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December 26, 2018

PG&E Letter DCL-18-100

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

10 CFR 50.90

Diablo Canyon Units 1 and 2
Docket No. 50-275, OL-DPR-80
Docket No. 50-323, OL-DPR-82
License Amendment Request 18-02
License Amendment Request to Revise Technical Specification 5.6.5b, "Core Operating Limits Report (COLR)" for Full Spectrum Loss-of-Coolant Accident Methodology

Dear Commissioners and Staff:

Pursuant to 10 CFR 50.90, Pacific Gas and Electric Company (PG&E) hereby requests approval of the enclosed proposed amendment to the Diablo Canyon Power Plant (DCPP) Facility Operating License Nos. DPR-80 and DPR-82 for Units 1 and 2, respectively. The enclosed license amendment request (LAR) proposes to modify Technical Specification (TS) 5.6.5b, "Core Operating Limits Report (COLR)," to replace the existing NRC-approved loss-of-coolant accident (LOCA) methodologies with the NRC-approved LOCA methodology contained in WCAP-16996-P-A, Revision 1, "Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (FULL SPECTRUM LOCA Methodology)," that was used for LOCA reanalysis for DCPP.

This LAR completes PG&E's actions to perform a reanalysis in accordance with 10 CFR 50.46(a)(3)(ii) as described in PG&E Letters DCL-11-082, for Unit 1, and DCL-12-102, for Unit 2.

The enclosure provides a detailed description and technical evaluation of the proposed change, including PG&E's determination that the proposed changes involve no significant hazards.

The following attachments are included in the enclosure:

A001
NRR

- Attachment 1 provides markup pages of existing TS to show the proposed change.
- Attachment 2 provides retyped TS pages.
- Attachment 3 provides the TS Bases markups.
- Attachment 4 provides the Updated Final Safety Analysis Report (UFSAR) changes associated with the LOCA reanalysis.
- Attachment 5 contains a Westinghouse Authorization Letter, CAW-18-4798, with an accompanying affidavit; a Proprietary Information Notice; and a Copyright Notice.
- Attachment 6 contains the non-proprietary version of Attachment 7.
- Attachment 7 (proprietary) includes the Westinghouse report, "Compliance with the Limitations and Conditions in the Revised NRC Final Safety Evaluation (FSE) for WCAP-16996-A, Revision 1."

Attachment 7 contains information proprietary to Westinghouse Electric Company LLC ("Westinghouse"). Accordingly, Attachment 5 includes a Westinghouse Authorization Letter, CAW-18-4798; an accompanying affidavit; a Proprietary Information Notice; and a Copyright Notice. The affidavit is signed by Westinghouse, the owner of the information. The affidavit sets forth the basis on which the Westinghouse proprietary information contained in Attachment 7 may be withheld from public disclosure by the Commission, and addresses with specificity the considerations listed in 10 CFR 2.390(b)(4).

PG&E requests that the Westinghouse proprietary information be withheld from public disclosure in accordance with 10 CFR 2.390.

Correspondence with respect to the copyright or proprietary aspects of the application for withholding related to the Westinghouse proprietary information or the Westinghouse affidavit provided in Attachment 5 to the Enclosure should reference Westinghouse Letter CAW-18-4798 and be addressed to Edmond J. Mercier, Manager, Fuels Licensing and Regulatory Support, Westinghouse Electric Company, 1000 Westinghouse Drive, Building 2 Suite 256, Cranberry Township, Pennsylvania 16066.

The changes requested in this LAR are not required to address an immediate safety concern. PG&E requests approval of this LAR by no later than December 31, 2019. PG&E requests the license amendments be made effective upon NRC issuance, to be implemented within 120 days from the date of issuance.

PG&E makes no new or revised regulatory commitments (as defined by NEI 99-04) in this letter.

In accordance with DCPP administrative procedures and the DCPP Quality Assurance Program, the proposed amendment has been reviewed by the Plant Staff Review Committee.

Pursuant to 10 CFR 50.91(b)(1), PG&E is sending a copy of this proposed amendment to the California Department of Public Health.

If you have any questions or require additional information, please contact Mr. Hossein Hamzehee at (805) 545-4720.

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

Executed on December 26, 2018.

Sincerely,

Paula Gerfen
Station Director

kjse/4328
Enclosure
cc: Diablo Distribution
cc/enc: Kriss M. Kennedy, NRC Region IV Administrator
Christopher W. Newport, NRC Senior Resident Inspector
Balwant K. Singal, NRC Senior Project Manager
Gonzalo L. Perez, Branch Chief, California Department of Public Health

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CAW-18-4798

August 29, 2018

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

Subject: **License Amendment Request to Revise Technical Specification 5.6.5b, "Core Operating Limits Report (COLR)" for Full Spectrum Loss of Coolant Accident Methodology – Attachment 7, "Compliance with the Limitations and Conditions in the Revised NRC Final Safety Evaluation (FSE) for WCAP-16996-A, Rev. 1" (Proprietary)**

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-18-4798 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 Pacific Gas & Electric Company (PG&E).

Correspondence with respect to the proprietary aspects of the Application for Withholding or the Westinghouse Affidavit should reference CAW-18-4798 and should be addressed to Edmond J. Mercier, Manager, Fuels Licensing and Regulatory Support, Westinghouse Electric Company, 1000 Westinghouse Drive, Building 2, Suite 256, Cranberry Township, PA 16066.

Edmond J. Mercier, Manager
Fuels Licensing and Regulatory Support

CAW-18-4798

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF BUTLER:

I, Edmond J. Mercier, 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: 8/29/2018

Edmond J. Mercier, Manager
Fuels Licensing and Regulatory Support

(1) I am Manager, Fuels Licensing and 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 "License Amendment Request to Revise Technical Specification 5.6.5b, 'Core Operating Limits Report (COLR)' for Full Spectrum Loss of Coolant Accident Methodology – Attachment 7, 'Compliance with the Limitations and Conditions in the Revised NRC Final Safety Evaluation (FSE) for WCAP-16996-A, Rev. 1' (Proprietary)," for submittal to the Commission, being transmitted by PG&E letter. The proprietary information as submitted by Westinghouse is that associated with PG&E's request for NRC approval of Revised Technical Specification 5.6.5b, "Core Operating Limits Report (COLR)" for implementation of the Full Spectrum Loss of Coolant Accident Methodology, and may be used only for that purpose.

(a) This information is part of that which will enable Westinghouse to provide

commercial support for NRC review and approval of License Amendment Requests for implementation of the Full Spectrum Loss of Coolant Accident Methodology.

(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 implementation of the Full Spectrum Loss of Coolant Accident 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.

PROPRIETARY INFORMATION NOTICE

Transmitted herewith are proprietary and non-proprietary versions of a document, furnished to the NRC in connection with requests for generic and/or plant-specific review and approval.

In order to conform to the requirements of 10 CFR 2.390 of the Commission's regulations concerning the protection of proprietary information so submitted to the NRC, the information which is proprietary in the proprietary versions is contained within brackets, and where the proprietary information has been deleted in the non-proprietary versions, only the brackets remain (the information that was contained within the brackets in the proprietary versions having been deleted). The justification for claiming the information so designated as proprietary is indicated in both versions by means of lower case letters (a) through (f) located as a superscript immediately following the brackets enclosing each item of information being identified as proprietary or in the margin opposite such information. These lower case letters refer to the types of information Westinghouse customarily holds in confidence identified in Sections (4)(ii)(a) through (4)(ii)(f) of the Affidavit accompanying this transmittal pursuant to 10 CFR 2.390(b)(1).

COPYRIGHT NOTICE

Evaluation of the Proposed Change

License Amendment Request 18-02, License Amendment Request to Revise Technical Specification 5.6.5b, "Core Operating Limits Report (COLR)" for Full Spectrum Loss-of-Coolant Accident Methodology

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ATTACHMENTS:

1. Technical Specification Markup Pages
2. Technical Specification Retyped Pages
3. Technical Specification Bases Markups
4. UFSAR Update Changes
5. Westinghouse Authorization Letter, CAW-18-4798, with an accompanying affidavit; a Proprietary Information Notice; and a Copyright Notice
6. Compliance with the Limitations and Conditions in the Revised NRC Final Safety Evaluation (FSE) for WCAP-16996-A, Rev. 1 (Non-Proprietary)
7. Compliance with the Limitations and Conditions in the Revised NRC Final Safety Evaluation (FSE) for WCAP-16996-A, Rev. 1 (Proprietary)

EVALUATION

1. SUMMARY DESCRIPTION

This submittal is a request to amend the Diablo Canyon Power Plant (DCPP) Facility Operating Licenses DPR-80 and DPR-82 for Units 1 and 2, respectively. The proposed license amendment request (LAR) requests approval to revise Technical Specification (TS) 5.6.5b, "Core Operating Limits Report (COLR)." The proposed change revises TS 5.6.5b to replace the current NRC-approved Loss-of-Coolant Accident (LOCA) methodologies with a single, newer NRC-approved LOCA methodology, the Full Spectrum LOCA Evaluation Model (FSLOCA EM), that is contained in WCAP-16996-P-A, Rev. 1, "Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (FULL SPECTRUM LOCA Methodology)," and was used for LOCA reanalysis for DCPP.

This LAR is a change to the emergency core cooling system (ECCS) LOCA EM for DCPP. It does not involve any changes to fuel type, peaking factors, fuel structural analysis, or boric acid precipitation methodology.

This LAR completes PG&E's action to perform an ECCS LOCA reanalysis in accordance with 10 CFR 50.46(a)(3)(ii) as described in PG&E Letter DCL-11-082, "10 CFR 50.46 Annual Report of Emergency Core Cooling System Evaluation Model Changes for 2010," dated July 19, 2011, for Unit 1 and PG&E Letter DCL-12-102, "10 CFR 50.46 30-Day Notification Report of Significant Emergency Core Cooling System Evaluation Model Changes that Affect Peak Cladding Temperature," dated October 18, 2012, for Unit 2.

2. DETAILED DESCRIPTION

The proposed changes to TS 5.6.5b reflect the NRC-approved LOCA methodology that was used for the LOCA reanalyses for DCPP. Attachment 1 to this Enclosure provides markup pages of the existing TS to show the proposed change. Attachment 2 provides the retyped TS pages. Attachment 3 provides the Updated Final Safety Analysis Report (UFSAR) changes associated with the LOCA reanalysis.

The proposed TS changes are described below.

TS 5.6.5b currently includes the following five NRC-approved LOCA methodologies (numbered as 5.6.5b.4 through 5.6.5b.8):

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

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 (Westinghouse Proprietary)

WCAP-12945-P-A, Westinghouse Code Qualification Document for Best-Estimate Loss of Coolant Analysis, June 1996 (Westinghouse Proprietary)

WCAP-12945-P-A, Addendum 1-A, Revision 0, "Method for Satisfying 10 CFR 50.46 Reanalysis Requirements for Best Estimate LOCA Evaluation Models," December 2004. (Westinghouse Proprietary) (Unit 1 Only)

WCAP-16009-P-A, Revision 0, Realistic Large-Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM), January 2005. (Westinghouse Proprietary) (Unit 2 Only)

TS 5.6.5b is revised to replace the above with the following NRC-approved LOCA methodology:

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

3. TECHNICAL EVALUATION

3.1. Compliance with the Limitations and Conditions in the Revised NRC Final Safety Evaluation (FSE) for Westinghouse WCAP-16996-A, Rev. 1

See Attachments 6 and 7 to this Enclosure.

3.2. Changes and Corrections to the FSLOCA EM in Westinghouse WCAP-16996-P-A, Rev. 1, as Identified in Westinghouse Letter LTR-NRC-18-30

The changes and corrections that have been made to the FSLOCA EM in Westinghouse WCAP-16996-P-A, Rev. 1, after the NRC issued the revised FSE (as identified in Westinghouse Letter LTR-NRC-18-30, Reference 2), are summarized below.

The final code version in the NRC-approved FSLOCA EM was WCOBRA/TRAC-TF2 Version 1.3. WCOBRA/TRAC-TF2 Version 1.4 was utilized in the analyses for DCPP Units 1 and 2. The changes from Version 1.3 to Version 1.4 are as follows:

General Code Maintenance

Various updates were made to enhance the usability of the code. These updates do not impact the code-calculated results.

Enhancement to Pump Momentum Equation at Low Pump Speed

A change was made to allow for transition from a homogeneous flow condition to a condition that allows slip at the pump impeller (and counter-current flow) when reactor coolant pump speed approaches zero. Under this condition, the PIPE momentum equations are solved rather than the simplified PUMP momentum equation at the pump-impeller cell face. A smooth transition between the two sets of momentum equations is ensured by ramping the interfacial drag during the transition. This improvement in the pump component momentum equations can allow liquid to flow back from the cold leg into the crossover leg, which has a negligible impact on the Region II analyses and a penalizing (i.e., conservative) impact on the Region I analyses.

Conservation of Non-Condensable Gas

An imbalance in the non-condensable gas mass that could occur under certain conditions (most likely in loop components during accumulator injection) in WCOBRA/TRAC-TF2 Version 1.3 was identified. Since a vapor/liquid mixture cannot exist below 32°F and WCOBRA/TRAC-TF2 does not implement the vapor property functions for temperatures below 32°F, whenever the energy content of the combined gas phase falls below 32°F, the Version 1.3 code reset the gas phase temperature to be 32°F, causing unbalanced mass and energy for the combined gas phase. This imbalance was corrected in WCOBRA/TRAC-TF2 Version 1.4, and the analyses for DCPP Units 1 and 2 were performed with the corrected code version.

There were also various errata identified in WCAP-16996-P-A, Rev. 1. These errata do not impact the WCOBRA/TRAC-TF2 coding. As such, there is no impact on the DCPP Unit 1 and 2 analyses.

3.3. 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 not exceeded:

Peak Cladding Temperature (10 CFR 50.46(b)(1)) – The analysis Peak Cladding Temperature (PCT) corresponds to a bounding estimate of the 95th percentile PCT at the 95-percent confidence level. Since the resulting PCT is less than 2,200°F, the analyses with the FSLOCA EM confirm that 10 CFR 50.46 acceptance criterion (b)(1), i.e., that PCT not exceed 2,200°F, is satisfied.

The results are shown in Tables 4 and 5 for DCPP Unit 1 and Unit 2, respectively.

Maximum Cladding Oxidation (10 CFR 50.46(b)(2)) – The analysis Maximum Local Oxidation (MLO) corresponds to a bounding estimate of the 95th percentile MLO at the 95-percent confidence level. Since the resulting MLO is less than 17 percent when converting the time-at-temperature to an equivalent cladding reacted using the Baker-Just correlation and adding the pre-transient corrosion, the analysis confirms that 10 CFR 50.46 acceptance criterion (b)(2), i.e., that the MLO of the cladding not exceed 17 percent of the total cladding thickness before oxidation, is satisfied.

The results are shown in Tables 4 and 5 for DCPP Unit 1 and Unit 2, respectively.

Maximum Hydrogen Generation (10 CFR 50.46(b)(3)) – The analysis Core-Wide Oxidation CWO corresponds to a bounding estimate of the 95th percentile CWO at the 95-percent confidence level. Since the resulting CWO is less than 1 percent, the analysis confirms that 10 CFR 50.46 acceptance criterion (b)(3), i.e., that CWO not exceed 1 percent of the total hypothetical amount, is satisfied.

The results are shown in Tables 4 and 5 for DCPP Unit 1 and Unit 2, respectively.

Coolable Geometry (10 CFR 50.46(b)(4)) – This criterion 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 DCPP Units 1 and 2 as shown in Tables 4 and 5.

Section 32.1 of the NRC-approved FSLOCA EM (Reference 1) documents 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 the combined LOCA and seismic loads was calculated to not occur for DCPP Units 1 and 2, based on performance of the fuel assembly structural analysis that, in turn, was based on a pipe break that considers the application of leak-before-break as approved for DCPP in License Amendment (LA) 221 for the Unit 1 Operating License (OL) and LA 223 for the Unit 2 OL (Reference 8).

Long-Term Cooling (10 CFR 50.46(b)(5)) – This criterion 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).

In summary, based on the analysis results for the small-break LOCA (SBLOCA, Region I) and large-break LOCA (LBLOCA, Region II) presented in Table 4 for DCPP Unit 1 and Table 5 for DCPP Unit 2, and the discussions above relative to the criteria in 10 CFR 50.46(b)(4) and (b)(5), it is concluded that DCPP Units 1 and 2 would continue to comply with the criteria in 10 CFR 50.46 upon approval of this LAR.

3.4. Comparison to Results from Analyses of Record (AORs)

The existing SBLOCA and LBLOCA AOR results for DCPP Units 1 and 2 are presented in Tables 6 and 7, respectively.

The SBLOCA AOR results for DCPP Unit 1 and DCPP Unit 2 originate from analyses performed with an evaluation model developed according to Appendix K of 10 CFR Part 50. The FSLOCA EM is a best-estimate plus uncertainty method, which relaxes some of the conservatisms required for EMs developed to Appendix K of 10 CFR Part 50. The improvement in the SBLOCA analysis PCT results is primarily attributed to the more realistic treatment of various phenomena within the FSLOCA EM, most notably the decay heat modeling; the prior analyses assumed decay heat based on 1.2 times ANSI/ANS 5.1-1971 (Reference 5), whereas the analyses with the FSLOCA EM are based on ANSI/ANS 5.1-1979 (Reference 6).

The SBLOCA MLO results are substantially higher for the analyses with the FSLOCA EM than in the AORs. This is primarily attributed to the AOR results' only considering the oxidation accrued during the LOCA transient, whereas the results from the FSLOCA EM include the steady-state corrosion. The CWO results for both the AORs and the analyses with the FSLOCA EM indicate little-to-no core-wide oxidation during the SBLOCA transient.

The improvement in the LBLOCA PCT results are attributed to differences between the FSLOCA EM and the legacy evaluation models, including, but not limited to, the following: improvements to the statistical analysis method (elimination of the superposition penalty [DCPP Unit 1 only] and larger sample size [both units]); improvements to the fuel temperature calculation; improvements to the axial power shape methodology; and improvements to the swelling, burst, and blockage models.

The LBLOCA MLO results for the AOR versus the results from analysis with the FSLOCA EM for DCPP Unit 1 are relatively similar, while the results for DCPP Unit 2 are much higher for the analysis with the FSLOCA EM. The AOR results only consider the oxidation accrued during the LOCA transient, whereas the results from the FSLOCA EM include the steady-state corrosion. The increase in the MLO for DCPP Unit 2 is primarily attributed to the contribution of the steady-state corrosion. The DCPP Unit 1 AOR MLO result is based on a LOCA transient with a cladding temperature in excess of 2,200°F for which the time-at-temperature has been artificially increased (both artifacts of the AOR evaluation model). The excessive transient cladding temperature and artificial increase in time-at-temperature from the AOR evaluation model tend to offset the inclusion of the steady-state corrosion within the FSLOCA EM, and the resulting MLO difference is therefore much smaller than that observed for Unit 2.

The LBLOCA CWO results for the AOR and the analysis with the FSLOCA EM are relatively low for DCPP Unit 2. For DCPP Unit 1, the AOR CWO result is based on the same transient as the local oxidation result with a cladding temperature in excess of 2,200°F. As such, the reduction in CWO for the analysis with the FSLOCA EM is largely attributed to the use of a transient with lower temperature, as predicted by differences in the evaluation models.

4. REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

Section 182.a of the Atomic Energy Act requires applicants for nuclear power plant operating licenses to include TS as part of the license. The Commission's regulatory requirements related to the content of the TS are contained in 10 CFR

50.36, "Technical specifications." The TS requirements in 10 CFR 50.36 include the following categories: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation (LCOs); (3) surveillance requirements; (4) design features; and (5) administrative controls.

However, the rule does not specify the particular requirements to be included in a plant's TS. Under 10 CFR 50.36(c)(2)(ii), a limiting condition for operation must be included in TS for any item meeting one or more of the following four criteria:

Criterion 1: Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

Criterion 2: A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

Criterion 3: A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

Criterion 4: A structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.

The proposed change does not impact the TS safety limits, limiting safety system settings, and limiting control settings; LCOs; surveillance requirements; or design features.

The proposed replacement NRC-approved LOCA methodology will be included in the Administrative Controls section of the DCPP TS and would be used to determine a core operating limit. The use of the proposed NRC-approved LOCA methodology will continue to ensure that the plant is operated in a safe manner. Therefore, the proposed change is consistent with the Administrative Controls requirement of 10 CFR 50.36(c)(5).

NRC Generic Letter (GL) 88-16, "Removal of Cycle-Specific Parameter Limits from Technical Specifications," dated October 4, 1988 (Reference 7), provides that it is acceptable for licensees to control reactor physics parameter limits by specifying an NRC-approved calculation methodology. These parameter limits may be removed from the TS and placed in a cycle-specific COLR that is

required to be submitted to the NRC every operating cycle or each time it is revised.

Consistent with the guidance in NRC GL 88-16, DCPP Unit 1 and 2 TS 5.6.5, "Core Operating Limits Report (COLR)" requires the following:

- An NRC-approved methodology is to be used to determine the core operating limits listed in TS 5.6.5a;
- The specific NRC-approved methodologies used to determine the core operating limits are to be listed in TS 5.6.5b; and
- The COLR, including any midcycle revisions or supplements, is to be provided upon issuance for each reload cycle to the NRC, per TS 5.6.5d.

The COLR is defined in Section 1.1 of the TS and the reporting requirements in TS 5.6.5 require that a COLR be submitted to the NRC each operating cycle, or each time the COLR is revised. The GL also required that the TS include a list of references of the NRC-approved methodologies that are used to determine the cycle-specific core operating limits. TS 5.6.5b identifies the NRC-approved analytical methodologies that are used to determine the core operating limits for DCPP Units 1 and 2. Upon approval of the LAR, the guidance in the GL continues to be met since the proposed change will continue to specify the NRC-approved methodologies used to determine the core operating limits.

Therefore, this LAR, which proposes to replace the previous NRC-approved LOCA methodologies with the NRC-approved LOCA methodology in WCAP-16996-P-A, Rev. 1, which was used for the DCPP LOCA reanalysis, satisfies NRC GL 88-16.

4.2 Precedent

None; this is the first application of the FSLOCA EM that was approved by the NRC in WCAP-16996-P-A, Rev. 1.

4.3 Significant Hazards Consideration

PG&E has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below.

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

Response: No.

The proposed change revises TS 5.6.5b to replace the current NRC-approved Loss-of-Coolant Accident (LOCA) methodologies listed in TS 5.6.5b with another NRC-approved methodology contained in WCAP-16996-P-A, Rev. 1, "Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (FULL SPECTRUM LOCA Methodology)."

The proposed changes to the TS 5.6.5b core operating limits methodologies, consists of replacing the current five LOCA methodologies with a newer, single NRC-approved methodology (the FSLOCA EM). The NRC review of the FSLOCA EM concluded that the analytical methods are acceptable as a replacement for the current LOCA analytical methods listed in TS 5.6.5b.

The proposed change does not affect the design or function of any plant structures, systems, and components (SSCs). Thus, the proposed change does not affect plant operation, design features, or the capability of any SSC to perform its safety function. In addition, the proposed change does not affect any previously evaluated accidents in the UFSAR, or any SSCs, operating procedures, and administrative controls that have the function of preventing or mitigating any accident previously evaluated in the UFSAR. Thus, the proposed use of the FSLOCA EM will continue to assure that the plant operates in the same safe manner as before and will not involve an increase in the probability of an accident.

The analyses results determined by use of the proposed new methodology will not increase the reactor power level or the core fission product inventory, and will not change any transport assumptions or the shutdown margin requirements of the DCPP TS. As such, DCPP will continue to operate within the power distribution limits and shutdown margins required by the TS and within the assumptions of the safety analyses described in the UFSAR. As such, the proposed changes do not involve a significant increase in the consequences of an accident.

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

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

   Response: No.

The proposed change revises TS 5.6.5b to replace the current NRC-approved Loss-of-Coolant Accident (LOCA) methodologies listed in TS 5.6.5b with a single, newer NRC-approved methodology contained in WCAP-16996-P-A, Rev. 1, "Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (FULL SPECTRUM LOCA Methodology)." The NRC review of the FSLOCA EM concluded that the analytical methods are acceptable as a replacement for the current LOCA analytical methods listed in TS 5.6.5b.

The proposed change provides revised analytical methods and does not change any system functions or maintenance activities. The change does not involve physical alteration of the plant; that is, no new or different type of equipment will be installed. The change does not impact the ability of any SSC to perform its safety function consistent with the assuptions of the safety analyses and continues to assure the plant is operated within safe limits. As such, the proposed change does not create new failure modes or mechanisms that are not identifiable during testing, and no new accident precursors are generated.

Therefore, the proposed change does not create the possibility of a new or different accident from any accident previously evaluated.

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

   Response: No.

   The margin of safety is established through equipment design, operating parameters, and the setpoints at which automatic actions are initiated. The proposed change does not physically alter safety-related systems, nor does it affect the way in which safety-related systems perform their functions. The setpoints at which protective actions are initiated are not altered by the proposed changes. Therefore, sufficient equipment remains available to actuate upon demand for the purpose of mitigating an analyzed event. The NRC has reviewed and approved the new methodology for the intended use in lieu of the current methodologies; thus, the margin of safety is not reduced due to this change.

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

Based on the above evaluation, PG&E concludes that the proposed change does not involve a significant hazards consideration under the standards

set forth in 10 CFR 50.92(c), and accordingly, a finding of "no significant hazards consideration" is justified.

4.4 Conclusions

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

5. ENVIRONMENTAL CONSIDERATION

PG&E has evaluated the proposed amendment and has determined that the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6. REFERENCES

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

2. LTR-NRC-18-30, "10 CFR 50.46 Annual Notification and Reporting for 2017," July 2018.

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

4. LTR-NRC-18-50, '"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)," July 2018.

5. ANSI/ANS-5.1-1971, Proposed ANS Standard "Decay Energy Release Rates Following Shutdown of Uranium-Fueled Thermal Reactors," 1971.

6. ANSI/ANS-5.1-1979, "Decay Heat Power in Light Water Reactors," 1979.

7. NRC Generic Letter 88-16, "Removal of Cycle-Specific Parameter Limits from Technical Specifications," October 1988.

8. Amendment No. 221 to Facility Operating License No. DPR-80 and Amendment No. 223 to Facility Operating License No. DPR-82 for Diablo Canyon Power Plant Units No. 1 and 2, "Diablo Canyon Power Plant, Units 1 and 2 - Issuance of Amendments re: Application of Leak-Before-Break in the Fuel Structural Analysis Results to Satisfy 10 CFR 50.46(b)(4) (CAC Nos. MF4988 and MF4989)," dated December 3, 2015.

7. TABLES

Table 1: DCPP Unit 1 Decay Heat Uncertainty Multiplier for Analysis Results

	Case	Multiplier (units of sigma)	Multiplier (absolute units) (1)
SBLOCA (Region I)	PCT	+1.6373σ	8.29%
	MLO	+0.7894σ	4.01%
	CWO	N/A (2)	N/A (2)
LBLOCA (Region II) Offsite Power Available	PCT	+0.0429σ	0.21%
	MLO	+1.2279σ	6.24%
	CWO	+0.9678σ	4.69%
LBLOCA (Region II) Loss-of-Offsite Power	PCT	+0.1253σ	0.60%
	MLO	+0.9842σ	5.01%
	CWO	+0.9678σ	4.69%

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 SBLOCA (Region I) CWO case since the analysis result for all runs is 0.0%.

Table 2: DCPP Unit 2 Decay Heat Uncertainty Multiplier for Analysis Results

	Case	Multiplier (units of sigma)	Multiplier (absolute units) (1)
SBLOCA (Region I)	PCT	+1.3310σ	6.46%
	MLO	+0.7716σ	3.93%
	CWO	N/A (2)	N/A (2)
LBLOCA (Region II) Offsite Power Available	PCT	+1.3295σ	6.52%
	MLO	+0.4590σ	2.33%
	CWO	+0.9239σ	4.47%
LBLOCA (Region II) Loss-of-Offsite Power	PCT	+0.5458σ	2.60%
	MLO	+1.0527σ	5.35%
	CWO	+0.9515σ	4.60%

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 SBLOCA (Region I) CWO case since the analysis result for all runs is 0.0%.

Table 3. Sampled Parameters Analyzed for DCPP Unit 1 and Unit 2

	Parameter	As-Analyzed Value or Range
1.	Core power	≤ 3469 MWt ± 0.3% Uncertainty
2.	Maximum total core peaking factor (F_Q), including uncertainties	2.58
3.	Maximum hot channel enthalpy rise factor ($F_{\Delta H}$), including uncertainties	1.65
4.	Vessel average temperature (T_{AVG})	Unit 1: 565.0 – 5.5°F ≤ T_{avg} ≤ 577.3 + 5.0°F Unit 2: 565.0 – 5.5°F ≤ T_{avg} ≤ 577.6 + 5.0°F
5.	Pressurizer pressure	2250 – 60 psia ≤ P_{RCS} ≤ 2250 + 60 psia
6.	Safety injection temperature	46°F ≤ SI Temp ≤ 90°F
7.	Accumulator temperature	85°F ≤ T_{ACC} ≤ 120°F
8.	Accumulator water volume	814 ft^3 ≤ V_{ACC} ≤ 886 ft^3
9.	Accumulator pressure	579 psig ≤ P_{ACC} ≤ 664 psig

Table 4. DCPP Unit 1 Analysis Results with the FSLOCA Evaluation Model

Outcome	SBLOCA (Region I) Value	LBLOCA (Region II) Value (OPA)	LBLOCA (Region II) Value (LOOP)
95/95 PCT	1,099°F	1,632°F	1,676°F
95/95 MLO	9.5%	9.5%	9.5%
95/95 CWO	0.0%	0.10%	0.15%

Table 5. DCPP Unit 2 Analysis Results with the FSLOCA Evaluation Model

Outcome	SBLOCA (Region I) Value	LBLOCA (Region II) Value (OPA)	LBLOCA (Region II) Value (LOOP)
95/95 PCT	1,012°F	1,558°F	1,574°F
95/95 MLO	9.5%	9.5%	9.5%
95/95 CWO	0.0%	0.03%	0.04%

Table 6. DCPP Unit 1 Analysis-of-Record Results
NOTRUMP and Code Qualification Document Evaluation Models

Outcome	SBLOCA Value	LBLOCA Value (Reflood 1)	LBLOCA Value (Reflood 2)
PCT	1,391°F	1,900°F	1,860°F
MLO	0.4%	11%	
CWO	0.06%	0.89%	

Table 7. DCPP Unit 2 Analysis-of-Record Results
NOTRUMP and Automated Statistical Treatment of Uncertainty Method Evaluation Models

Outcome	SBLOCA Value	LBLOCA Value
PCT	1,288°F	1,872°F
MLO	0.2%	1.6%
CWO	0.03%	0.17%

Technical Specification Markup Pages

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

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

1. WCAP-10216-P-A, Relaxation of Constant Axial Offset Control F_Q Surveillance Technical Specification, (Westinghouse Proprietary),

2. WCAP-9272-P-A, Westinghouse Reload Safety Evaluation Methodology, (Westinghouse Proprietary),

3. WCAP-8385, Power Distribution Control and Load Following Procedures, (Westinghouse Proprietary),

~~4. WCAP-10054-P-A, Westinghouse Small Break LOCA ECCS Evaluation Model Using the NOTRUMP Code, August 1985 (Westinghouse Proprietary),~~

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

~~5. 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 (Westinghouse Proprietary),~~

5. Not used.

~~6. WCAP-12945-P-A, Westinghouse Code Qualification Document for Best-Estimate Loss of Coolant Analysis, June 1996 (Westinghouse Proprietary),~~

6. Not used.

~~7. WCAP-12945-P-A, Addendum 1-A, Revision 0, "Method for Satisfying 10 CFR 50.46 Reanalysis Requirements for Best Estimate LOCA Evaluation Models," December 2004. (Westinghouse Proprietary) (Unit 1 Only),~~

7. Not used.

~~8. WCAP-16009-P-A, Revision 0, Realistic Large-Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM), January 2005. (Westinghouse Proprietary) (Unit 2 Only),~~

8. Not used.

9. WCAP-8567-P-A, "Improved Thermal Design Procedure,"

10. WCAP-16045-P-A, "Qualification of the Two-Dimensional Transport Code PARAGON," and

11. WCAP-16045-P-A, Addendum 1-A, "Qualification of the NEXUS Nuclear Data Methodology."

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Technical Specification Retyped Pages

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5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

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2. WCAP-9272-P-A, Westinghouse Reload Safety Evaluation Methodology, (Westinghouse Proprietary),
3. WCAP-8385, Power Distribution Control and Load Following Procedures, (Westinghouse Proprietary),
4. WCAP-16996-P-A, Revision 1, "Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (FULL SPECTRUM LOCA Methodology),"
5. Not used.
6. Not used.
7. Not used.
8. Not used.
9. WCAP-8567-P-A, "Improved Thermal Design Procedure,"
10. WCAP-16045-P-A, "Qualification of the Two Dimensional Transport Code PARAGON," and
11. WCAP-16045-P-A, Addendum 1-A, "Qualification of the NEXUS Nuclear Data Methodology."

