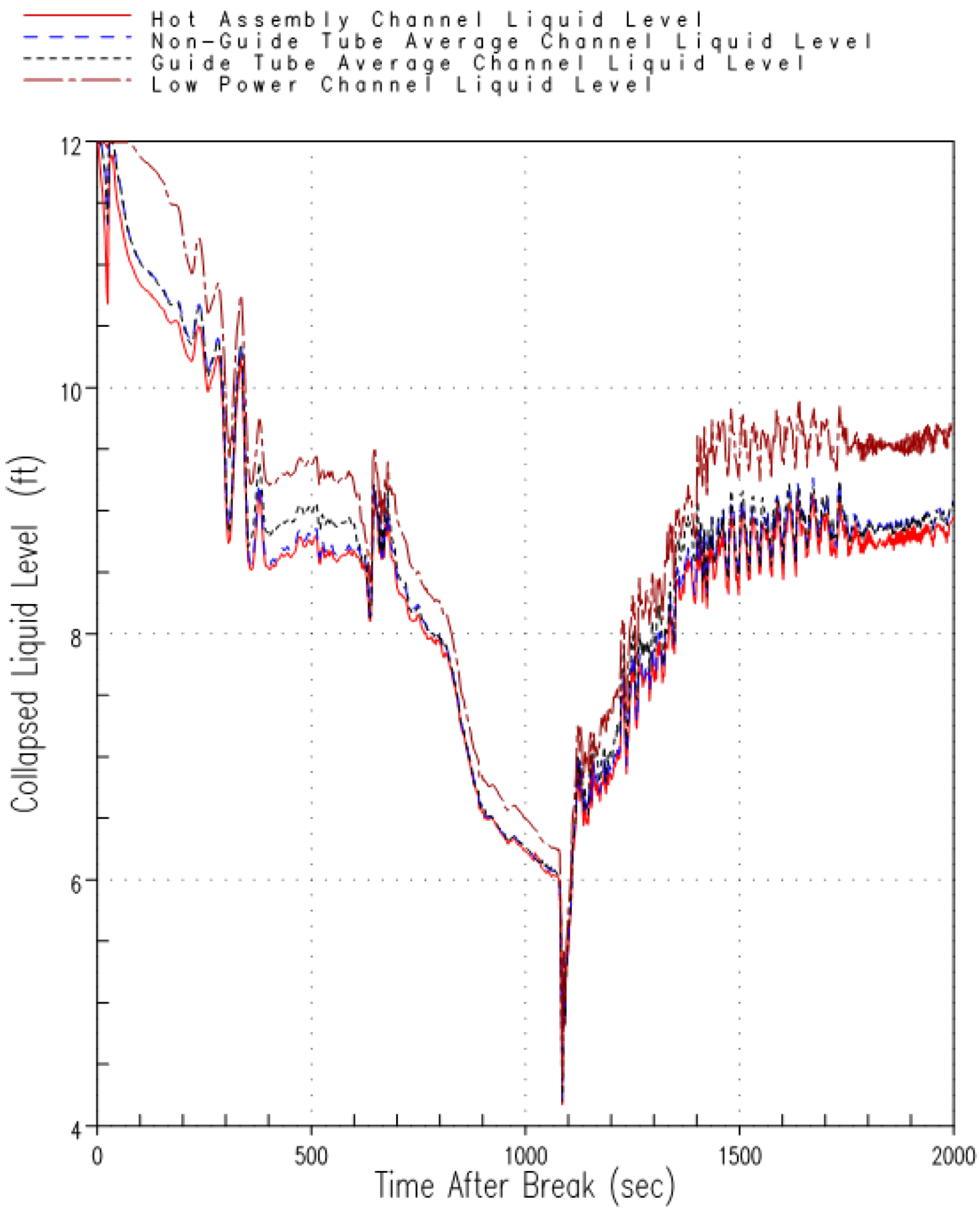

Figure 7A: Comanche Peak Unit 1 Core Collapsed Liquid Levels (Relative to Bottom of Active Fuel) for the Region I Analysis PCT Case

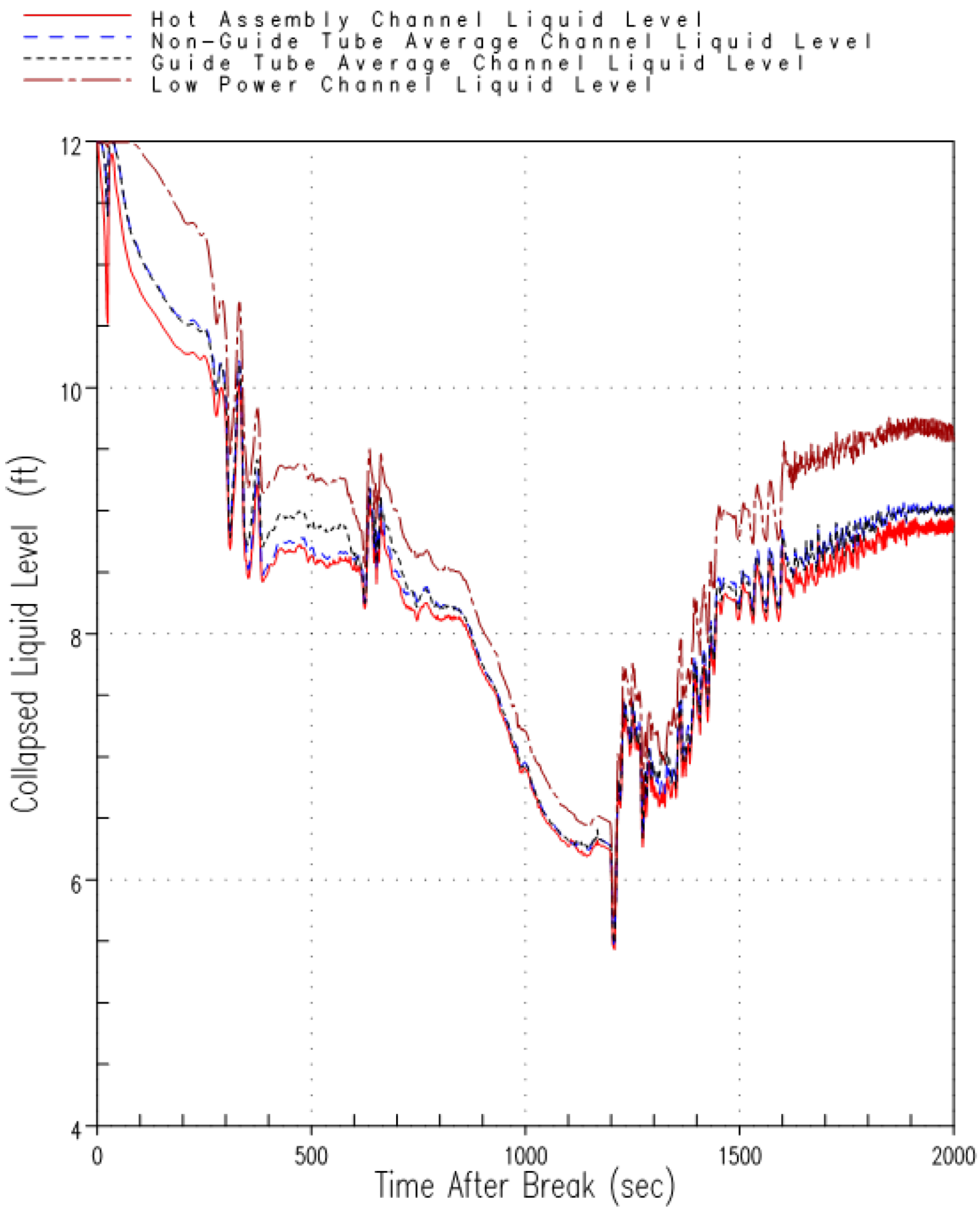

Figure 7B: Comanche Peak Unit 2 Core Collapsed Liquid Levels (Relative to Bottom of Active Fuel) for the Region I Analysis PCT Case

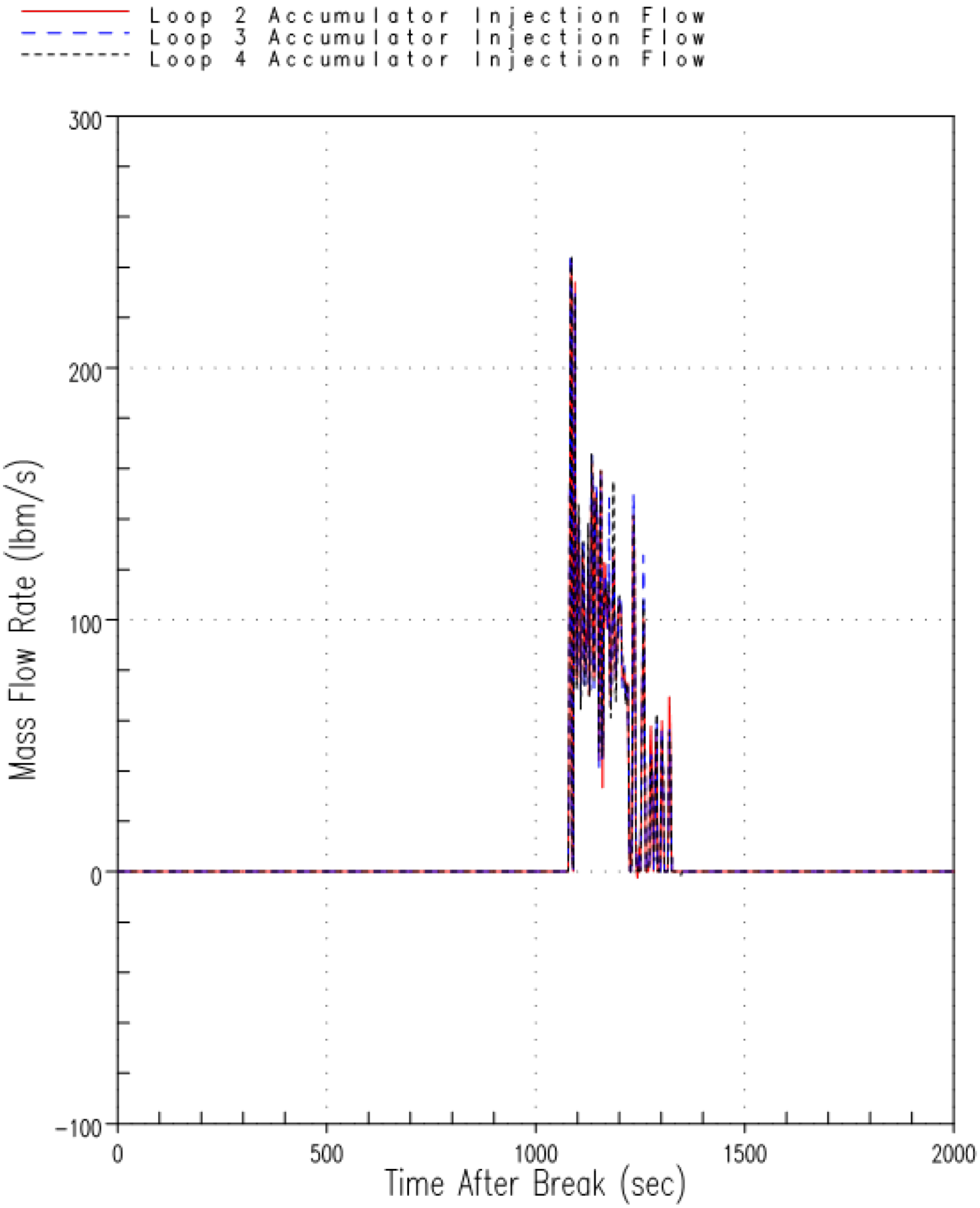

Figure 8A: Comanche Peak Unit 1 Accumulator Injection Flow for the Region I Analysis PCT Case

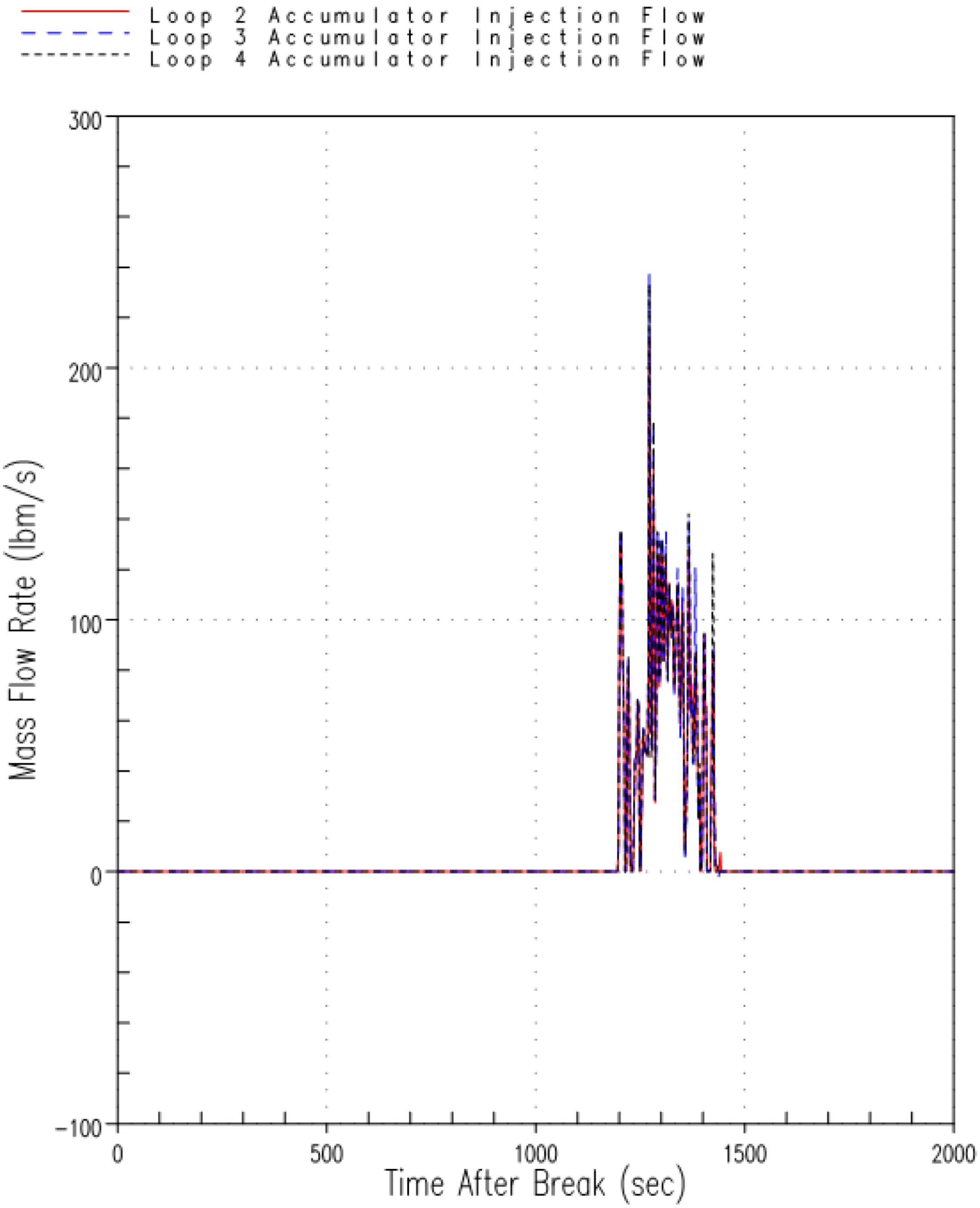

Figure 8B: Comanche Peak Unit 2 Accumulator Injection Flow for the Region I Analysis PCT Case

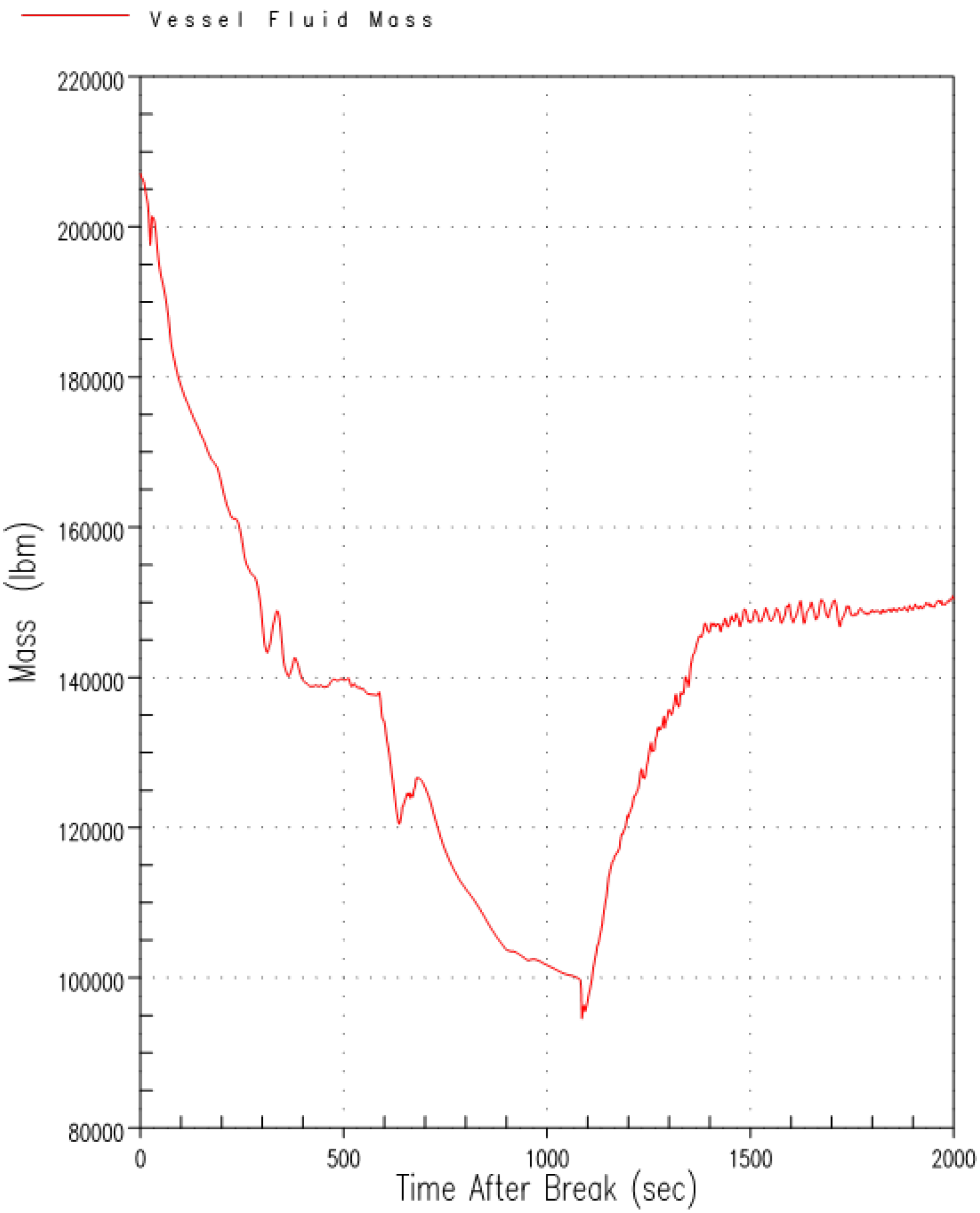

Figure 9A: Comanche Peak Unit 1 Vessel Fluid Mass for the Region I Analysis PCT Case

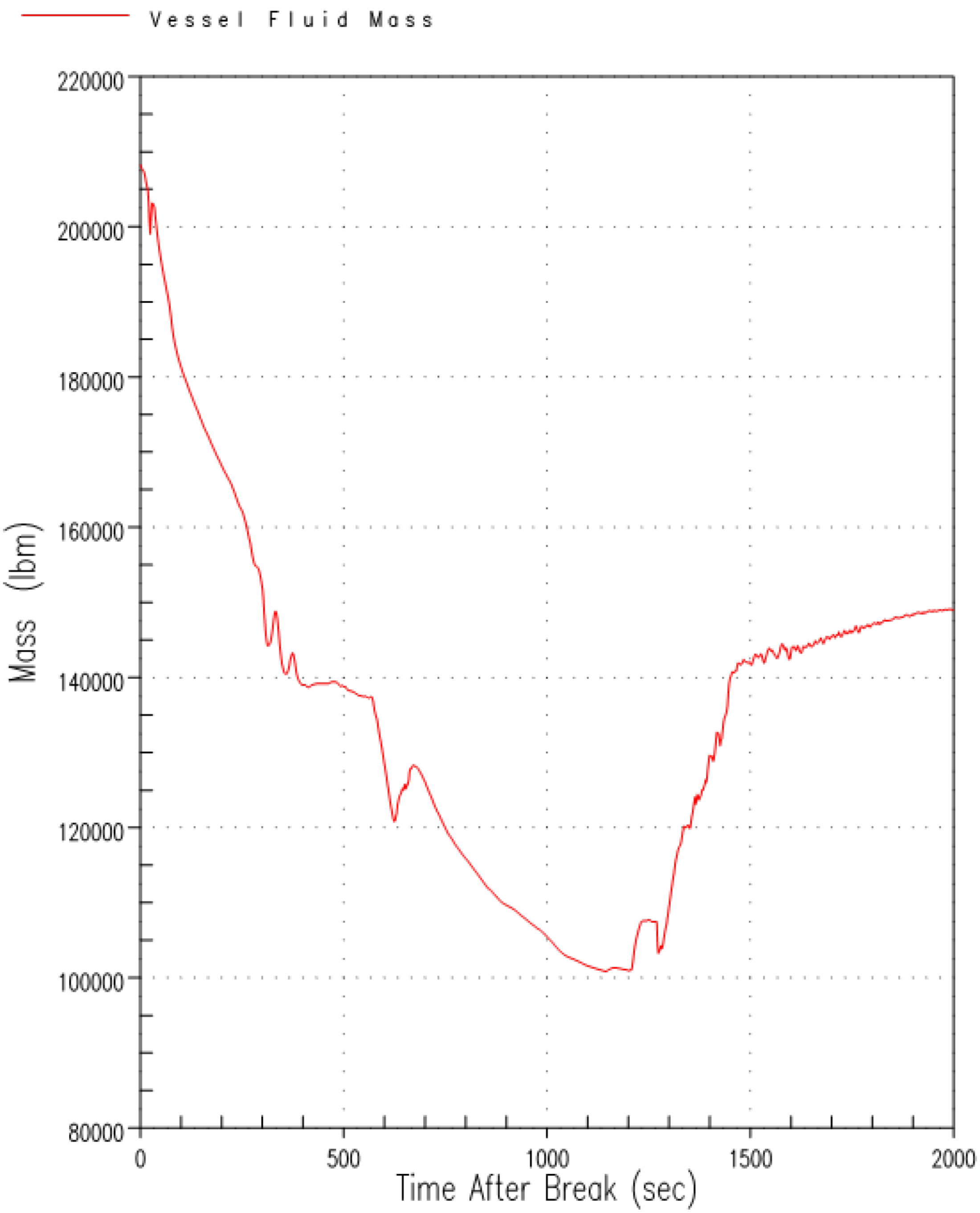

Figure 9B: Comanche Peak Unit 2 Vessel Fluid Mass for the Region I Analysis PCT Case

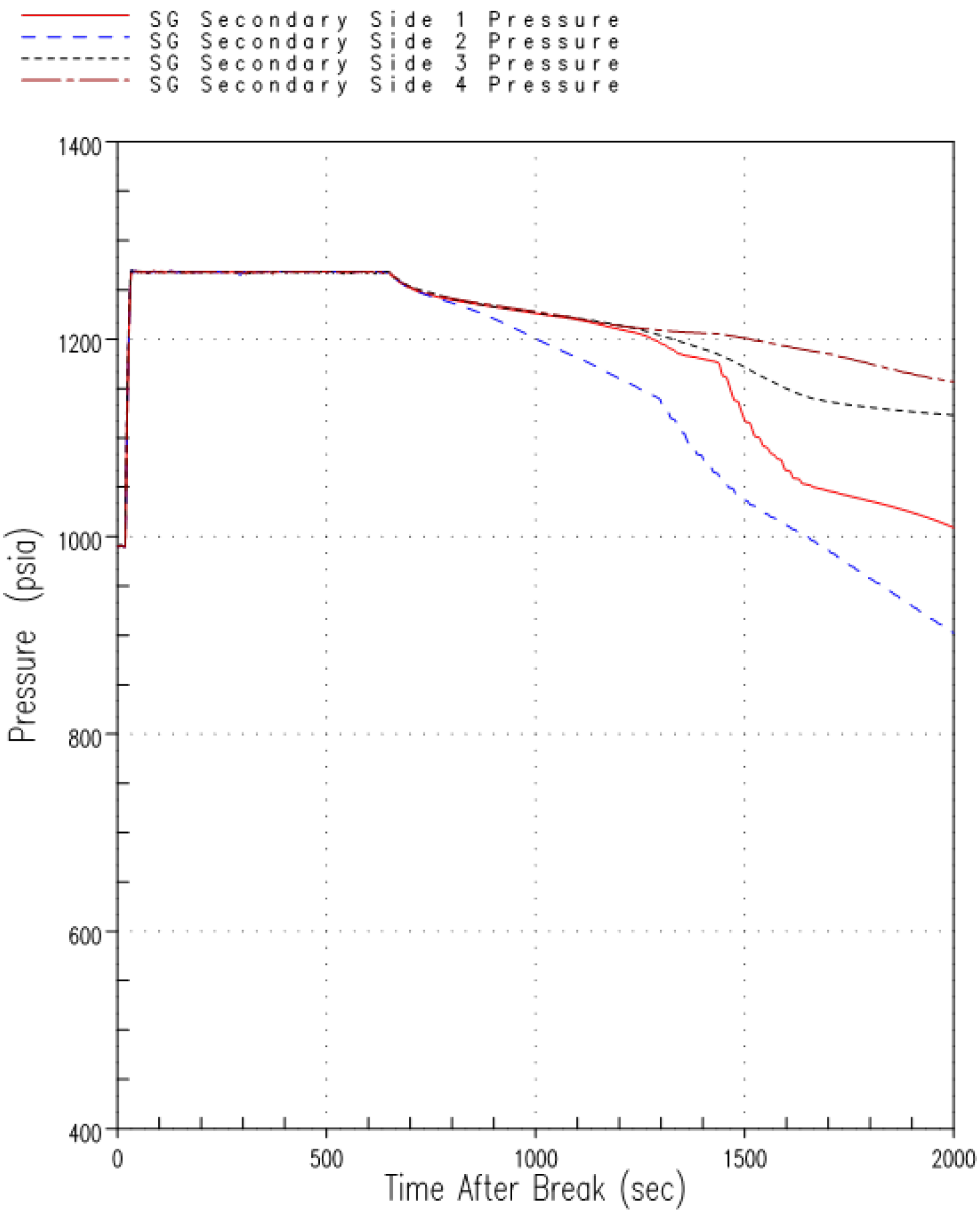

Figure 10A: Comanche Peak Unit 1 Steam Generator Secondary Side Pressure for the Region I Analysis PCT Case

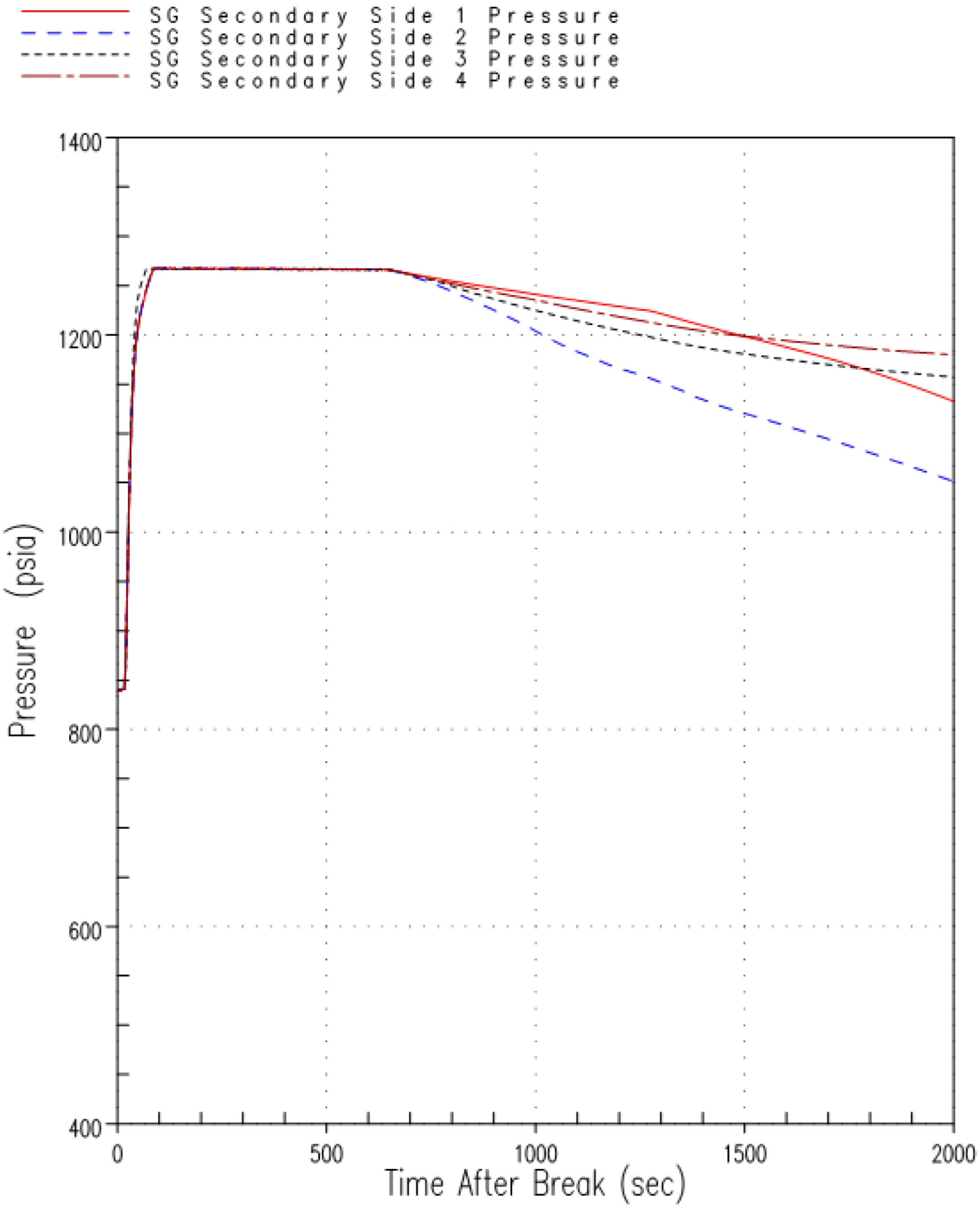

Figure 10B: Comanche Peak Unit 2 Steam Generator Secondary Side Pressure for the Region I Analysis PCT Case

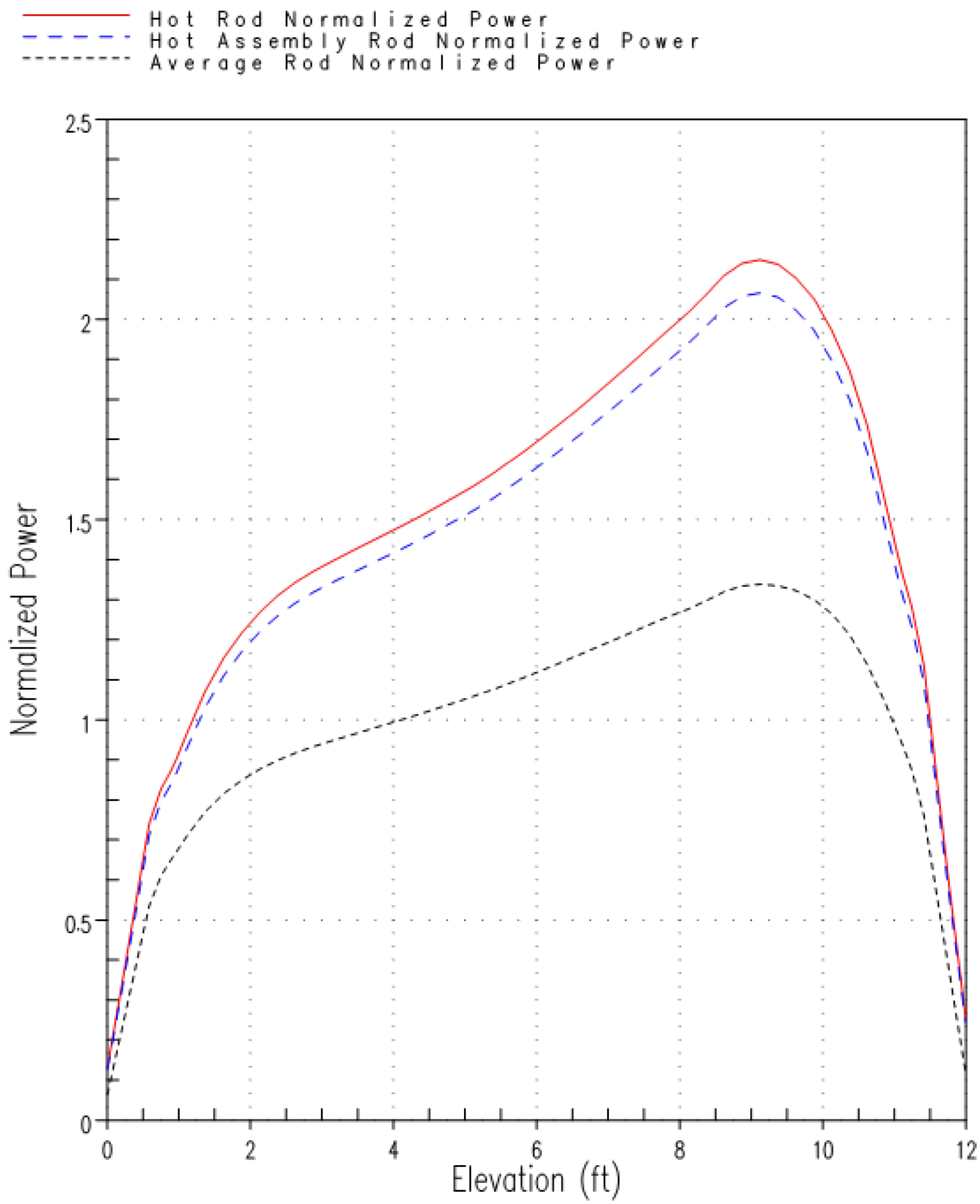

Figure 11A: Comanche Peak Unit 1 Normalized Core Power Shapes for the Region I Analysis PCT Case

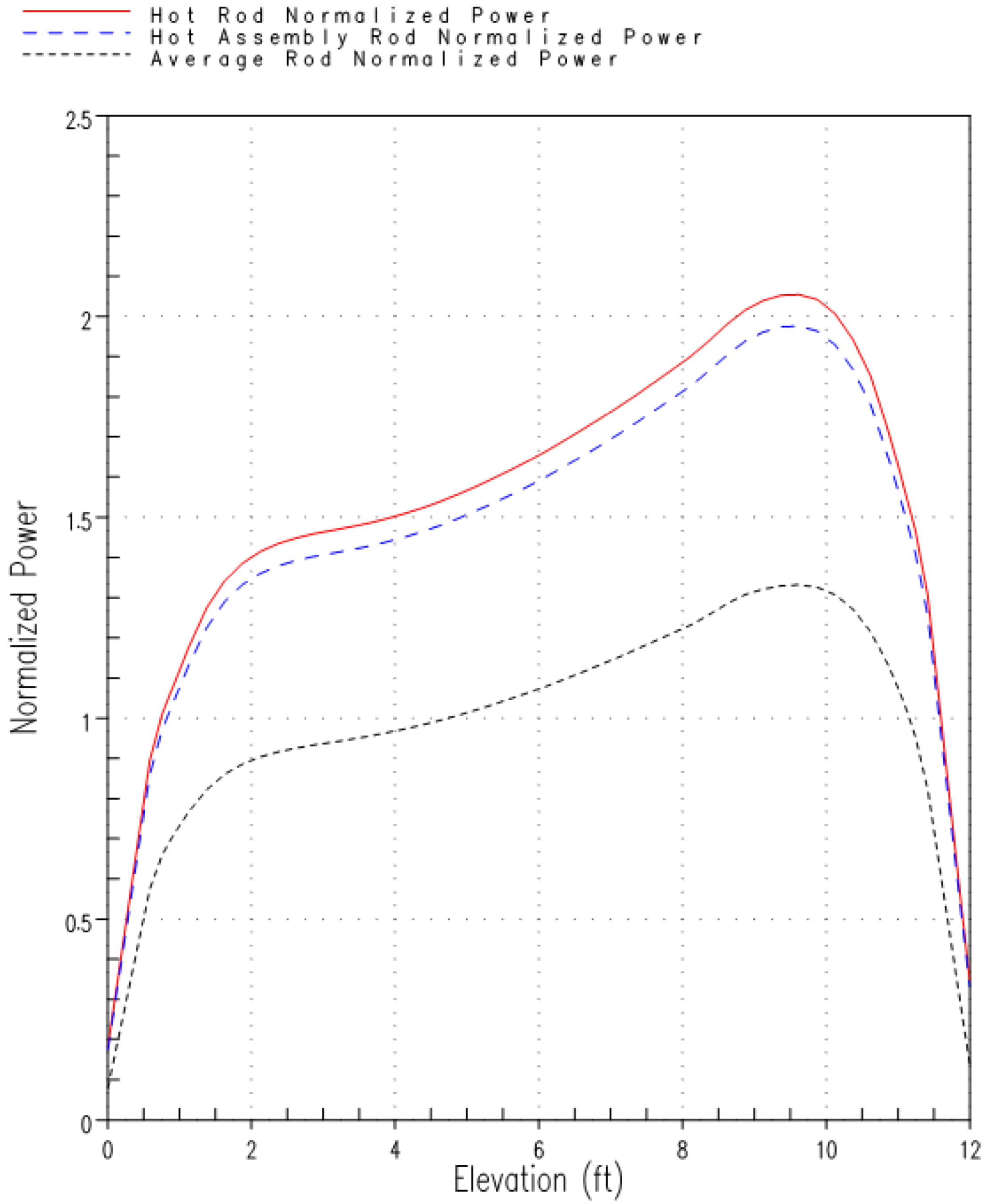

Figure 11B: Comanche Peak Unit 2 Normalized Core Power Shapes for the Region I Analysis PCT Case

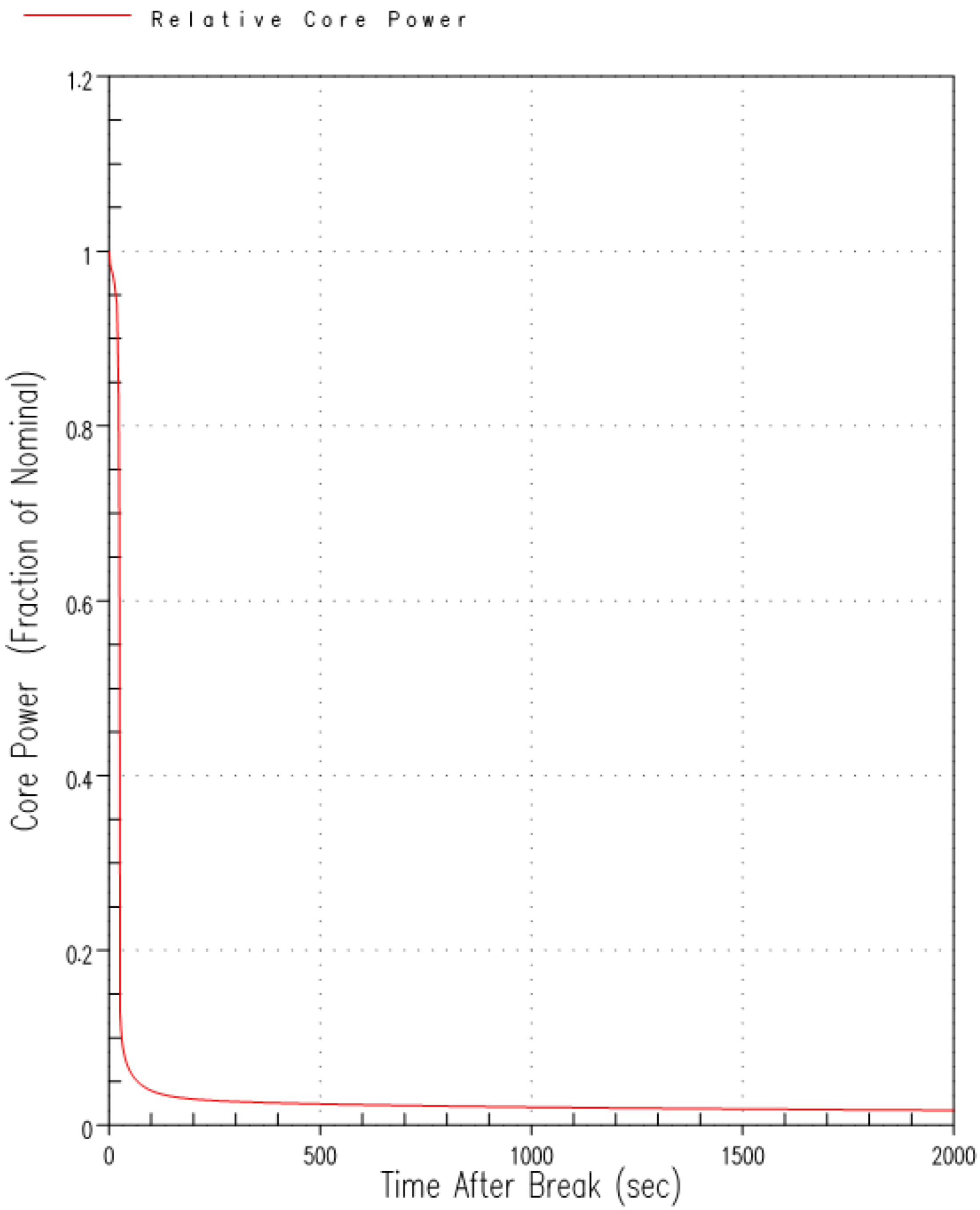

Figure 12A: Comanche Peak Unit 1 Relative Core Power for the Region I Analysis PCT Case

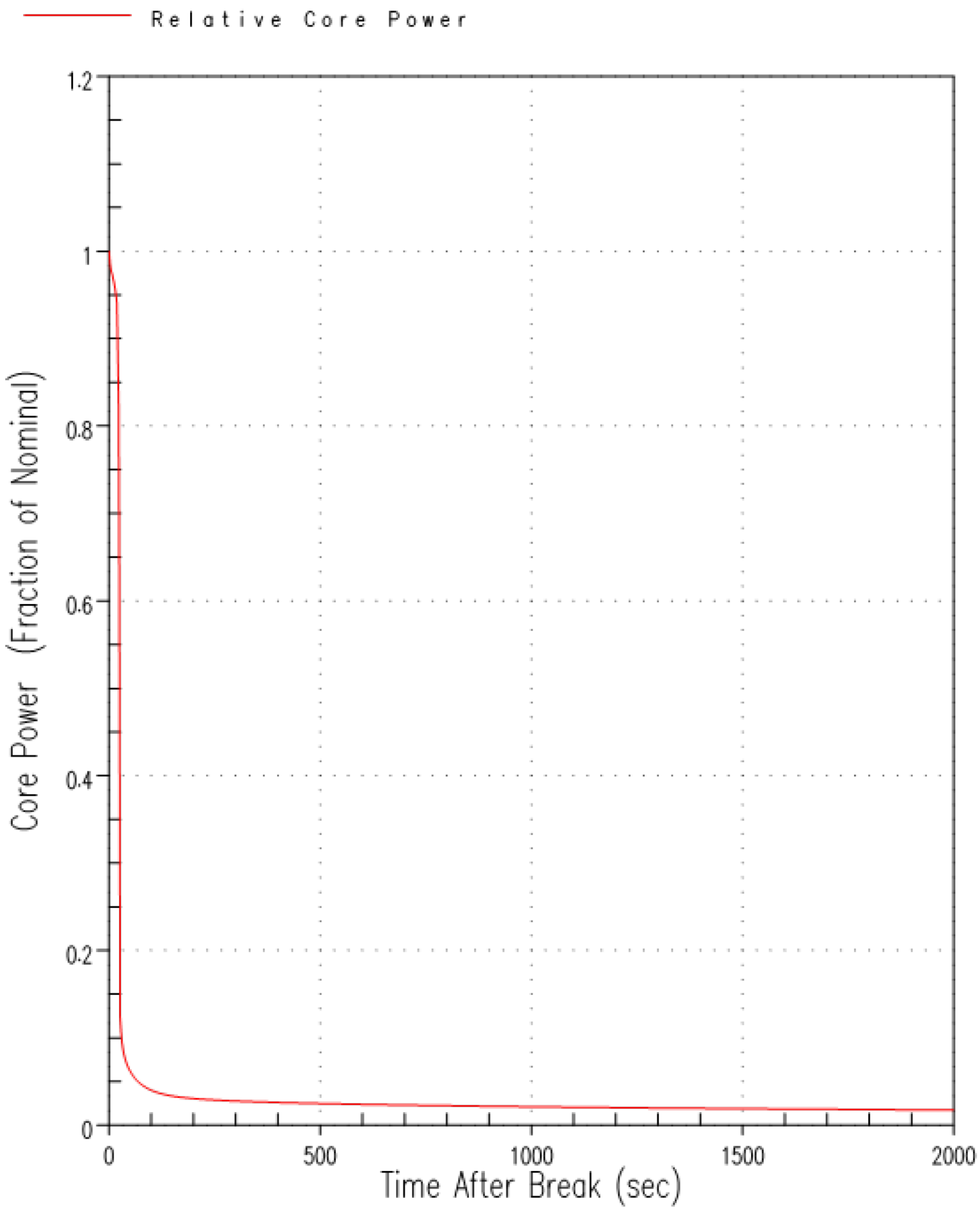

Figure 12B: Comanche Peak Unit 2 Relative Core Power for the Region I Analysis PCT Case