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Technical Specification Bases Markups

NOTE: Some header and footer information reflecting Westinghouse source documentation has been retained in this Attachment. This is for PG&E internal use.

and $F^N_{\Delta H}$ are analyzed in

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 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. 6, "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. Measurements are generally taken with the core at, or near, steady state 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 10 CFR 50.36(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 highest probability for a DNB condition.

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, $PF_{\Delta H}$, specified in the COLR for 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 $F^{N\,RTP}_{\Delta H}$ assures compliance with the LCO at all power levels.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The accumulators do not discharge above the pressure of their nitrogen cover gas (579 to 664 psig.) At higher pressures the ECCS centrifugal charging pumps and SI pumps injection becomes solely responsible for terminating the temperature increase.

This LCO helps to ensure that the following acceptance criteria established for the ECCS by 10 CFR 50.46 (Ref. 2) that are applicable for the accumulators 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;
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 cladding surrounding the plenum volume, were to react; and
d. Core is maintained in a coolable geometry.

Since the accumulators discharge during the blowdown and reflood phase of a LOCA, they do not contribute to the long term cooling requirements of 10 CFR 50.46, though their water volume is credited as part of the long term cooling inventory.

For both the large and small break LOCA analyses, a ~~nominal~~ contained accumulator water volume (814 cubic feet to 886 cubic feet) 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 is a peak clad temperature penalty. Depending on the~~ In the NRC-approved methodology used to analyze large breaks, an increase in water volume may result in 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~~ credits the nominal line water volume from the accumulator to the check valve. The safety analysis assumes values of ≥814 cubic feet and ≤886 cubic feet. The ~~implementation of these values is performed accounting for~~ uncertainty values used in the LOCA analyses bound instrument uncertainty.

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.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

A reduction below the accumulator LCO minimum boron concentration would produce a subsequent reduction in the available containment recirculation sump boron concentration for post LOCA shutdown and an increase in the sump pH. The maximum boron concentration is used in determining the cold leg to hot leg recirculation injection switchover time and minimum sump pH.

The large and small break LOCA analyses ~~are performed at the~~ use a ~~minimum~~ nitrogen cover pressure range (579 psig ~~), since sensitivity analyses have demonstrated that higher nitrogen cover pressure results in a computed peak clad temperature benefit.~~ to 664 psig). The maximum nitrogen cover pressure limit (664 psig) provides margin to assure inadvertent relief valve actuation does not occur.

These analysis-assumed pressures are specified in the SRs. Volumes are shown on the control board indicators as % readings on accumulator narrow range level instruments. Adjustments to the analysis parameters for instrument inaccuracies or other reasons are applied to determine the acceptance criteria used in the plant surveillance procedures. These adjustments assure the assumed analyses parameters are maintained.

The effects on containment mass and energy releases from the accumulators are accounted for in the appropriate analyses (Refs. 1 and 3).

The accumulators satisfy Criterion 2 and Criterion 3 of 10 CFR 50.36(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 LOCA, the ECCS acceptance criteria of 10 CFR 50.46 (Ref. 2) could be violated.

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

(continued)

or without

BASES

APPLICABLE SAFETY ANALYSES (continued)

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 to limit 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 the injection phase for mitigation of a small break LOCA event. This event establishes the flow and discharge head for the design point of the CCPs. The SGTR and MSLB events also credit the CCPs. The OPERABILITY requirements for the ECCS are based on the following LOCA analysis assumptions:

a. A large break LOCA event, with loss of offsite power and a single failure disabling one ~~RHR pump~~ (all EDG trains are assumed to operate ~~due to~~ requirements for modeling full active containment ECCS train 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 break LOCA. It also ensures that the centrifugal charging and SI pumps will deliver sufficient water and boron during a small break LOCA to maintain core subcriticality. For smaller break 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 10 CFR 50.36(c)(2)(ii).

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BASES

APPLICABLE SAFETY ANALYSES (continued)

LOCA

Volume

Insufficient water in the RWST could result in insufficient borated water inventory in the containment recirculation sump when the transfer to the recirculation phase occurs. The deliverable volume limit is set by the LBLOCA and containment analyses. The RWST minimum contained water volume of 455,300 gallons (93.6% level uncorrected for instrument uncertainty) is required to fully submerge the sump strainer at the initiation of recirculation mode for a LBLOCA. The sump strainer does not need to be fully submerged during a SBLOCA (Reference 7).

For the RWST, the deliverable volume is less than the total volume contained since, due to the design of the tank, the ECCS suction nozzle elevation is above the bottom of the tank, so more water can be contained than can be delivered. The contained water volume limit includes an allowance for water not usable because of tank discharge location or other physical characteristics.

The accident analysis minimum RWST low level volume of 250,000 gallons is used in the LOCA analyses to calculate a conservative switchover to sump recirculation time.

Boration

During accident conditions, the RWST provides a source of borated water to the ECCS and CS System pumps. Improper boron concentrations could result in a reduction of SDM or excessive boric acid precipitation in the core following a LOCA, as well as excessive caustic stress corrosion of mechanical components and systems inside the containment. The minimum boron concentration limit ensures that the spray and the containment recirculation sump solutions, after mixing with the sodium hydroxide from the spray additive tank, will not exceed the maximum pH values. The maximum boron concentration limit ensures that the containment recirculation sump solution will not be less than the minimum pH requirement. The design basis transients and applicable safety analyses concerning each of these systems are discussed in the Diablo Canyon UFSAR. These analyses are used to assess changes to the RWST in order to evaluate their effects in relation to the acceptance limits in the analyses.

For a large-break LOCA analysis, the RWST minimum contained water volume of 455,300 gallons (93.6% level uncorrected for uncertainty), and the lower boron concentration limit of 2300 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 upper limit on boron concentration of 2500 ppm is used to determine the maximum allowable time to initiate hot leg recirculation following a LOCA. The purpose of initiating hot leg recirculation is to avoid boron precipitation in the core following the accident when the break is in the cold leg.

(continued)

BASES

APPLICABLE SAFETY ANALYSIS

Boration (continued)

The use of minimum containment backpressure in the LOCA analysis results in a conservative calculation of Peak Clad Temperature (PCT). The basis for this conclusion is the effect that the containment pressure has on the core reflood rate. A lower containment pressure has the effect of reducing the density of the steam exiting the break, which increases the differential pressure provided by the downcomer head (this phenomena is sometimes referred to as steam binding). Thus, a higher downcomer mixture level is required to maintain the same reflood rate as before. The additional time required to establish the downcomer head translates into a reduction in the reflood rate in the core. When the downcomer has completely filled, the equilibrium reflood rate for the low containment pressure case would be less than that calculated for a high containment pressure case. This reduction in reflood rate results in a reduction in heat transfer and ultimately an increase in the calculated PCT. Thus, the regulations require that a low containment pressure be calculated in the large-break LOCA analysis.

~~When calculating containment back-pressure for LOCA peak clad temperature analysis, the CS temperature is assumed to be equal to the RWST minimum temperature limit of 35°F. If the minimum temperature limit is violated, the CS further reduces containment pressure, which decreases the core reflood as explained in the preceding paragraph.~~ For the containment response following a MSLB, the lower limit on boron concentration is used to maximize the total energy release to containment.

At the time hot leg switchover is performed, there is the potential following a cold leg LOCA that boron-diluted liquid from the containment sump will displace the boron-concentrated liquid in the core. To compensate for this momentary reduction of boron in the core, control rod insertion and the boron equivalent worth of Xenon at hot leg switch over (HLSO) have been credited after a cold leg LOCA to provide negative reactivity necessary to assure core subcriticality.

a low RWST temperature will reduce the containment back pressure during the large-break LOCA transient.

Temperature

The primary reason ~~for the TS minimum~~ RWST temperature is to ensure the water will be above freezing. In addition, ~~the LOCA analysis SATAN code assumes the containment spray temperature to be equal to the RWST TS temperature limit of 35 degrees F. Low water temperature can affect the analysis model of containment spray to result in a reduction of containment pressure, which affects core reflood and increases peak clad temperature.~~

The LOCA analysis models the containment spray temperature within the range of 46°F to 90°F, based on historical plant data. However, the safety injection flow spilled into containment is modeled at the minimum RWST temperature limit of 35°F.

BASES

APPLICABLE SAFETY ANALYSES (continued)	The SLB event bounds the LOCA event from the containment peak pressure standpoint (Ref. 1). The initial pressure condition used in the containment analysis was 16 psia (1.3 psig). This resulted in a maximum peak pressure from a SLB of 42.41 psig. The containment analysis (Ref. 1) shows that the maximum peak calculated containment pressure, results from the limiting SLB at 70% power. The maximum containment pressure resulting from the worst case SLB does not exceed the containment design pressure, 47 psig. The containment was also designed for an external pressure load equivalent to -3.5 psig. The inadvertent actuation of the Containment Spray System was analyzed to determine the resulting reduction in containment pressure (sudden cooling of -1.8 psid). The initial pressure condition used in this analysis was -1.7 psig. LCO 3.6.4 limits the operation of containment to equal to or less than -1.0 psig. This resulted in a minimum pressure inside containment of -2.8 psig, which is less 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 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)~~. 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.

(continued)

BASES (continued)

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 4 hours. The Required Action is necessary to return operation to within the bounds of the containment analysis. The 4 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 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 based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.
REFERENCES	1. UFSAR, Section 6.2. 2. ~~10 CFR 50, Appendix K.~~ Deleted.

BASES

APPLICABLE SAFETY ANALYSES (continued)	The analysis and evaluation show that under the worst case scenario, the highest peak containment pressure is 42.41 psig (experienced during an MSLB at 70% power) compared to an allowable 47 psig. The analysis shows that the peak containment temperature is 312.4°F (experienced during an MSLB at 0% power) and is compared to the environmental qualifications of plant equipment. Both results meet the intent of the design basis (refer to the Bases for LCO 3.6.4, "Containment Pressure," and LCO 3.6.5 for a detailed discussion). The analyses and evaluations assume a unit specific power level of 102% for the LOCA with one containment spray train and two CFCUs operating. The limiting case MSLB analyses and evaluations are based upon a unit specific power level of 0% or 70% with two containment spray trains and three CFCUs operating. The peak pressure case assumes a failure of the feedwater regulating valve in the faulted loop, and the peak temperature case assumes a failure of the MSIV in the faulted loop. Initial (pre-accident) containment conditions of 120°F and 1.3 psig are assumed. 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 (Ref. 2)~~.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.6.8

This SR requires verification that each CFCU actuates upon receipt of an actual or simulated safety injection signal. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

SR 3.6.6.9

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. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

SR 3.6.6.10

The CFCUs are designed to start or restart in low speed upon receipt of an SI signal. This SR ensures that this feature is functioning properly. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. UFSAR, Section 6.2.2
2. ~~10 CFR 50, Appendix K~~ Deleted.
3. UFSAR, Section 6.2.3
4. UFSAR, Section 6.2.5
5. ASME Code for Operation and Maintenance of Nuclear Power Plants, 2004 Edition including 2005 and 2006 Addenda
6. License Amendment 89/88, April 16, 1996
7. Calculation STA-075, "Minimum ECCS Flow and Minimum Recirculation Spray Flow During the Sump Recirculation Phases"
8. License Amendment 202/203, December 31, 2008
9. WCAP-16294-NP-A, Rev. 1, "Risk-Informed Evaluation of Changes to Technical Specification Required Action Endstates for Westinghouse NSSS PWRs," June 2010
10. UFSAR, Appendix 3.1A

(continued)

UFSAR Update Changes

(for information only)

NOTES: The changes to UFSAR LOCA Sections 15.3.1, 15.4.1, and 15.4.10 are included first, followed by changes to the other miscellaneous UFSAR sections.

Some header and footer information reflecting Westinghouse source documentation has been retained in this Attachment. This is for PG&E internal use.

Insert 1 (at the end of Section 15.4.10):

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

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

77. "Information Notice 98-29: Predicted Increase in Fuel Rod Cladding Oxidation," USNRC, August 1998.

78. "Transient and Accident Analysis Methods," Regulatory Guide 1.203, USNRC, December 2005.

79. "U.S. Nuclear Regulatory Commission 10 CFR 50.46 Annual Notification and Reporting for 2017," LTR-NRC-18-30, July 2018.

15.3.1 LOSS OF REACTOR COOLANT FROM SMALL RUPTURED PIPES OR FROM CRACKS IN LARGE PIPES THAT ACTUATES EMERGENCY CORE COOLING SYSTEM

Small break loss-of-coolant accident (SBLOCA) analyses have been performed for Diablo Canyon Power Plant Units 1 and 2 using the **FULL SPECTRUM**™ loss-of-coolant accident (**FSLOCA**™) evaluation model (EM).

The break sizes considered in the Westinghouse FULL SPECTRUM LOCA EM include what traditionally are defined as Small and Large Break LOCA (LBLOCA). The traditional SBLOCA analysis is referred to as the Region I analysis, and the traditional LBLOCA analysis is referred to as the Region II analysis.

The FSLOCA EM analyses for Diablo Canyon Power Plant Units 1 and 2 are discussed in Section 15.4.1 since the LBLOCA (Region II) analysis is more limiting than the SBLOCA (Region I) analysis.

15.3.1.1 Acceptance Criteria

15.3.1.1.1 10 CFR Part 50, Section 50.46, Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors

(1) *Peak cladding temperature.* The calculated maximum fuel element cladding temperature shall not exceed 2200°F.

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

(3) *Maximum hydrogen generation.* The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react. This reduces the potential for explosive hydrogen/oxygen mixtures inside containment.

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

(5) *Long-term cooling.* After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

15.3.1.1.2 Radiological Criteria

The radiological consequences of a SBLOCA are within the applicable guidelines and limits specified in 10 CFR Part 100 detailed in Section 15.5.11.

15.4.1 MAJOR REACTOR COOLANT SYSTEM PIPE RUPTURES (LOCA)

Analyses with the FULL SPECTRUM loss-of-coolant accident (FSLOCA) evaluation model (EM) have been completed for the Diablo Canyon Power Plant Units 1 and 2. Section 15.4.1 discusses the LOCA analyses.

15.4.1.1 Acceptance Criteria

15.4.1.1.1 10 CFR Part 50, Section 50.46, Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors

It must be demonstrated that there is a high level of probability that the limits set forth in 10 CFR 50.46 are met. The acceptance criteria are listed below:

(1) *Peak cladding temperature.* The calculated maximum fuel element cladding temperature shall not exceed 2200 °F.

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

(3) *Maximum hydrogen generation.* The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

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

(5) *Long-term cooling.* After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

15.4.1.1.2 Radiological Criteria

(1) The resulting potential exposures to individual members of the public and to the general population shall be lower than the applicable guidelines and limits specified in 10 CFR Part 100.

15.4.1.2 FULL SPECTRUM LOCA Evaluation Model

15.4.1.2.1 Introduction

The FULL SPECTRUM LOCA EM (Reference 75) 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 FULL SPECTRUM LOCA 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 in two regions. Region I analyzes 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 FULL SPECTRUM LOCA 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 76), which explicitly models TCD and is benchmarked to high burnup data in Reference 76. The fuel pellet thermal conductivity model in the WCOBRA/TRAC-TF2 code used in the FSLOCA EM explicitly accounts for pellet thermal conductivity degradation.

Three 10 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 1) via statistical methods. The MLO is defined as the sum of pre-transient corrosion and transient oxidation consistent with the position in Reference 77. 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 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 75). Since the Diablo Canyon Power Plant units are Westinghouse designed 4-loop plants with cold leg ECCS injection, the approved method is applicable to Diablo Canyon Power Plant Units 1 and 2.

Section 15.4.1.3 summarizes the application of the Westinghouse FSLOCA EM to Diablo Canyon Power Plant Units 1 and 2. The application of the FSLOCA EM to Diablo Canyon Power Plant Units 1 and 2 is consistent with the NRC approved methodology in WCAP-16996-P-A, Revision 1 (Reference 75) with the corrections and changes reported in LTR-NRC-18-30 (Reference 79) pursuant to 10 CFR 50.46. The application of the FSLOCA EM is also applicable for the Diablo Canyon Power Plant Units 1 and 2 plant design and operating conditions.

The major plant parameter and analysis assumptions used in the Diablo Canyon Power Plant analyses with the FSLOCA EM are provided in Table 15.4.1.2-1A for Unit 1, Table 15.4.1.2-1B for Unit 2, and Tables 15.4.1.2-2 to 15.4.1.2-7 for both Units 1 and 2.

15.4.1.2.2 Method of Analysis

15.4.1.2.2.1 FULL SPECTRUM LOCA Evaluation Model Development

In 1988, the NRC Staff amended the requirements of 10 Code of Federal Regulations (CFR) 50.46 (References 1 and 54) 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. Under the amended rules, best-estimate thermal-hydraulic models may be used in place of models with Appendix K features. After the rule change, Westinghouse developed and received approval for

a best-estimate LBLOCA EM, which is discussed in WCAP-12945-P-A (Reference 56). The EM is referred to as the Code Qualification Document (CQD). Then, Westinghouse developed and received approval for a best-estimate LBLOCA EM discussed in WCAP-16009-P-A (Reference 69), referred to as the Automated Statistical Treatment of Uncertainty Method (ASTRUM). Both the CQD and ASTRUM EMs were developed following Regulatory Guide (RG) 1.157 (Reference 55).

When the FSLOCA EM was being developed, the NRC issued RG 1.203 (Reference 78) 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.

Due to the significant differences between the Unit 1 and Unit 2 reactor vessel internals, plant-specific vessel models were developed and evaluated. The significant differences between the units are summarized below:

Unit 1	Unit 2
"Top Hat"-Upper Support Plate	Flat Upper Support Plate
Domed Lower Support Plate	Flat Lower Support Plate
Thermal Shield	Neutron Pads
Diffuser Plate	No Diffuser Plate
Downflow	Upflow
T-Hot Upper Head Temperature	Returned to T-Cold Upper Head Temperature

Subsequent subsections describe the respective results from the Unit 1 and Unit 2 FSLOCA EM analyses.

15.4.1.2.2.2 WCOBRA/TRAC-TF2 COMPUTER CODE

The FULL SPECTRUM LOCA EM (Reference 75) 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 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 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.

15.4.1.3 FSLOCA EM Analysis

15.4.1.3.1 SBLOCA (Region I) Analysis

Identification of Cause and Accident Description

A LOCA is defined as a rupture of the RCS piping or of any line connected to the system. This includes small pipe ruptures, historically a 3/8-inch diameter opening up to and including a break size of 1.0 square foot, that result in break flow rates greater than the makeup flow rate from either Centrifugal Charging Pump 1 (CCP1) or CCP2. The coolant that would be released to containment contains fission products.

The small pipe break analysis for Diablo Canyon Power Plant Units 1 and 2 considers a spectrum of break sizes to ensure that the small breaks with a likelihood of the most severe core uncovery are captured within the analysis.

Following the postulated break, a depressurization of the RCS and a loss of coolant would result. A reactor trip occurs when the pressurizer low pressure trip setpoint is reached. Safety injection (SI) is actuated when the low pressurizer pressure SI setpoint is reached. The consequences of the accident are mitigated in two ways:

1. The reactor trip and borated water injection complement void formation in the core resulting in a rapid reduction in nuclear power to a residual level corresponding to the delayed fission and fission product decay.

2. The injection of borated water ensures sufficient flooding of the core to prevent excessive fuel rod cladding temperatures.

Before the break occurs, the plant is in an equilibrium condition, i.e., the heat generated in the core is being removed via the steam generator into the secondary system. During the blowdown phase, heat from decay, hot reactor vessel internals, and the reactor vessel continues to be transferred to the RCS. The heat transfer between the RCS and the secondary system may be in either direction depending on the relative temperatures between the RCS and the steam generator secondary side. In the case of continued heat addition to the steam generator secondary side, the pressure increases and is relieved by the main steam safety valves (MSSVs). Make-up to the steam generator secondary side is automatically provided by the auxiliary feedwater pumps.

The low pressurizer pressure SI signal terminates normal feedwater flow by closing the main feedwater line isolation valves and initiates auxiliary feedwater flow by starting auxiliary feedwater pumps. The SI signal also initiates SI; for an SBLOCA, the RCS pressure can remain high for an extended period, and the high pressure SI system functions to partially mitigate the RCS inventory lost through the break.

When the RCS depressurizes below the accumulator cover pressure, the accumulators begin to inject water into the reactor coolant loops and recover the RCS inventory, effectively ending the transient.

15.4.1.3.1.1 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 until safety injection is initiated on the low pressurizer pressure SI setpoint. After reaching this setpoint and applying the safety injection delays in Tables 15.4.1.2-1A and 15.4.1.2-1B, 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 as identified in Table 15.4.1.2-7.

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 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 heat up.

Core Recovery

The RCS pressure continues to decrease, and once it reaches that of the accumulator gas pressure, the introduction of additional ECCS water from the accumulators replenishes the vessel inventory and recovers the core mixture level. The transient is considered over as the break flow is compensated by the injected flow.

15.4.1.3.1.2A Unit 1 Analysis Results

The Diablo Canyon Power Plant Unit 1 SBLOCA (Region I) analysis was performed in accordance with the NRC approved methodology in Reference 75 with the corrections and changes reported in Reference 79. The transient that produced the analysis PCT result is a cold leg break with a break diameter of 3.4-inches. The most limiting ECCS single failure of one ECCS train is assumed in the analysis as identified in Table 15.4.1.2-1A. Control rod drop is modeled for breaks less than 1 square foot assuming a 2 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 loss-of-offsite power (LOOP) transients. Safety injection is actuated when the low pressurizer pressure SI setpoint is reached after the delay.

The results of the Diablo Canyon Power Plant Unit 1 SBLOCA (Region I) uncertainty analysis are summarized in Table 15.4.1.3-1A.

Table 15.4.1.3-2A contains a sequence of events for the transient that produced the SBLOCA (Region I) analysis PCT result. Figures 15.4.1.3-1A through 15.4.1.3-12A illustrate the calculated key transient response parameters for this transient.

15.4.1.3.1.2B Unit 2 Analysis Results

The Diablo Canyon Power Plant Unit 2 SBLOCA (Region I) analysis was performed in accordance with the NRC approved methodology in Reference 75 with the corrections and changes reported in Reference 79. The transient that produced the analysis PCT result is a cold leg break with a break diameter of 3.6-inches. The most limiting ECCS single failure of one ECCS train is assumed in the analysis as identified in Table 15.4.1.2-1B. Control rod drop is modeled for breaks less than 1 square foot assuming a 2 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. Safety injection is actuated when the low pressurizer pressure SI setpoint is reached after the delay.

The results of the Diablo Canyon Power Plant Unit 2 SBLOCA (Region I) uncertainty analysis are summarized in Table 15.4.1.3-1B.

Table 15.4.1.3-2B contains a sequence of events for the transient that produced the SBLOCA (Region I) analysis PCT result. Figures 15.4.1.3-1B through 15.4.1.3-12B illustrate the calculated key transient response parameters for this transient.

15.4.1.3.2 LBLOCA (Region II) Analysis

Identification of Cause and Accident Description

Should a large break occur, rapid depressurization of the RCS to a pressure nearly equal to the containment pressure occurs in approximately 40 seconds, with a nearly complete loss of RCS inventory. Rapid voiding in the core shuts down the reactor. The SI system is actuated when the low pressurizer pressure setpoint (SI signal) is reached, and the accumulators inject upon RCS depressurization below the accumulator cover pressure, mitigating the consequences of the accident in two ways:

1. The borated water injection complements void formation in causing a rapid reduction in power to a residual level corresponding to fission product decay heat. The level of RCS mixed boron concentration is sufficient to ensure that the post-LOCA core remains subcritical. However, no credit is taken for the insertion of control rods to shut down the reactor in the large break analysis.

2. The injection of borated water provides core cooling and prevents excessive fuel rod cladding temperatures.

Before the break occurs, the reactor is assumed to be in a full power equilibrium condition, i.e., the heat generated in the core is being removed by the steam generators. At the beginning of the blowdown phase, the entire RCS contains sub-cooled liquid which transfers heat from the core by forced convection with some nucleate boiling. During blowdown, heat from fission product decay and stored energy in the fuel pellets continues to be transferred to the fuel rod cladding. After the break occurs, departure from nucleate boiling occurs.

The heat transfer between the RCS and the secondary system may be in either direction, based on the progression of the transient and the relative fluid temperatures. In the case of the large break LOCA, the primary pressure rapidly decreases below the secondary system pressure, and the steam generators become an additional heat source.

As the RCS pressure decreases to the accumulator cover pressure, injection of accumulator liquid into the cold leg begins. However, significant accumulator inventory is lost out of the break due to the phenomenon of emergency core cooling bypass. After the initial surge of accumulator inventory is lost out of the break, core bypass breaks down and the remaining accumulator liquid refills the lower portion of the reactor vessel. Reflood of the core and eventual quench of the fuel rods is accomplished by the injection of water from the RWST.

15.4.1.3.2.1 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.

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. 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 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 vessel metal mass stored energy and core decay heat, and leads to an increase in the vessel fluid mass. Eventually the core inventory increases enough that liquid entrainment is able to quench all the fuel assemblies in the core.

15.4.1.3.2.2A Unit 1 Analysis Results

The Diablo Canyon Power Plant Unit 1 LBLOCA (Region II) analysis was performed in accordance with the NRC approved methodology in Reference 75 with the corrections and changes reported in Reference 79. The analysis was performed assuming both offsite power available (OPA) and LOOP, and the results of both of the LOOP and OPA analyses are compared to the 10 CFR 50.46 limits. The transient that produced the more limiting analysis PCT result relative to the offsite power assumption is a LOOP DEG break of the cold leg. The most limiting ECCS single failure of one ECCS train is assumed in the analysis as identified in Table 15.4.1.2-1A. The results of the Diablo Canyon Power Plant Unit 1 LBLOCA (Region II) OPA and LOOP uncertainty analyses are summarized in Table 15.4.1.3-1A.

Table 15.4.1.3-3A contains a sequence of events for the transient that produced the more limiting analysis PCT result relative to the offsite power assumption. Figures 15.4.1.3-13A through 15.4.1.3-26A illustrate the key major response parameters for this transient.

The containment pressure is calculated for each LOCA transient in the analysis using the COCO code (Reference 61). 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 15.4.1.2-2, 15.4.1.2-3 and 15.4.1.2-4) 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 75. The containment model for COCO calculates a conservatively low containment pressure, including the effects of all the installed pressure reducing systems and processes such as assuming all trains of containment spray are operable and assuming fan cooler operation. The containment backpressure for the transient that produced the analysis PCT result is provided in Figure 15.4.1.3-21A.

15.4.1.3.2.2B Unit 2 Analysis Results

The Diablo Canyon Power Plant Unit 2 LBLOCA (Region II) analysis was performed in accordance with the NRC approved methodology in Reference 75 with the corrections and changes reported in Reference 79. The analysis was performed assuming both OPA and LOOP, and the results of both of the LOOP and OPA analyses are compared to the 10 CFR 50.46 limits. The transient that produced the more limiting analysis PCT result relative to the offsite power assumption is a LOOP DEG break of the cold leg. The most limiting ECCS single failure of one ECCS train is assumed in the analysis as identified in Table 15.4.1.2-1B. The results of the Diablo Canyon Power Plant Unit 1 LBLOCA (Region II) OPA and LOOP uncertainty analyses are summarized in Table 15.4.1.3-1B.

Table 15.4.1.3-3B contains a sequence of events for the transient that produced the more limiting analysis PCT result. Figures 15.4.1.3-13B through 15.4.1.3-26B illustrate the key major response parameters for this transient.

The containment pressure is calculated for each LOCA transient in the analysis using the COCO code (Reference 61). 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 15.4.1.2-2, 15.4.1.2-3 and 15.4.1.2-4) 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 75. The containment model for COCO calculates a conservatively low containment pressure, including the effects of all the installed pressure reducing systems and processes such as assuming all trains of containment spray are operable and assuming fan cooler operation. The containment backpressure for the transient that produced the analysis PCT result is provided in Figure 15.4.1.3-21B.

15.4.1.3.3 10 CFR 50.46 Requirements

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 analyses with the FSLOCA EM confirm that 10 CFR 50.46 acceptance criterion (b)(1), i.e., "Peak Cladding Temperature less than 2,200°F," is demonstrated.

The results are shown in Tables 15.4.1.3-1A and 15.4.1.3-1B for Diablo Canyon Power Plant Units 1 and 2, respectively.

(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 analyses confirm that 10 CFR 50.46 acceptance criterion (b)(2), i.e., "Maximum Local Oxidation of the cladding less than 17 percent," is demonstrated.

The results are shown in Tables 15.4.1.3-1A and 15.4.1.3-1B for Diablo Canyon Power Plant Units 1 and 2, respectively.

(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 analyses confirm that 10 CFR 50.46 acceptance criterion (b)(3), i.e., "Core-Wide Oxidation less than 1 percent," is demonstrated.

The results are shown in Tables 15.4.1.3-1A and 15.4.1.3-1B for Diablo Canyon Power Plant Units 1 and 2, respectively.

(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 the Diablo Canyon Power Plant Units 1 and 2 as previously discussed.

It is discussed in Section 32.1 of the NRC-approved FSLOCA EM (Reference 75) 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 Diablo Canyon Power Plant Units 1 and 2.

(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 75).

Based on the analysis results for SBLOCA (Region I) and LBLOCA (Region II) presented in Table 15.4.1.3-1A for Unit 1 and Table 15.4.1.3-1B for Unit 2, it is concluded that Diablo Canyon Power Plant Units 1 and 2 comply with the criteria in 10 CFR 50.46.