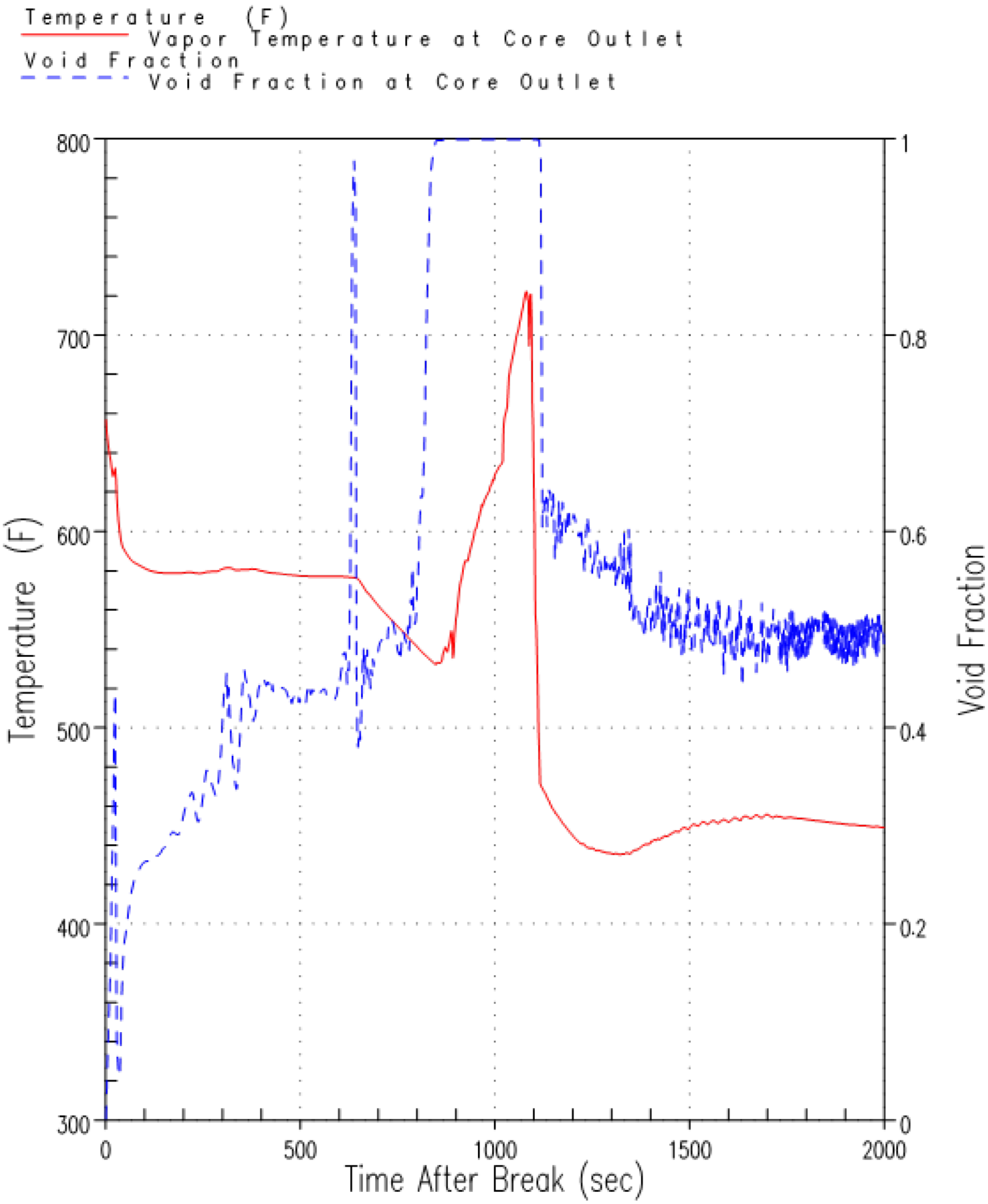

Figure 13A: Comanche Peak Unit 1 Vapor Temperature and Void Fraction at Core Outlet (Hot Assembly Channel) for the Region I Analysis PCT Case

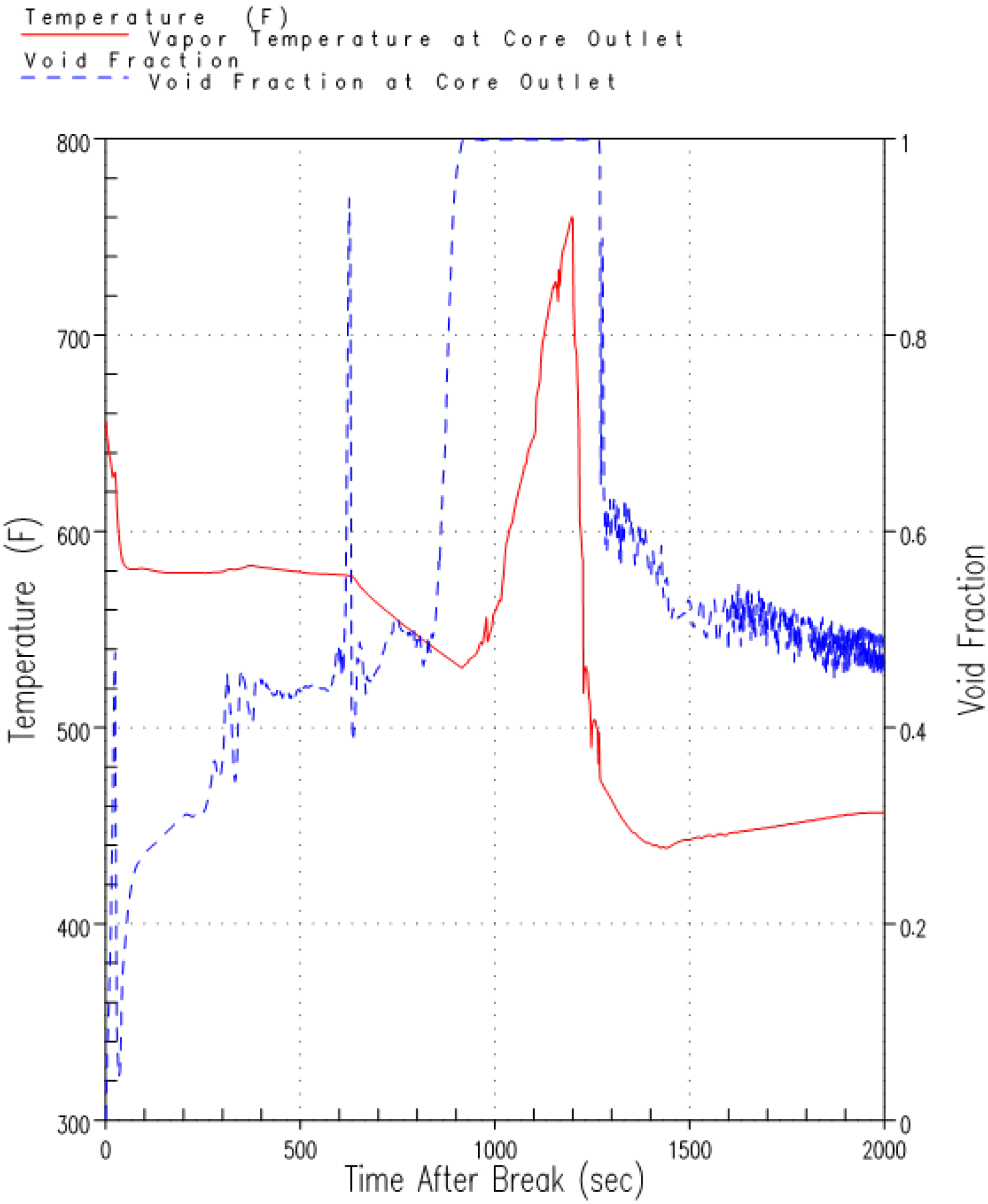

Figure 13B: Comanche Peak Unit 2 Vapor Temperature and Void Fraction at Core Outlet (Hot Assembly Channel) for the Region I Analysis PCT Case

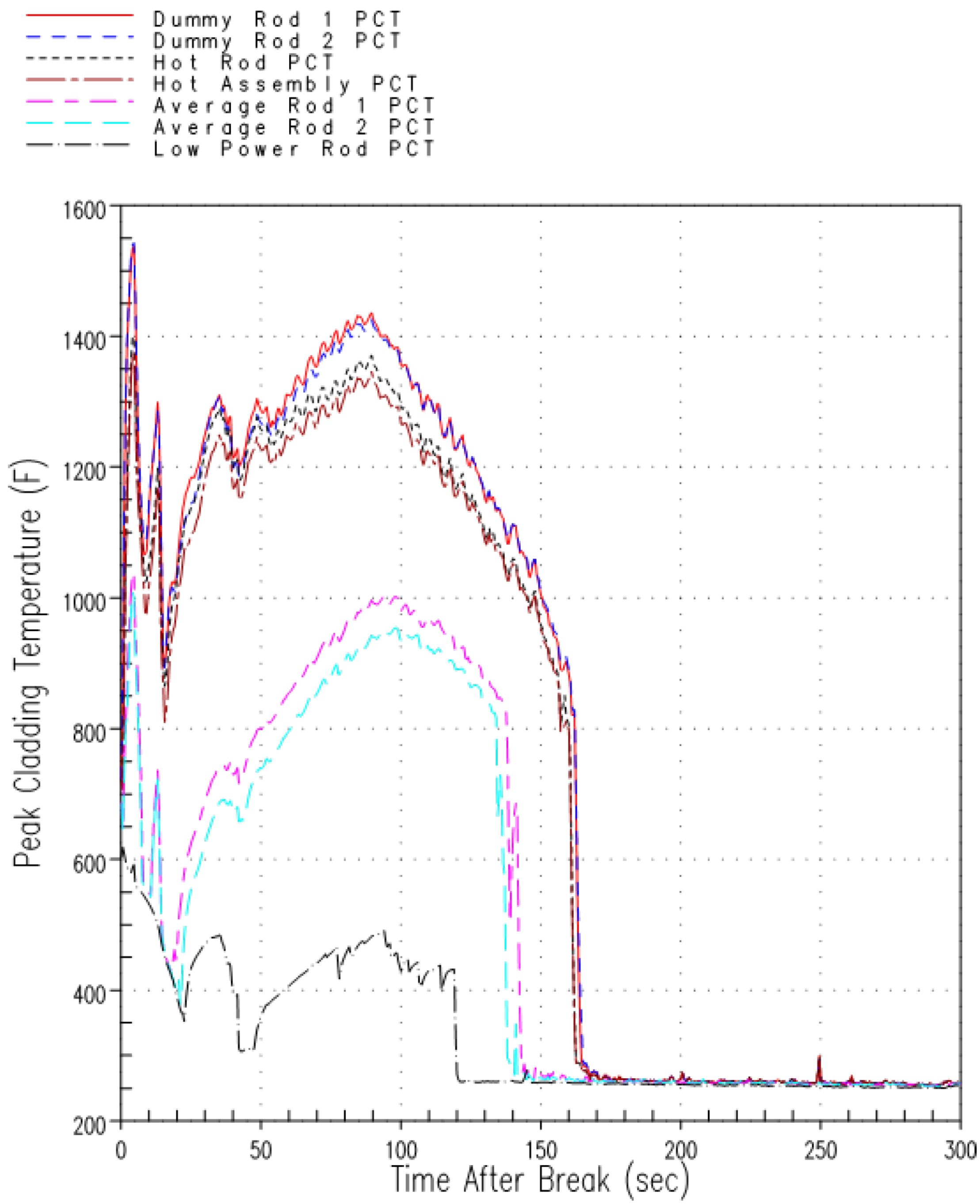

Figure 14A: Comanche Peak Unit 1 Peak Cladding Temperature for all Rods for the Region II Analysis PCT Case

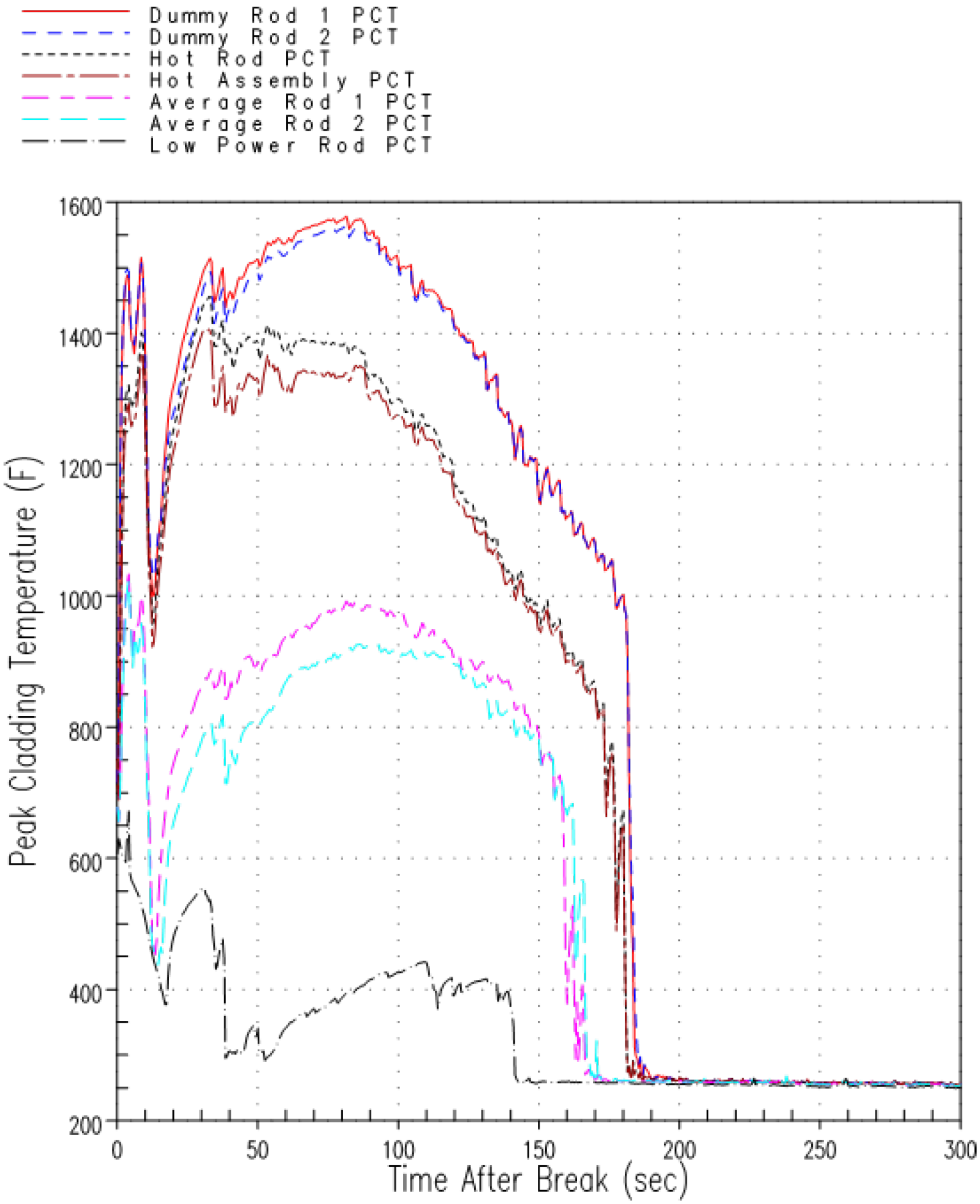

Figure 14B: Comanche Peak Unit 2 Peak Cladding Temperature for all Rods for the Region II Analysis PCT Case

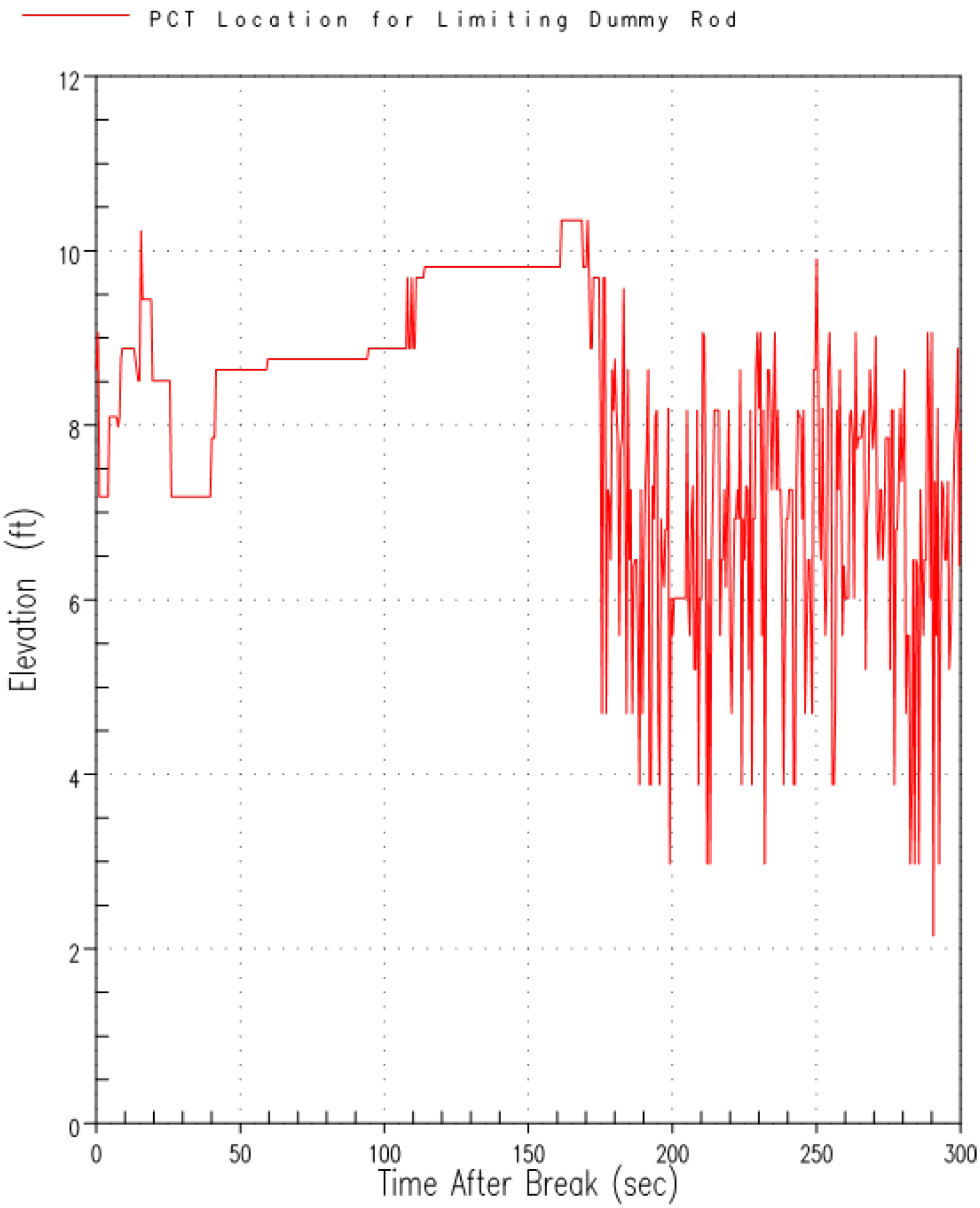

Figure 15A: Comanche Peak Unit 1 Peak Cladding Temperature Elevation (Relative to Bottom of Active Fuel) for the Region II Analysis PCT Case

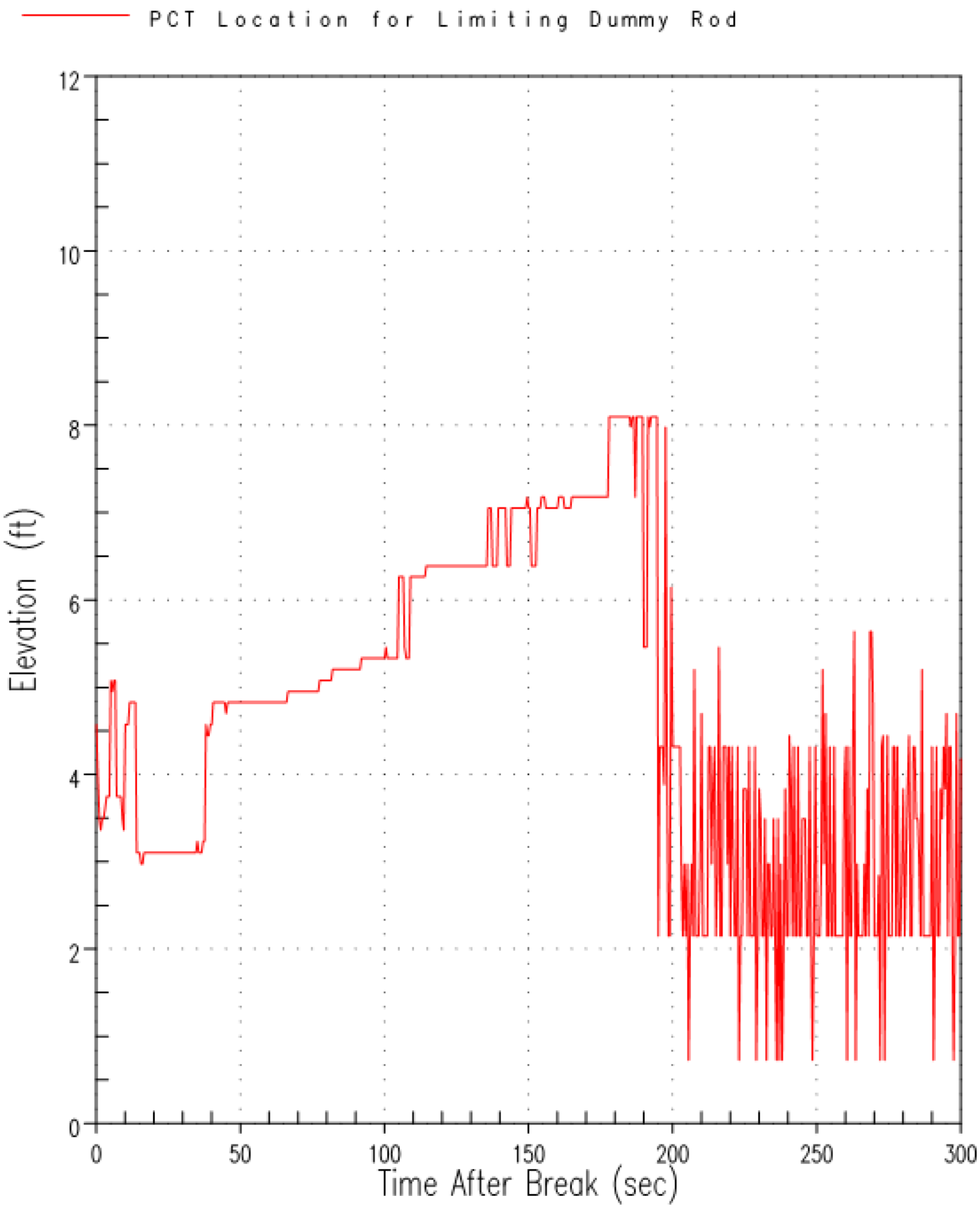

Figure 15B: Comanche Peak Unit 2 Peak Cladding Temperature Elevation (Relative to Bottom of Active Fuel) for the Region II Analysis PCT Case

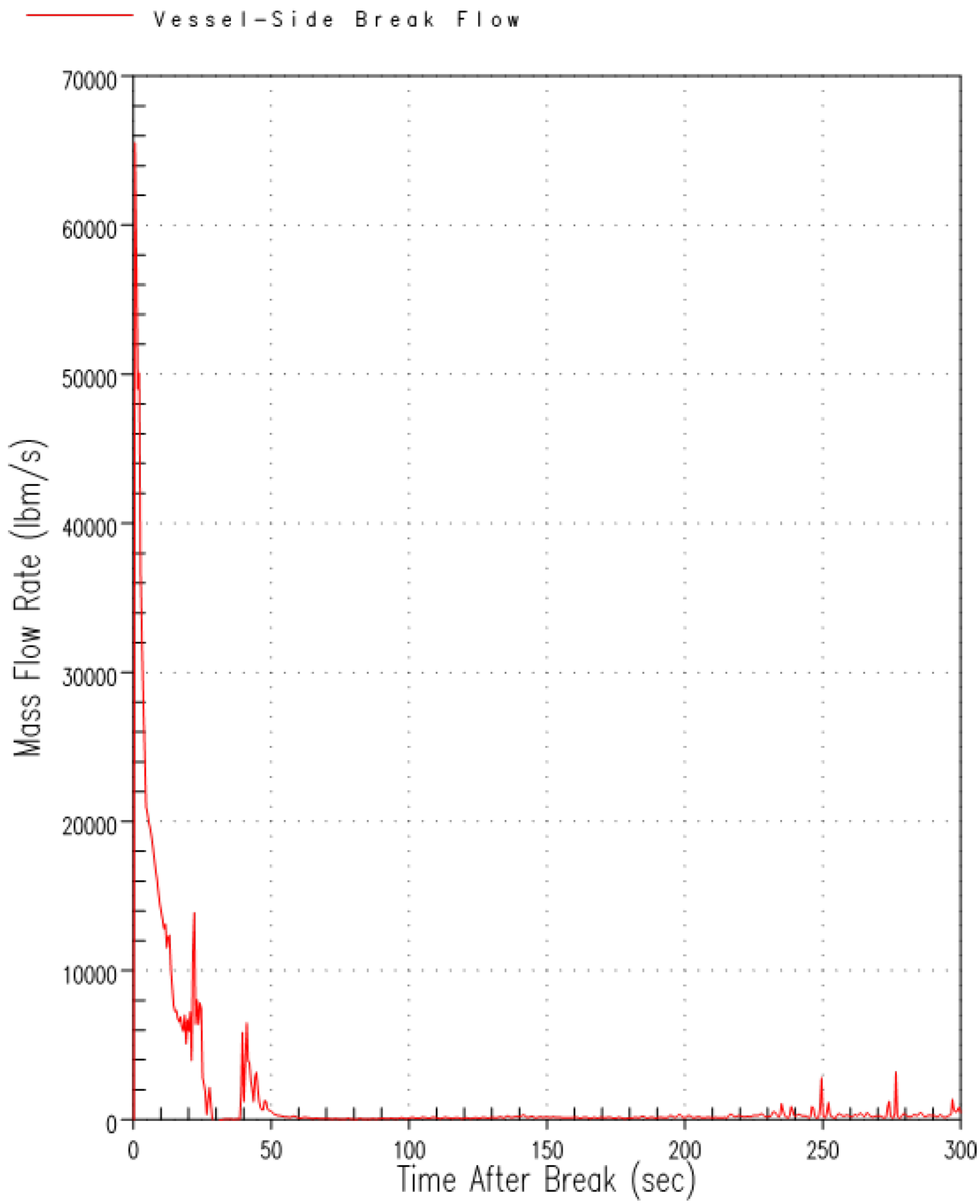

Figure 16A: Comanche Peak Unit 1 Vessel-Side Break Mass Flow Rate for the Region II Analysis PCT Case

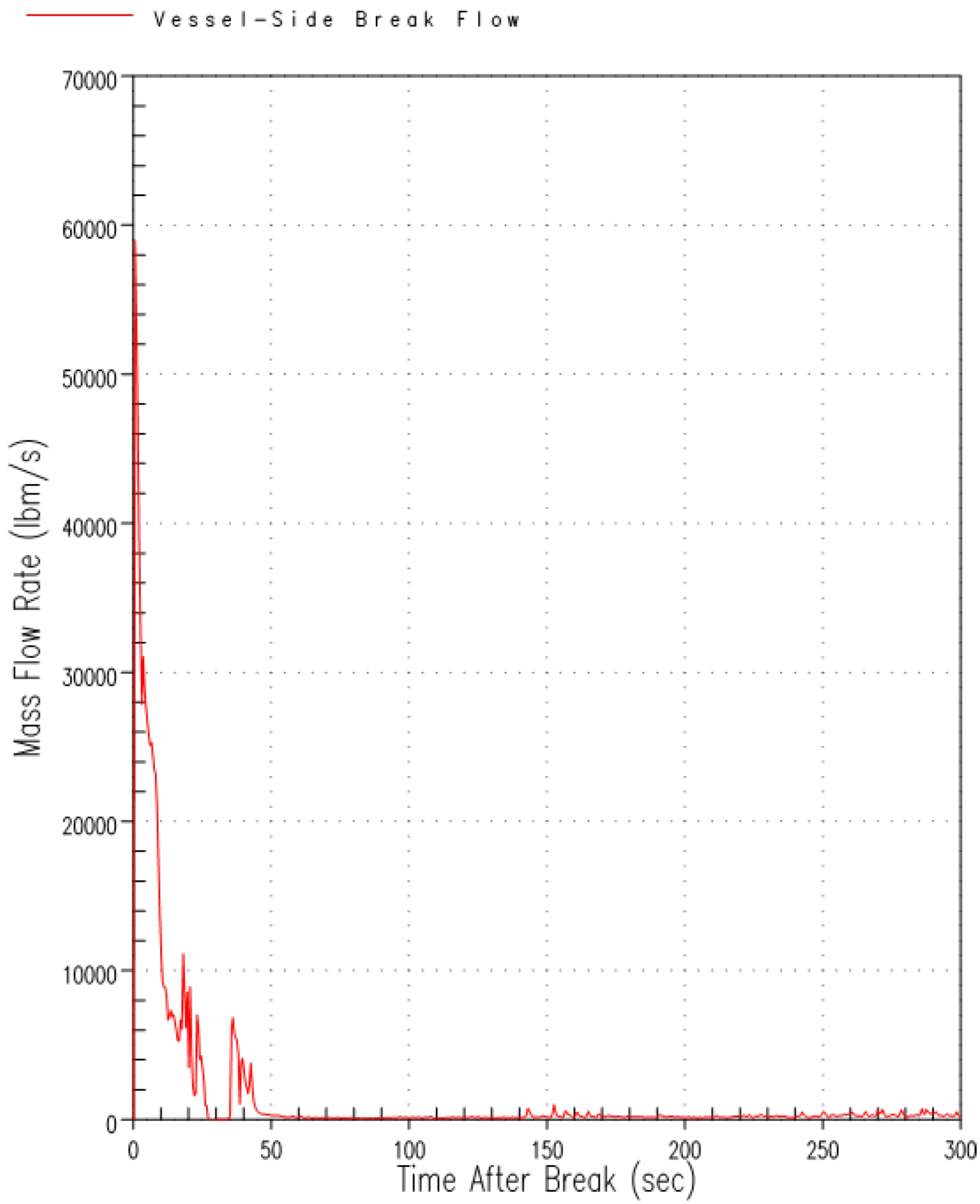

Figure 16B: Comanche Peak Unit 2 Vessel-Side Break Mass Flow Rate for the Region II Analysis PCT Case

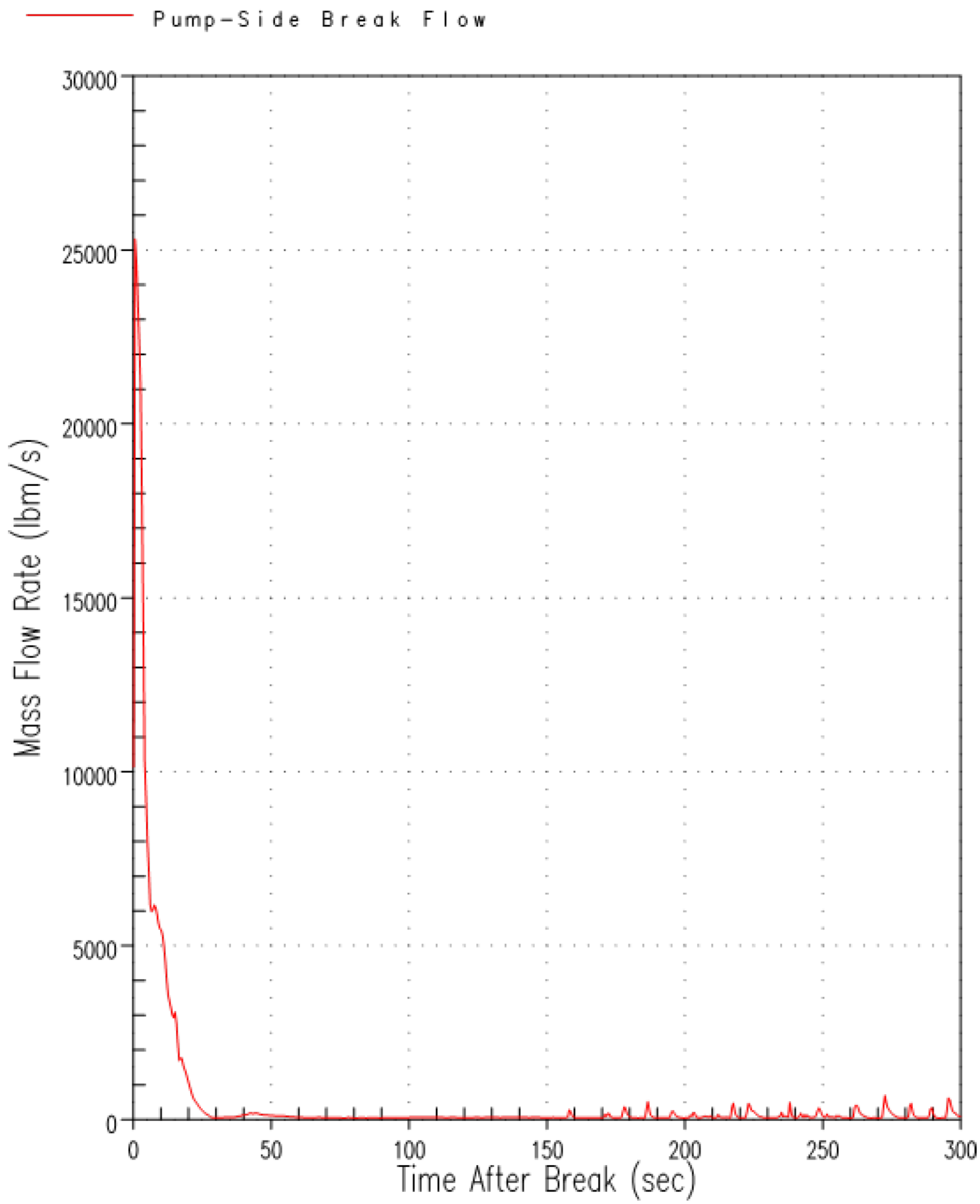

Figure 17A: Comanche Peak Unit 1 Pump-Side Break Mass Flow Rate for the Region II Analysis PCT Case

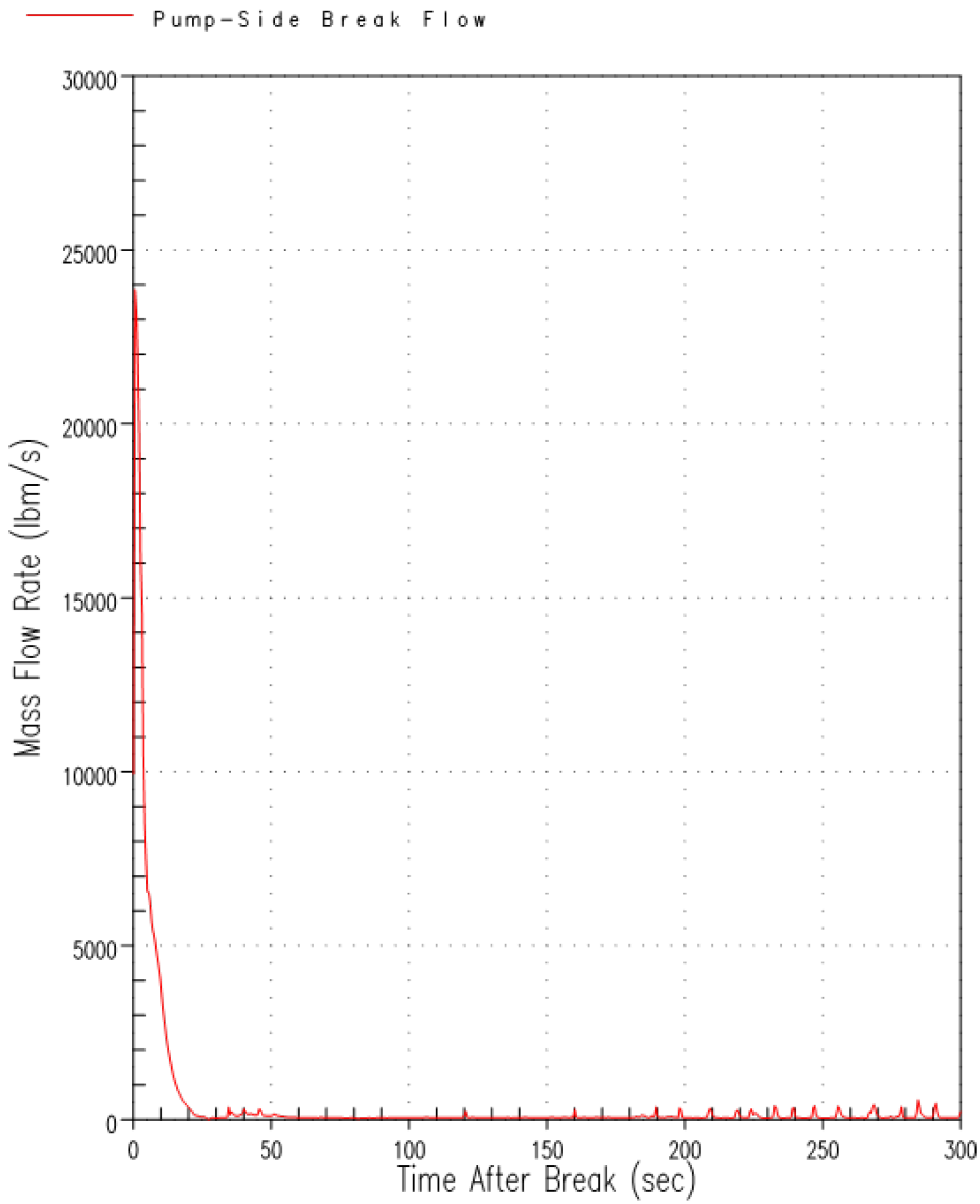

Figure 17B: Comanche Peak Unit 2 Pump-Side Break Mass Flow Rate for the Region II Analysis PCT Case

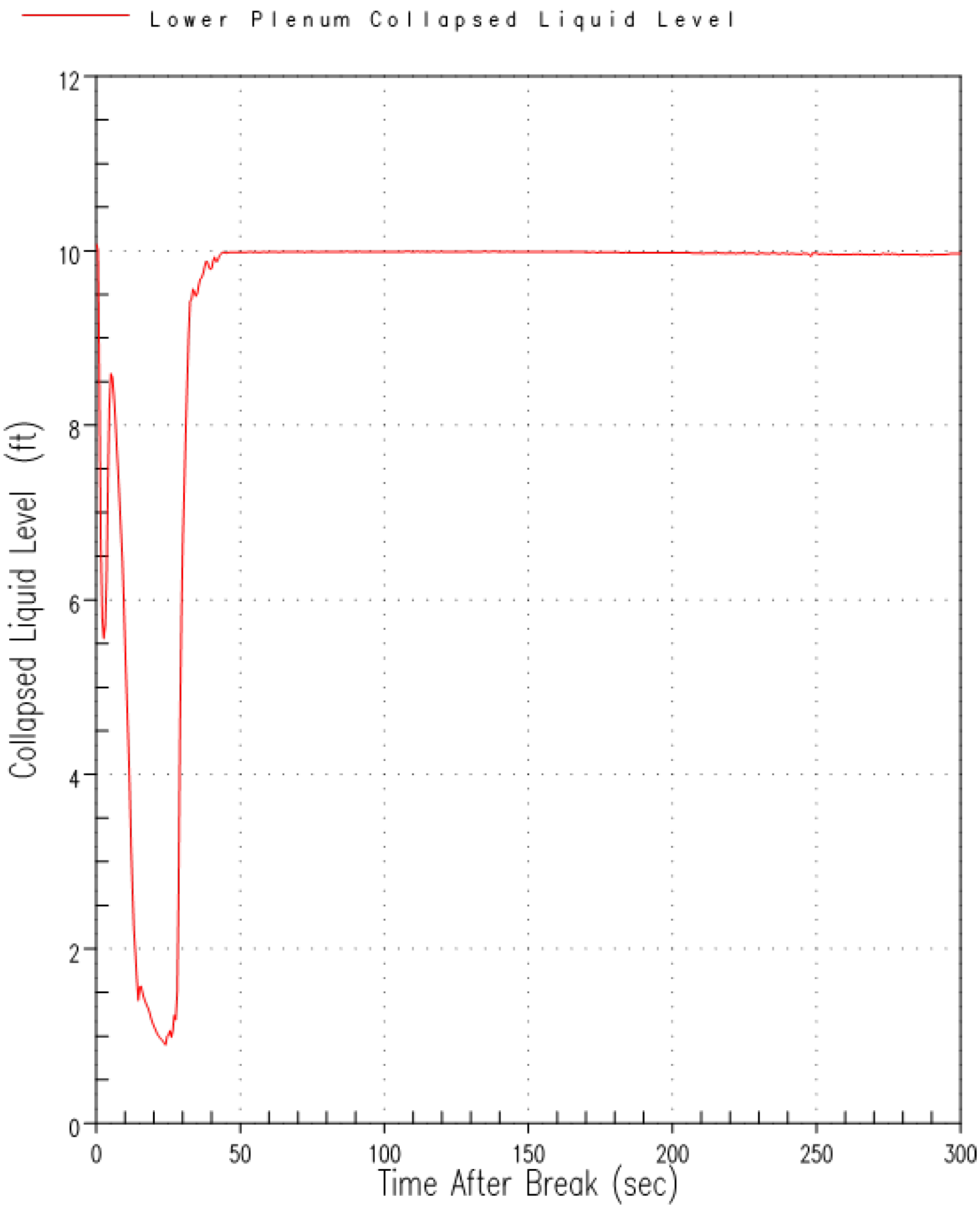

Figure 18A: Comanche Peak Unit 1 Lower Plenum Collapsed Liquid Level (Relative to Inside Bottom of Vessel) for the Region II Analysis PCT Case

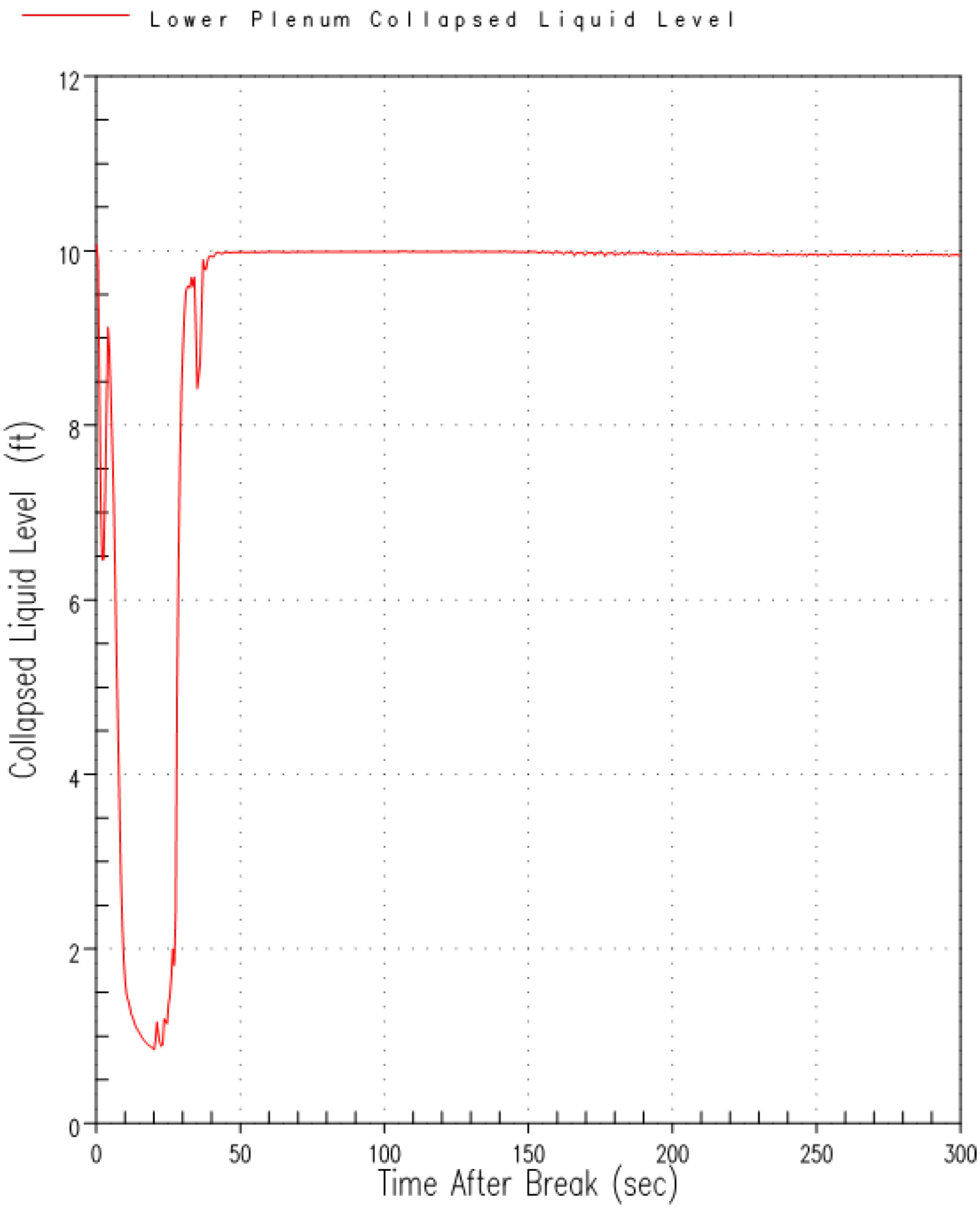

Figure 18B: Comanche Peak Unit 2 Lower Plenum Collapsed Liquid Level (Relative to Inside Bottom of Vessel) for the Region II Analysis PCT Case