Table 15.4.1.2-1A. Plant Operating Range Analyzed and Key Parameters for Diablo Canyon Power Plant Unit 1

Parameter		As-Analyzed Value or Range
1.0	**Core Parameters**	
	a) Core power	≤ 3469 MWt ± 0.3% Uncertainty
	b) Fuel type	17x17 VANTAGE5 fuel, ZIRLO® Cladding Material , Non-IFBA and IFBA
	c) Maximum total core peaking factor (F_Q), including uncertainties	2.58
	d) Maximum hot channel enthalpy rise factor ($F_{\Delta H}$), including uncertainties	1.65
	e) Axial flux difference (AFD) band at 100% power, %	+10 / -14
2.0	**Reactor Coolant System Parameters**	
	a) Thermal design flow (TDF)	87,700 gpm/loop
	b) Vessel average temperature (T_{AVG})	565.0 – 5.5°F ≤ T_{avg} ≤ 577.3 + 5.0°F
	c) Pressurizer pressure	2250 – 60 psia ≤ P_{RCS} ≤ 2250 + 60 psia
	d) Reactor coolant pump (RCP) model and power	Model 93A, 6000 hp
3.0	**Containment Parameters**	
	a) Containment modeling	SBLOCA (Region I): Constant pressure equal to initial containment pressure LBLOCA (Region II): Calculated for each transient using transient-specific mass and energy releases and the information in Tables 15.4.1.2-2, 15.4.1.2-3 and 15.4.1.2-4
4.0	**Steam Generator (SG) and Secondary Side Parameters**	
	a) Steam generator tube plugging level	≤ 15%
	b) Main steam safety valve (MSSV) nominal set pressures, uncertainty and accumulation	Table 15.4.1.2-7
	c) Main feedwater temperature	435°F (the maximum value from the range is used)

Table 15.4.1.2-1A. Plant Operating Range Analyzed and Key Parameters for Diablo Canyon Power Plant Unit 1

Parameter		As-Analyzed Value or Range
	d) Main feedwater isolation time from signal to full valve closure	9.0 seconds
	e) Auxiliary feedwater temperature	Nominal (80°F)
	f) Auxiliary feedwater flow rate	97.5 gpm/SG
	g) Auxiliary feedwater delivery delay time	60 seconds
5.0	**Safety Injection (SI) Parameters**	
	a) Single failure configuration	ECCS: Loss of one train of pumped ECCS LBLOCA (Region II) containment pressure: All SI trains
	b) Safety injection temperature	46°F ≤ SI Temp ≤ 90°F
	c) Low pressurizer pressure safety injection safety analysis limit	1695 psia
	d) Initiation delay time from low pressurizer pressure SI setpoint to full SI flow	≤ 42 seconds (OPA) or 67 seconds (LOOP)
	e) Safety injection flow	Minimum flows in Table 15.4.1.2-5 (SBLOCA, Region I) or Table 15.4.1.2-6 (LBLOCA, Region II)
6.0	**Accumulator Parameters**	
	a) Accumulator temperature	$85°F \le T_{ACC} \le 120°F$
	b) Accumulator water volume	$814\ ft^3 \le V_{ACC} \le 886\ ft^3$
	c) Accumulator pressure	$579\ psig \le P_{ACC} \le 664\ psig$
	d) Accumulator boron concentration	≥ 2200 ppm
7.0	**Reactor Protection System Parameters**	
	a) Low pressurizer pressure reactor trip signal processing time	≤ 2.0 seconds
	b) Low pressurizer pressure reactor trip setpoint	1860 psia

Table 15.4.1.2-1B. Plant Operating Range Analyzed and Key Parameters for Diablo Canyon Power Plant Unit 2

Parameter		As-Analyzed Value or Range
1.0	**Core Parameters**	
	a) Core power	≤ 3469 MWt ± 0.3% Uncertainty
	b) Fuel type	17x17 VANTAGE5 fuel, ZIRLO® Cladding Material , Non-IFBA and IFBA
	c) Maximum total core peaking factor (F_Q), including uncertainties	2.58
	d) Maximum hot channel enthalpy rise factor ($F_{\Delta H}$), including uncertainties	1.65
	e) Axial flux difference (AFD) band at 100% power, %	+10 / -14
2.0	**Reactor Coolant System Parameters**	
	a) Thermal design flow (TDF)	88,500 gpm/loop
	b) Vessel average temperature (T_{AVG})	565.0 – 5.5°F ≤ T_{avg} ≤ 577.6 + 5.0°F
	c) Pressurizer pressure	2250 – 60 psia ≤ P_{RCS} ≤ 2250 + 60 psia
	d) Reactor coolant pump (RCP) model and power	Model 93A, 6000 hp
3.0	**Containment Parameters**	
	a) Containment modeling	SBLOCA (Region I): Constant pressure equal to initial containment pressure LBLOCA (Region II): Calculated for each transient using transient-specific mass and energy releases and the information in Tables 15.4.1.2-2, 15.4.1.2-3 and 15.4.1.2-4
4.0	**Steam Generator (SG) and Secondary Side Parameters**	
	a) Steam generator tube plugging level	≤ 15%
	b) Main steam safety valve (MSSV) nominal set pressures, uncertainty and accumulation	Table 15.4.1.2-7
	c) Main feedwater temperature	435°F (the maximum value from the range is used)
	d) Main feedwater isolation time from signal to full valve closure	9.0 seconds
	e) Auxiliary feedwater temperature	Nominal (80°F)
	f) Auxiliary feedwater flow rate	97.5 gpm/SG

Table 15.4.1.2-1B. Plant Operating Range Analyzed and Key Parameters for Diablo Canyon Power Plant Unit 2

Parameter		As-Analyzed Value or Range
	g) Auxiliary feedwater delivery delay time	60 seconds
5.0	**Safety Injection (SI) Parameters**	
	a) Single failure configuration	ECCS: Loss of one train of pumped ECCS LBLOCA (Region II) containment pressure: All SI trains
	b) Safety injection temperature	46°F ≤ SI Temp ≤ 90°F
	c) Low pressurizer pressure safety injection safety analysis limit	1695 psia
	d) Initiation delay time from low pressurizer pressure SI setpoint to full SI flow	≤ 42 seconds (OPA) or 67 seconds (LOOP)
	e) Safety injection flow	Minimum flows in Table 15.4.1.2-5 (SBLOCA, Region I) or Table 15.4.1.2-6 (LBLOCA, Region II)
6.0	**Accumulator Parameters**	
	a) Accumulator temperature	85°F ≤ T_{ACC} ≤ 120°F
	b) Accumulator water volume	814 ft^3 ≤ V_{ACC} ≤ 886 ft^3
	c) Accumulator pressure	579 psig ≤ P_{ACC} ≤ 664 psig
	d) Accumulator boron concentration	≥ 2200 ppm
7.0	**Reactor Protection System Parameters**	
	a) Low pressurizer pressure reactor trip signal processing time	≤ 2.0 seconds
	b) Low pressurizer pressure reactor trip setpoint	1860 psia

Table 15.4.1.2-2. Containment Data Used for LBLOCA (Region II) Calculation of Containment Pressure for Diablo Canyon Power Plant Units 1 and 2

Parameter	Value
Maximum containment net free volume	2.63×10^6 ft^3
Minimum initial containment temperature at full power operation	85°F
Minimum RWST temperature for containment spray	46°F ≤ RWST Temp ≤ 90°F
Minimum RWST temperature for broken loop spilling SI	35°F
Minimum containment outside air / ground temperature	32.4°F
Minimum initial containment pressure at normal full power operation	13.7 psia
Minimum containment spray pump initiation delay from containment high pressure signal time	≥ 40.8 seconds (OPA) or 46.8 seconds (LOOP)
Maximum containment spray flow rate from all pumps	6800 gpm
Minimum recirculation spray pump initiation time under OPA conditions	Recirculation spray flow initiated immediately after RWST Low Level setpoint is reached
Maximum number of containment fan coolers in operation during LOCA transient	5
Minimum fan cooler initiation delay time	≥ 0.0 seconds (OPA) or 15.0 seconds (LOOP)
Maximum heat removal rate per fan cooler as a function on containment temperature	Table 15.4.1.2-3
Maximum number of containment venting lines (including purge lines, pressure relief lines or any others) which can be OPEN at onset of transient at full power operation	1
Maximum effective valve diameter of each containment venting line	12 inches
Maximum containment pressure setpoint for venting valve closure	20 psia
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 15.4.1.2-4

Table 15.4.1.2-3. Fan Cooler Performance Data Used for LBLOCA (Region II) Calculation of Containment Pressure for Diablo Canyon Power Plant Units 1 and 2

Containment Temperature (°F)	Heat Removal Rate (MBTU/hr)	Heat Removal Rate (BTU/sec)
100	14.64	4067.3
130	29.26	8128.2
160	44.27	12296.9
190	58.58	16273.1
210	67.85	18847.6
215	70.14	19483.8
240	115.35	32041.8
270	160.79	44665.1
300	196.14	54484.5

Table 15.4.1.2-4. Containment Heat Sink Data Used for LBLOCA (Region II) Calculation of Containment Pressure for Diablo Canyon Power Plant Units 1 and 2

Wall	Area (ft^2)	Thickness (in)	Material
1	65,749	42.0	Concrete
2	24,054	12.0	Concrete
3	14,313	24.0	Concrete
4	48,183	12.0	Concrete
5	15,725	12.0	Concrete
6	20,493	108.0	Concrete
7	33,867	30.0	Concrete
8	8,525	1.68	Steel
9	4,015	1.92	Steel
10	1,771	6.99	Steel
11	45,396	0.565	Steel
12	24,090	0.088	Steel
13	10,597	0.22	Steel
14	8,470	0.088	Steel
15	23,438	0.102	Steel
16	20,266	0.071	Steel

17	26,050	0.708	Steel
18	33,000	0.127	Steel
19	11,004	0.773	Steel
20	99,616	0.375	Steel
21	1,530	1.596	Steel
22	21,022	1.098	Steel
23	6,755	0.745	Steel
24	792	0.96	Steel
25	9,737	0.144	Stainless Steel
26	943	0.654	Stainless Steel
27	1,373	0.642	Steel
28	575	3.0	Steel
29	17,542	0.75	Steel

Table 15.4.1.2-5. Safety Injection Flow Used for SBLOCA (Region I) Calculation for Diablo Canyon Power Plant Units 1 and 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	231	388	2847.6
34.7	229.6	385.2	2675.4
54.7	228.2	382.4	2492.5
74.7	226.8	379.6	2295.2
94.7	225.4	376.8	2078.6
114.7	224	374	1833.8
134.7	222.4	371	1542.5
154.7	220.8	368	1154.6
174.7	219.2	365	430
174.9	219.18	364.97	0
214.7	216	359	
314.7	208	344	
414.7	201	327	
514.7	193	310	
614.7	185	292	
714.7	176	274	
814.7	168	253	
914.7	159	230	
1014.7	150	206	
1114.7	131	177	
1214.7	112	141	
1314.7	91	98	
1414.7	69	36	
1414.71	69	0	
1514.7	46	0	
1614.7	22		
1614.71	0		
3000.0	0		

Table 15.4.1.2-6. Safety Injection Flow Used for LBLOCA (Region II) Calculation for Diablo Canyon Power Plant Units 1 and 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	231	388	2848.7
34.7	229.6	384.4	2306.5
54.7	228.2	380.8	1744.7
74.7	226.8	377.2	1195.8
94.7	225.4	373.6	826.9
114.7	224	370	376
126.9	223.02	367.8	0
214.7	216	352	
314.7	208	333	
414.7	201	313	
514.7	193	291	
614.7	185	268	
714.7	176	244	
814.7	168	216	
914.7	159	174	
1014.7	150	129	
1114.7	131	76	
1214.7	112	9	
1214.71	112	0	
1314.7	91	0	
1414.7	69		
1514.7	46		
1614.7	22		
1614.71	0		
3000	0		

Table 15.4.1.2-7. Steam Generator Main Steam Safety Valve Parameters for Diablo Canyon Power Plant Units 1 and 2

Stage	Set Pressure (psig)	Uncertainty (%)	Accumulation (psi)
1	1065.0	+3	+5
2	1078.0	+3	+5
3	1090.0	+3	+5
4	1103.0	+3	+5
5	1115.0	+3	+5

Table 15.4.1.3-1A. Diablo Canyon Power Plant Unit 1 Analysis Results with the FSLOCA EM

Outcome	SBLOCA (Region I) Value	LBLOCA (Region II) Value (OPA)	LBLOCA (Region II) Value (LOOP)
95/95 PCT	1,099 °F	1,632 °F	1,676°F
95/95 MLO	9.5 %	9.5 %	9.5 %
95/95 CWO	0.0 %	0.10 %	0.15 %

Table 15.4.1.3-1B. Diablo Canyon Power Plant Unit 2 Analysis Results with the FSLOCA EM

Outcome	SBLOCA (Region I) Value	LBLOCA (Region II) Value (OPA)	LBLOCA (Region II) Value (LOOP)
95/95 PCT	1,012 °F	1,558 °F	1,574°F
95/95 MLO	9.5 %	9.5 %	9.5 %
95/95 CWO	0.0 %	0.03 %	0.04 %

Table 15.4.1.3-2A. Diablo Canyon Power Plant Unit 1 Sequence of Events for SBLOCA (Region I) Analysis PCT Transient

Event	Time after Break (sec)
Start of Transient	0.0
Reactor Trip Signal	13.7
Safety Injection Signal	24.9
Safety Injection Begins	91.9
Loop Seal Clearing Occurs	608
Top of Core Uncovered	1,040
Accumulator Injection Begins	1,374
PCT Occurs	1,506
Top of Core Recovered	1,736

Table 15.4.1.3-2B. Diablo Canyon Power Plant Unit 2 Sequence of Events for SBLOCA (Region I) Analysis PCT Transient

Event	Time after Break (sec)
Start of Transient	0.0
Reactor Trip Signal	10.0
Safety Injection Signal	19.1
Safety Injection Begins	86.1
Loop Seal Clearing Occurs	407
Top of Core Uncovered	803
Accumulator Injection Begins	1,198
PCT Occurs	1,201
Top of Core Recovered	1,296

Table 15.4.1.3-3A. Diablo Canyon Power Plant Unit 1 Sequence of Events for LBLOCA (Region II) Analysis PCT Transient Assuming LOOP

Event	Time after Break (sec)
Start of Transient	0.0
Burst Occurs	3.1
Safety Injection Signal	5.4
Accumulator Injection Begins	11.0
End of Blowdown	15.0
Accumulator Empty	52.0
Safety Injection Begins	72.4
PCT Occurs	124
All Rods Quenched	238

Table 15.4.1.3-3B. Diablo Canyon Power Plant Unit 2 Sequence of Events for LBLOCA (Region II) Analysis PCT Transient Assuming LOOP

Event	Time after Break (sec)
Start of Transient	0.0
Safety Injection Signal	5.7
Burst Occurs	10.3
Accumulator Injection Begins	11.1
End of Blowdown	15.0
PCT Occurs	40.0
Accumulator Empty	54.0
Safety Injection Begins	72.7
All Rods Quenched	188

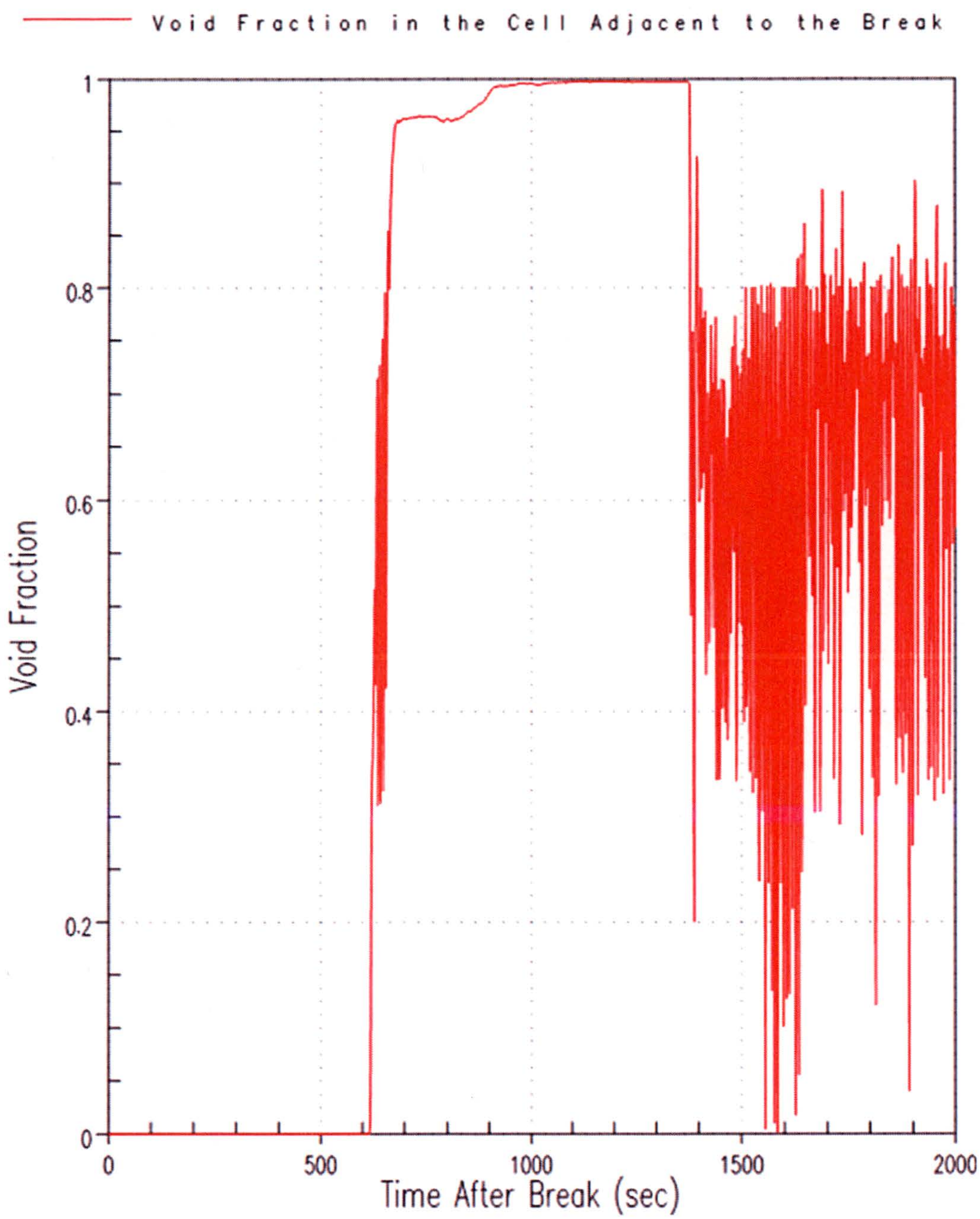

Figure 15.4.1.3-1A: Diablo Canyon Power Plant Unit 1 Break Flow Void Fraction for SBLOCA (Region I)

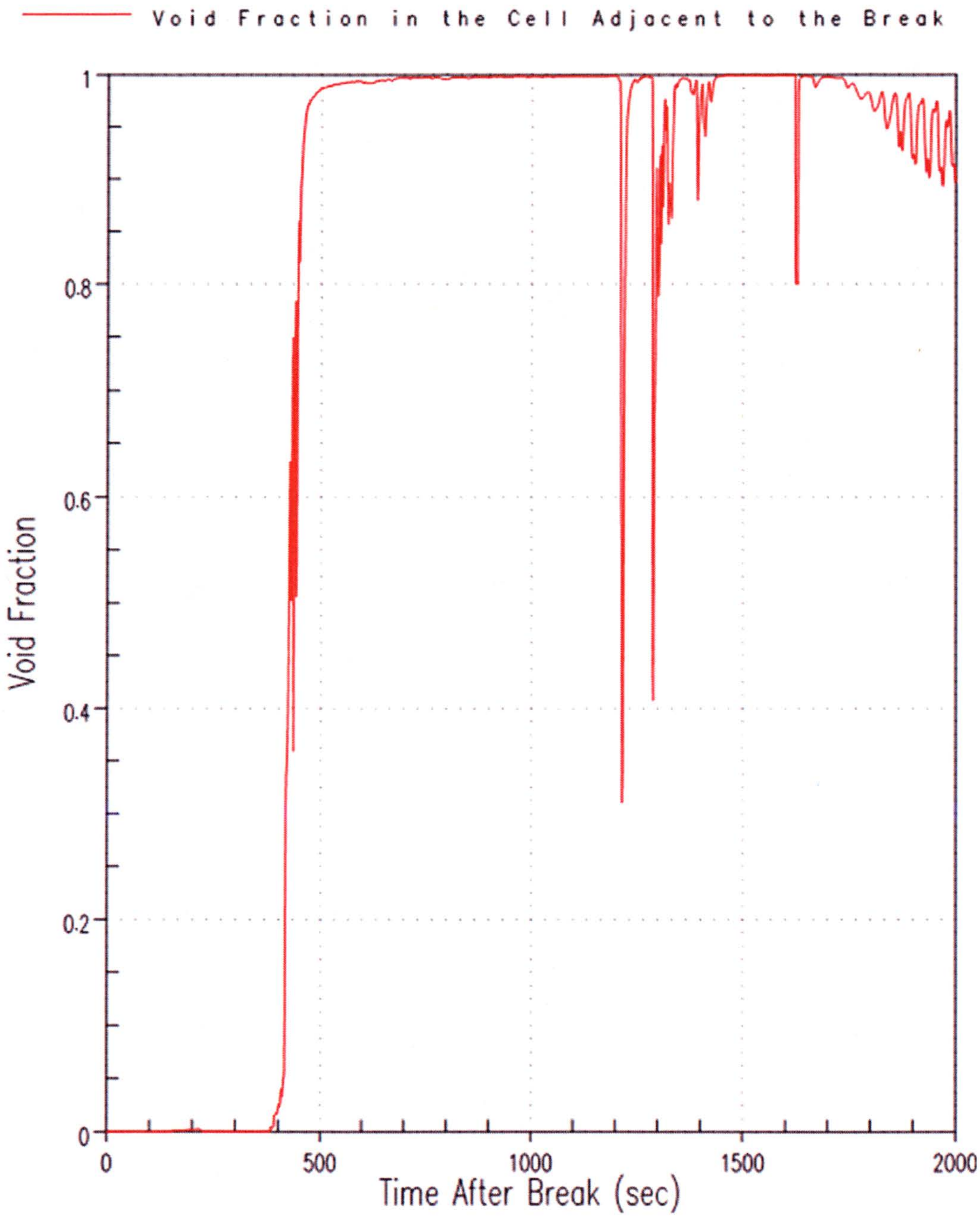

Figure 15.4.1.3-1B: Diablo Canyon Power Plant Unit 2 Break Flow Void Fraction for SBLOCA (Region I)

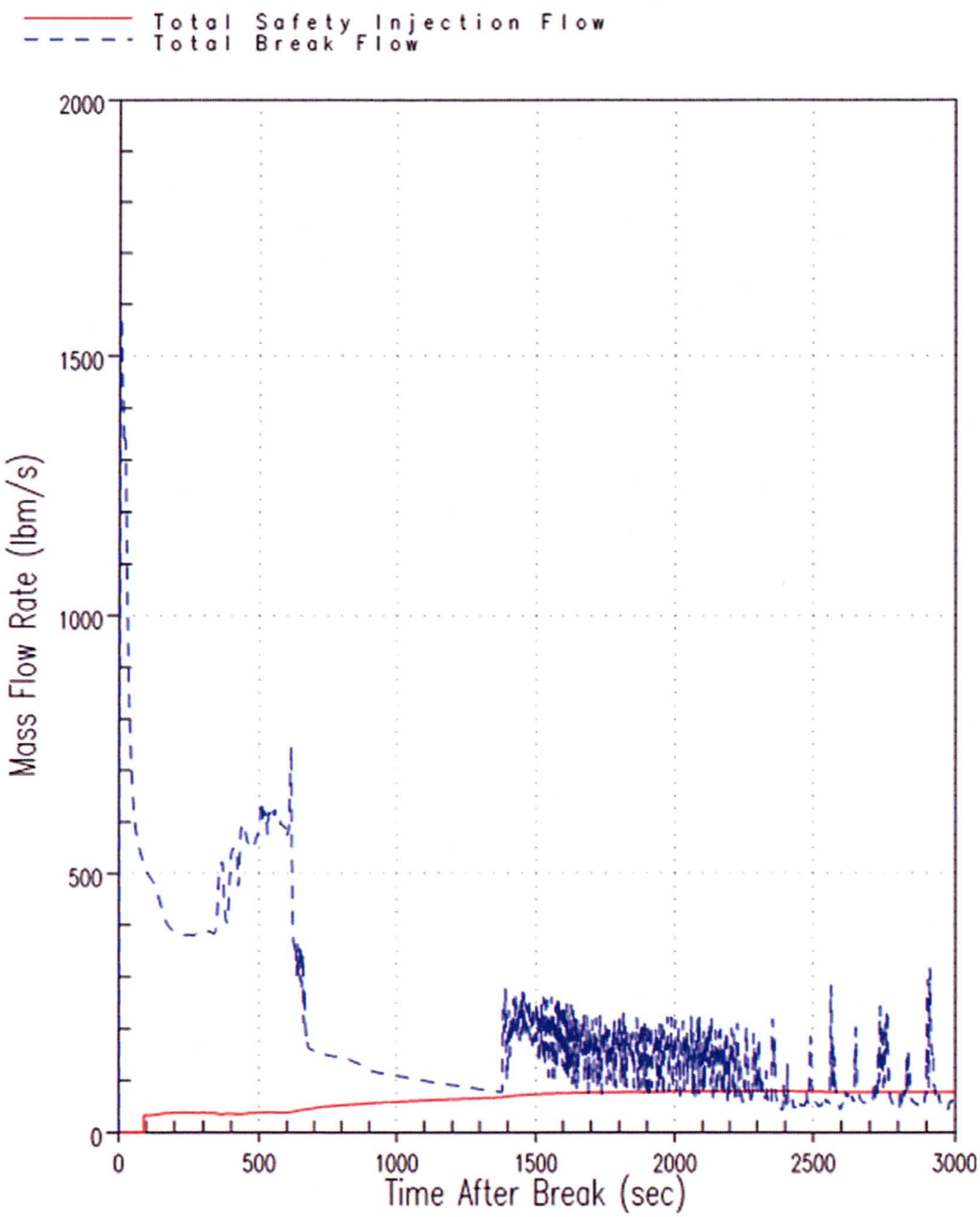

Figure 15.4.1.3-2A: Diablo Canyon Power Plant Unit 1 Total Safety Injection Flow and Total Break Flow for SBLOCA (Region I)

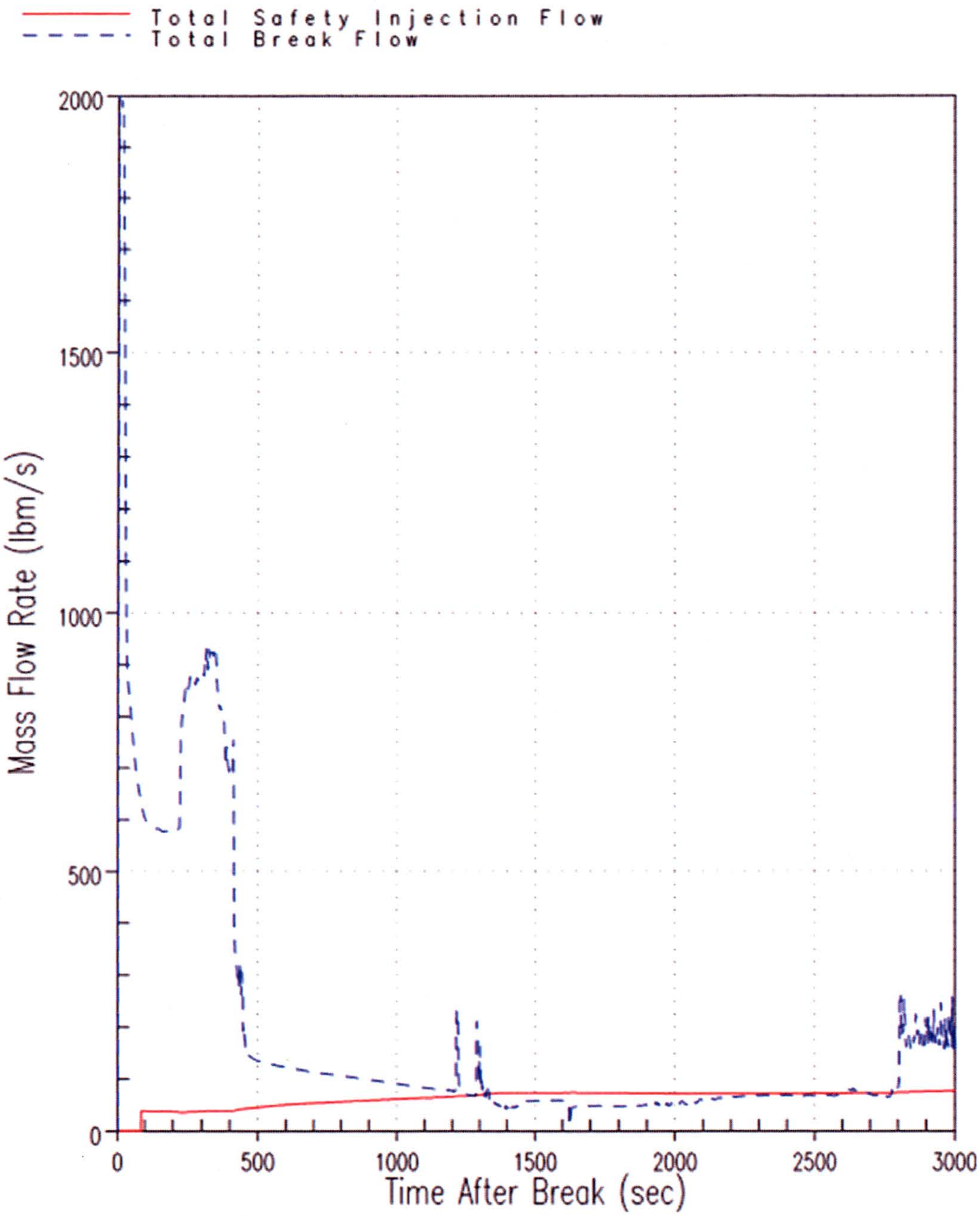

Figure 15.4.1.3-2B: Diablo Canyon Power Plant Unit 2 Total Safety Injection Flow and Total Break Flow for SBLOCA (Region I)

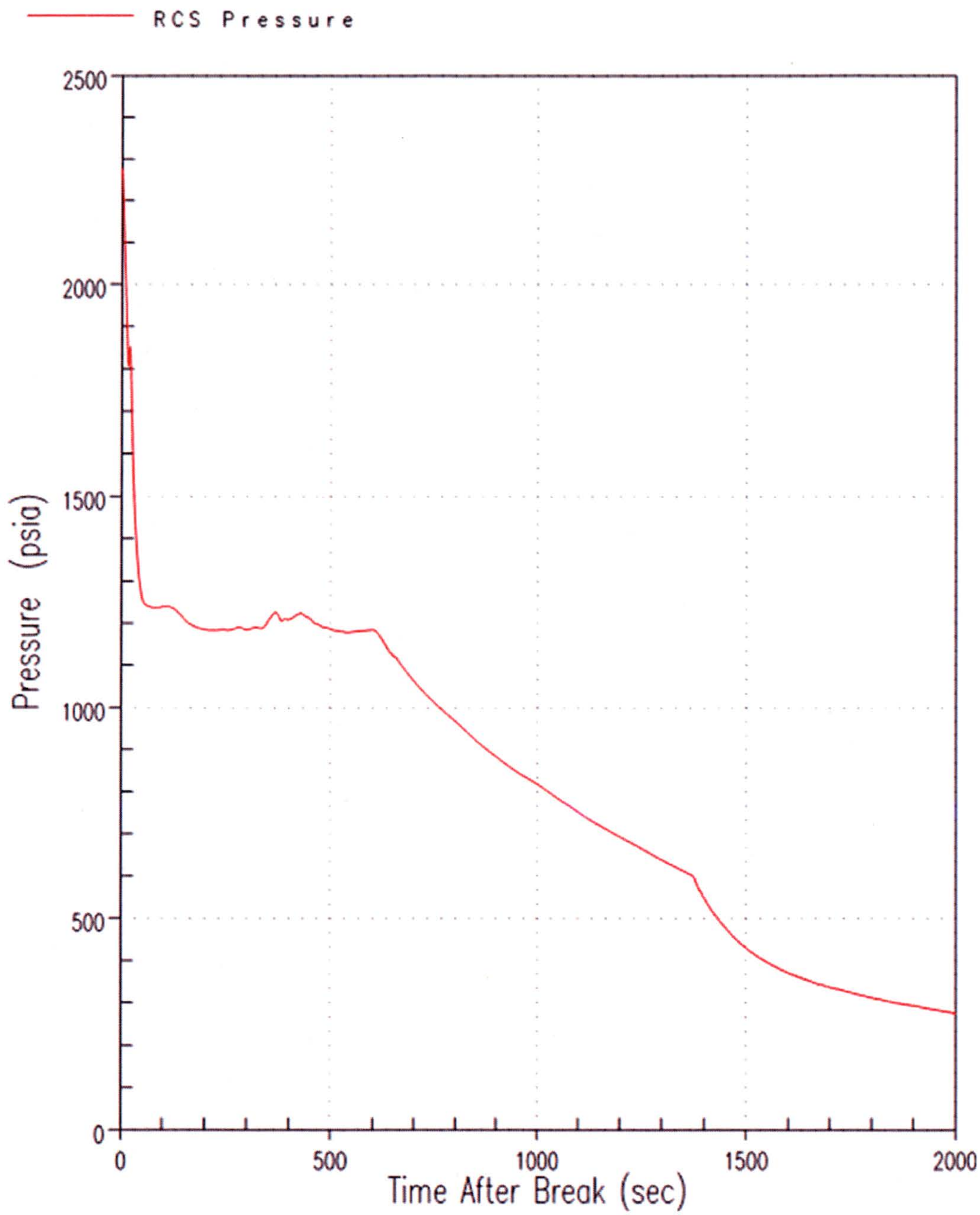

Figure 15.4.1.3-3A: Diablo Canyon Power Plant Unit 1 RCS Pressure for SBLOCA (Region I)