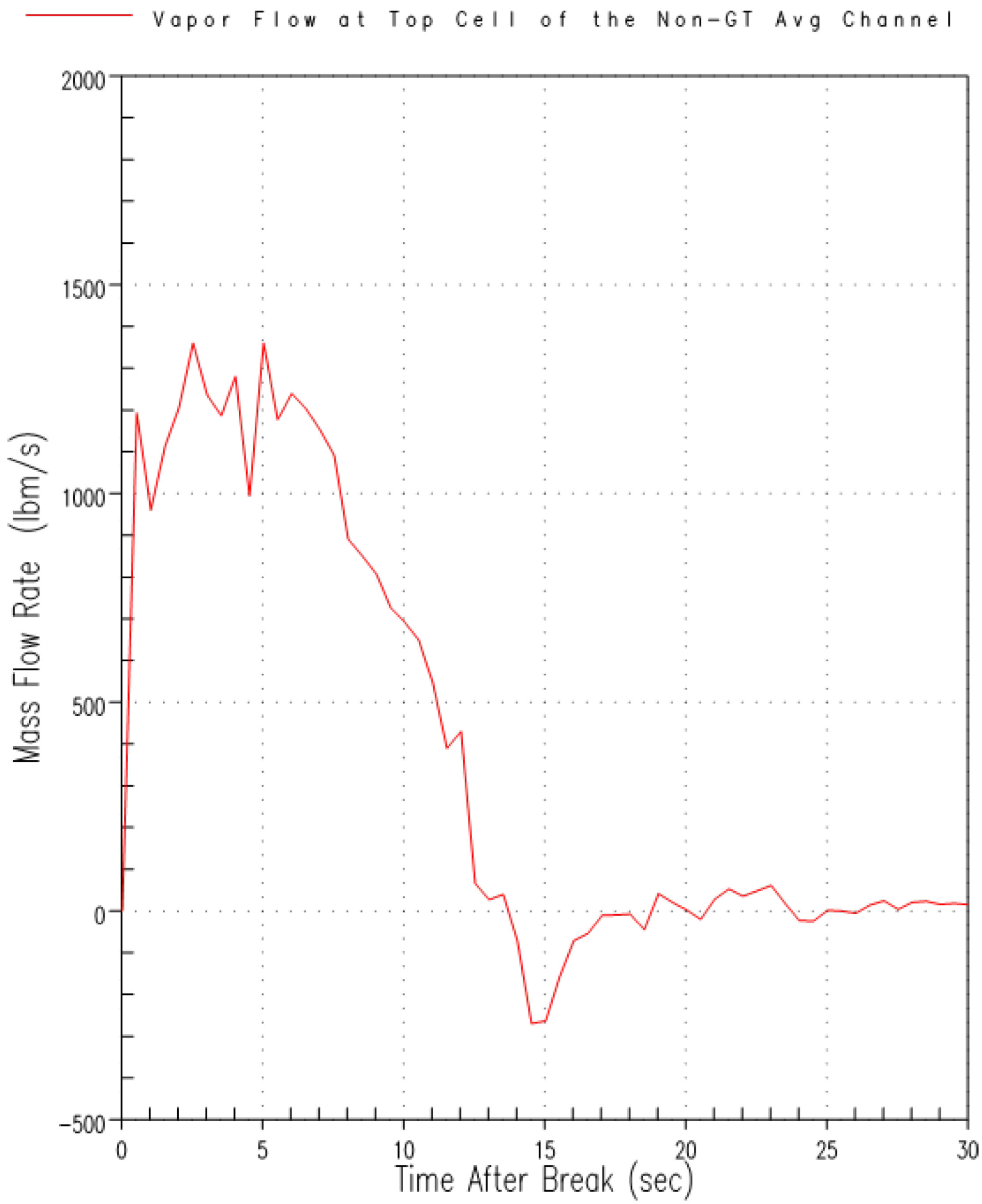

Figure 19A: Comanche Peak Unit 1 Vapor Mass Flow Rate at the Top Cell Face of the Core Average Channel not Under Guide Tubes for the Region II Analysis PCT Case

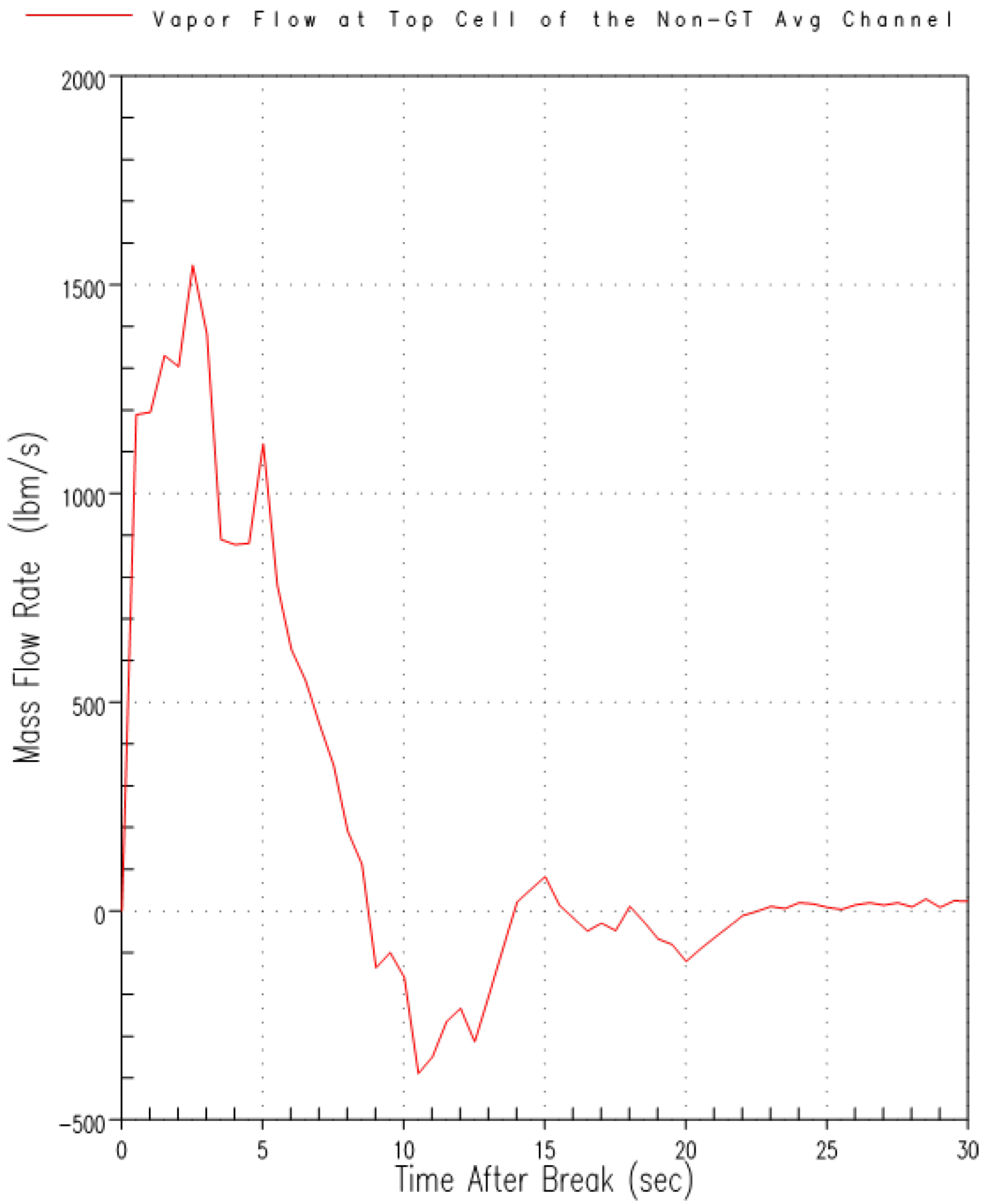

Figure 19B: Comanche Peak Unit 2 Vapor Mass Flow Rate at the Top Cell Face of the Core Average Channel not Under Guide Tubes for the Region II Analysis PCT Case

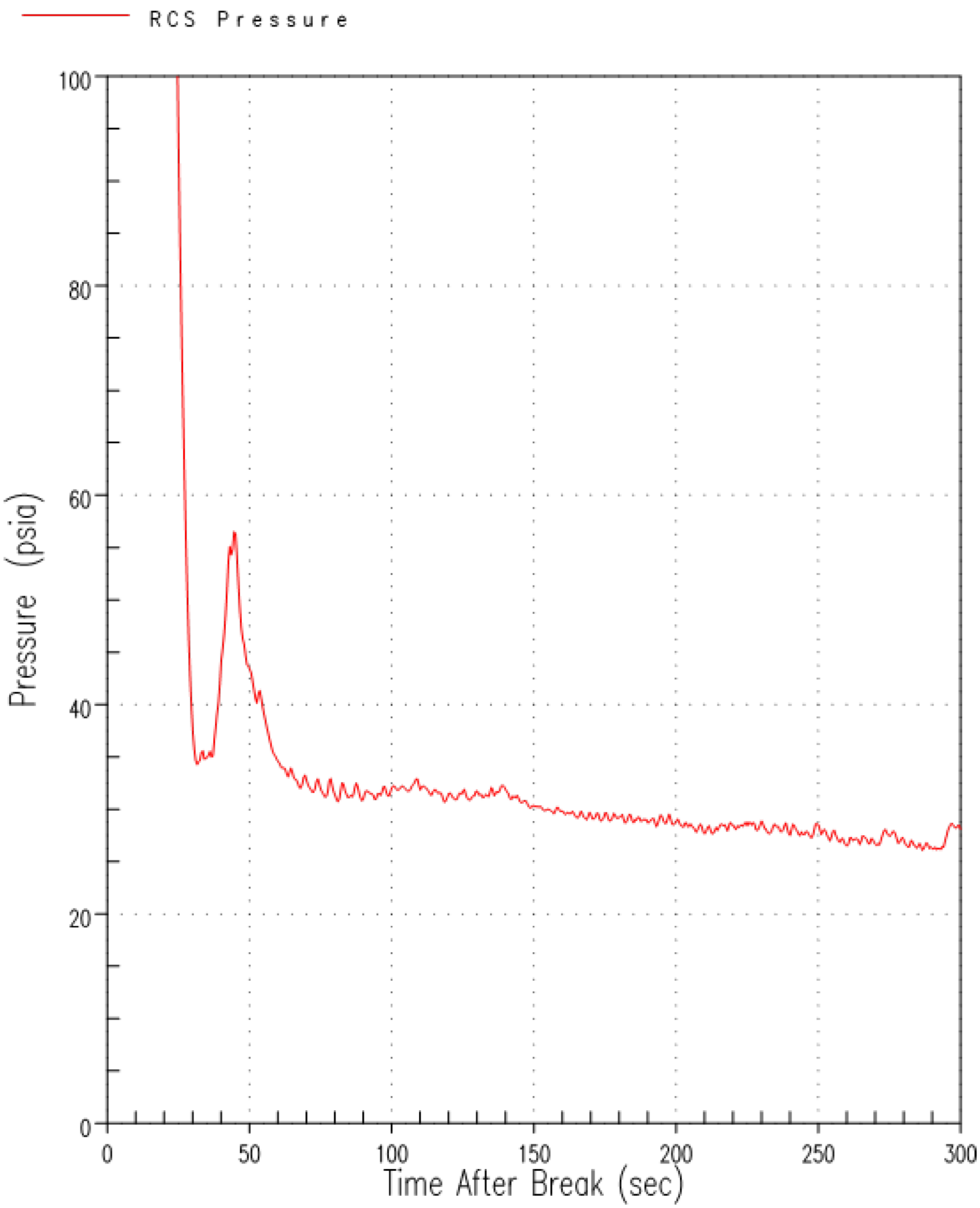

Figure 20A: Comanche Peak Unit 1 RCS Pressure for the Region II Analysis PCT Case

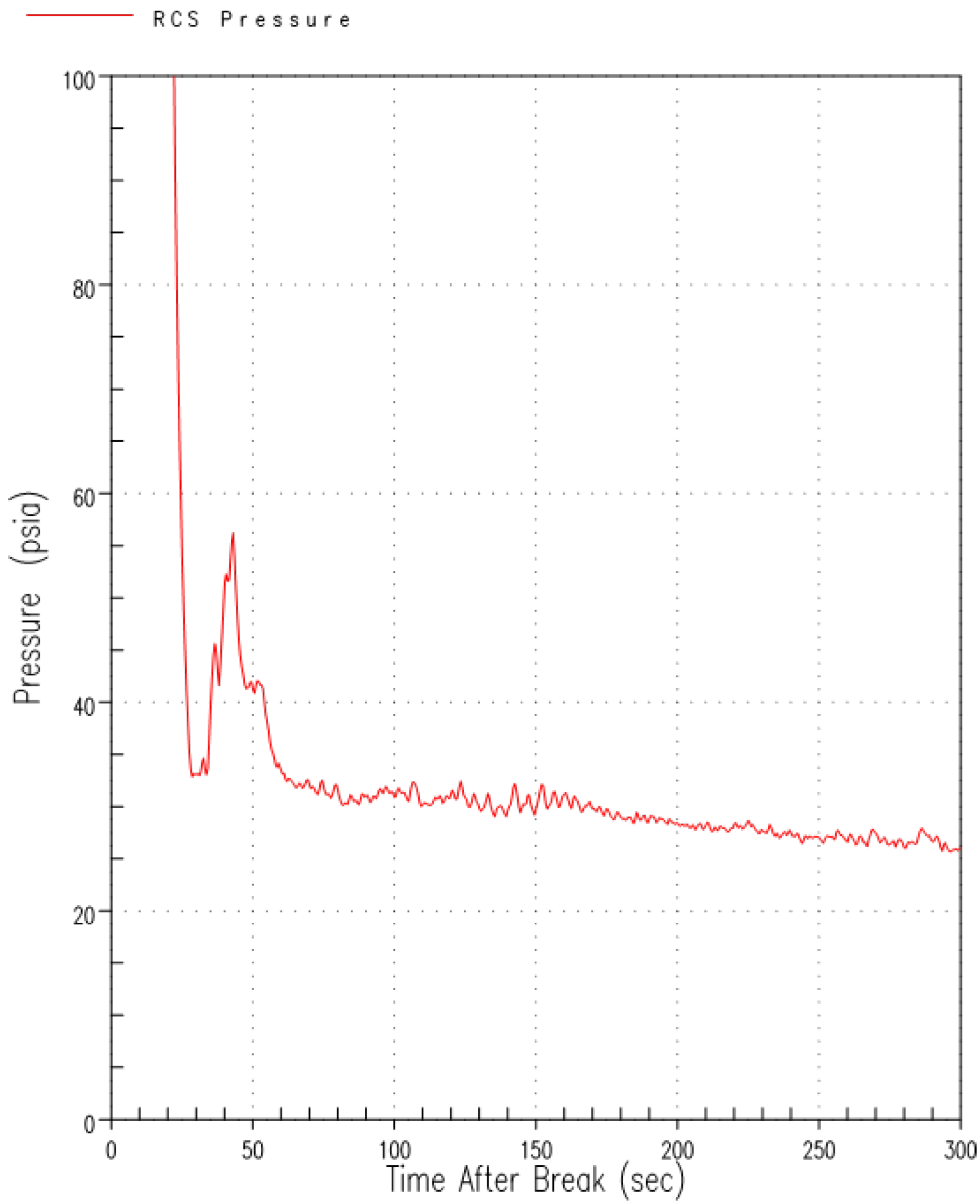

Figure 20B: Comanche Peak Unit 2 RCS Pressure for the Region II Analysis PCT Case

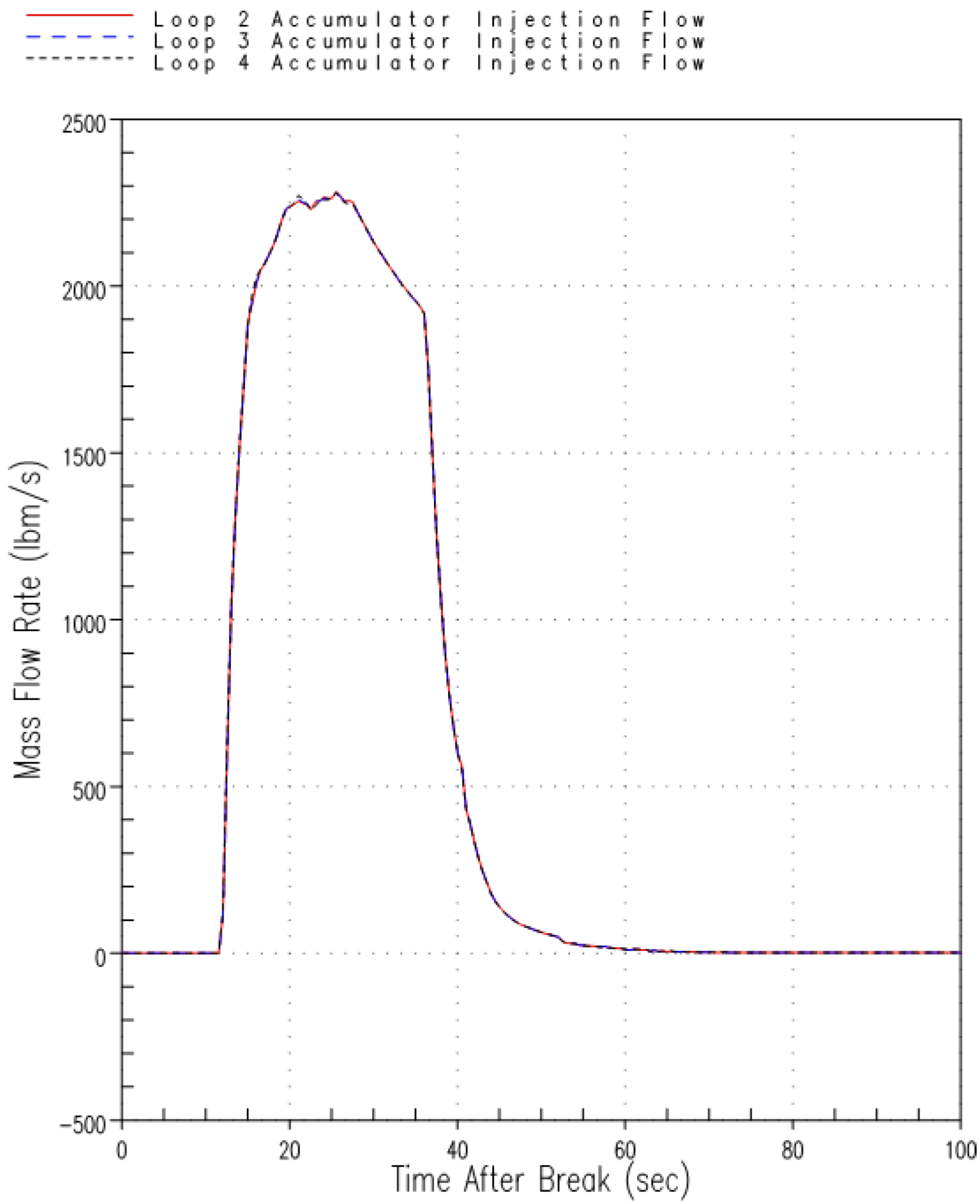

Figure 21A: Comanche Peak Unit 1 Accumulator Injection Flow per Loop for the Region II Analysis PCT Case

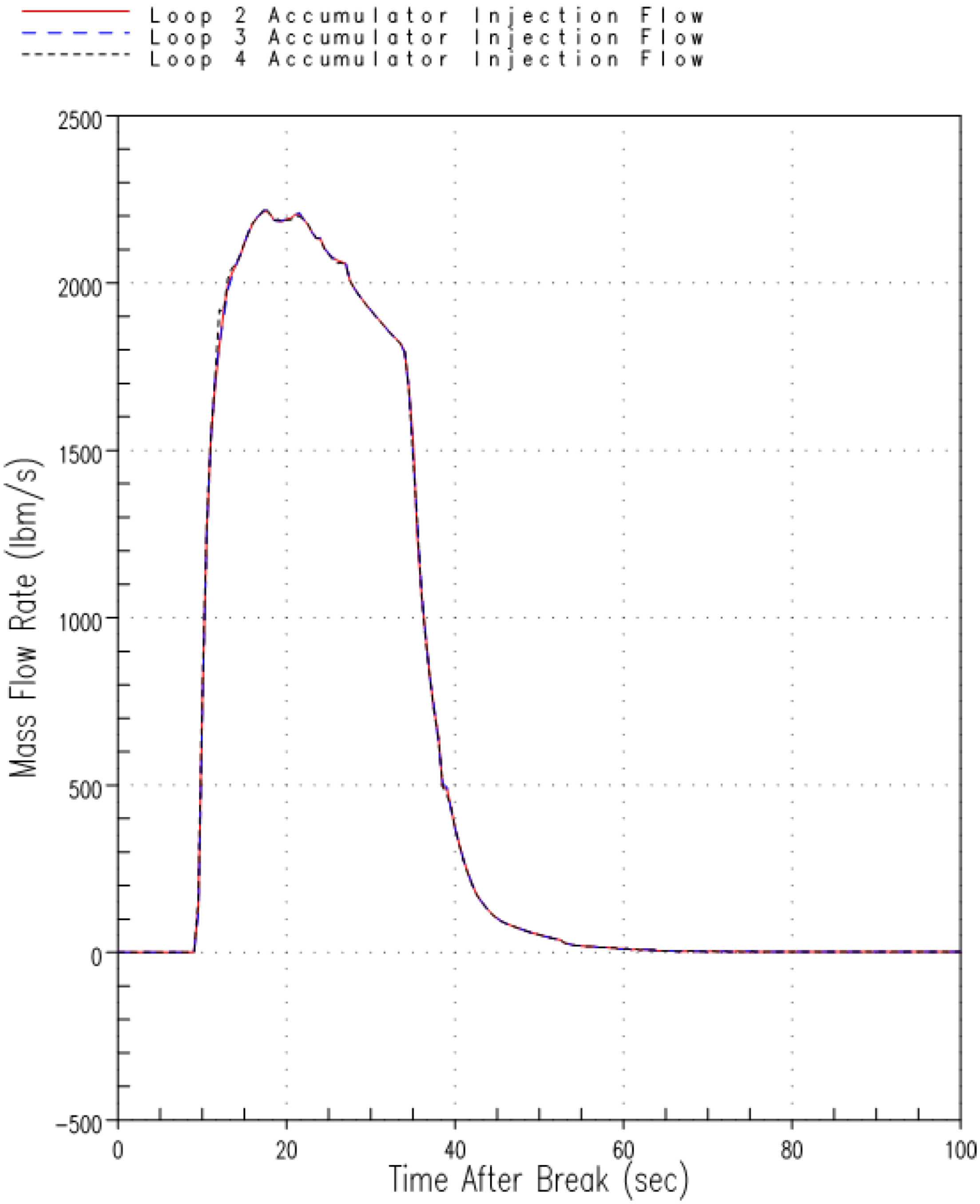

Figure 21B: Comanche Peak Unit 2 Accumulator Injection Flow per Loop for the Region II Analysis PCT Case

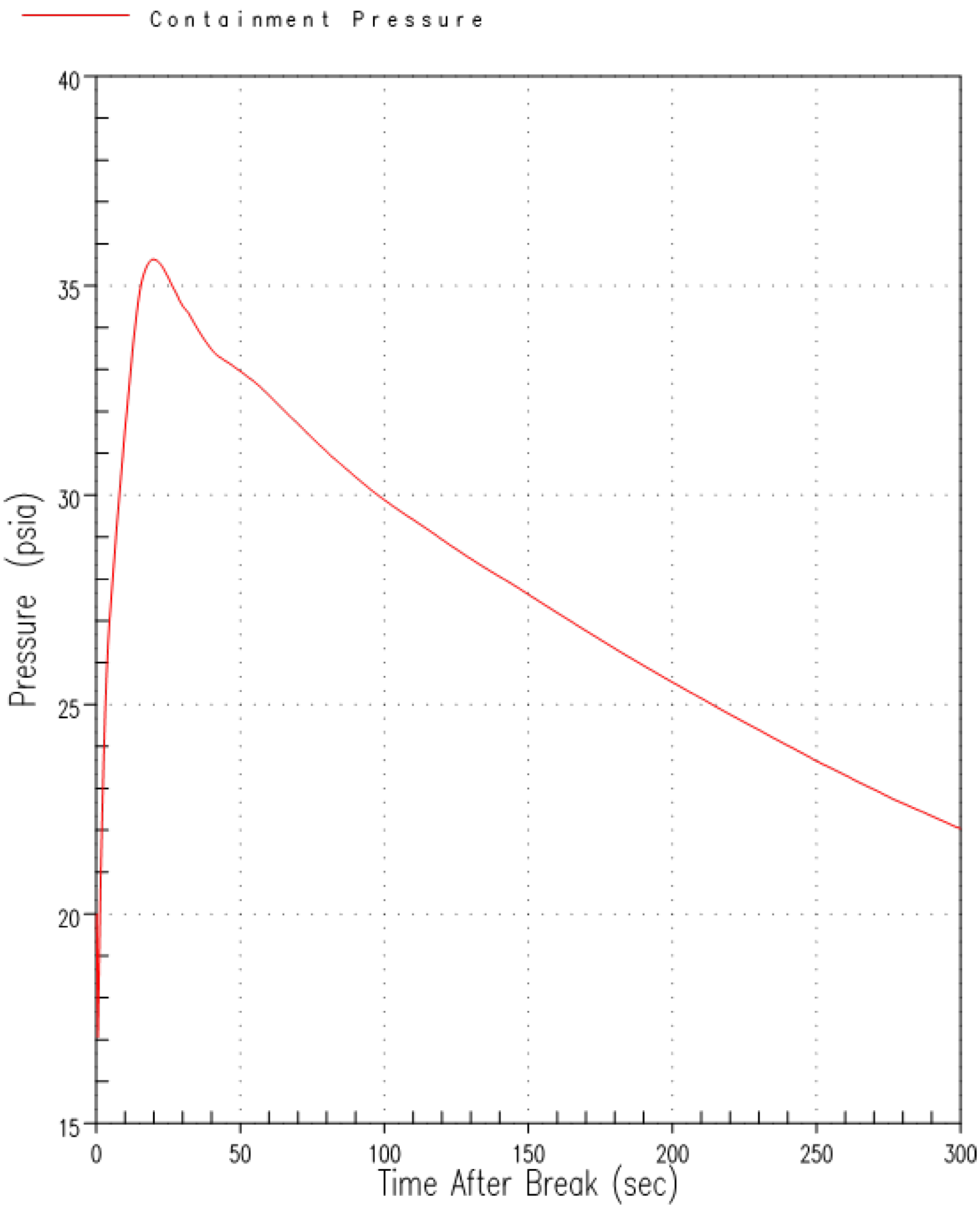

Figure 22A: Comanche Peak Unit 1 Containment Pressure for the Region II Analysis PCT Case

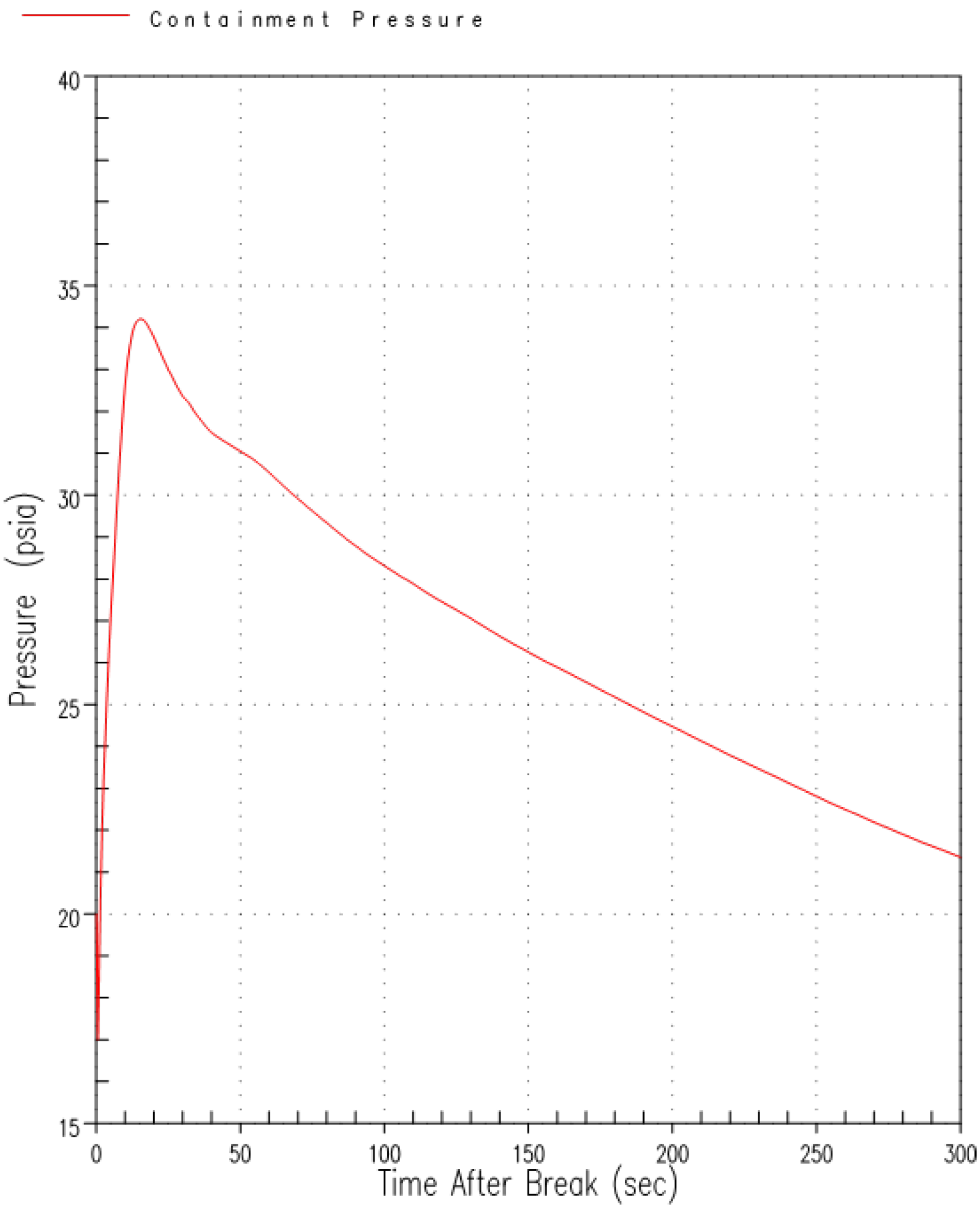

Figure 22B: Comanche Peak Unit 2 Containment Pressure for the Region II Analysis PCT Case

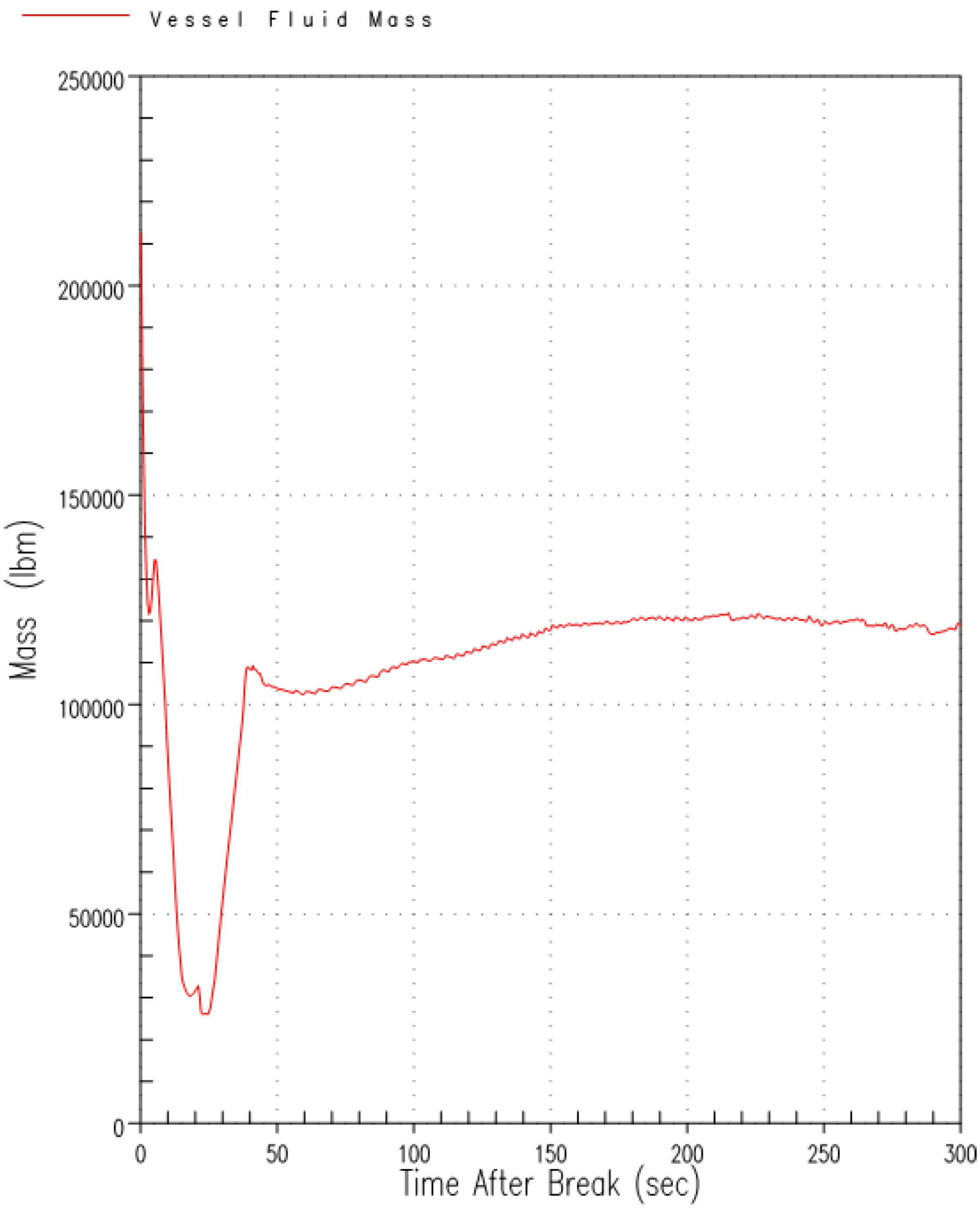

Figure 23A: Comanche Peak Unit 1 Vessel Fluid Mass for the Region II Analysis PCT Case

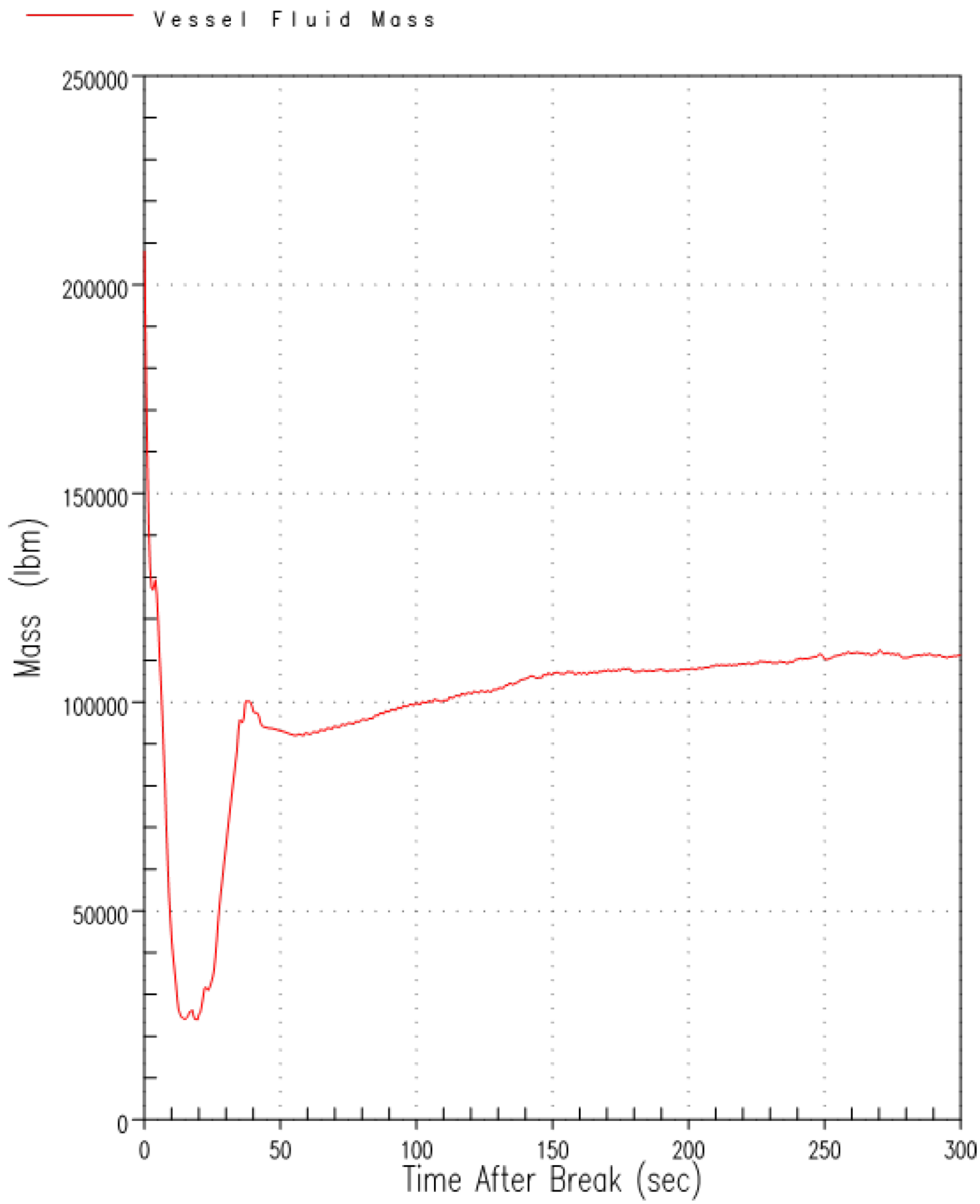

Figure 23B: Comanche Peak Unit 2 Vessel Fluid Mass for the Region II Analysis PCT Case

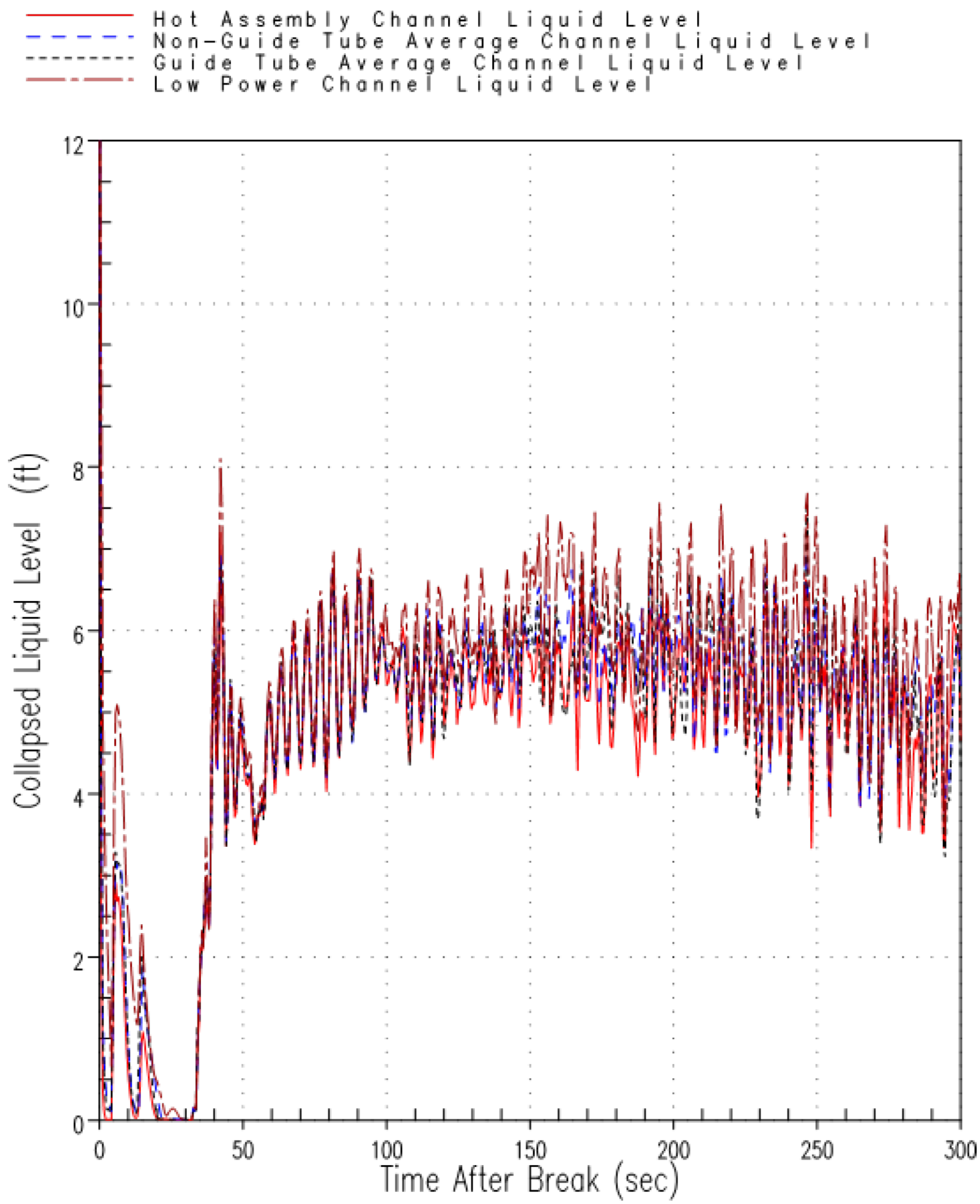

Figure 24A: Comanche Peak Unit 1 Collapsed Liquid Level for Each Core Channel (Relative to Bottom of Active Fuel) for the Region II Analysis PCT Case

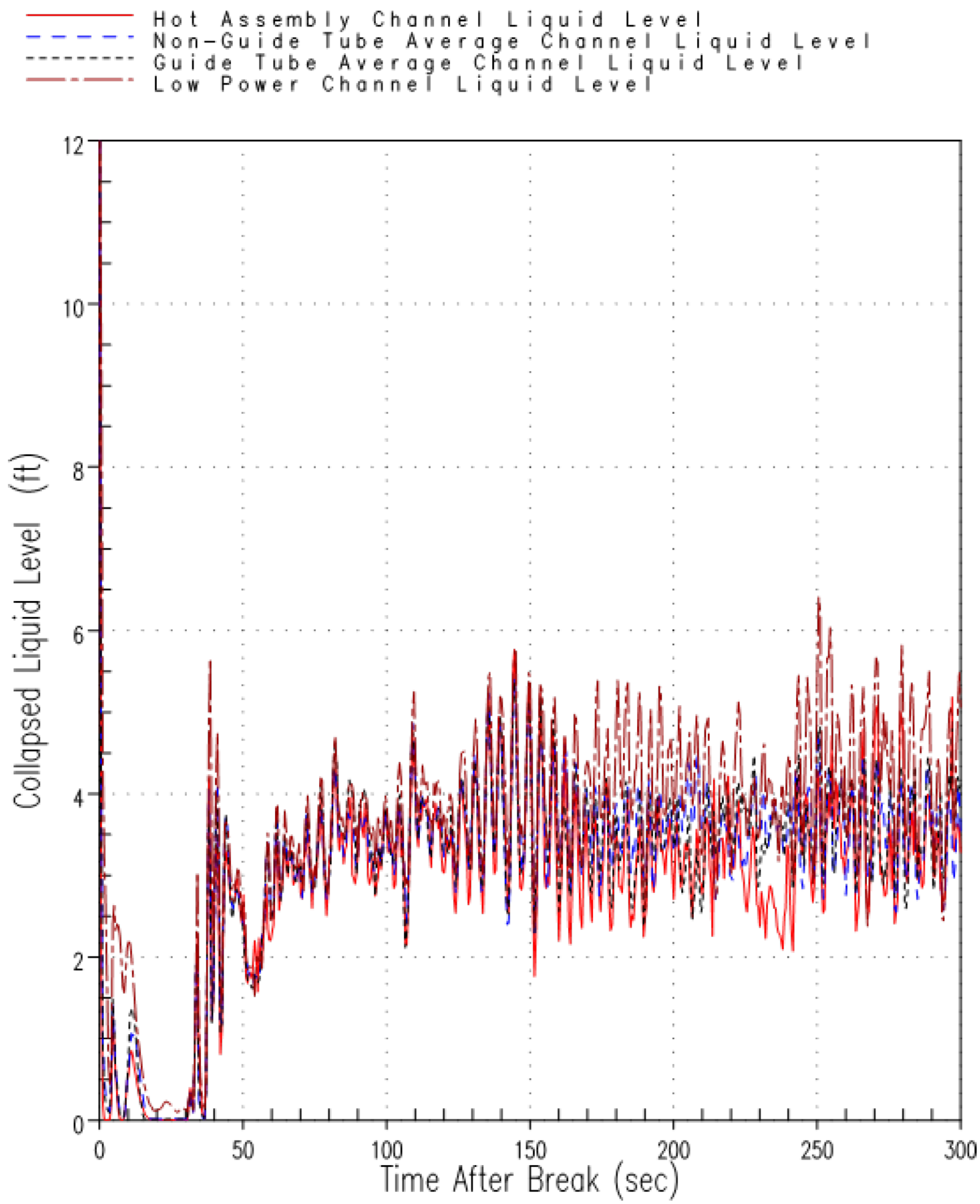

Figure 24B: Comanche Peak Unit 2 Collapsed Liquid Level for Each Core Channel (Relative to Bottom of Active Fuel) for the Region II Analysis PCT Case