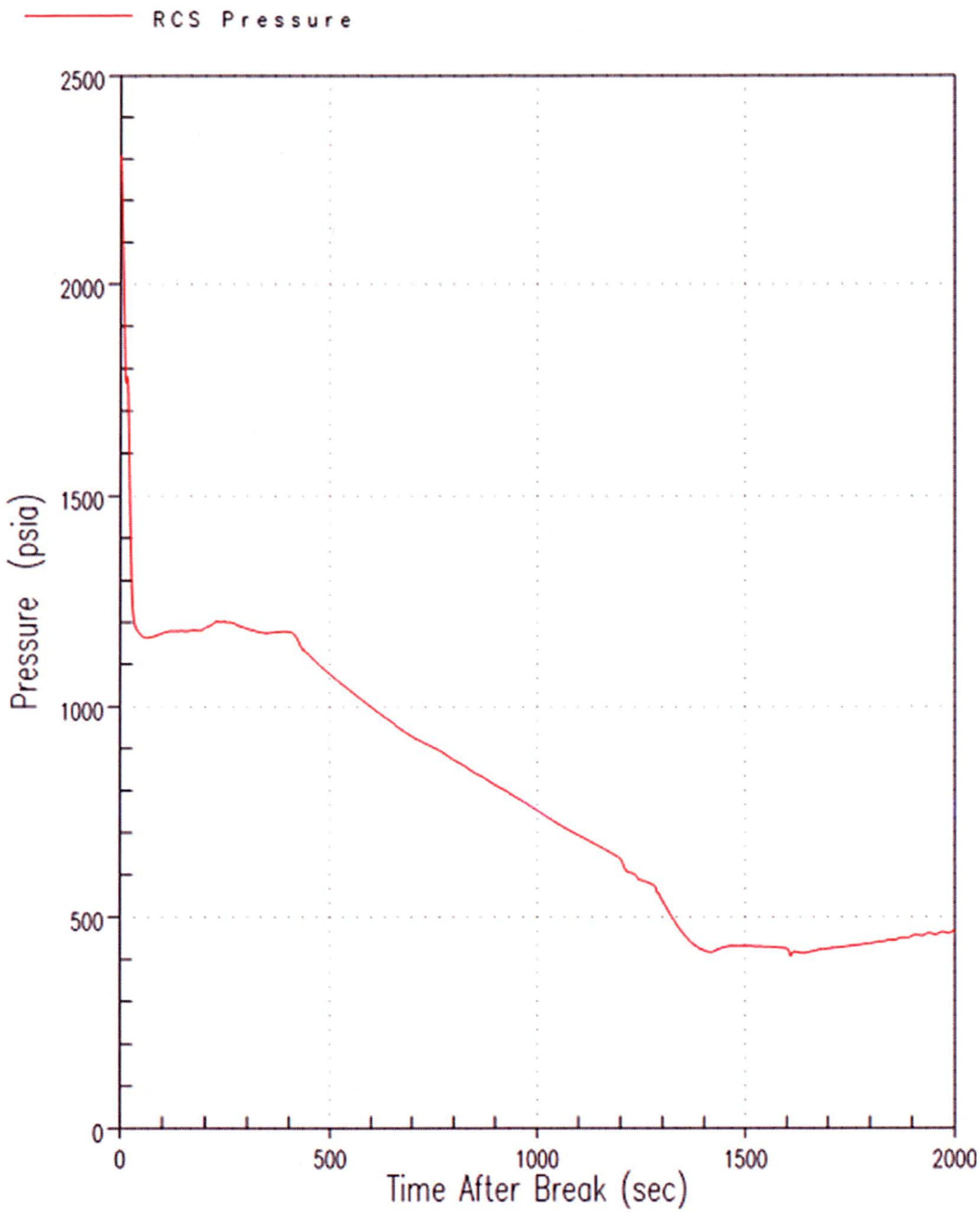

Figure 15.4.1.3-3B: Diablo Canyon Power Plant Unit 2 RCS Pressure for SBLOCA (Region I)

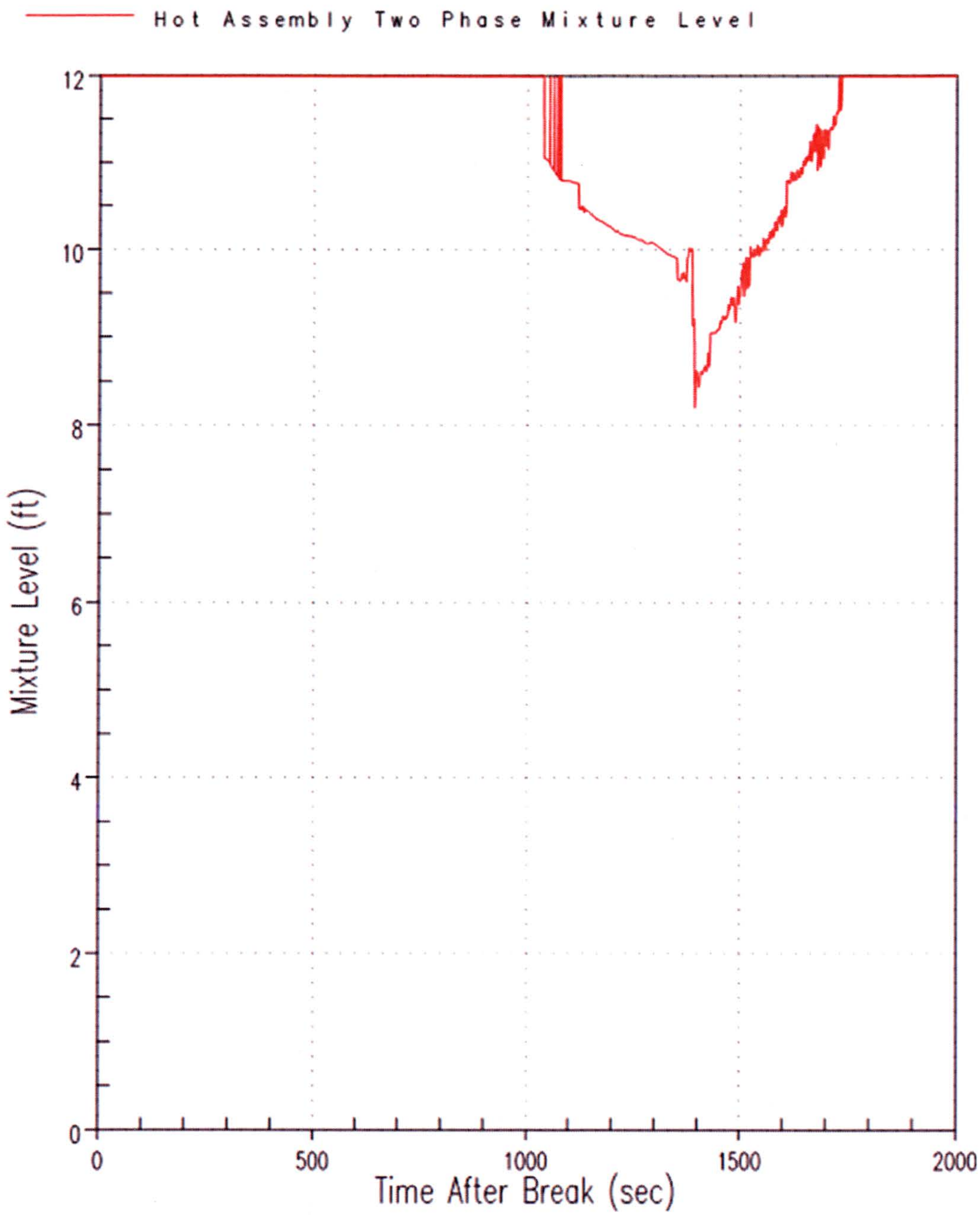

Figure 15.4.1.3-4A: Diablo Canyon Power Plant Unit 1 Hot Assembly Two-Phase Mixture Level for SBLOCA (Region I)

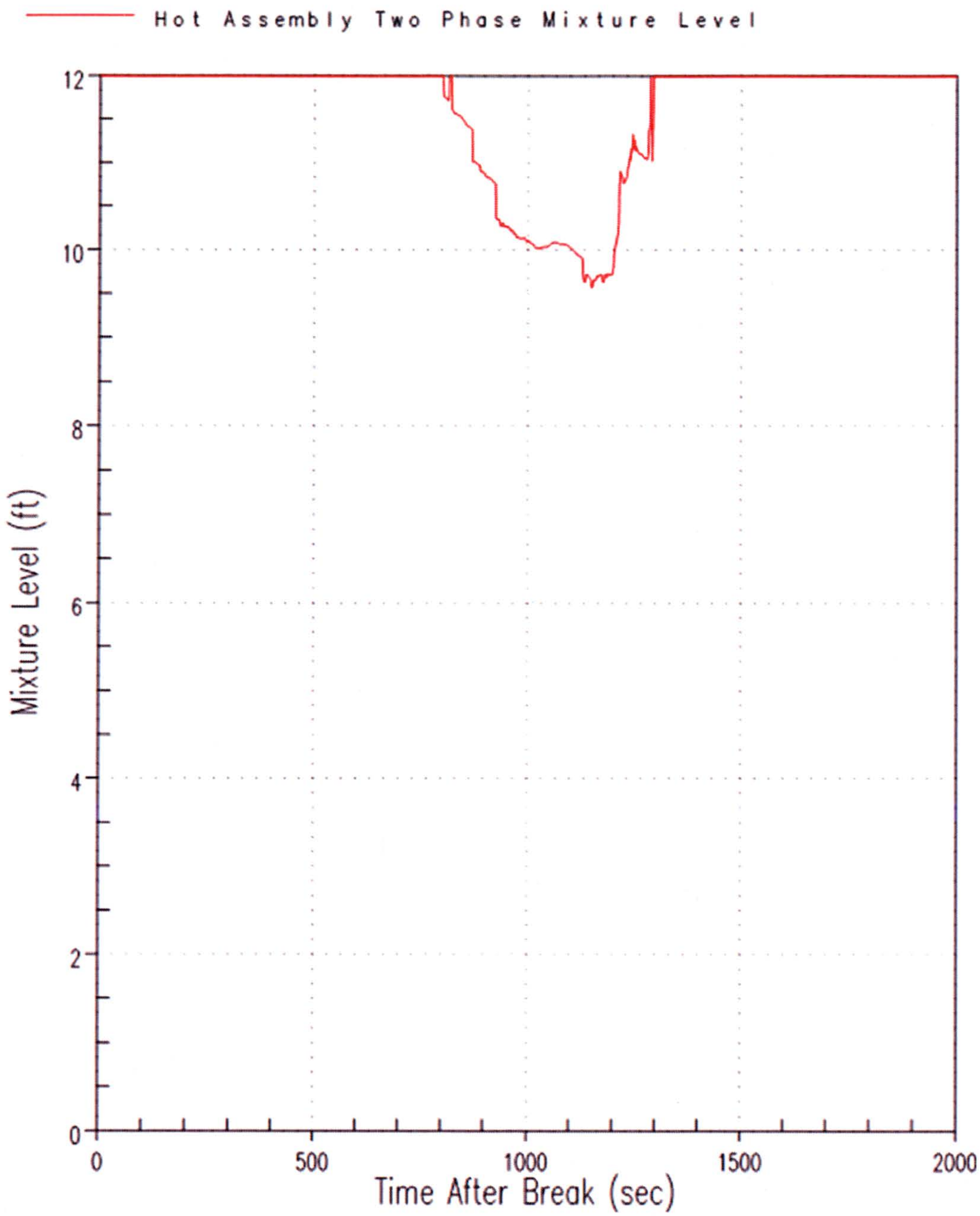

Figure 15.4.1.3-4B: Diablo Canyon Power Plant Unit 2 Hot Assembly Two-Phase Mixture Level for SBLOCA (Region I)

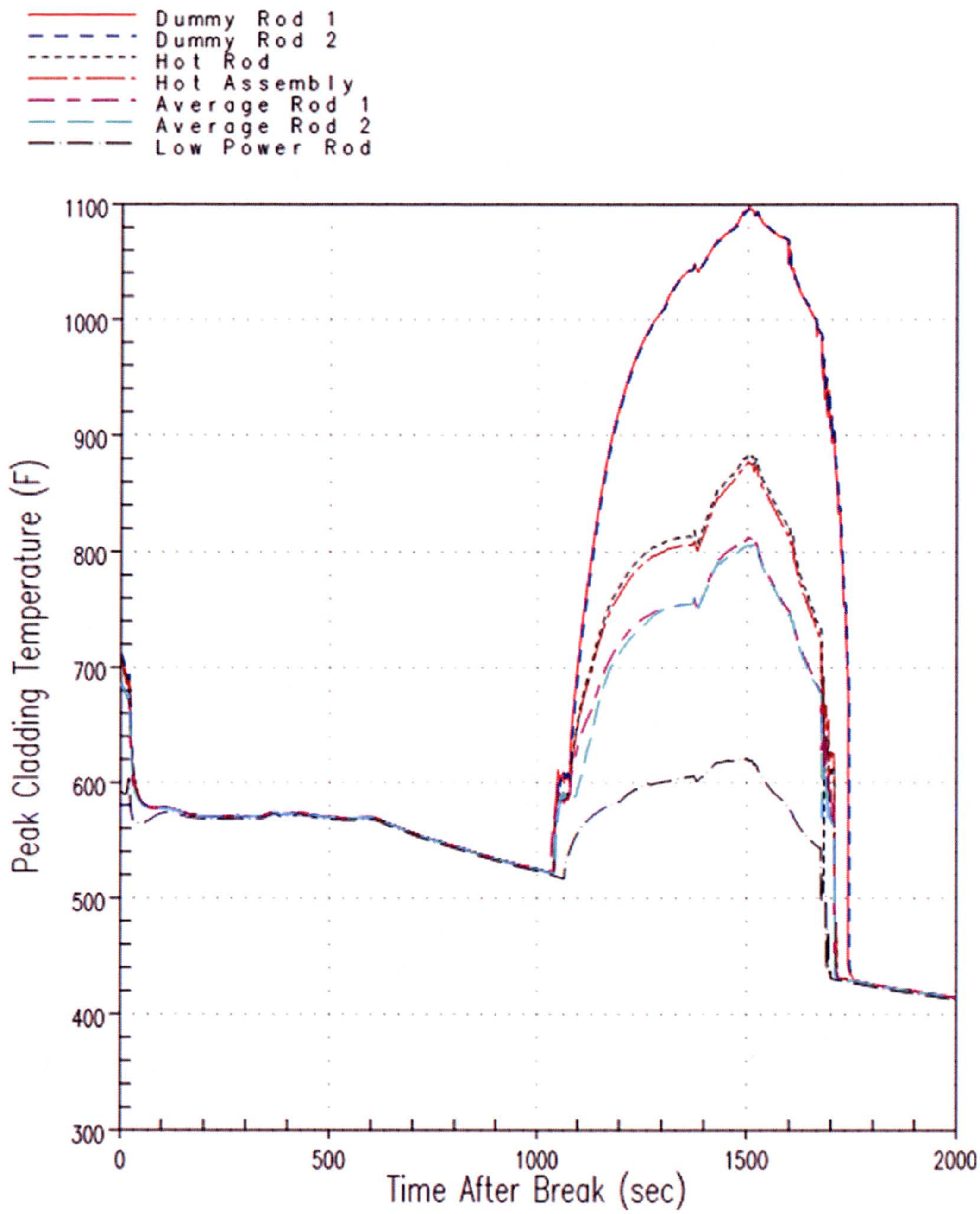

Figure 15.4.1.3-5A: Diablo Canyon Power Plant Unit 1 Peak Cladding Temperature for all Rods for SBLOCA (Region I)

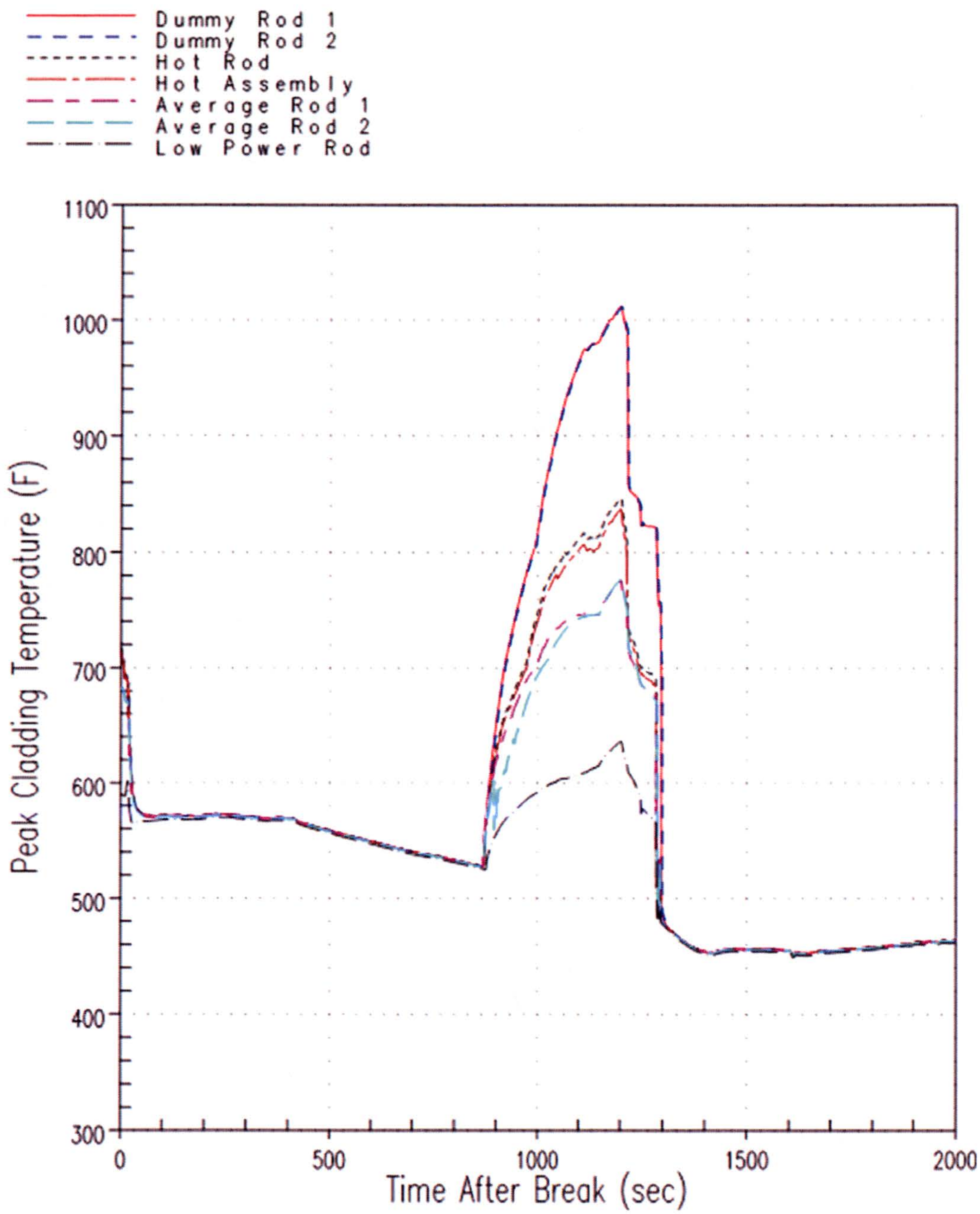

Figure 15.4.1.3-5B: Diablo Canyon Power Plant Unit 2 Peak Cladding Temperature for all Rods for SBLOCA (Region I)

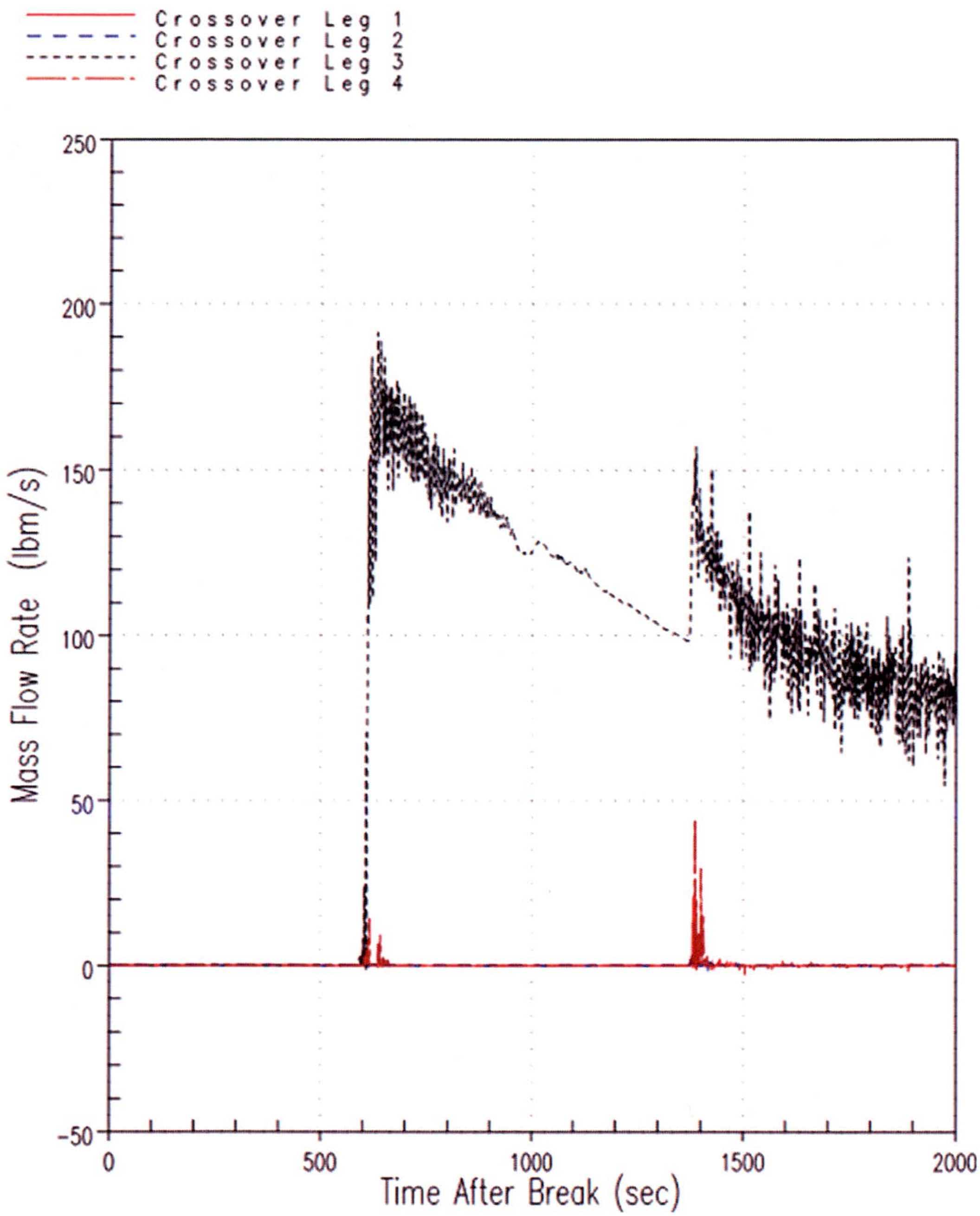

Figure 15.4.1.3-6A: Diablo Canyon Power Plant Unit 1 Vapor Mass Flow Rate through the Crossover Legs for SBLOCA (Region I)

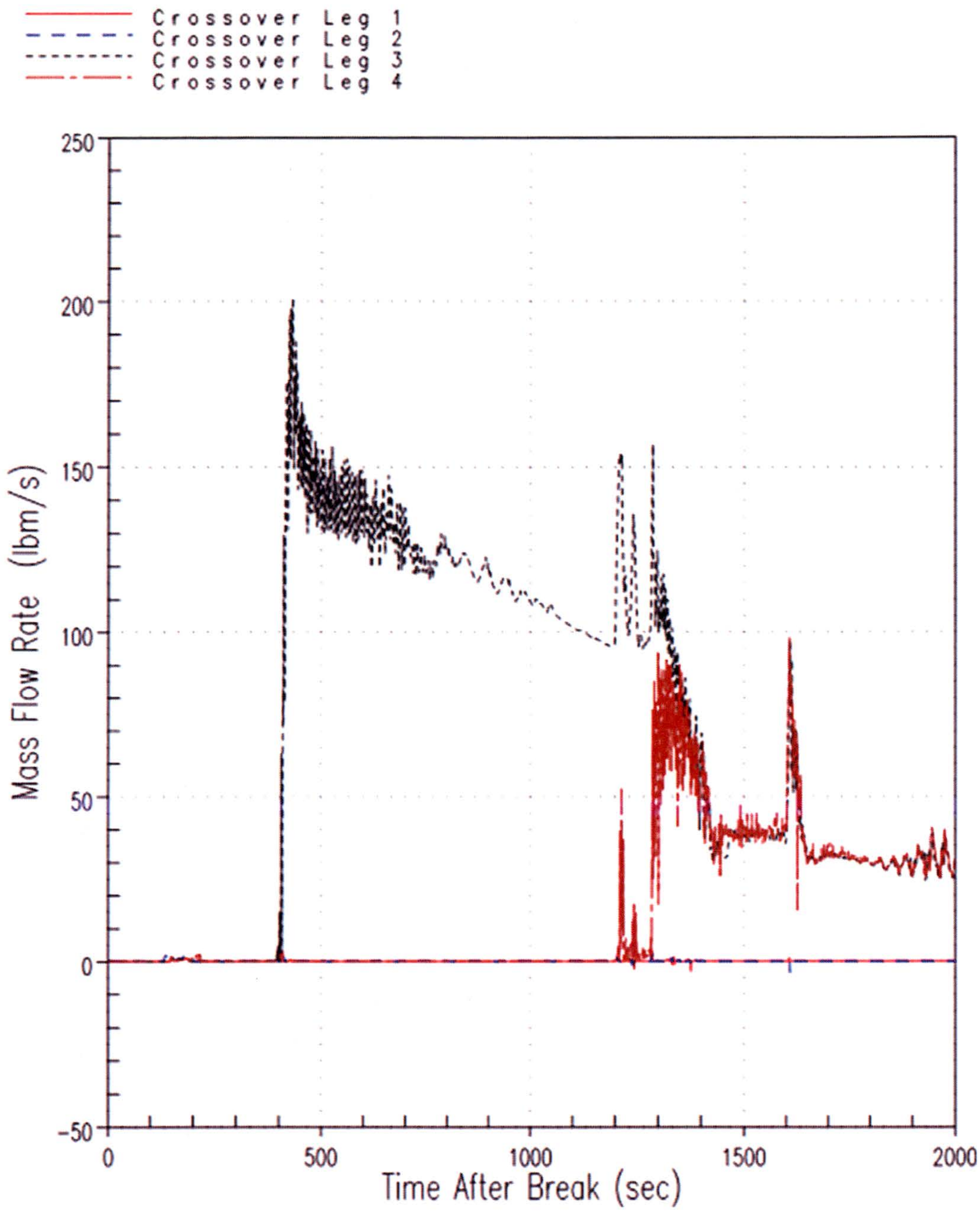

Figure 15.4.1.3-6B: Diablo Canyon Power Plant Unit 2 Vapor Mass Flow Rate through the Crossover Legs for SBLOCA (Region I)

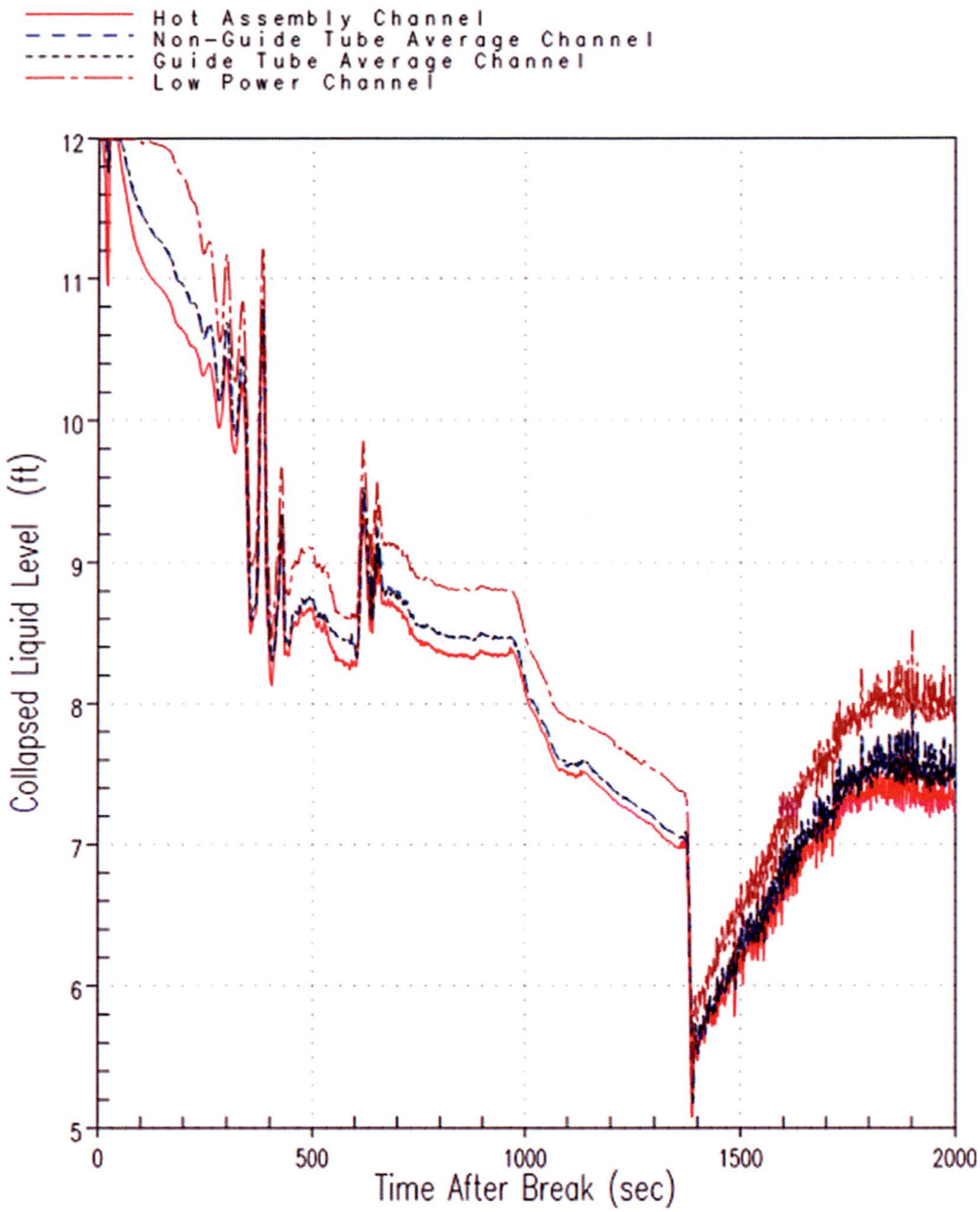

Figure 15.4.1.3-7A: Diablo Canyon Power Plant Unit 1 Core Collapsed Liquid Levels for SBLOCA (Region I)

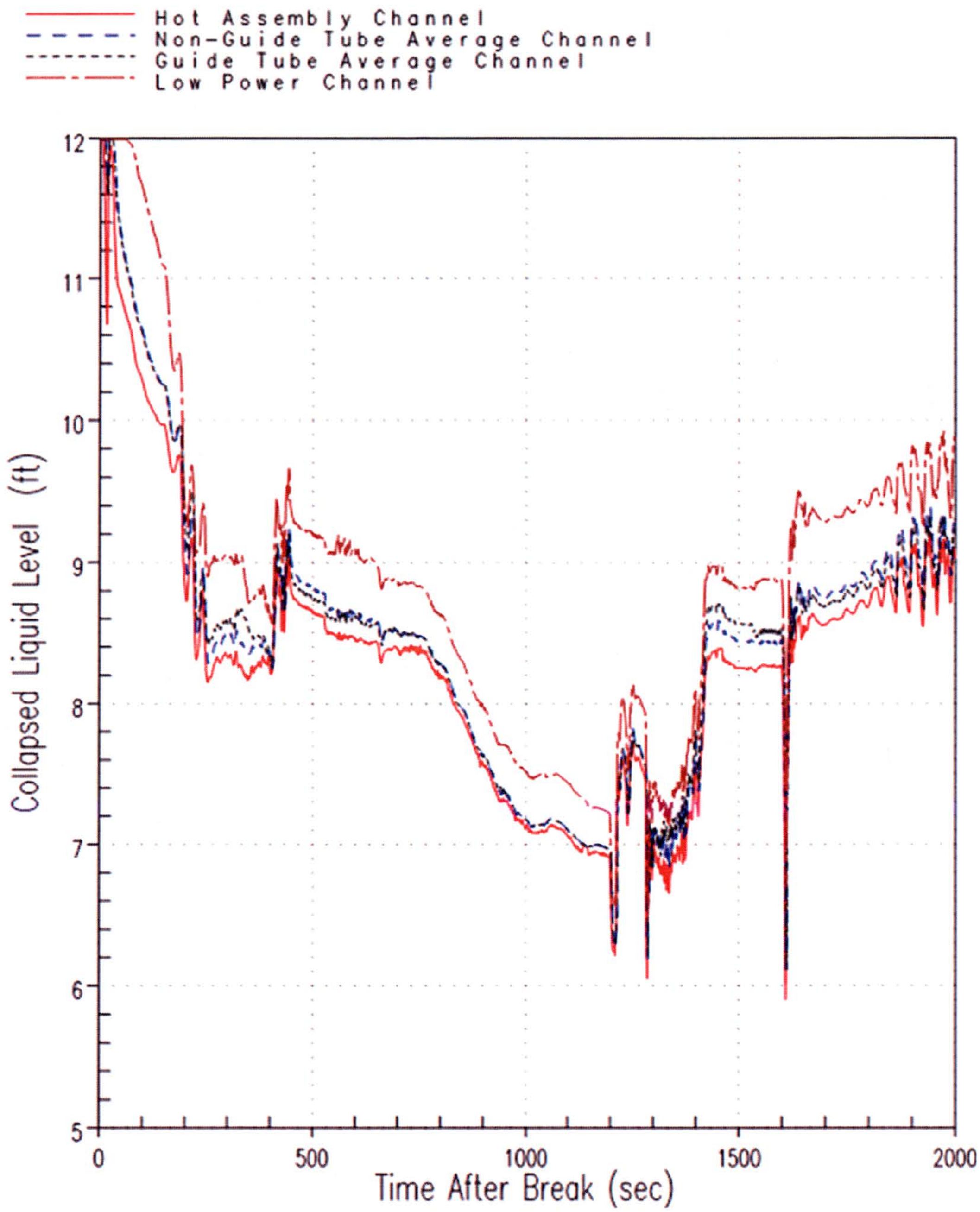

Figure 15.4.1.3-7B: Diablo Canyon Power Plant Unit 2 Core Collapsed Liquid Levels for SBLOCA (Region I)