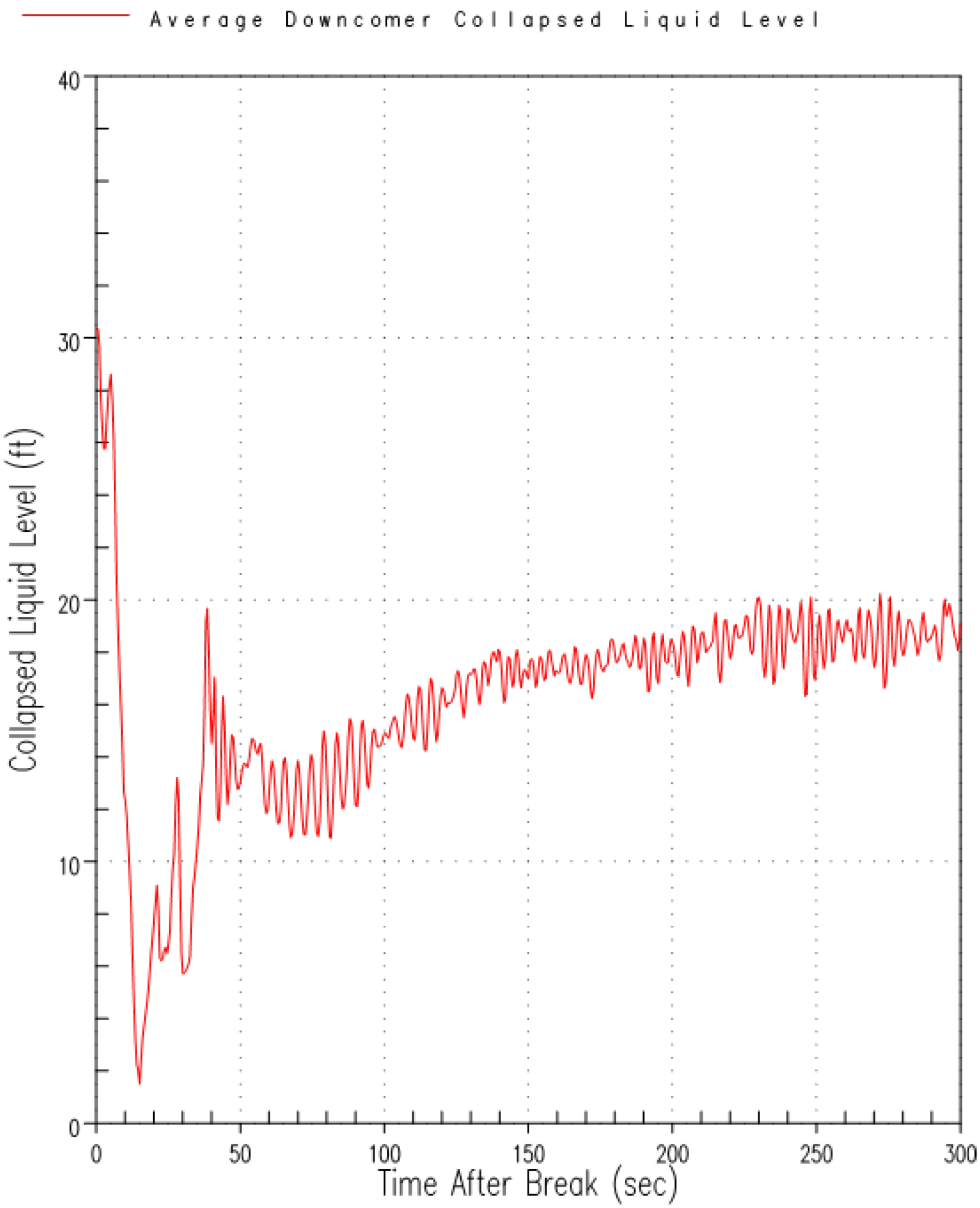

Figure 25A: Comanche Peak Unit 1 Average Downcomer Collapsed Liquid Level (Relative to Bottom of Upper Tie Plate) for the Region II Analysis PCT Case

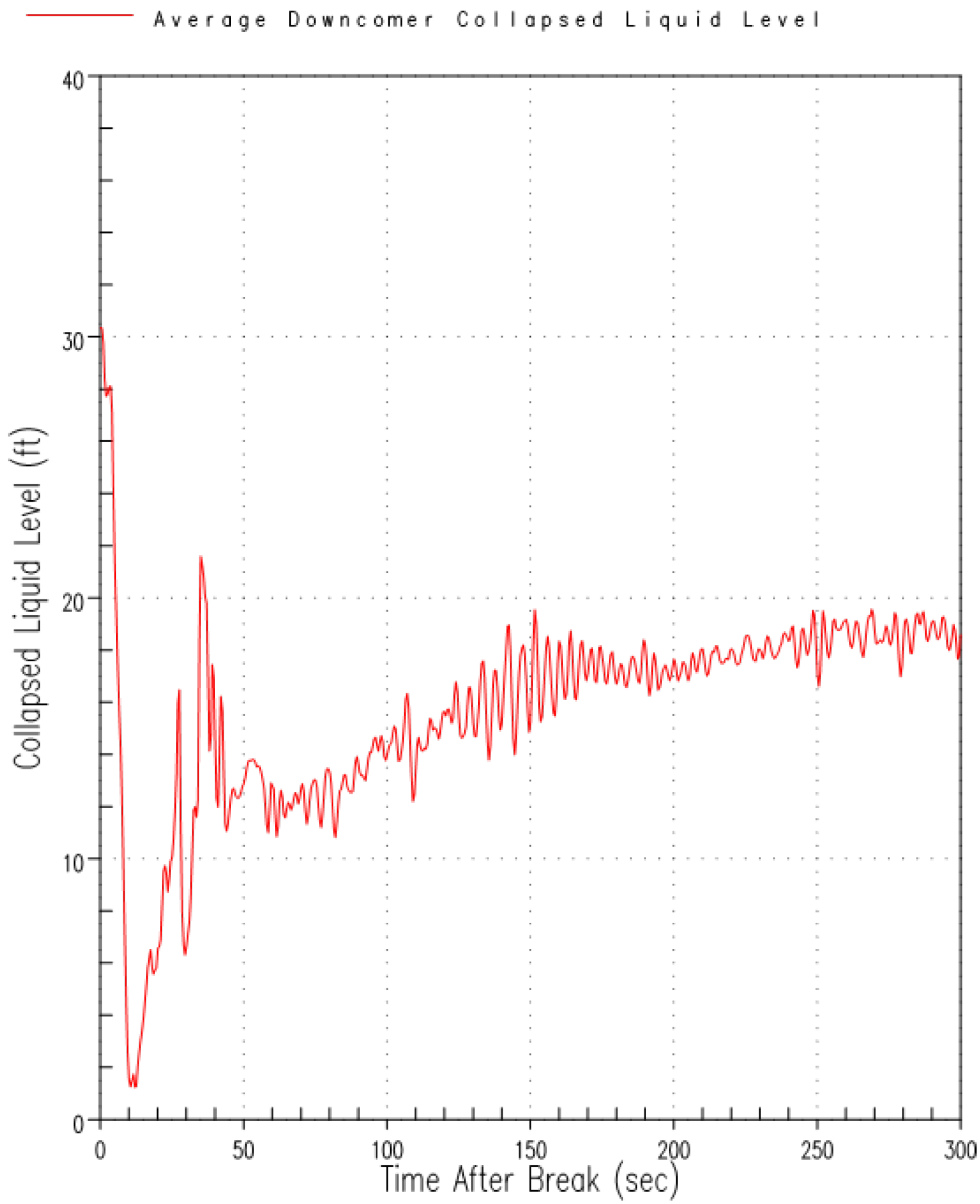

Figure 25B: Comanche Peak Unit 2 Average Downcomer Collapsed Liquid Level (Relative to Bottom of Upper Tie Plate) for the Region II Analysis PCT Case

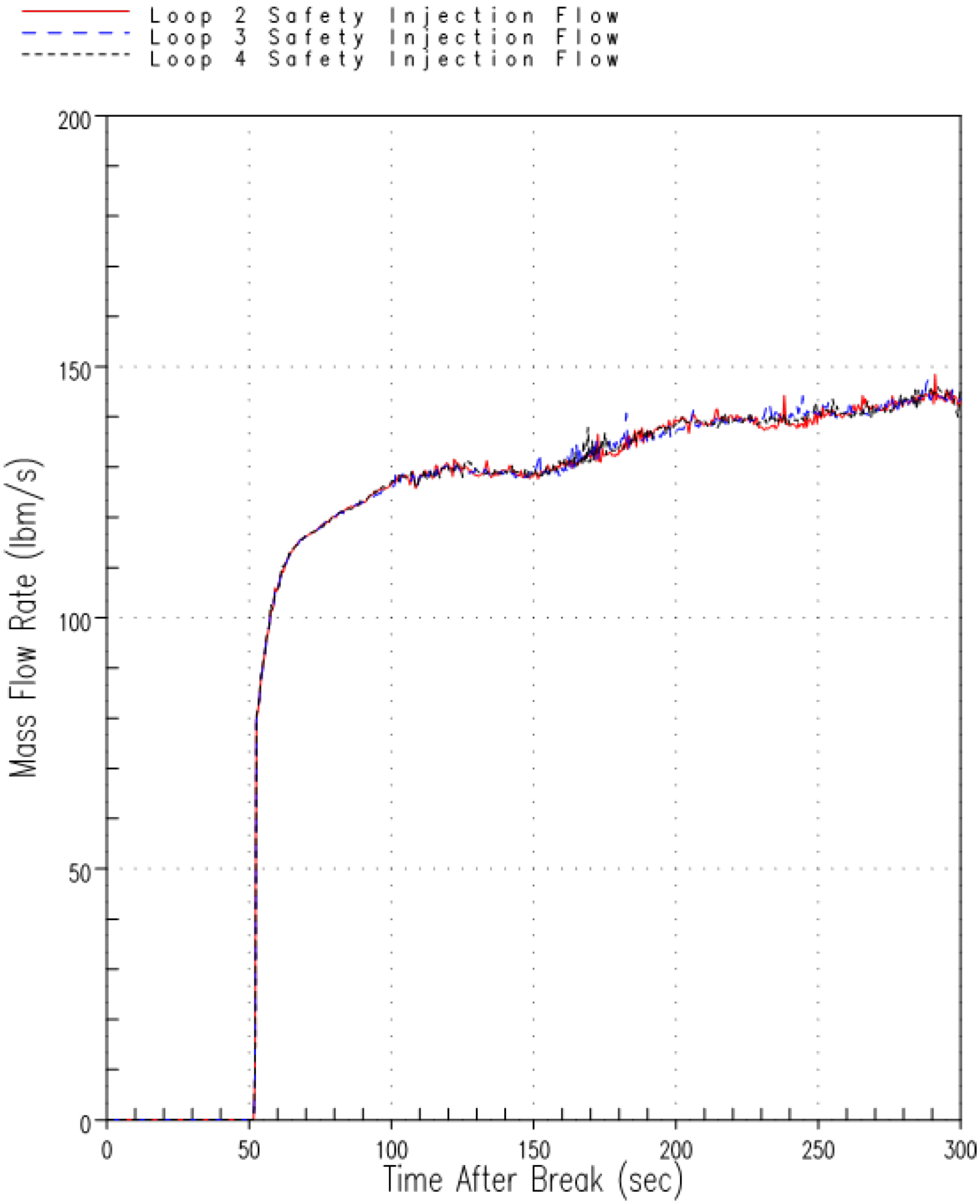

Figure 26A: Comanche Peak Unit 1 Total Safety Injection Flow Rate per Loop (not including Accumulator Flow) for the Region II Analysis PCT Case

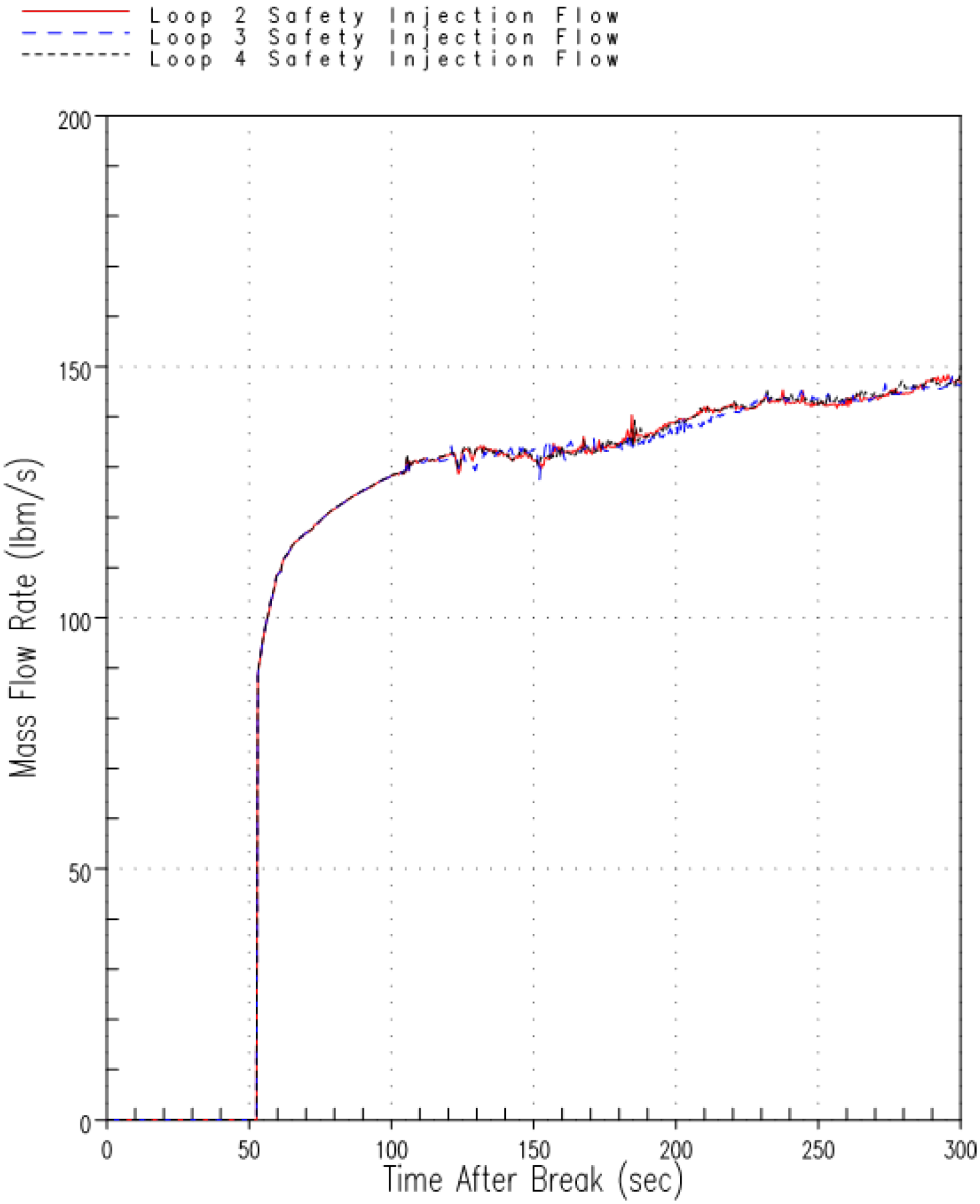

Figure 26B: Comanche Peak Unit 2 Total Safety Injection Flow Rate per Loop (not including Accumulator Flow) for the Region II Analysis PCT Case

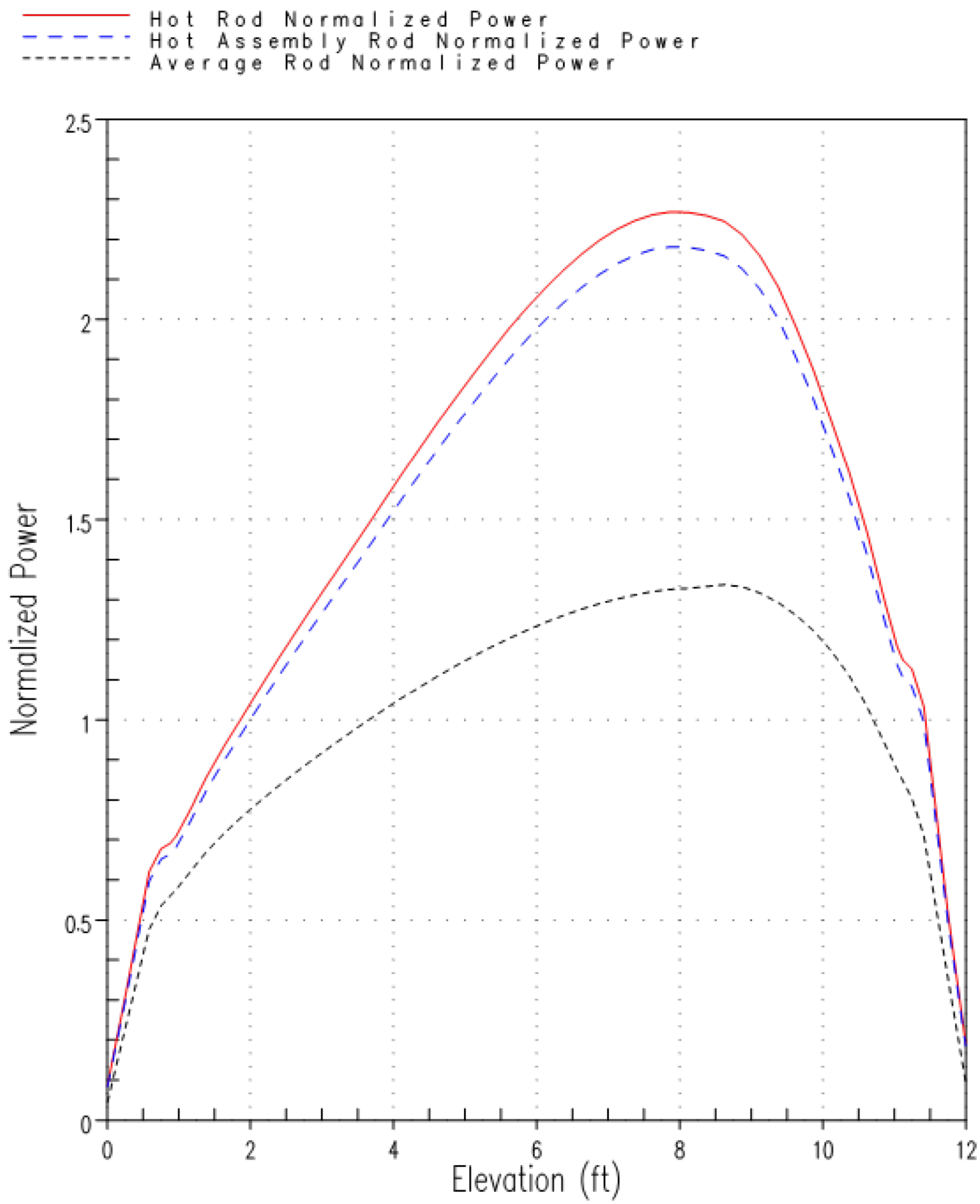

Figure 27A: Comanche Peak Unit 1 Normalized Core Power Shapes for the Region II Analysis PCT Case

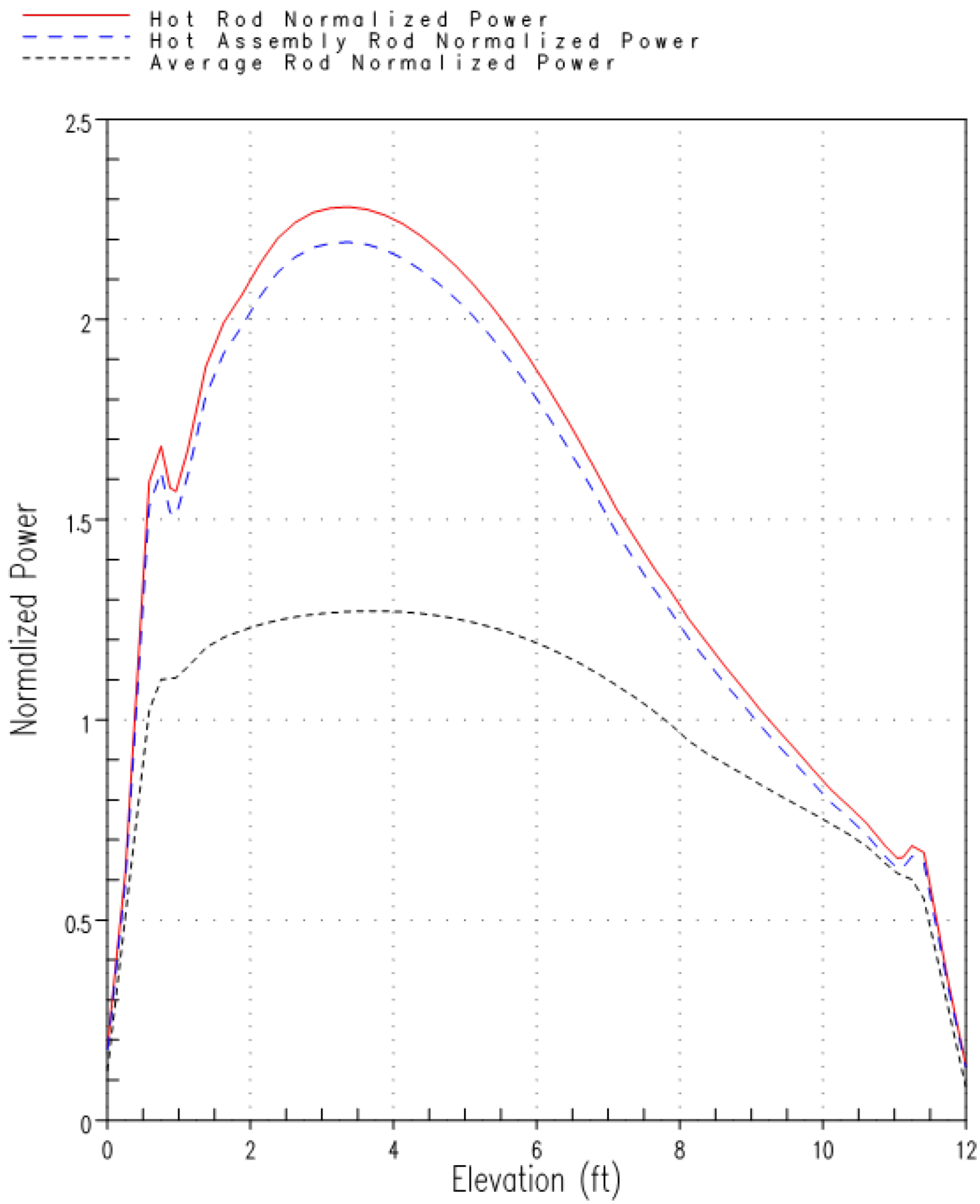

Figure 27B: Comanche Peak Unit 2 Normalized Core Power Shapes for the Region II Analysis PCT Case