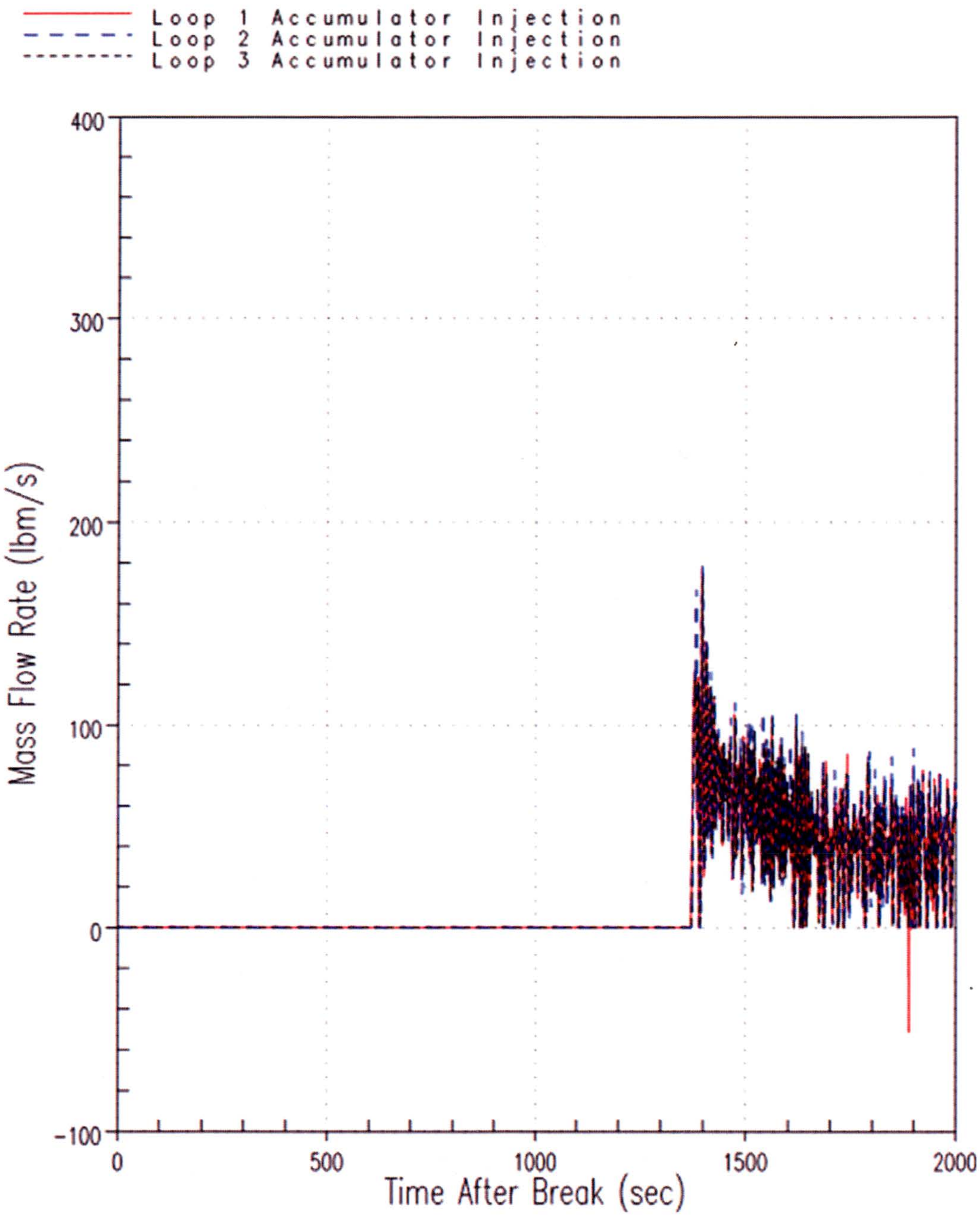

Figure 15.4.1.3-8A: Diablo Canyon Power Plant Unit 1 Accumulator Injection Flow for SBLOCA (Region I)

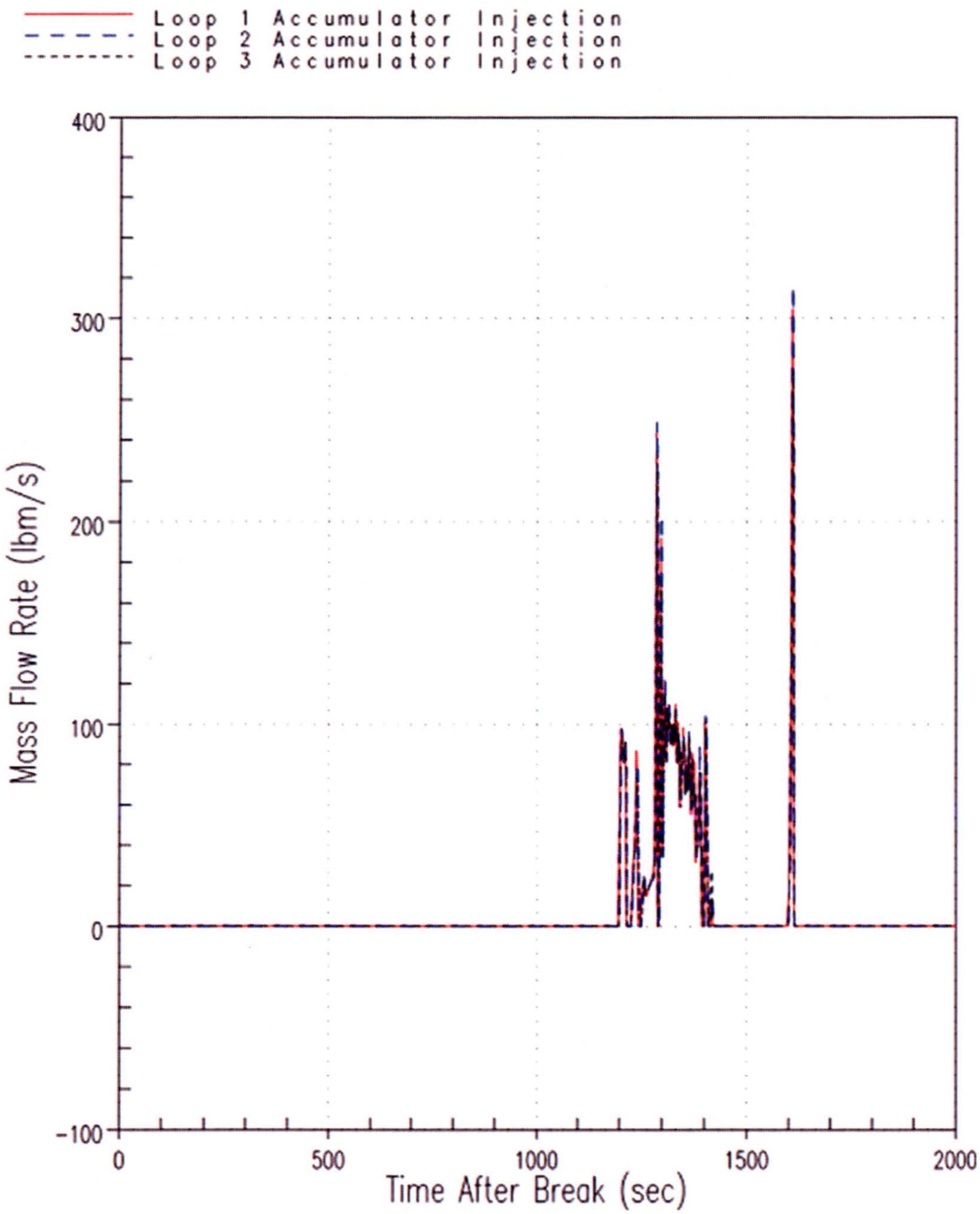

Figure 15.4.1.3-8B: Diablo Canyon Power Plant Unit 2 Accumulator Injection Flow for SBLOCA (Region I)

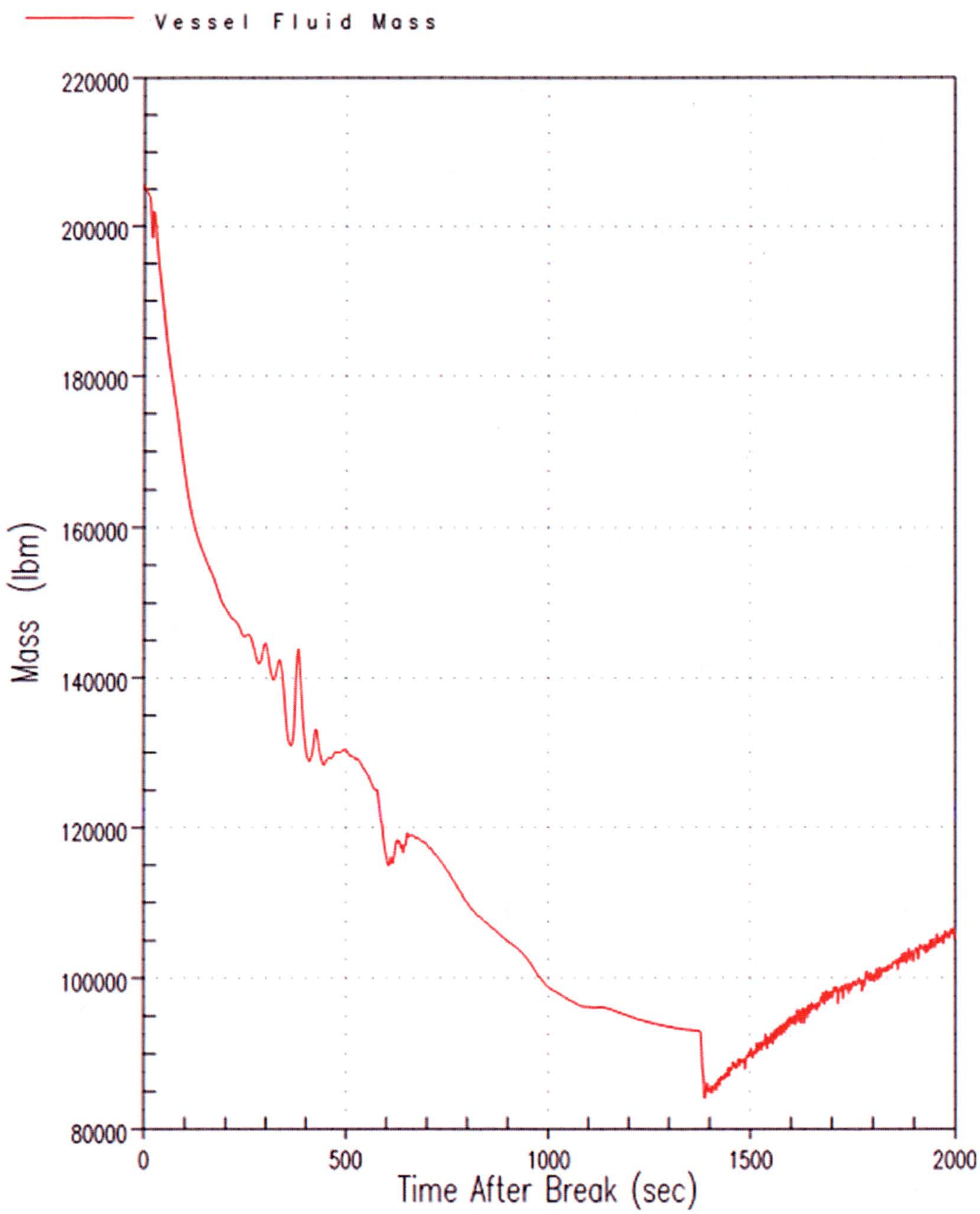

Figure 15.4.1.3-9A: Diablo Canyon Power Plant Unit 1 Vessel Fluid Mass for SBLOCA (Region I)

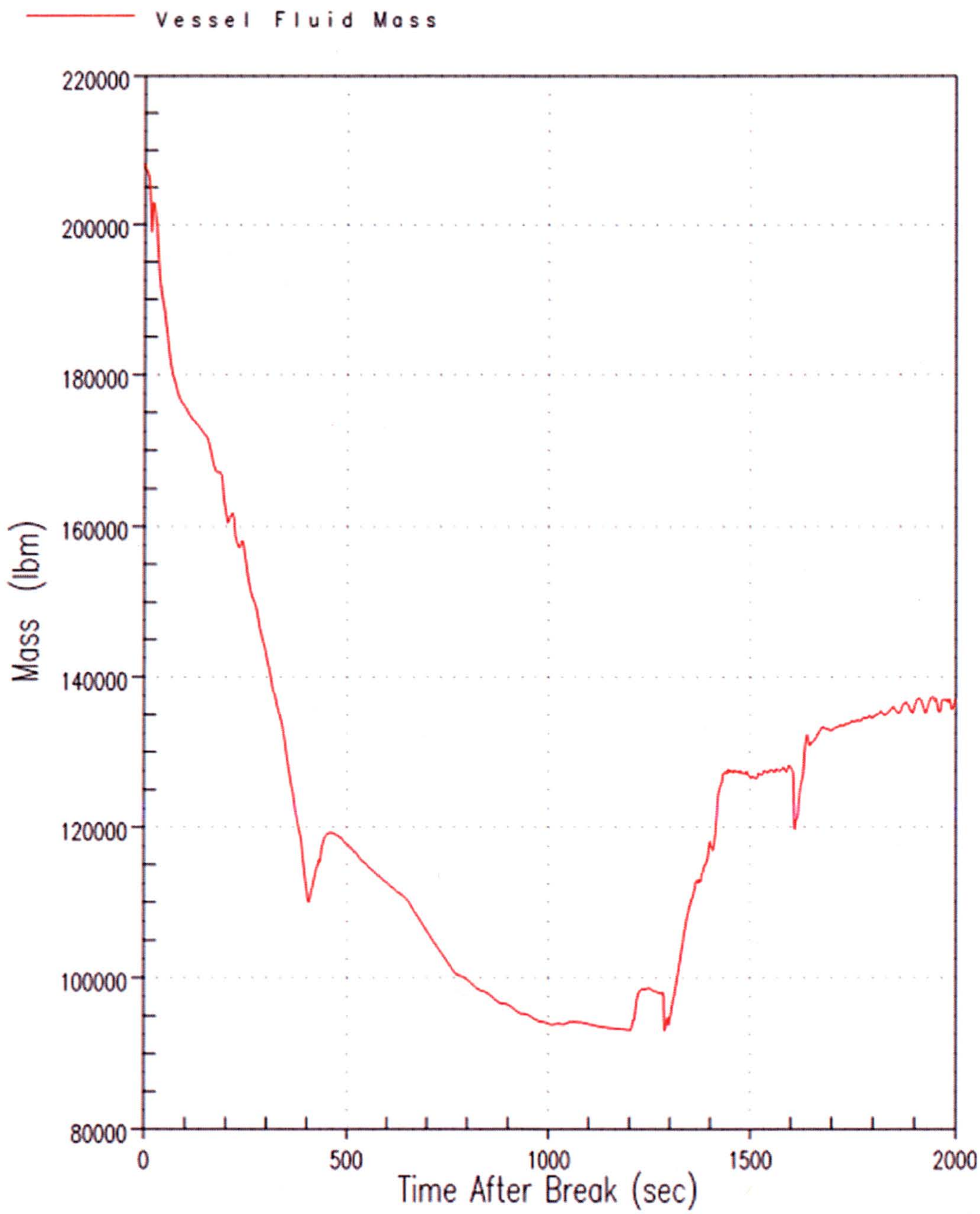

Figure 15.4.1.3-9B: Diablo Canyon Power Plant Unit 2 Vessel Fluid Mass for SBLOCA (Region I)

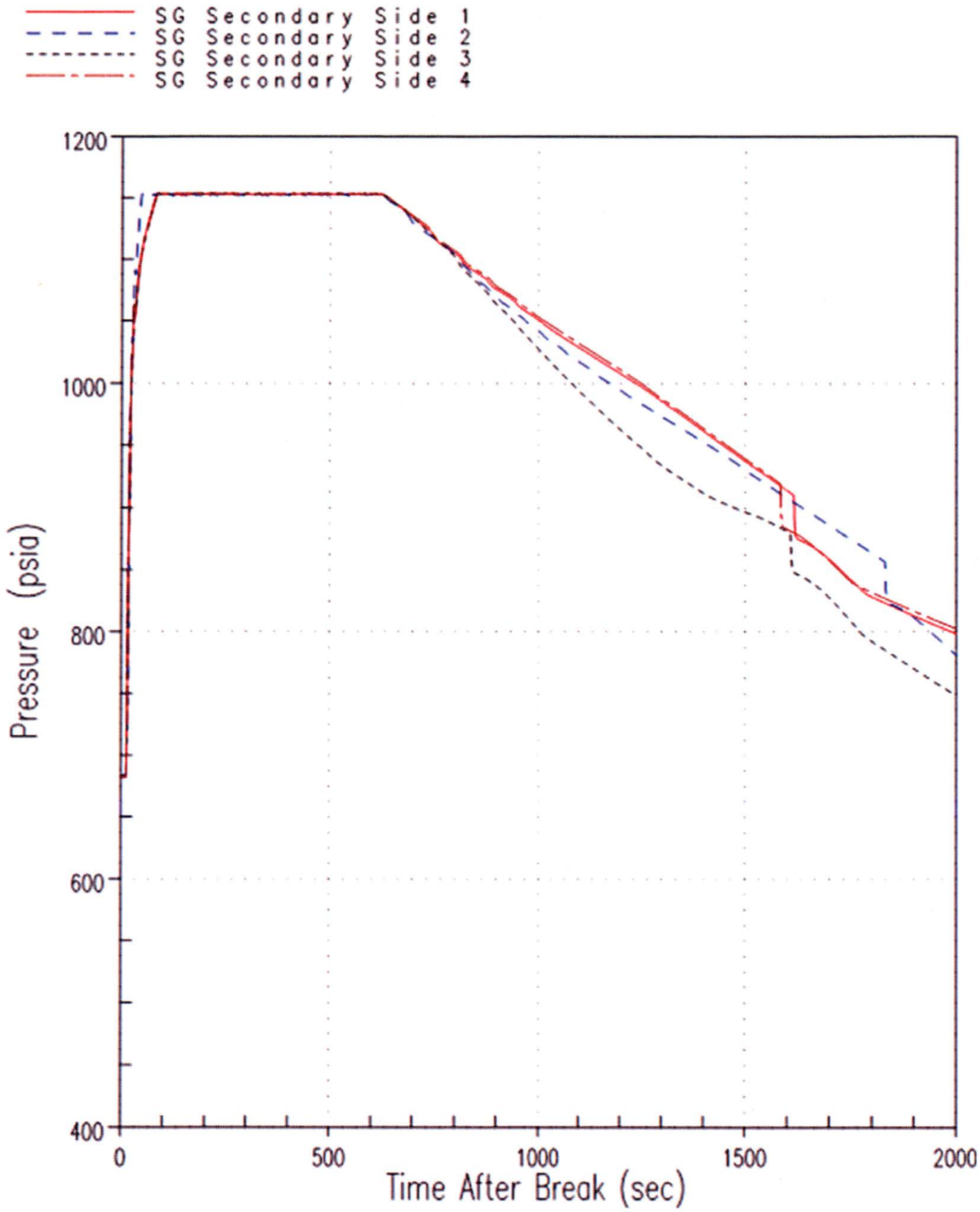

Figure 15.4.1.3-10A: Diablo Canyon Power Plant Unit 1 Steam Generator Secondary Side Pressure for SBLOCA (Region I)

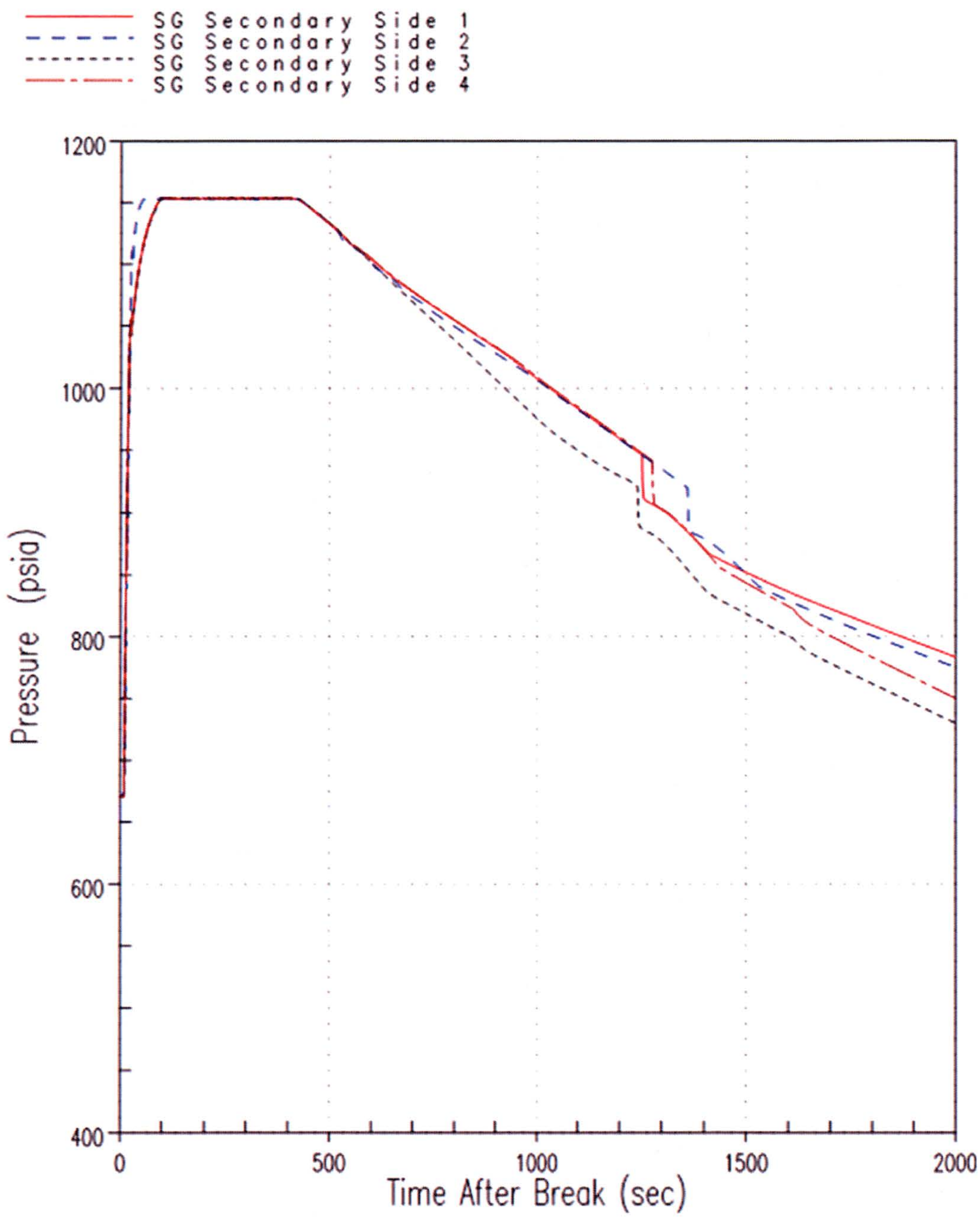

Figure 15.4.1.3-10B: Diablo Canyon Power Plant Unit 2 Steam Generator Secondary Side Pressure for SBLOCA (Region I)

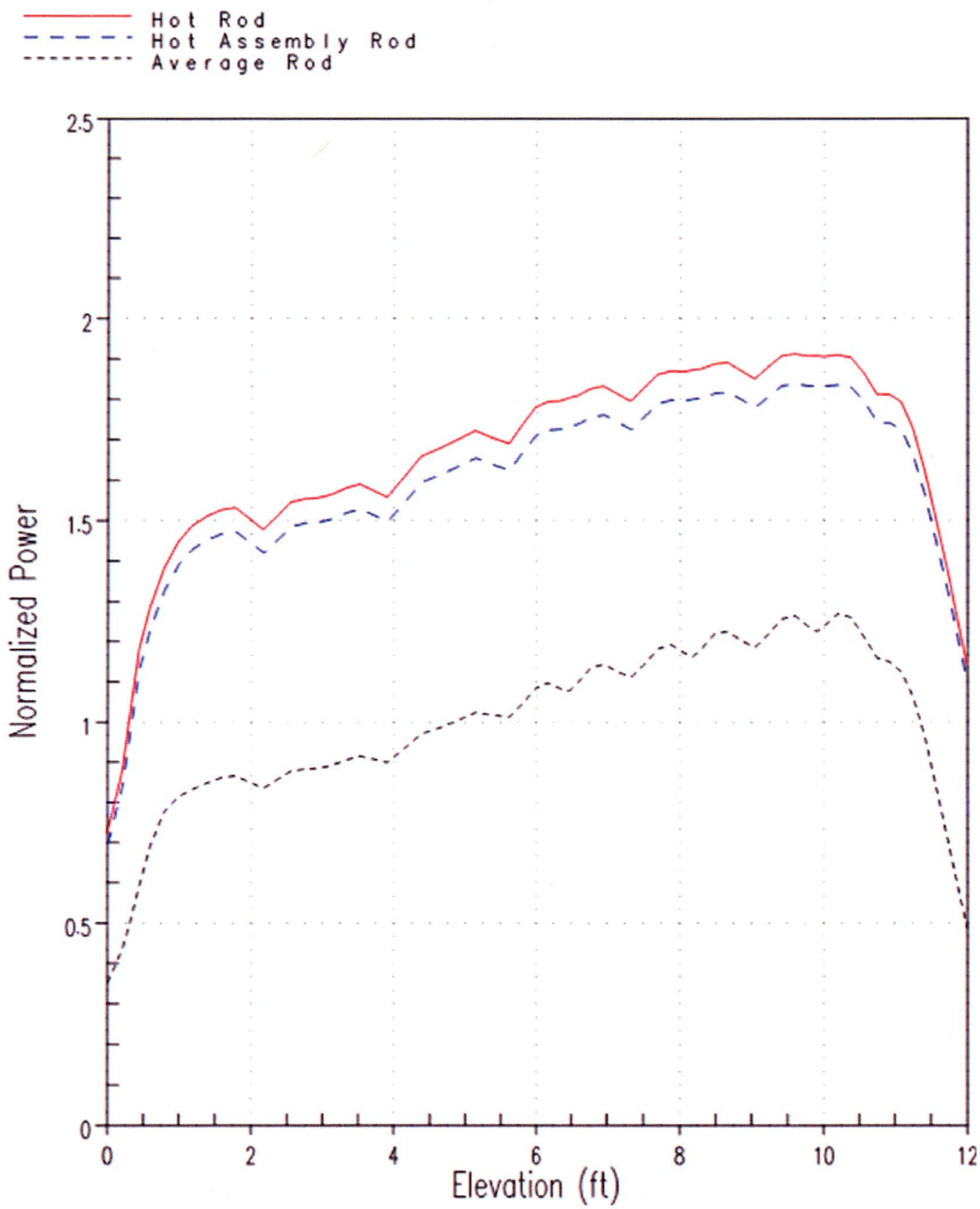

Figure 15.4.1.3-11A: Diablo Canyon Power Plant Unit 1 Normalized Core Power Shape for SBLOCA (Region I)

Note: The localized power decreases occur at grid elevations.

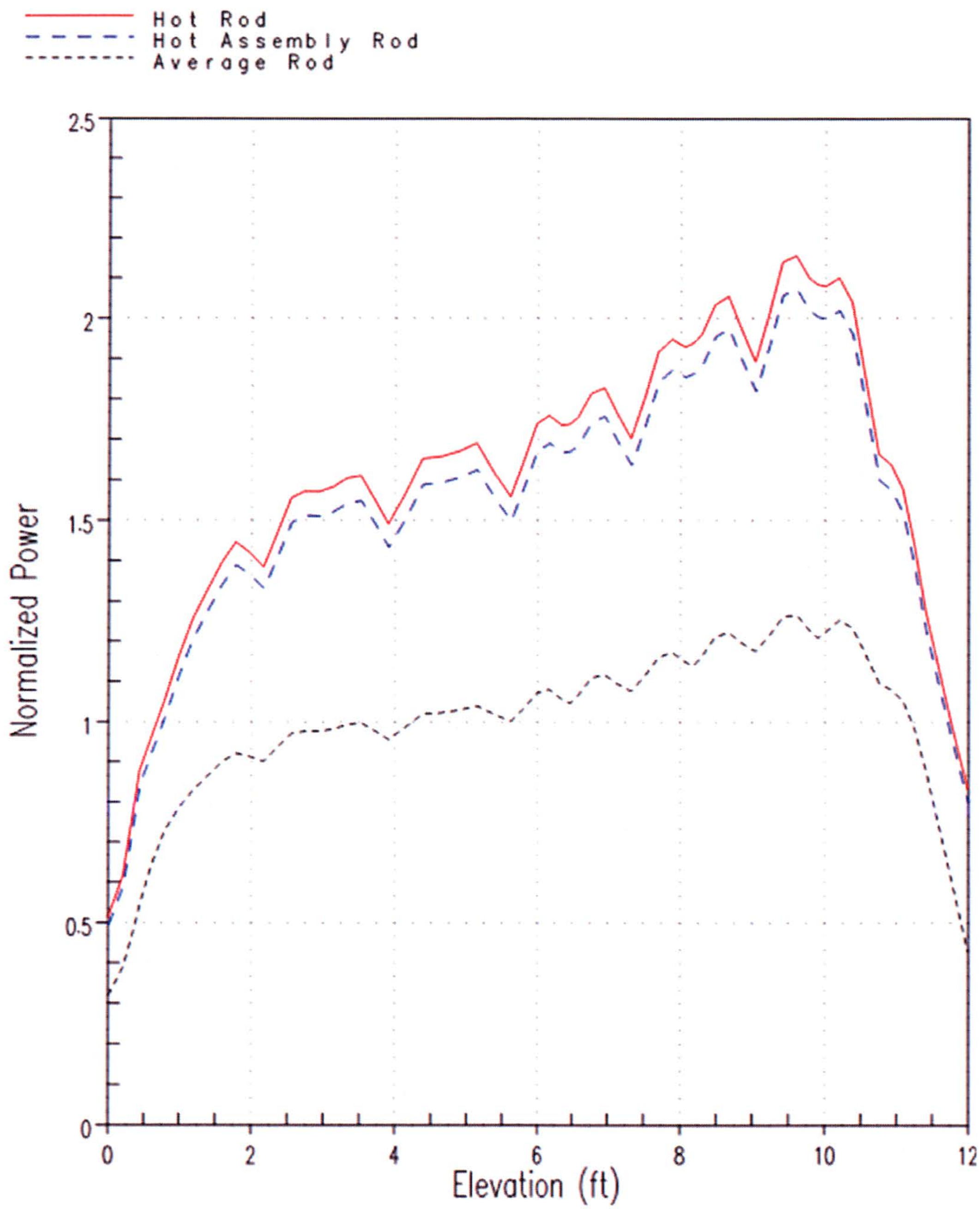

Figure 15.4.1.3-11B: Diablo Canyon Power Plant Unit 2 Normalized Core Power Shape for SBLOCA (Region I)

Note: The localized power decreases occur at grid elevations.

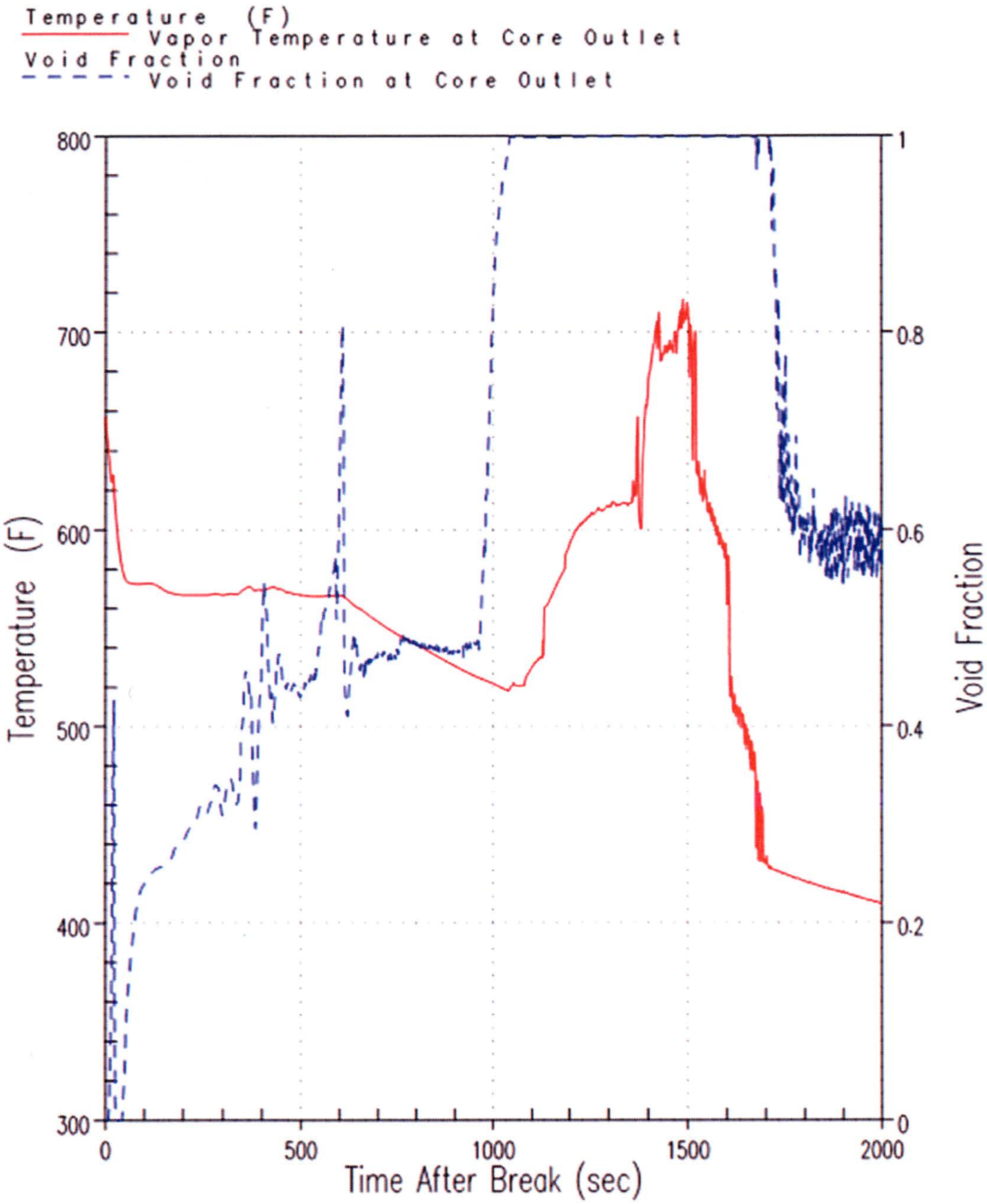

Figure 15.4.1.3-12A: Diablo Canyon Power Plant Unit 1 Vapor Temperature and Void Fraction at Core Outlet for SBLOCA (Region I)

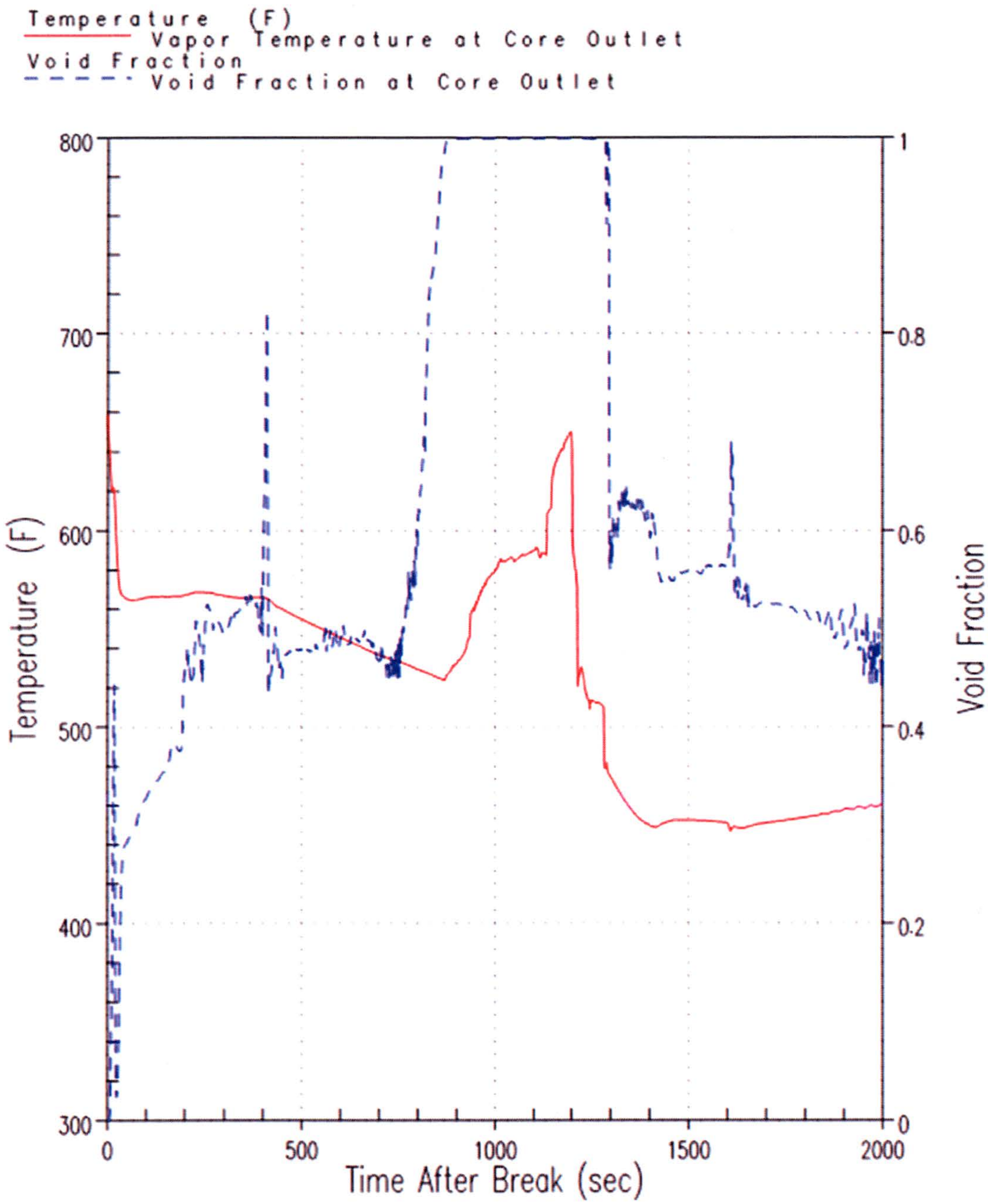

Figure 15.4.1.3-12B: Diablo Canyon Power Plant Unit 2 Vapor Temperature and Void Fraction at Core Outlet for SBLOCA (Region I)

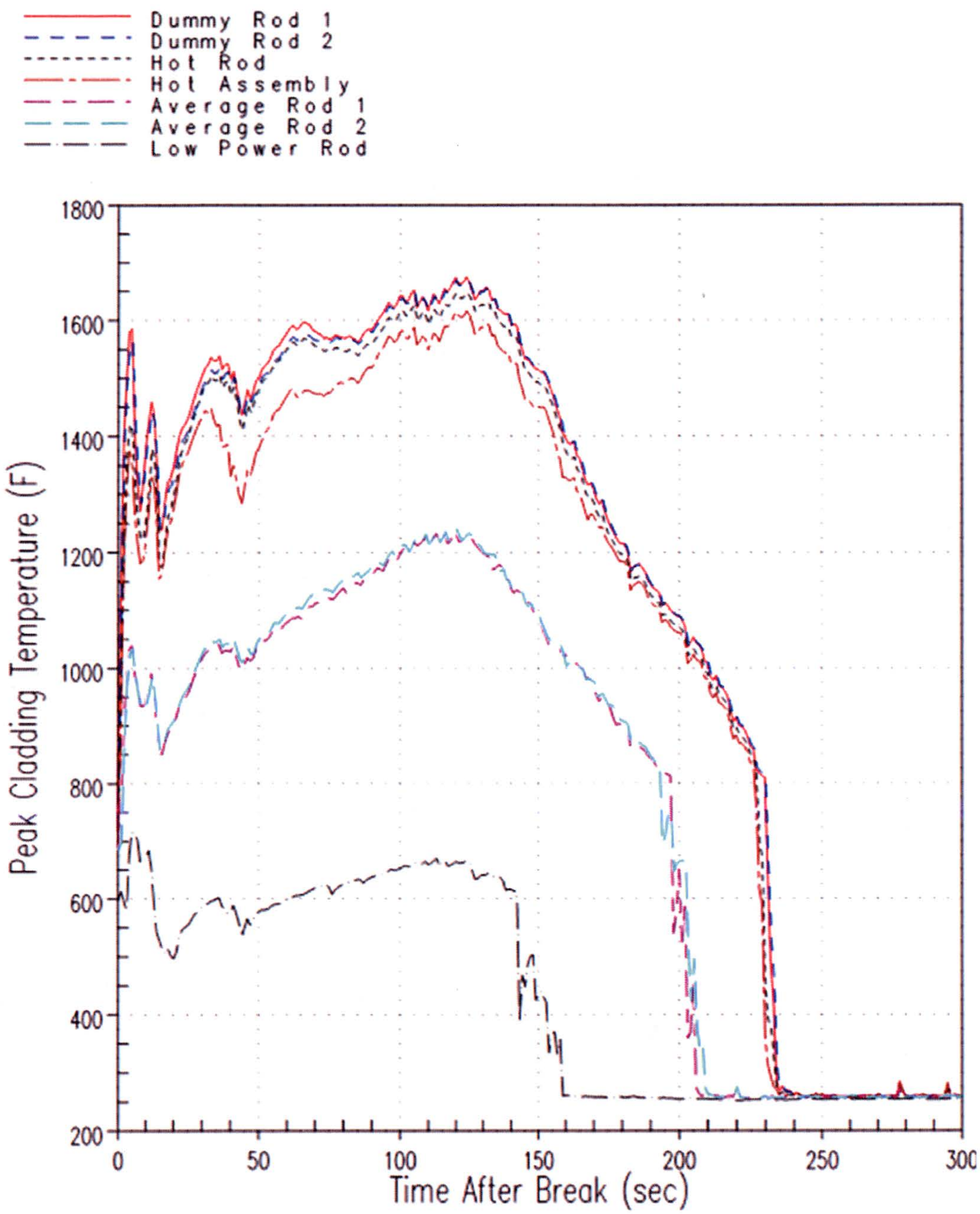

Figure 15.4.1.3-13A: Diablo Canyon Power Plant Unit 1 Peak Cladding Temperature for all Rods for LBLOCA (Region II)

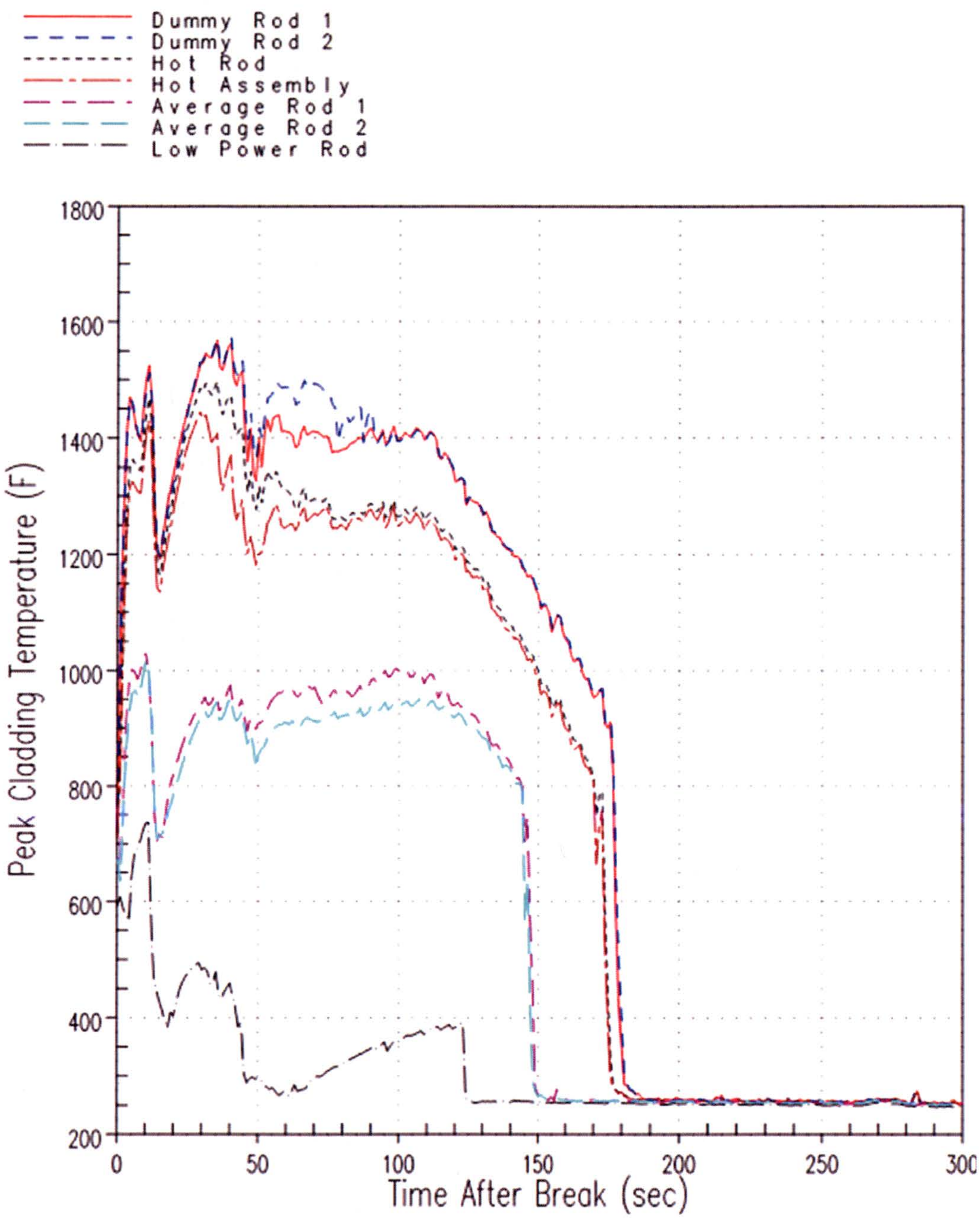

Figure 15.4.1.3-13B: Diablo Canyon Power Plant Unit 2 Peak Cladding Temperature for all Rods for LBLOCA (Region II)

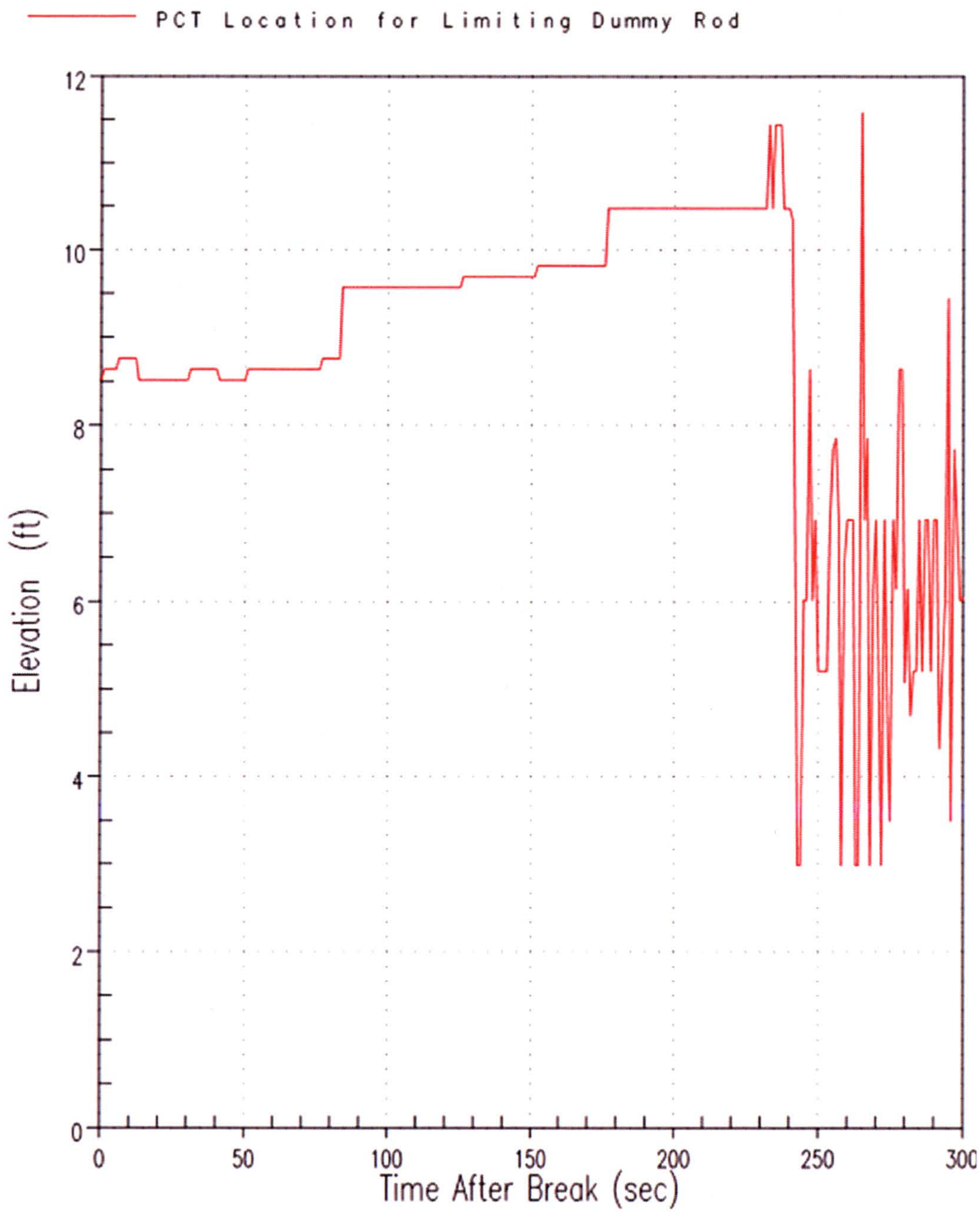

Figure 15.4.1.3-14A: Diablo Canyon Power Plant Unit 1 Peak Cladding Temperature Elevation for LBLOCA (Region II)

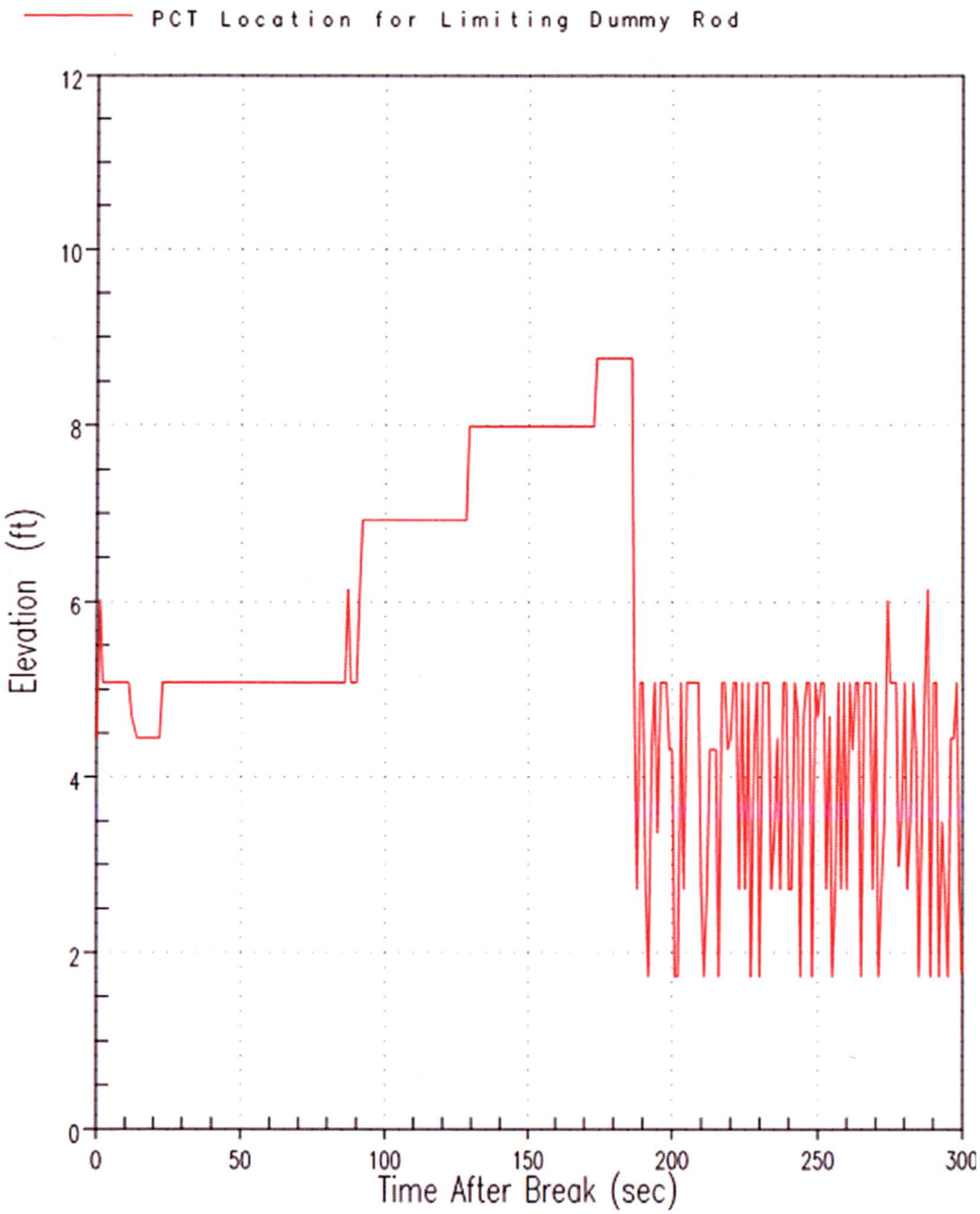

Figure 15.4.1.3-14B: Diablo Canyon Power Plant Unit 2 Peak Cladding Temperature Elevation for LBLOCA (Region II)

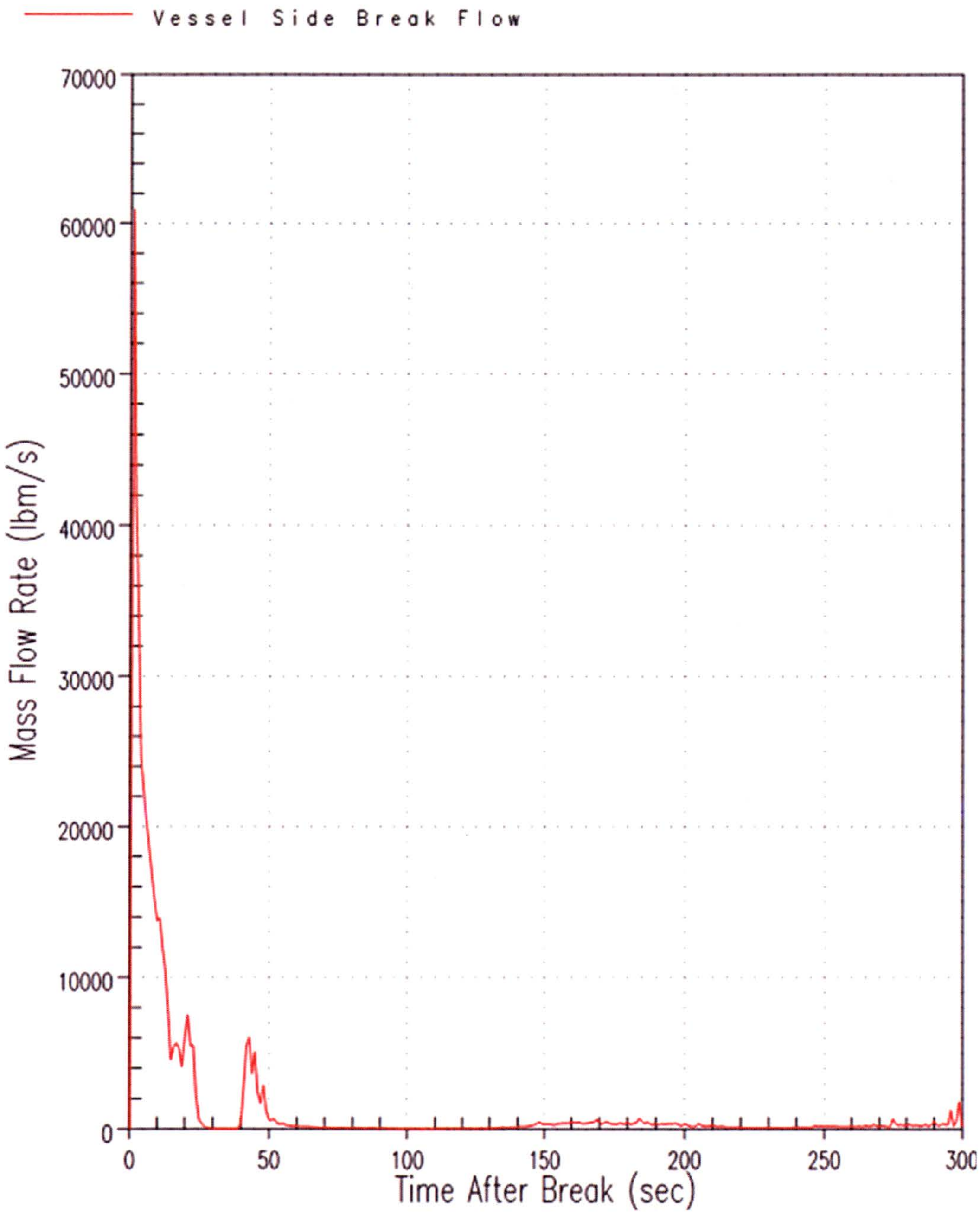

Figure 15.4.1.3-15A: Diablo Canyon Power Plant Unit 1 Vessel Side Break Mass Flow Rate for LBLOCA (Region II)

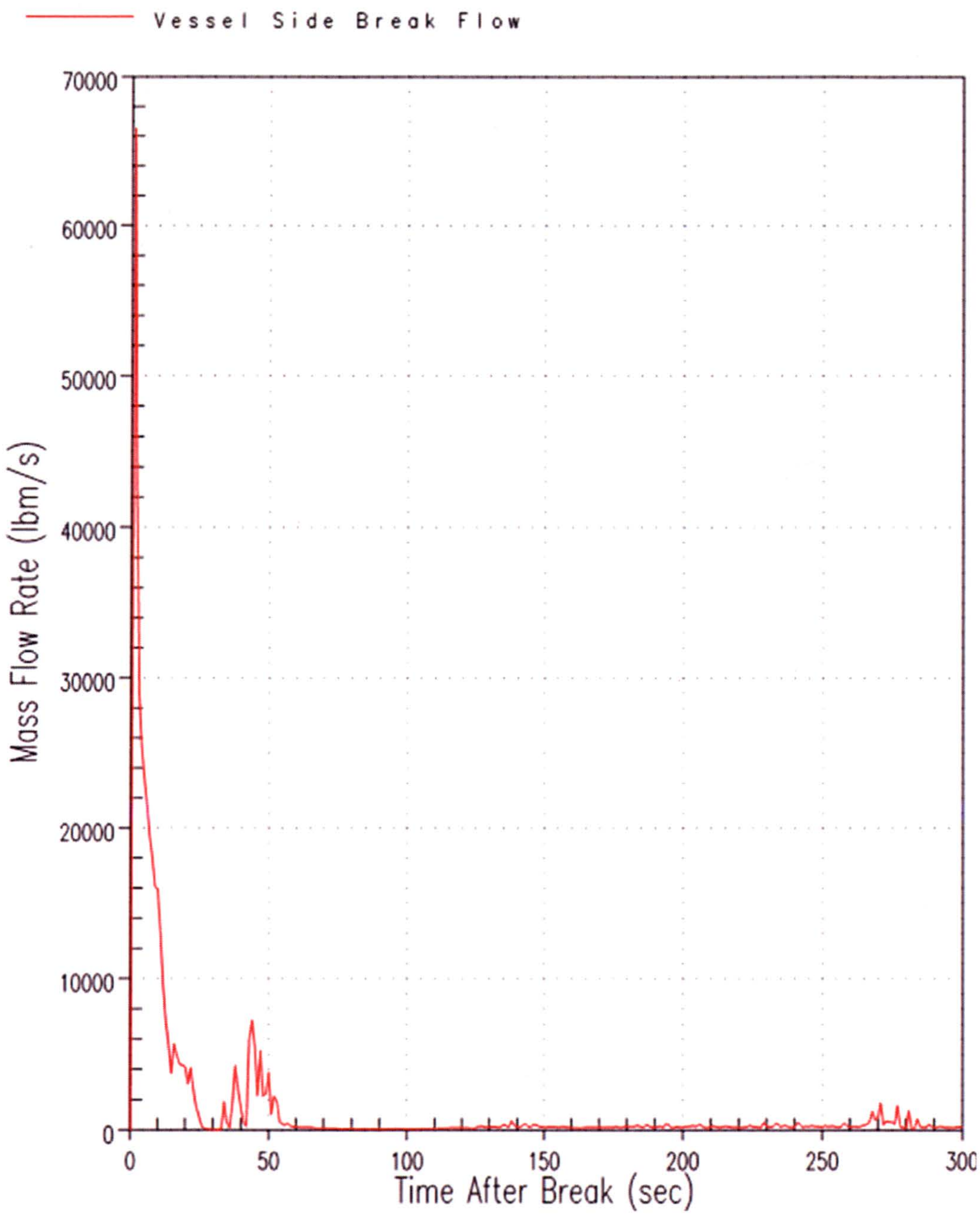

Figure 15.4.1.3-15B: Diablo Canyon Power Plant Unit 2 Vessel Side Break Mass Flow Rate for LBLOCA (Region II)