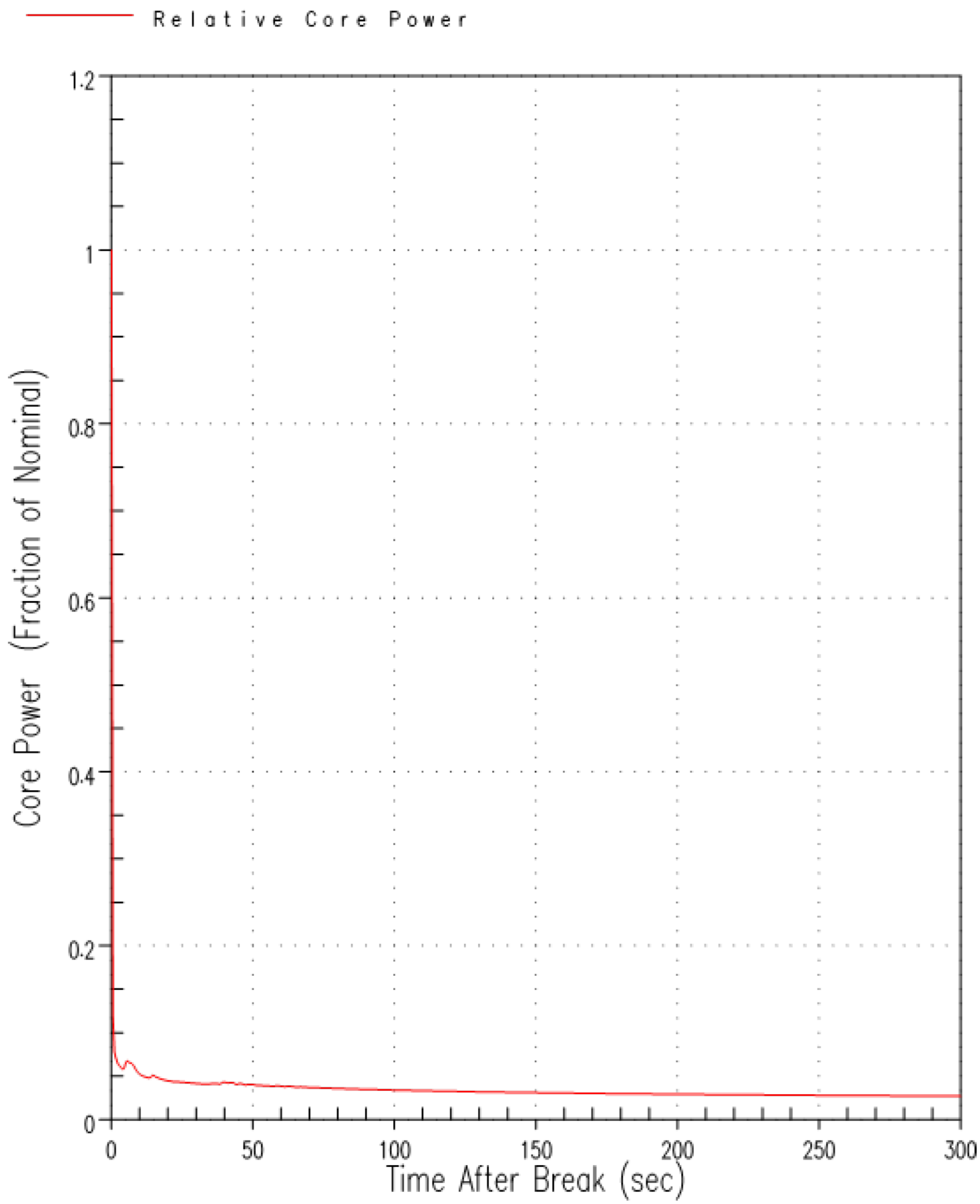

Figure 28A: Comanche Peak Unit 1 Relative Core Power for the Region II Analysis PCT Case

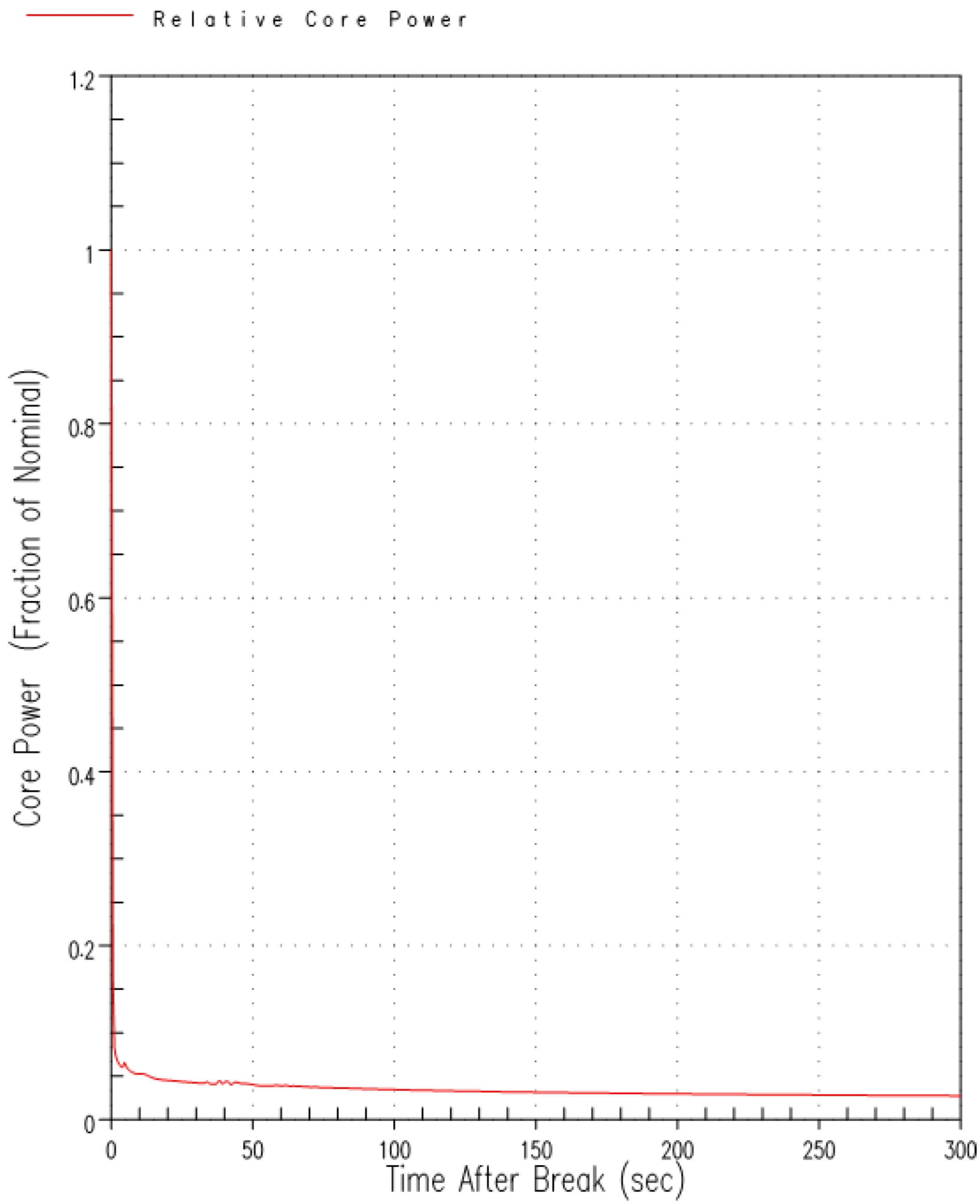

Figure 28B: Comanche Peak Unit 2 Relative Core Power for the Region II Analysis PCT Case

Westinghouse

Westinghouse Electric Company
1000 Westinghouse Drive
Cranberry Township, Pennsylvania 16066
USA

U.S. Nuclear Regulatory Commission
Document Control Desk
11555 Rockville Pike
Rockville, MD 20852

Direct tel: (412) 374-5541
Direct fax: (724) 940-8542
e-mail: hosackkl@westinghouse.com

CAW-19-4846

February 27, 2019

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

Subject: WPT-18138, P-Attachment, "Application of Westinghouse FULL SPECTRUM LOCA Evaluation Model to the Comanche Peak Units 1 and 2 Nuclear Plants"

The Application for Withholding Proprietary Information from Public Disclosure is submitted by Westinghouse Electric Company LLC ("Westinghouse"), pursuant to the provisions of paragraph (b)(1) of Section 2.390 of the Nuclear Regulatory Commission's ("Commission's") regulations. It contains commercial strategic information proprietary to Westinghouse and customarily held in confidence.

The proprietary information for which withholding is being requested in the above-referenced report is further identified in Affidavit CAW-19-4846 signed by the owner of the proprietary information, Westinghouse. The Affidavit, which accompanies this letter, sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR Section 2.390 of the Commission's regulations.

Accordingly, this letter authorizes the utilization of the accompanying Affidavit by Vistra Operations Company, LLC.

Correspondence with respect to the proprietary aspects of the Application for Withholding or the Westinghouse Affidavit should reference CAW-19-4846, and should be addressed to Camille T. Zozula, Manager, Facilities and Infrastructure Licensing, Westinghouse Electric Company, 1000 Westinghouse Drive, Building 2 Suite 256, Cranberry Township, Pennsylvania 16066.

Korey L. Hosack, Manager
Product Line Regulatory Support

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF BUTLER:

I, Korey L. Hosack, am authorized to execute this Affidavit on behalf of Westinghouse Electric Company LLC ("Westinghouse") and declare that the averments of fact set forth in this Affidavit are true and correct to the best of my knowledge, information, and belief.

Executed on: 20190227

Korey L. Hosack, Manager
Product Line Regulatory Support

(1) I am Manager, Product Line Regulatory Support, Westinghouse Electric Company LLC ("Westinghouse"), and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rule making proceedings, and am authorized to apply for its withholding on behalf of Westinghouse.

(2) I am making this Affidavit in conformance with the provisions of 10 CFR Section 2.390 of the Nuclear Regulatory Commission's ("Commission's") regulations and in conjunction with the Westinghouse Application for Withholding Proprietary Information from Public Disclosure accompanying this Affidavit.

(3) I have personal knowledge of the criteria and procedures utilized by Westinghouse in designating information as a trade secret, privileged or as confidential commercial or financial information.

(4) Pursuant to the provisions of paragraph (b)(4) of Section 2.390 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

(i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Westinghouse.

(ii) The information is of a type customarily held in confidence by Westinghouse and not customarily disclosed to the public. Westinghouse has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitute Westinghouse policy and provide the rational basis required.

Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential competitive advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of

Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage (e.g., by optimization or improved marketability).

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.

(e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

(f) It contains patentable ideas, for which patent protection may be desirable.

(iii) There are sound policy reasons behind the Westinghouse system which include the following:

(a) The use of such information by Westinghouse gives Westinghouse a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Westinghouse competitive position.

(b) It is information that is marketable in many ways. The extent to which such information is available to competitors diminishes the Westinghouse ability to sell products and services involving the use of the information.

(c) Use by our competitor would put Westinghouse at a competitive disadvantage by reducing his expenditure of resources at our expense.

(d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component may be the key to the entire puzzle, thereby depriving Westinghouse of a competitive advantage.

(e) Unrestricted disclosure would jeopardize the position of prominence of Westinghouse in the world market, and thereby give a market advantage to the competition of those countries.

(f) The Westinghouse capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.

(iv) The information is being transmitted to the Commission in confidence and, under the provisions of 10 CFR Section 2.390, is to be received in confidence by the Commission.

(v) The information sought to be protected is not available in public sources or available information has not been previously employed in the same original manner or method to the best of our knowledge and belief.

(vi) The proprietary information sought to be withheld in this submittal is that which is appropriately marked in WPT-18138, P-Attachment, "Application of Westinghouse FULL SPECTRUM LOCA Evaluation Model to the Comanche Peak Units 1 and 2 Nuclear Plants" (Proprietary), for submittal to the Commission, being transmitted by Vistra Operations Company, LLC letter. The proprietary information as submitted by Westinghouse is that associated with NRC's review of Vistra Operations Company, LLC application to revise the Comanche Peak Nuclear Plant Unit 1 and Unit 2 for the **FULL SPECTRUM™** LOCA (**FSLOCA™**) methodology, and may be used only for that purpose.

(a) This information is part of that which will enable Westinghouse to support the NRC's review of Vistra Operations Company, LLC application to FULL SPECTRUM LOCA.

(b) Further, this information has substantial commercial value as follows:

(i) Westinghouse plans to sell the use of similar information to its customers for the purpose of future NRC reviews regarding **FULL SPECTRUM** LOCA (**FSLOCA**) methodology.

(ii) Westinghouse can sell support and defense of industry guidelines and acceptance criteria for plant-specific applications.

(iii) The information requested to be withheld reveals the distinguishing aspects of a methodology which was developed by Westinghouse.

Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar technical evaluation justifications and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

The development of the technology described in part by the information is the result of applying the results of many years of experience in an intensive Westinghouse effort and the expenditure of a considerable sum of money.

In order for competitors of Westinghouse to duplicate this information, similar technical programs would have to be performed and a significant manpower effort, having the requisite talent and experience, would have to be expended.

Further the deponent sayeth not.

Supplemental Information for Evaluation of the Gamma Energy Redistribution Error for Comanche Peak Unit 1 (TBX) and Unit 2 (TCX) FSLOCA EM Analyses

Background

An error was discovered in the gamma redistribution multiplier on hot rod and hot assembly power (FGAMMA) used for the **FULL SPECTRUM™** Loss-of-Coolant Accident (**FSLOCA™**) evaluation model (EM), which results in an underestimation of the hot rod and hot assembly power by up to 5%. The underestimation of the hot rod and hot assembly power is expected to result in an underprediction in the calculated peak cladding temperature (PCT) and potentially the oxidation results for analyses performed using the FSLOCA EM. Therefore, the correction of this error must be assessed for completed analyses.

The Comanche Peak Unit 1 (TBX) and Unit 2 (TCX) FSLOCA EM analyses are documented in WCAP-18417-P [1] and WCAP-18418-P [2]. The purpose of this attachment is to provide supplemental information for the evaluation of the gamma energy redistribution error to be included in attachments 3 and 4 of this submittal.

Information for Insertion in Section 1.2.3

In Section 1.2.3 of attachment 3 and 4 of this submittal, the following paragraphs should be inserted at the end of the Compliance section for Limitation and Condition Number 2:

> The treatment for the uncertainty in the gamma energy redistribution is discussed on pages 29-75 and 29-76 of Reference 1, and the equation for the assumed increase in hot rod and hot assembly relative power is presented on page 29-76. The power increase in the hot rod and hot assembly due to energy redistribution in the application of the FSLOCA EM to Comanche Peak Unit 1 and Unit 2 was calculated incorrectly. This 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 uncertainty. The effect of the error correction was evaluated against the application of the FSLOCA EM to Comanche Peak Unit 1 and Unit 2.
>
> The error correction has only a limited impact on the power modeled for a single assembly in the core. As such, there is a negligible impact of the error correction on the system thermal-hydraulic response during the postulated LOCA.
>
> For Region I, 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 15°F for Comanche Peak Unit 1, leading to a final result of 1,032°F for the Region I analysis, and the correction of the error was estimated to increase the Region I analysis PCT by 3°F for Comanche Peak Unit 2, leading to a final result of 1,116°F for the Region I analysis.

For Region II, 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 different transient phases (i.e., blowdown versus reflood) based on the PWR sensitivity studies and existing power distribution sensitivity studies. Based on the results from the PWR sensitivity studies, the correction of the error is estimated to increase the Region II analysis PCT by 20°F for Comanche Peak Unit 1, leading to an analysis result of 1,566°F for the Region II analysis assuming loss-of-offsite power and 1,566°F for the Region II analysis assuming offsite power available. Based on the results from the PWR sensitivity studies, the correction of the error is estimated to increase the Region II analysis PCT by 31°F for Comanche Peak Unit 2, leading to an analysis result of 1,610°F for the Region II analysis assuming loss-of-offsite power and 1,600°F for the Region II analysis assuming offsite power available.

All of the analysis results including the error correction continue to maintain compliance with the 10 CFR 50.46 acceptance criteria.

Updated Tables 7A and 7B

Tables 7A and 7B in attachment 3 and 4 of this submittal, should be replaced with the following updated tables:

Table 7A. Comanche Peak Unit 1 Analysis Results with the FSLOCA EM

Outcome	Region I Value	Region II Value (LOOP)	Region II Value (OPA)
95/95 PCT[1]	1,017+15 = 1,032°F	1,546+20 = 1,566°F	1,546+20 = 1,566°F
95/95 MLO	8.96%	8.64%	8.64%
95/95 CWO	0.00%	0.02%	0.02%

Note:

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

Table 7B. Comanche Peak Unit 2 Analysis Results with the FSLOCA EM

Outcome	Region I Value	Region II Value (LOOP)	Region II Value (OPA)
95/95 PCT[1]	1,113+3 = 1,116°F	1,579+31 = 1,610°F	1,569+31 = 1,600°F
95/95 MLO	8.70%	8.82%	8.82%
95/95 CWO	0.00%	0.04%	0.04%
Note: 1. The PCT presented in the table shows the analysis-of-record result, which is the sum of the uncertainty analysis result plus the impact of the energy redistribution uncertainty error correction. The figures presenting the analysis results correspond to the uncertainty analysis results. The MLO and CWO were confirmed to maintain compliance with the 10 CFR 50.46 acceptance criteria with the error correction.			

References

1. WCAP-18417-P, “Engineering Summary Report of the Comanche Peak Unit 1 Loss-of-Coolant Accident (LOCA) Analysis with the FULL SPECTRUM LOCA (FSLOCA) Methodology,” February 2019.
2. WCAP-18418-P, “Engineering Summary Report of the Comanche Peak Unit 2 Loss-of-Coolant Accident (LOCA) Analysis with the FULL SPECTRUM LOCA (FSLOCA) Methodology,” March 2019.
3. WPT-18138, “Transmittal of the Comanche Peak Units 1 and 2 FULL SPECTRUM LOCA (FSLOCA) Evaluation Model (EM) Analyses License Amendment Request Input (Revision 0),” February 2019.

LOCA Peak Cladding Temperature (PCT) Summary

Plant Name: COMANCHE PEAK 1
Utility Name: Luminant
EM: FSLOCA
AOR Description: FULL SPECTRUM LOCA EM Large Break
Summary Sheet Status: Future

	PCT (°F)	Reference #	Note #
ANALYSIS-OF-RECORD	1566	1	(a)

AOR + ASSESSMENTS **PCT =** 1566.0 °F

REFERENCES

1 WCAP-18417-P, "Engineering Summary Report of the Comanche Peak Unit 1 Loss-of-Coolant Accident (LOCA) Analysis with the FULL SPECTRUM LOCA (FSLOCA) Methodology," February 2019.

NOTES:

(a) The analysis-of-record PCT reflects the uncertainty analysis that included the error in gamma energy redistribution uncertainty (1546°F) as well as the correction of the error (20°F).

LOCA Peak Cladding Temperature (PCT) Summary

Plant Name: COMANCHE PEAK 1
Utility Name: Luminant
EM: FSLOCA
AOR Description: FULL SPECTRUM LOCA EM Small Break
Summary Sheet Status: Future

	PCT (°F)	Reference #	Note #
ANALYSIS-OF-RECORD	1032	1	(a)

AOR + ASSESSMENTS **PCT =** 1032.0 °F

REFERENCES

1 WCAP-18417-P, “Engineering Summary Report of the Comanche Peak Unit 1 Loss-of-Coolant Accident (LOCA) Analysis with the FULL SPECTRUM LOCA (FSLOCA) Methodology,” February 2019.

NOTES:

(a) The analysis-of-record PCT reflects the uncertainty analysis that included the error in gamma energy redistribution uncertainty (1017°F) as well as the correction of the error (15°F).

Version: COMANCHE PEAK 1 TBX_LOCA-50.46_TBX_Base_FSLOCA_SBLOCA – 1.2 V.V

LOCA Peak Cladding Temperature (PCT) Summary

Plant Name: COMANCHE PEAK 2
Utility Name: Luminant
EM: FSLOCA
AOR Description: FULL SPECTRUM LOCA EM Large Break
Summary Sheet Status: Future

	PCT (°F)	Reference #	Note #
ANALYSIS-OF-RECORD	1610	1	(a)

AOR + ASSESSMENTS **PCT =** 1610.0 °F

REFERENCES

1 WCAP-18418-P, "Engineering Summary Report of the Comanche Peak Unit 2 Loss-of-Coolant Accident (LOCA) Analysis with the FULL SPECTRUM LOCA (FSLOCA) Methodology," March 2019.

NOTES:

(a) The analysis-of-record PCT reflects the uncertainty analysis that included the error in gamma energy redistribution uncertainty (1579°F) as well as the correction of the error (31°F).

LOCA Peak Cladding Temperature (PCT) Summary

Plant Name: COMANCHE PEAK 2
Utility Name: Luminant
EM: FSLOCA
AOR Description: FULL SPECTRUM LOCA EM Small Break
Summary Sheet Status: Future

	PCT (°F)	Reference #	Note #
ANALYSIS-OF-RECORD	1116	1	(a)

AOR + ASSESSMENTS **PCT =** 1116.0 °F

REFERENCES

1 WCAP-18418-P, "Engineering Summary Report of the Comanche Peak Unit 2 Loss-of-Coolant Accident (LOCA) Analysis with the FULL SPECTRUM LOCA (FSLOCA) Methodology," March 2019.

NOTES:

(a) The analysis-of-record PCT reflects the uncertainty analysis that included the error in gamma energy redistribution uncertainty (1113°F) as well as the correction of the error (3°F).

Version: COMANCHE PEAK 2 TCX_LOCA-50.46_TCX_Base_FSLOCA_SBLOCA – 1.3 V.V