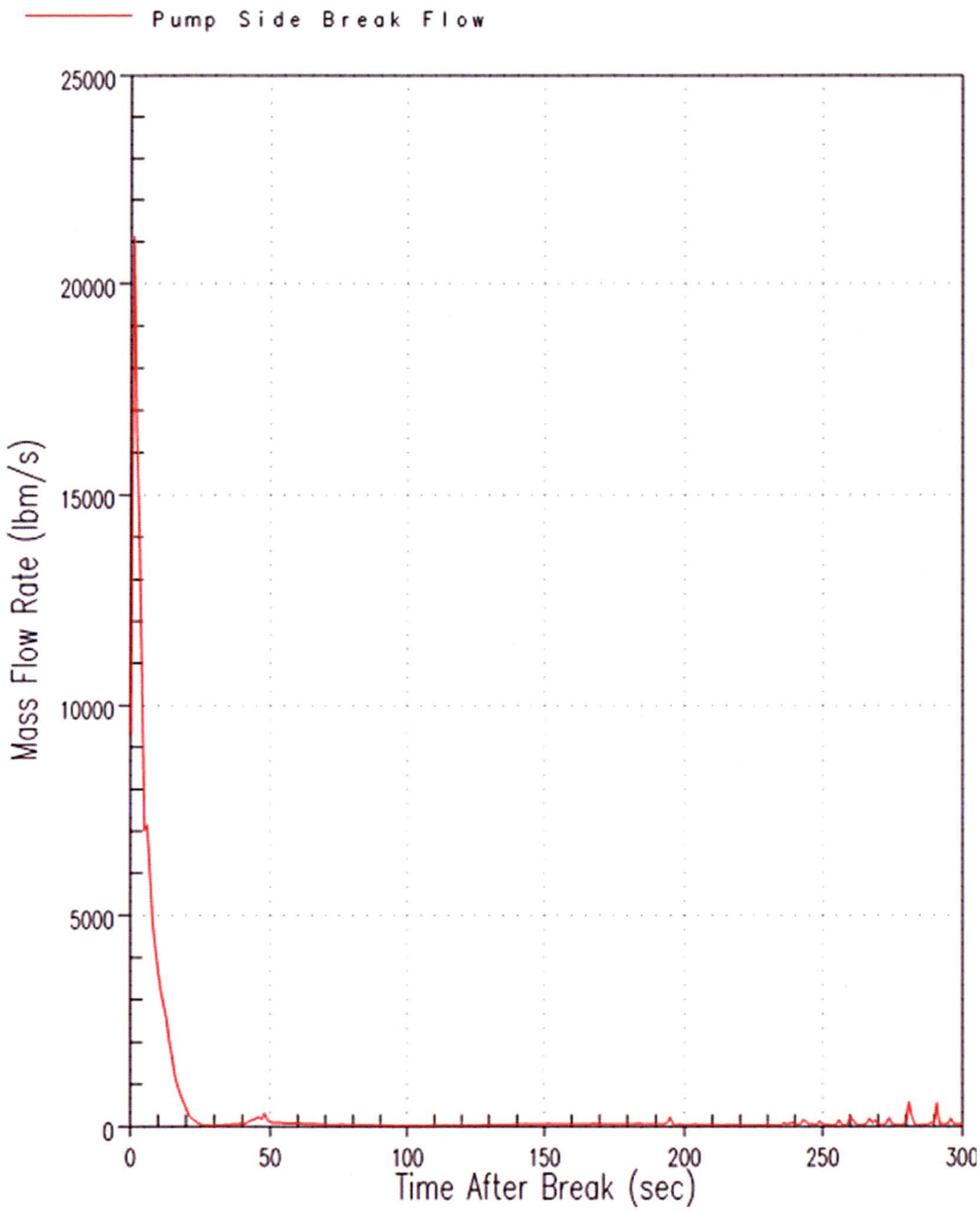

Figure 15.4.1.3-16A: Diablo Canyon Power Plant Unit 1 Pump Side Break Mass Flow Rate for LBLOCA (Region II)

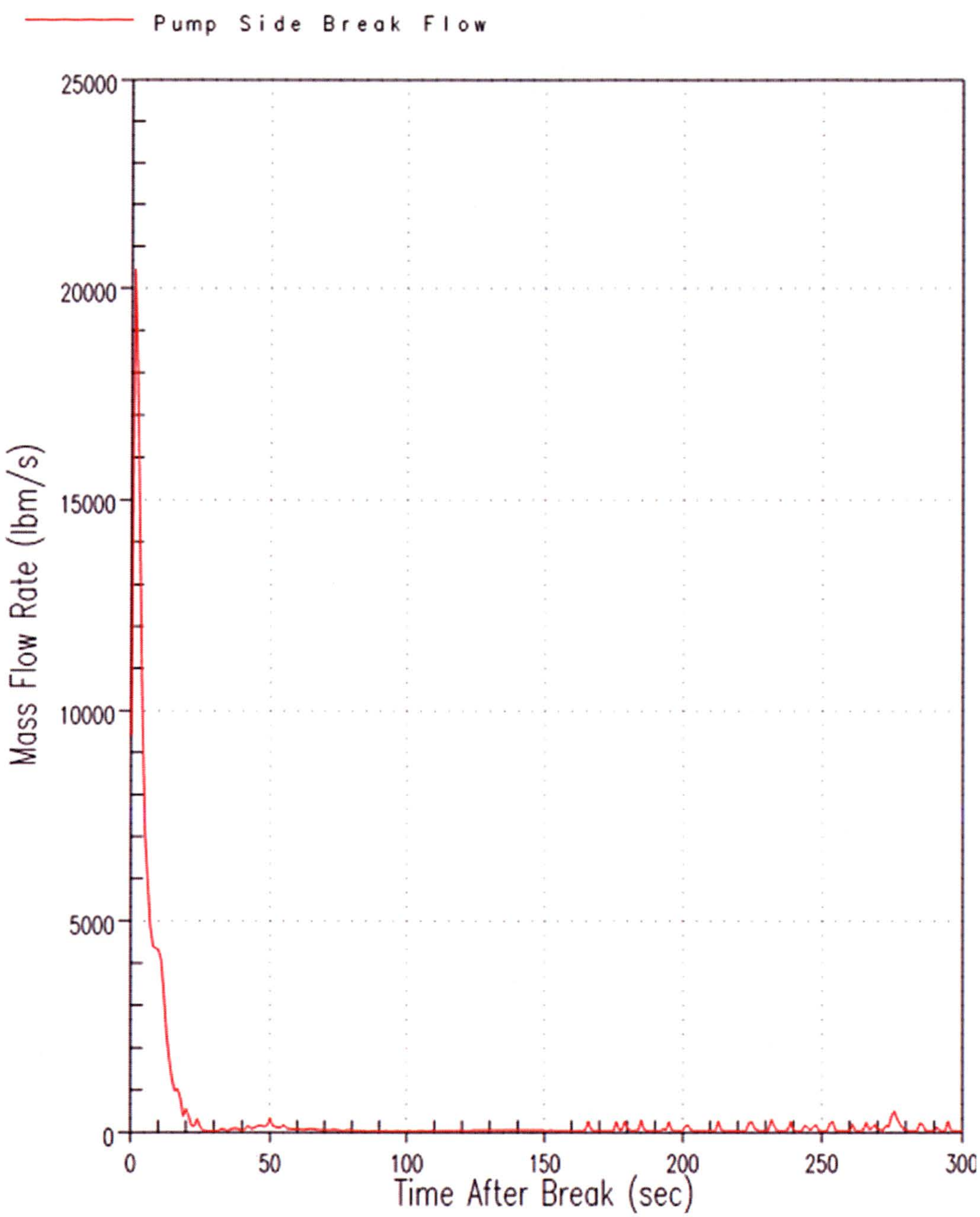

Figure 15.4.1.3-16B: Diablo Canyon Power Plant Unit 2 Pump Side Break Mass Flow Rate for LBLOCA (Region II)

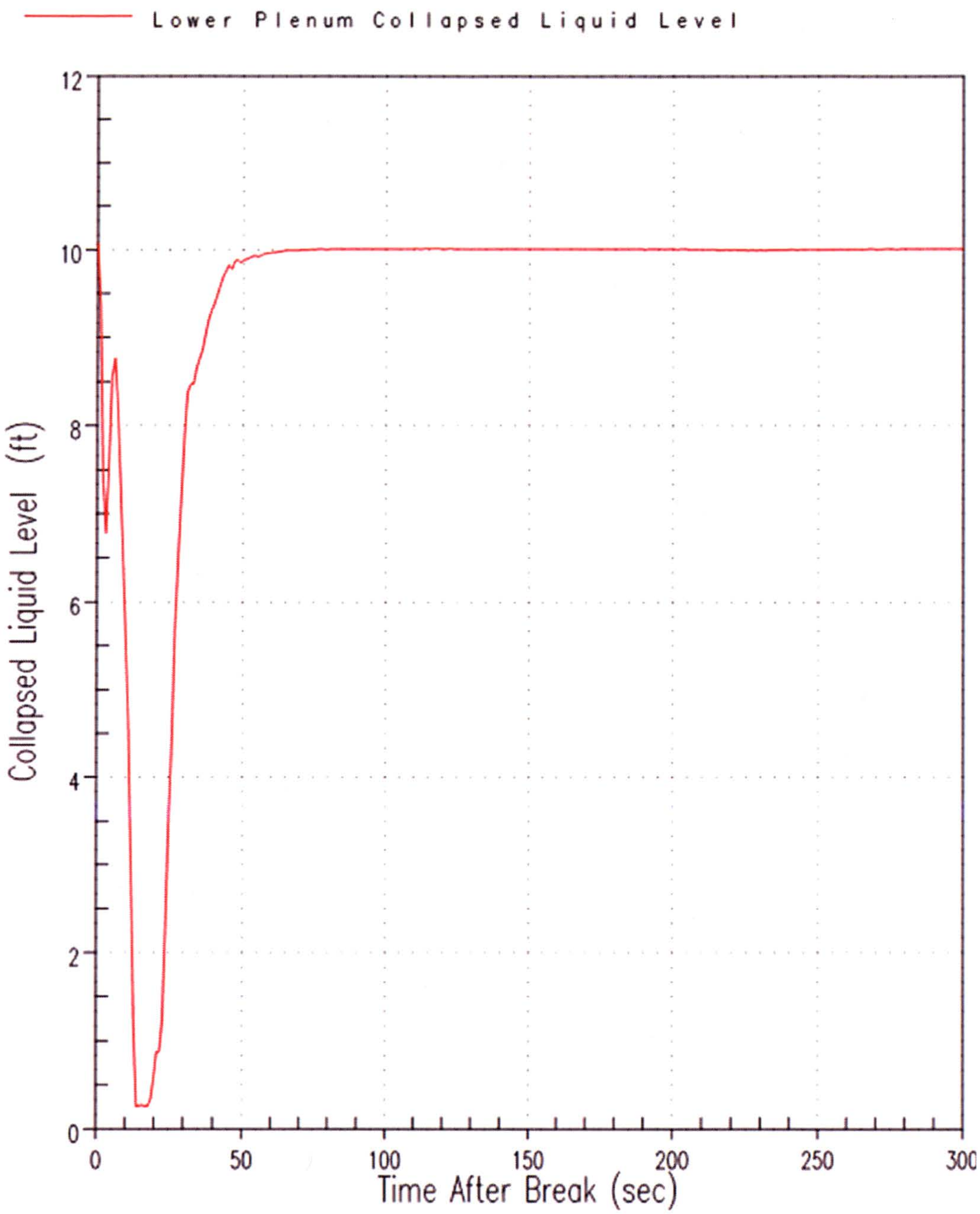

Figure 15.4.1.3-17A: Diablo Canyon Power Plant Unit 1 Lower Plenum Collapsed Liquid Level for LBLOCA (Region II)

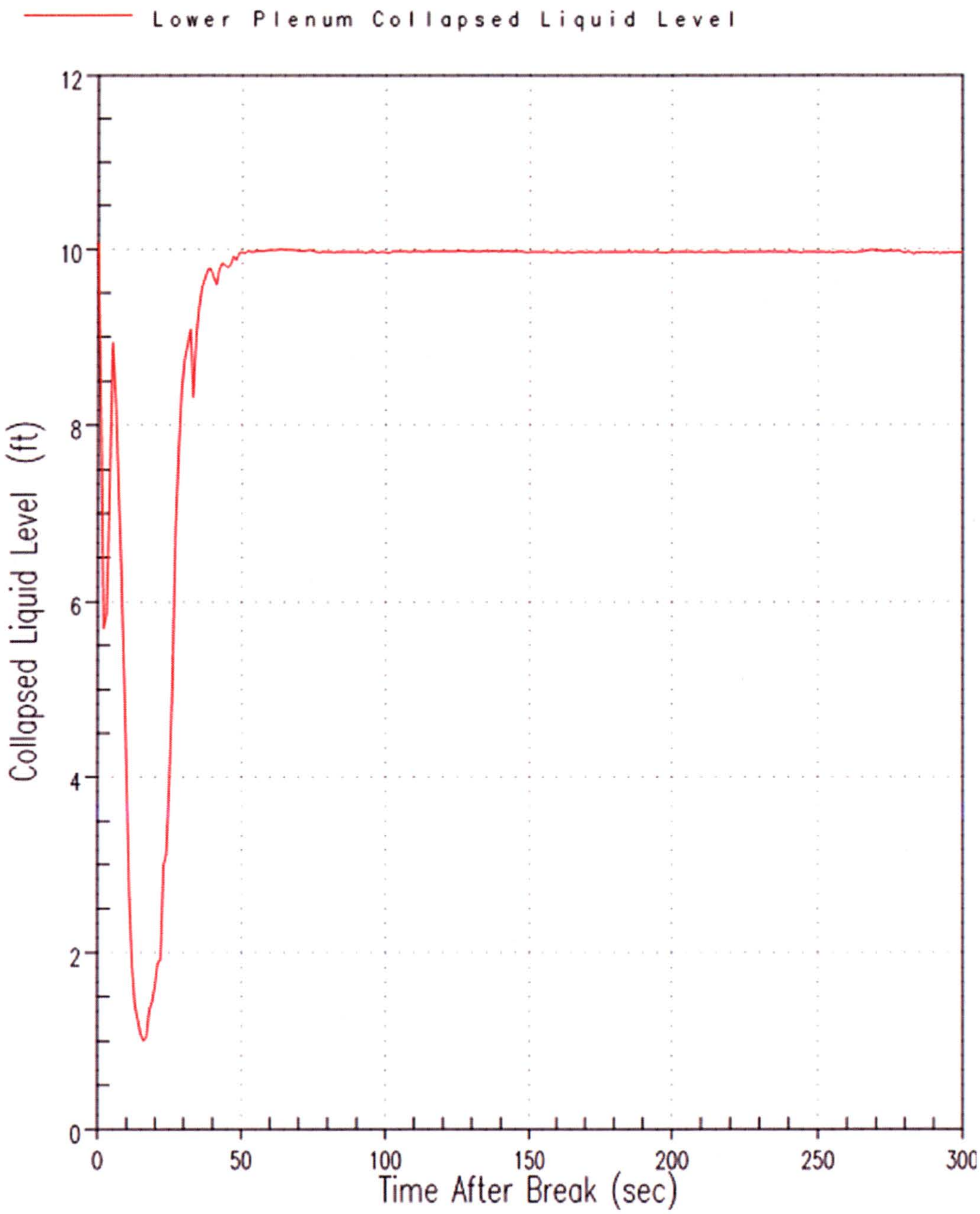

Figure 15.4.1.3-17B: Diablo Canyon Power Plant Unit 2 Lower Plenum Collapsed Liquid Level for LBLOCA (Region II)

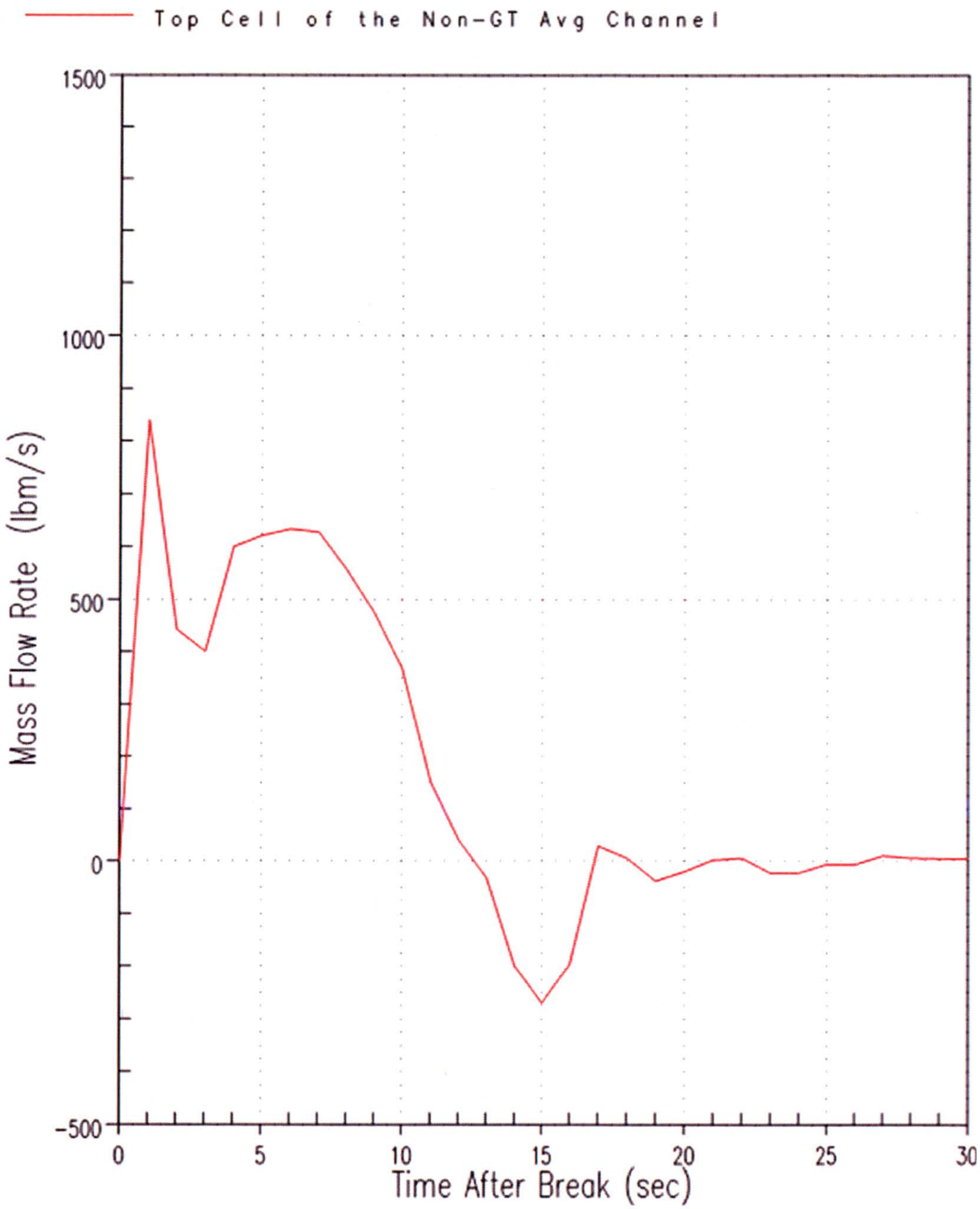

Figure 15.4.1.3-18A: Diablo Canyon Power Plant Unit 1 Vapor Mass Flow Rate at the Top Cell Face of the Core Average Channel not Under Guide Tubes for LBLOCA (Region II)

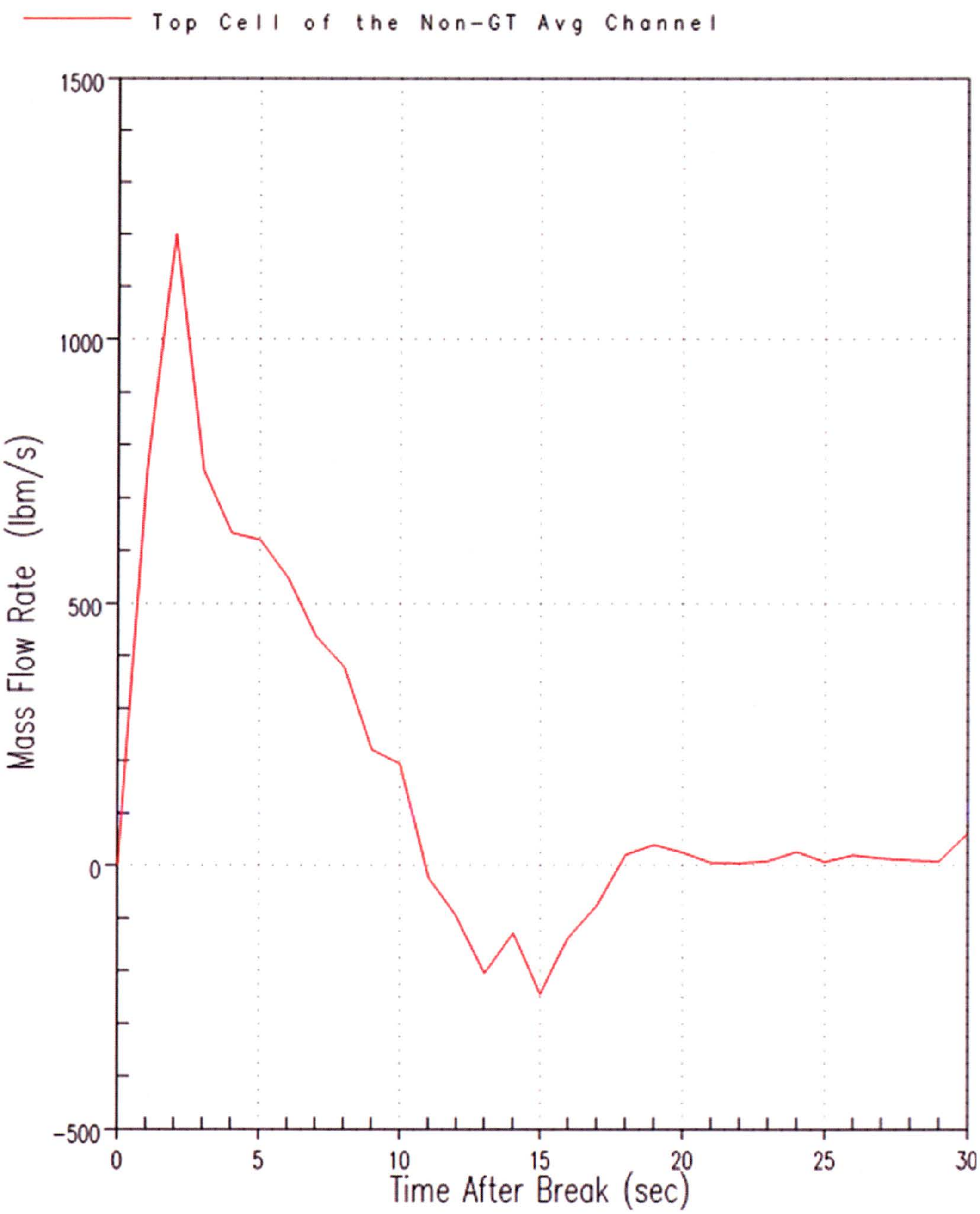

Figure 15.4.1.3-18B: Diablo Canyon Power Plant Unit 2 Vapor Mass Flow Rate at the Top Cell Face of the Core Average Channel not Under Guide Tubes for LBLOCA (Region II)

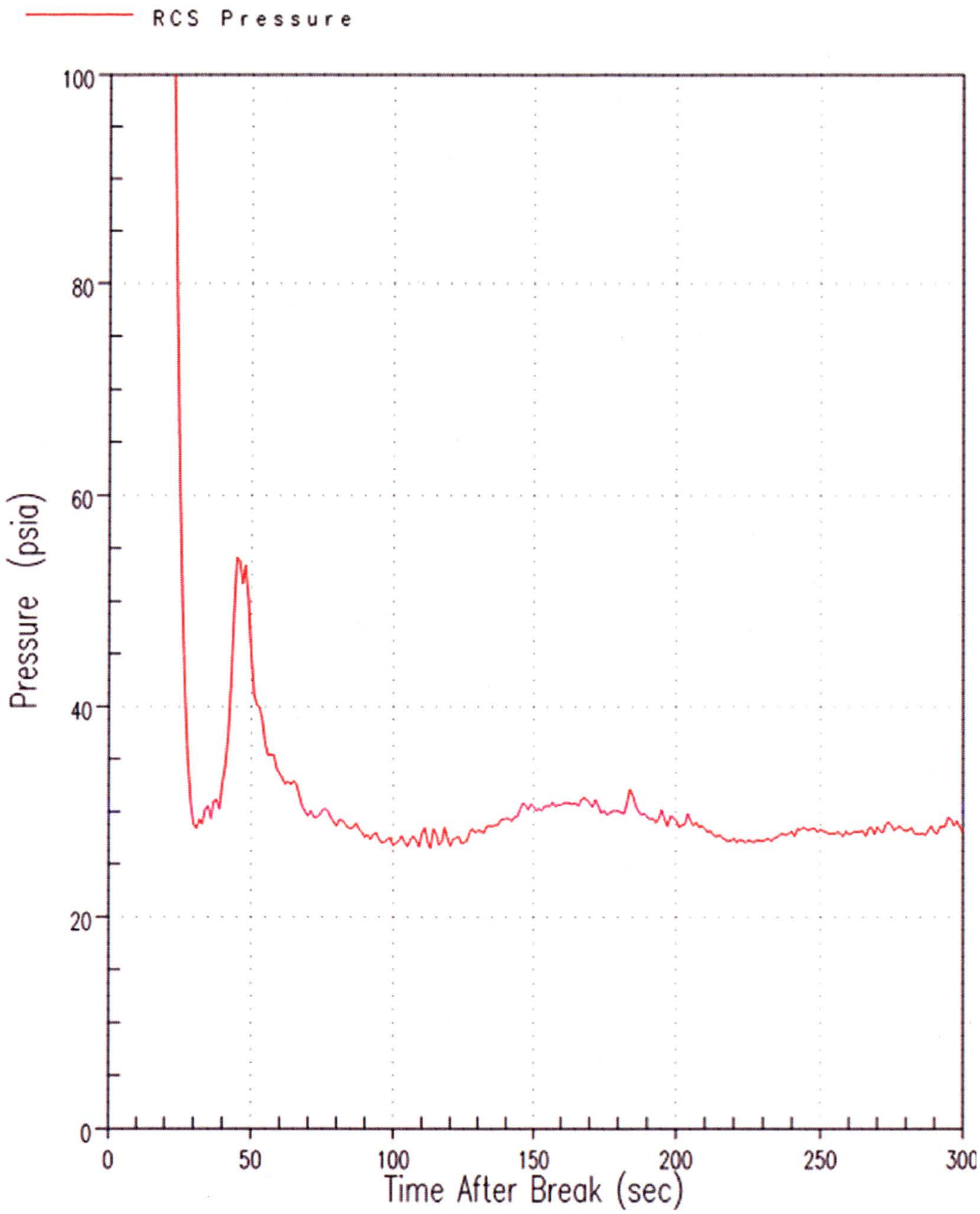

Figure 15.4.1.3-19A: Diablo Canyon Power Plant Unit 1 RCS Pressure for LBLOCA (Region II)

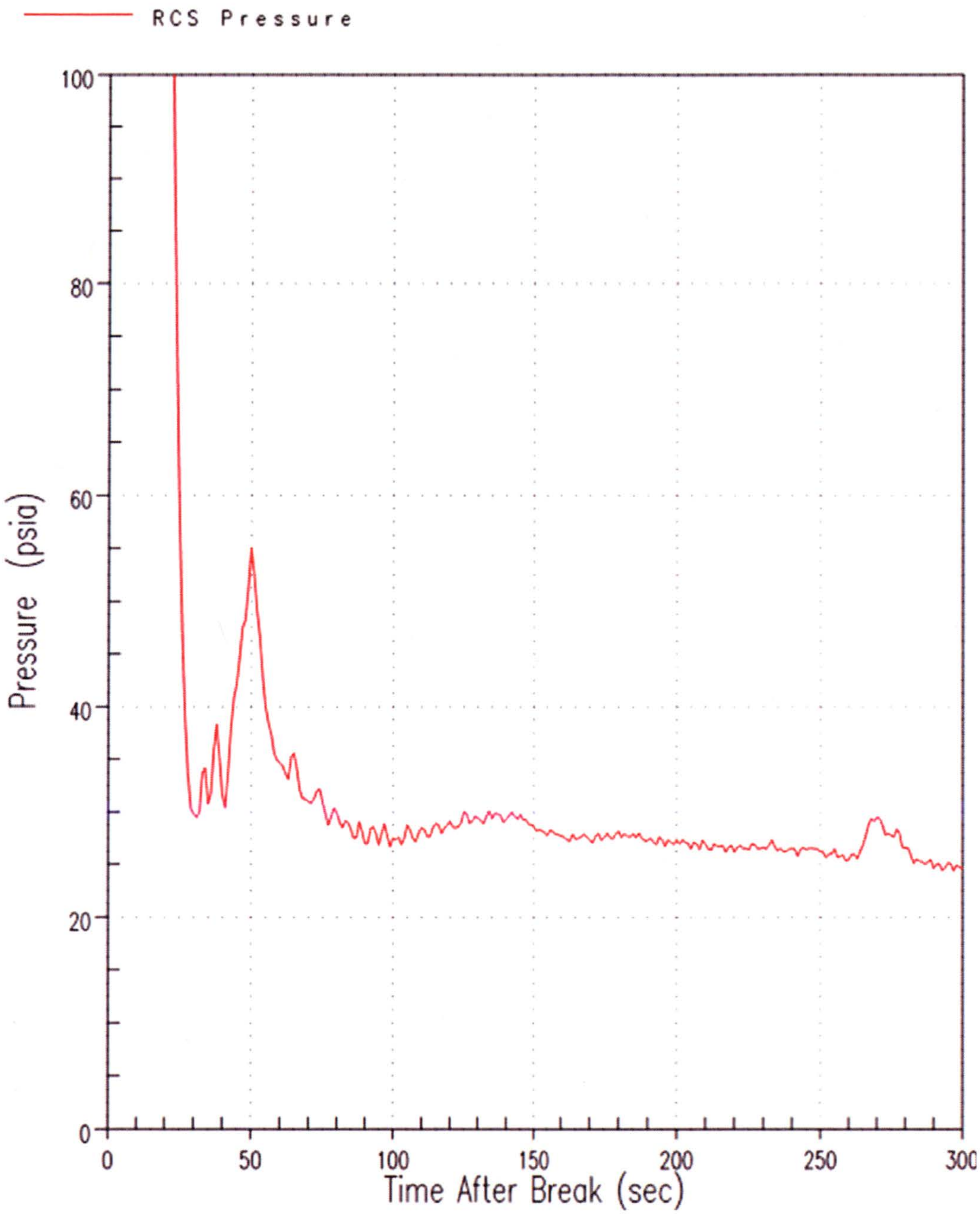

Figure 15.4.1.3-19B: Diablo Canyon Power Plant Unit 2 RCS Pressure for LBLOCA (Region II)

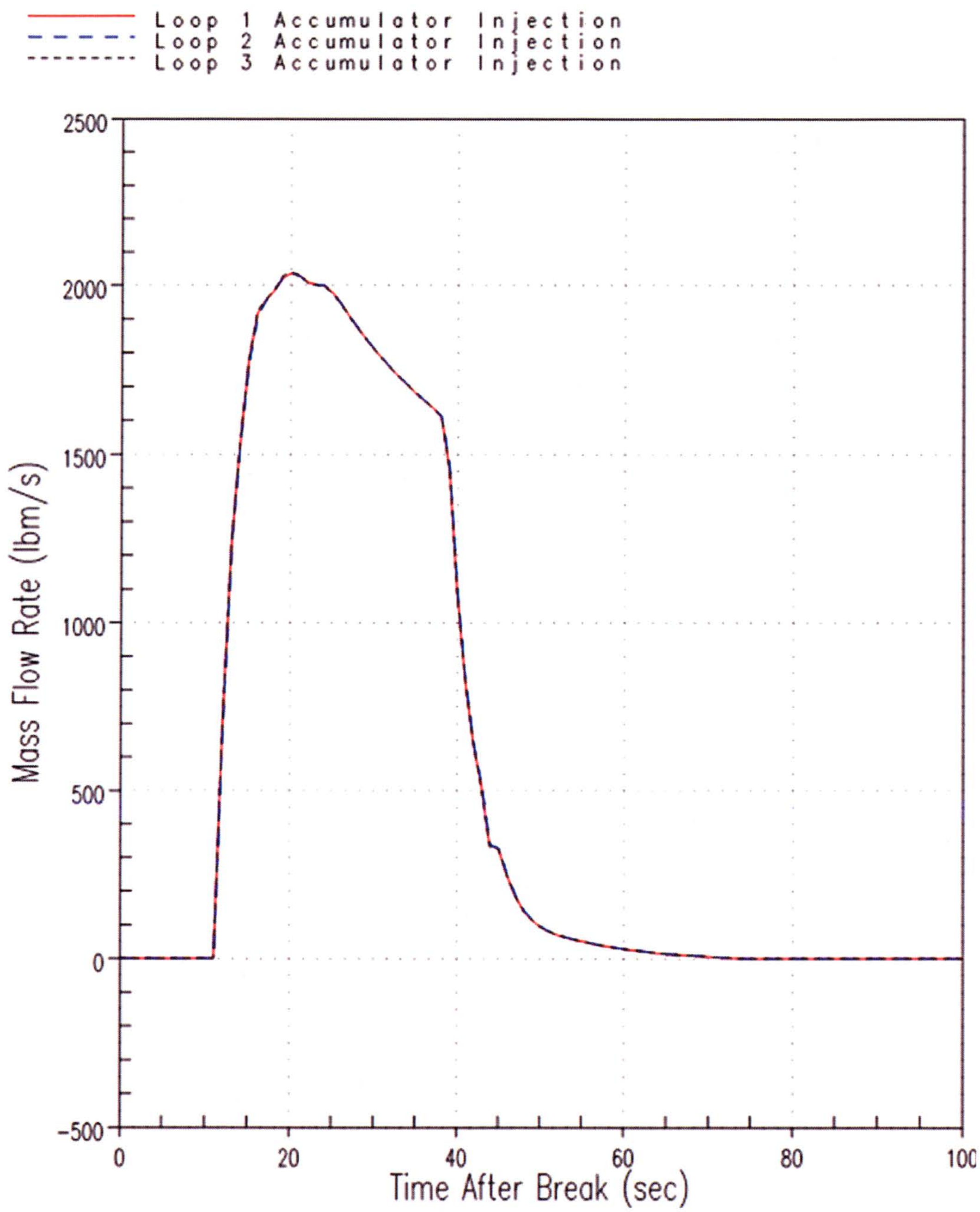

Figure 15.4.1.3-20A: Diablo Canyon Power Plant Unit 1 Accumulator Injection Flow per Loop for LBLOCA (Region II)

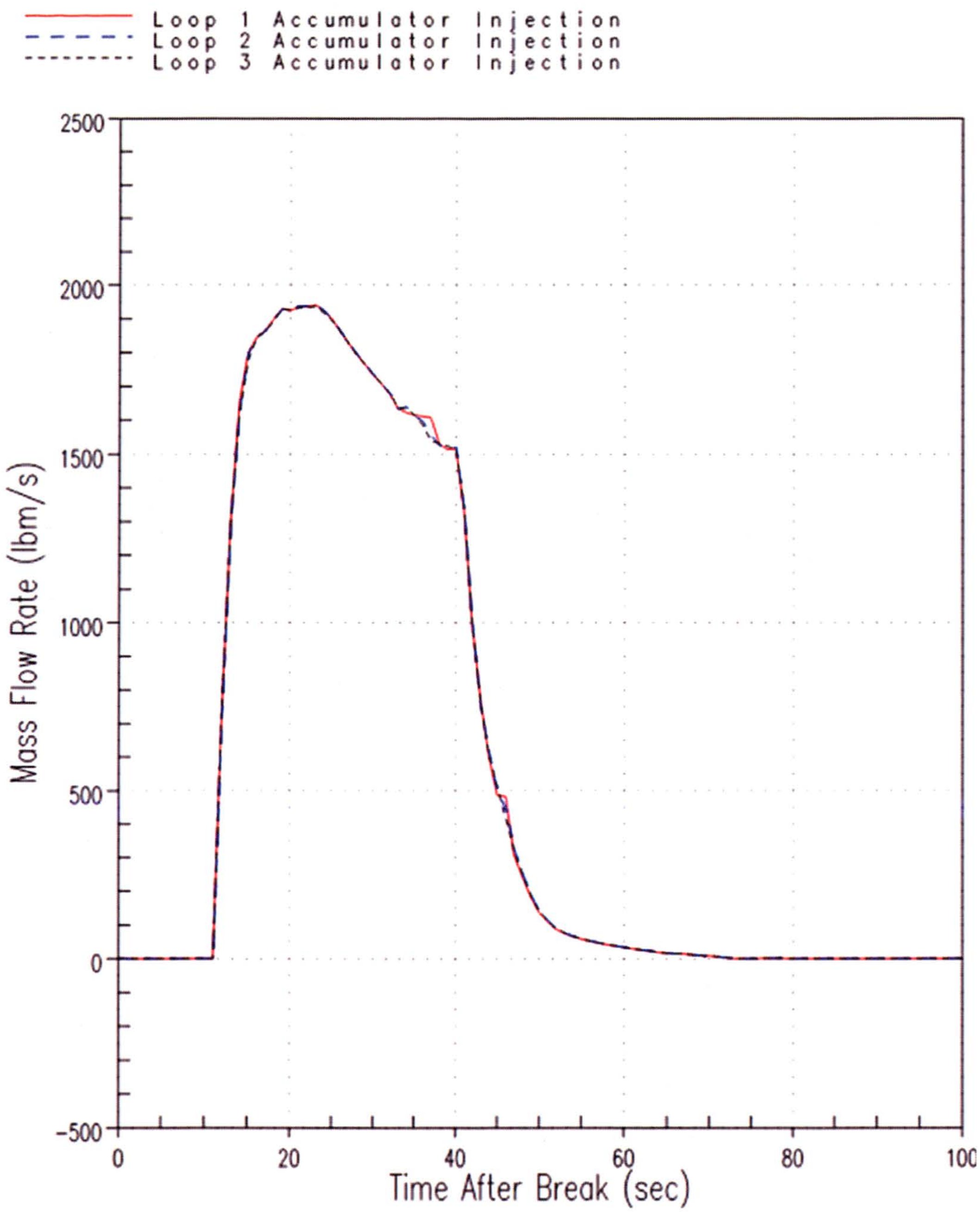

Figure 15.4.1.3-20B: Diablo Canyon Power Plant Unit 2 Accumulator Injection Flow per Loop for LBLOCA (Region II)

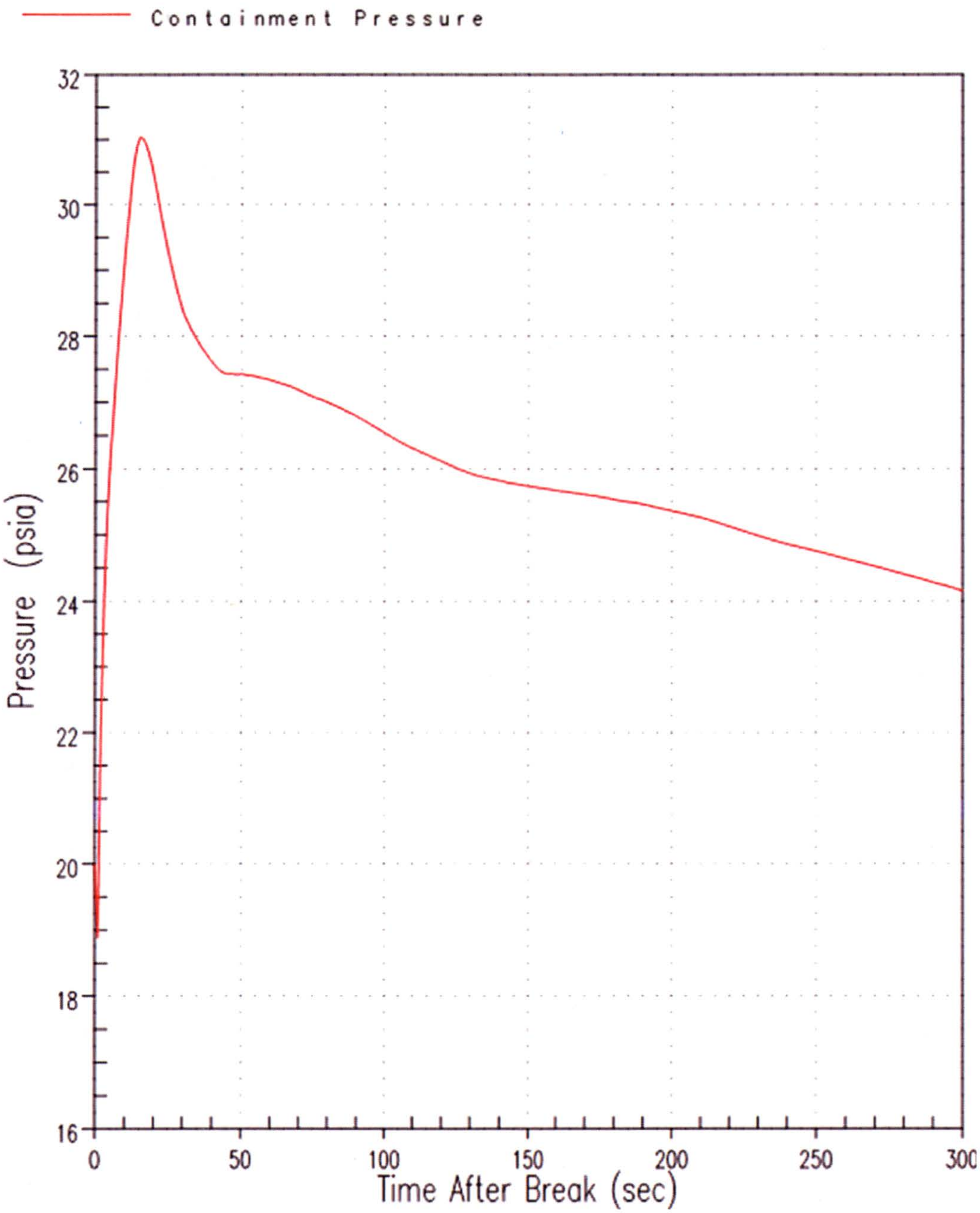

Figure 15.4.1.3-21A: Diablo Canyon Power Plant Unit 1 Containment Pressure for LBLOCA (Region II)

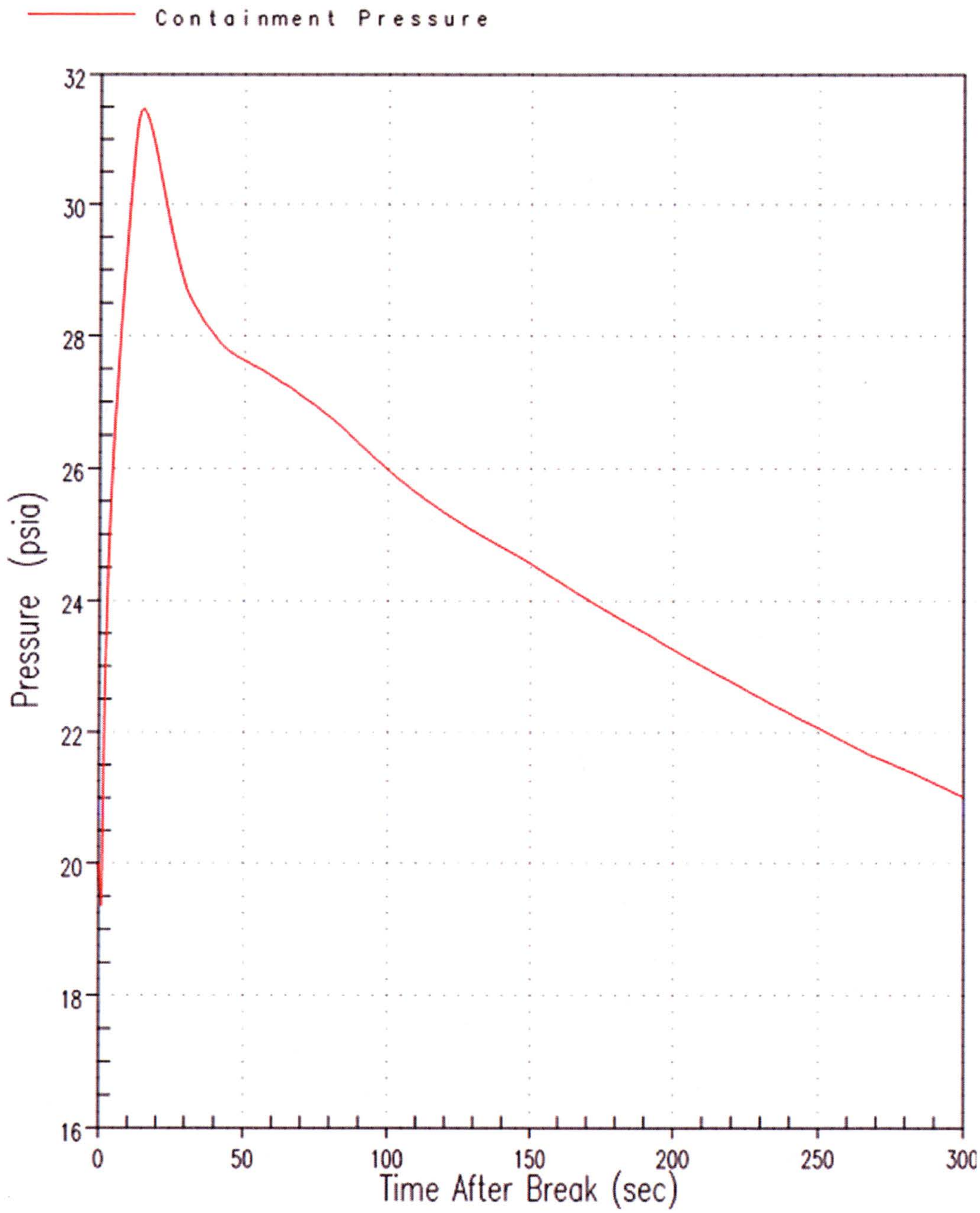

Figure 15.4.1.3-21B: Diablo Canyon Power Plant Unit 2 Containment Pressure for LBLOCA (Region II)

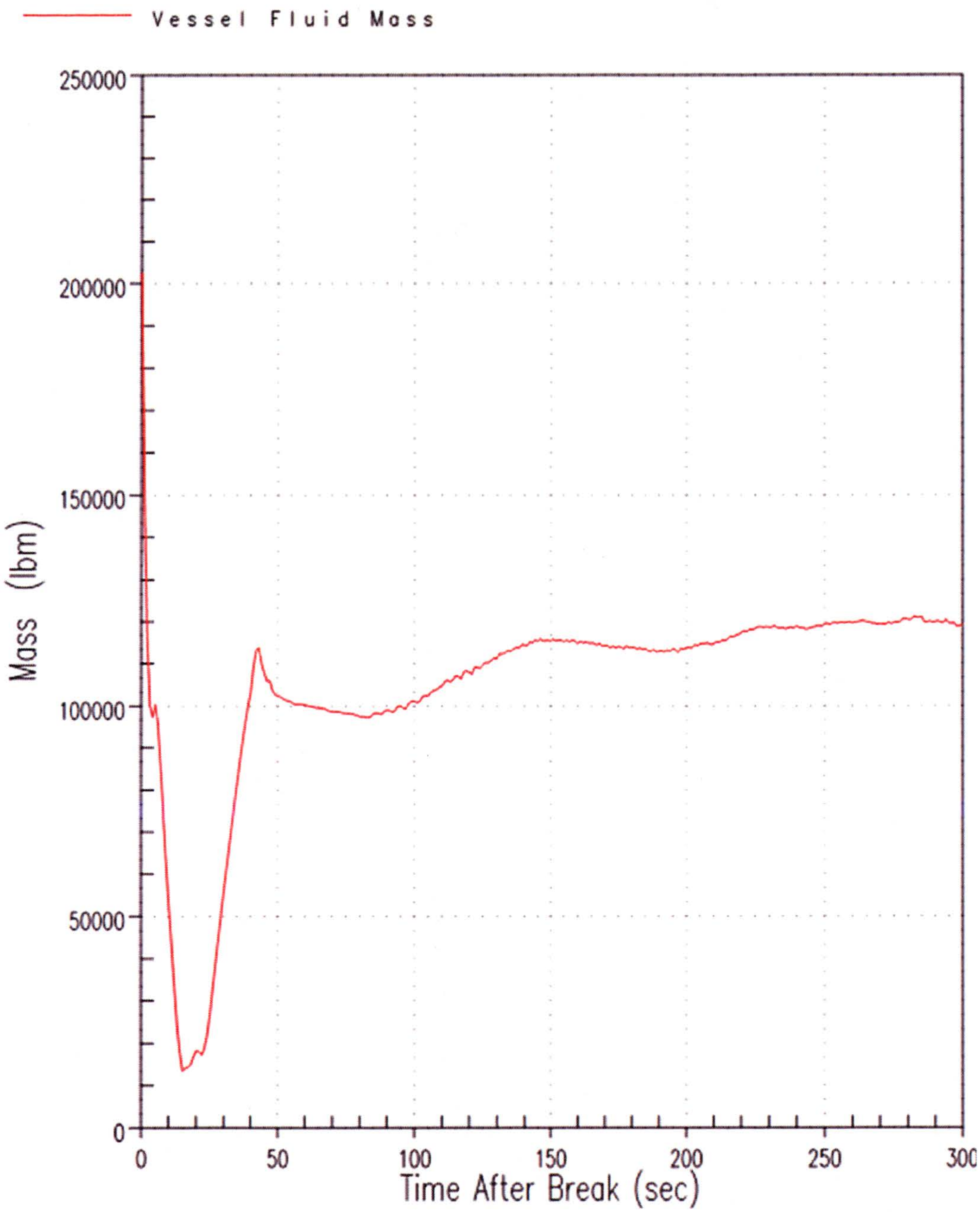

Figure 15.4.1.3-22A: Diablo Canyon Power Plant Unit 1 Vessel Fluid Mass for LBLOCA (Region II)

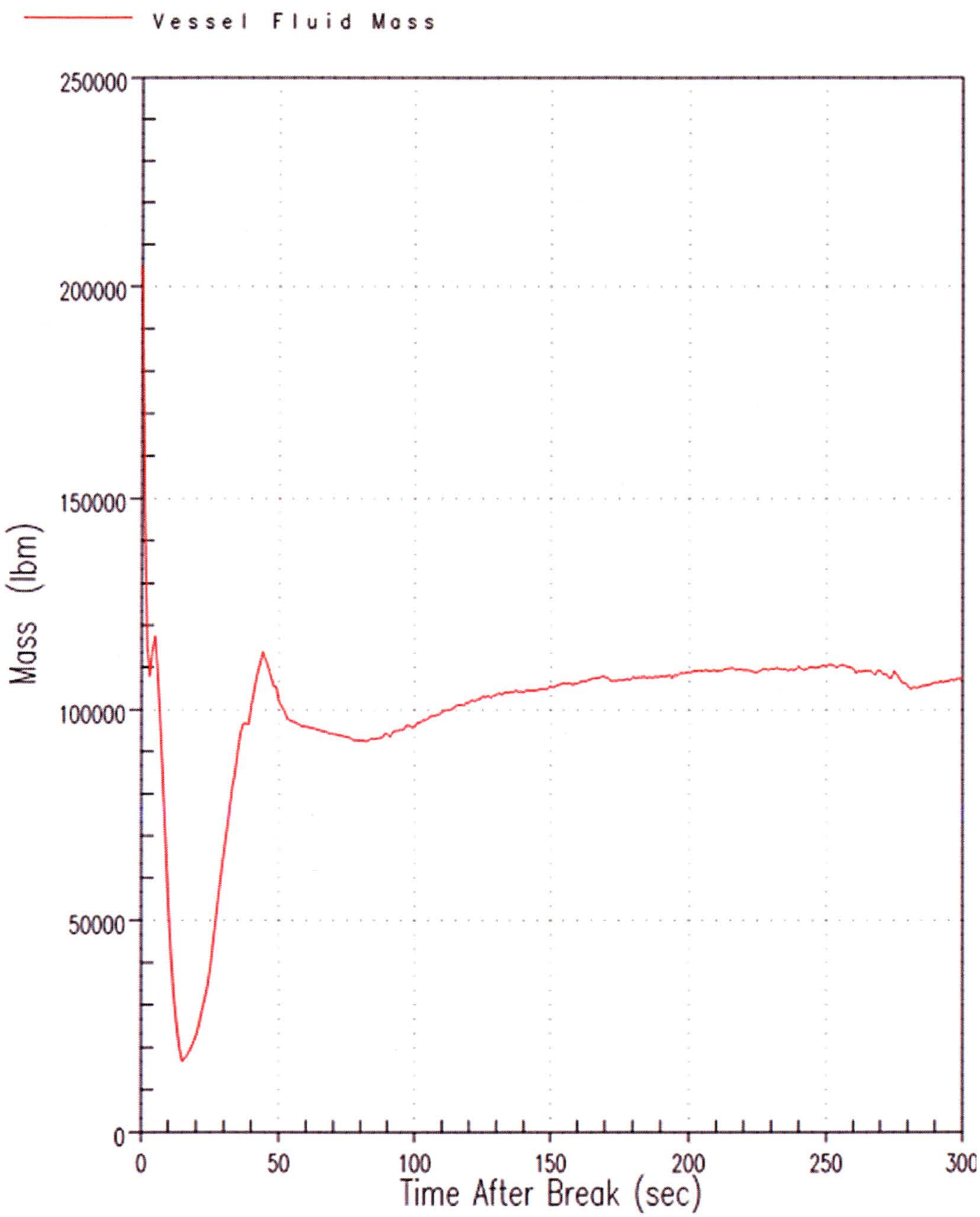

Figure 15.4.1.3-22B: Diablo Canyon Power Plant Unit 2 Vessel Fluid Mass for LBLOCA (Region II)

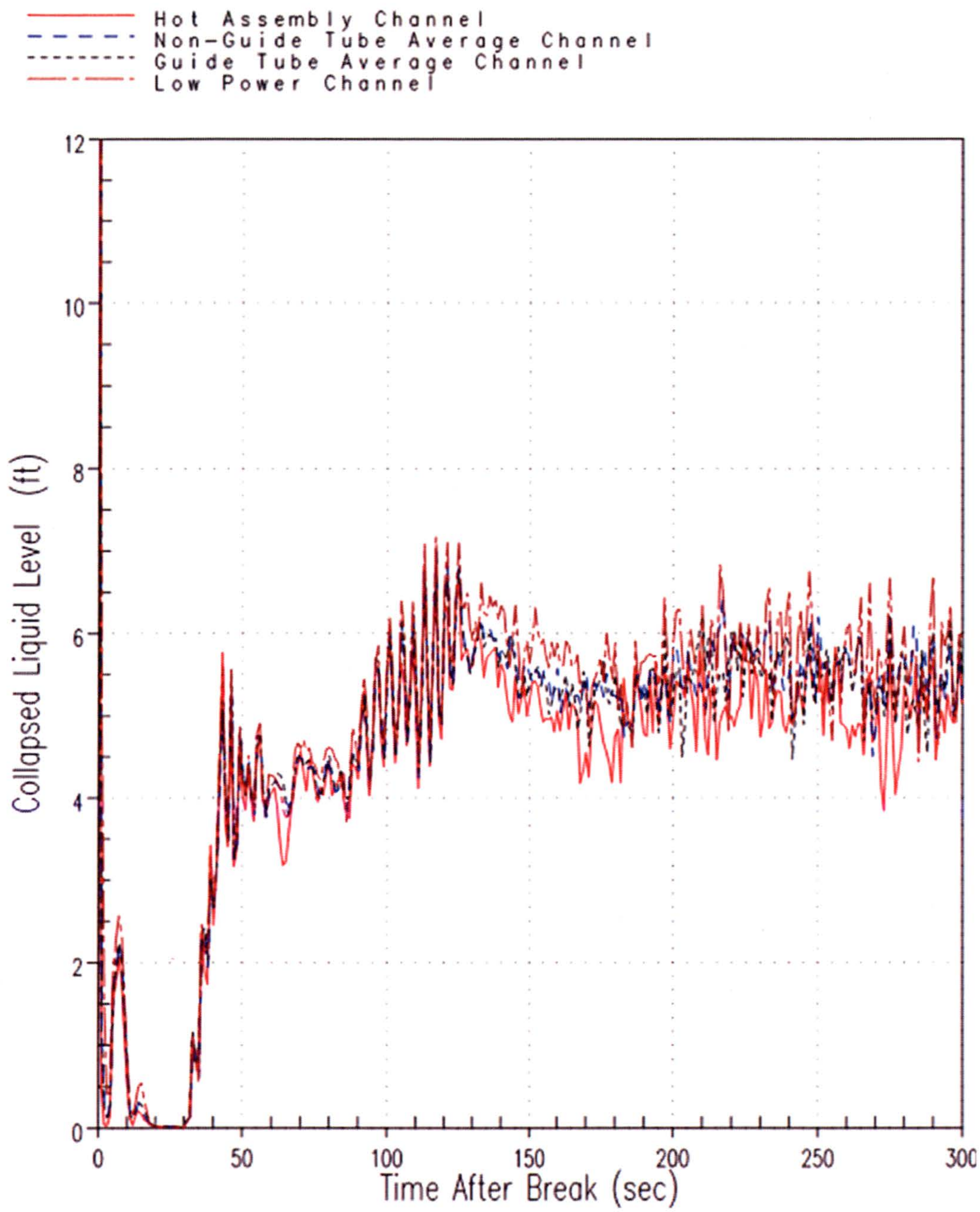

Figure 15.4.1.3-23A: Diablo Canyon Power Plant Unit 1 Collapsed Liquid Level for Each Core Channel for LBLOCA (Region II)

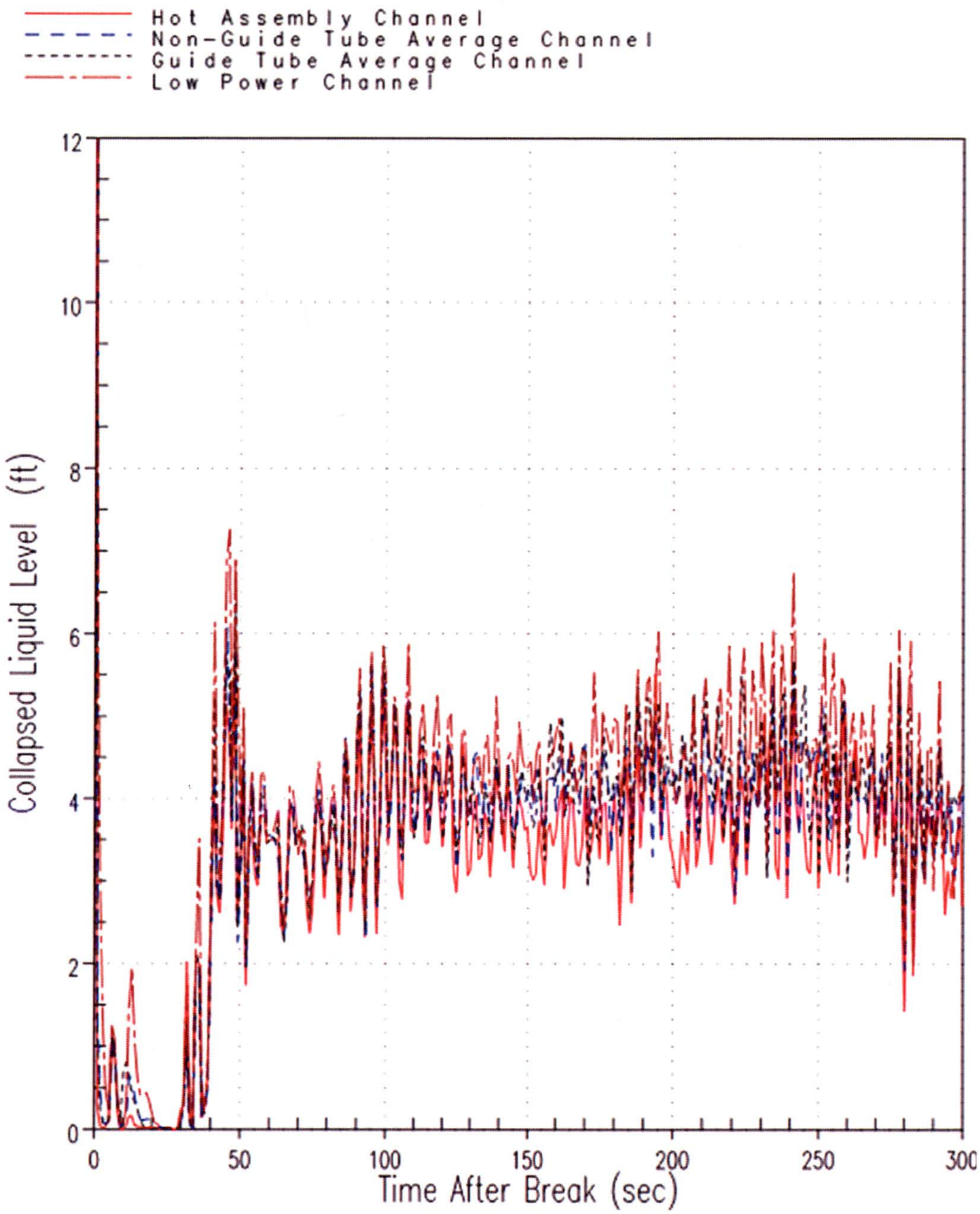

Figure 15.4.1.3-23B: Diablo Canyon Power Plant Unit 2 Collapsed Liquid Level for Each Core Channel for LBLOCA (Region II)

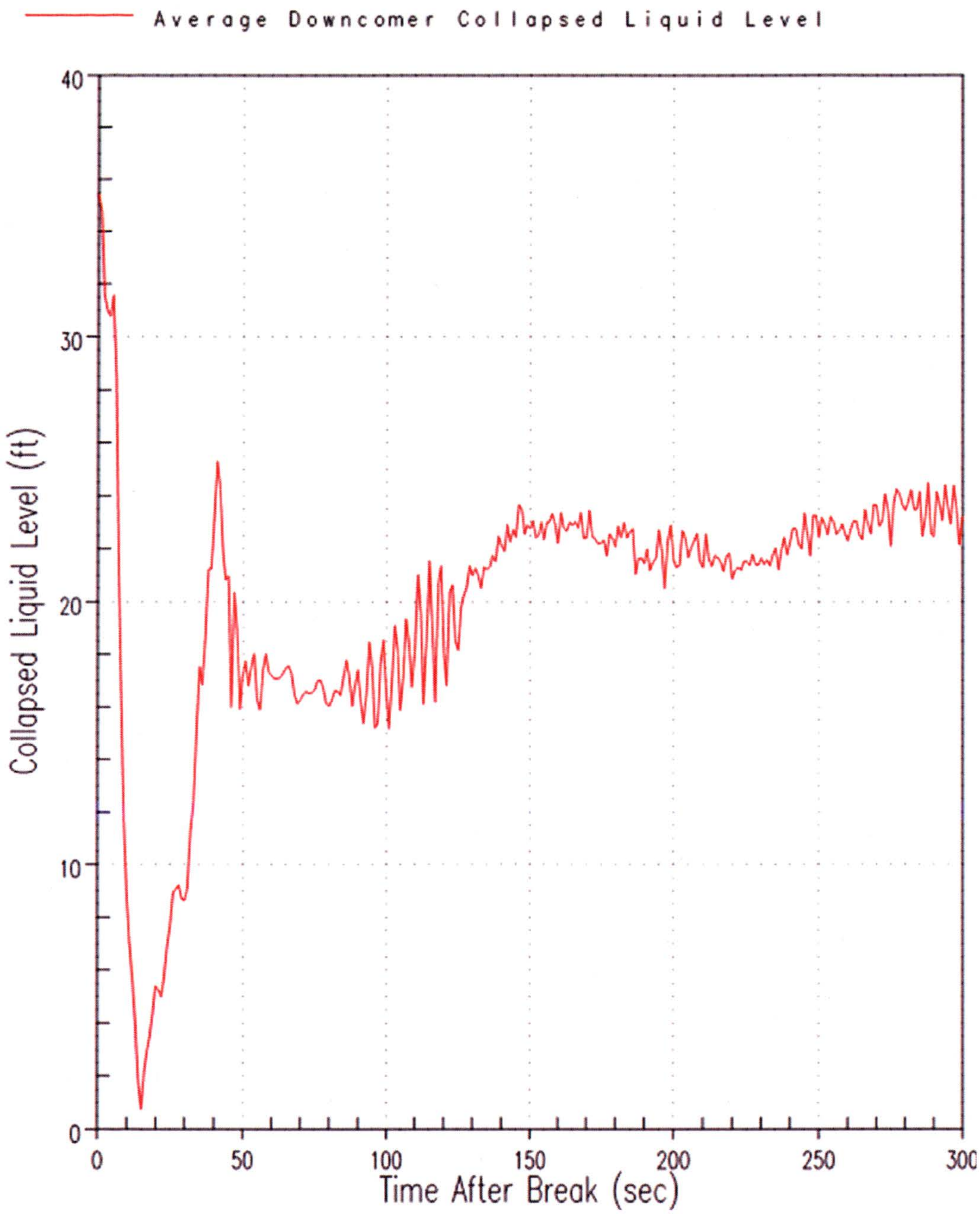

Figure 15.4.1.3-24A: Diablo Canyon Power Plant Unit 1 Average Downcomer Collapsed Liquid Level for LBLOCA (Region II)

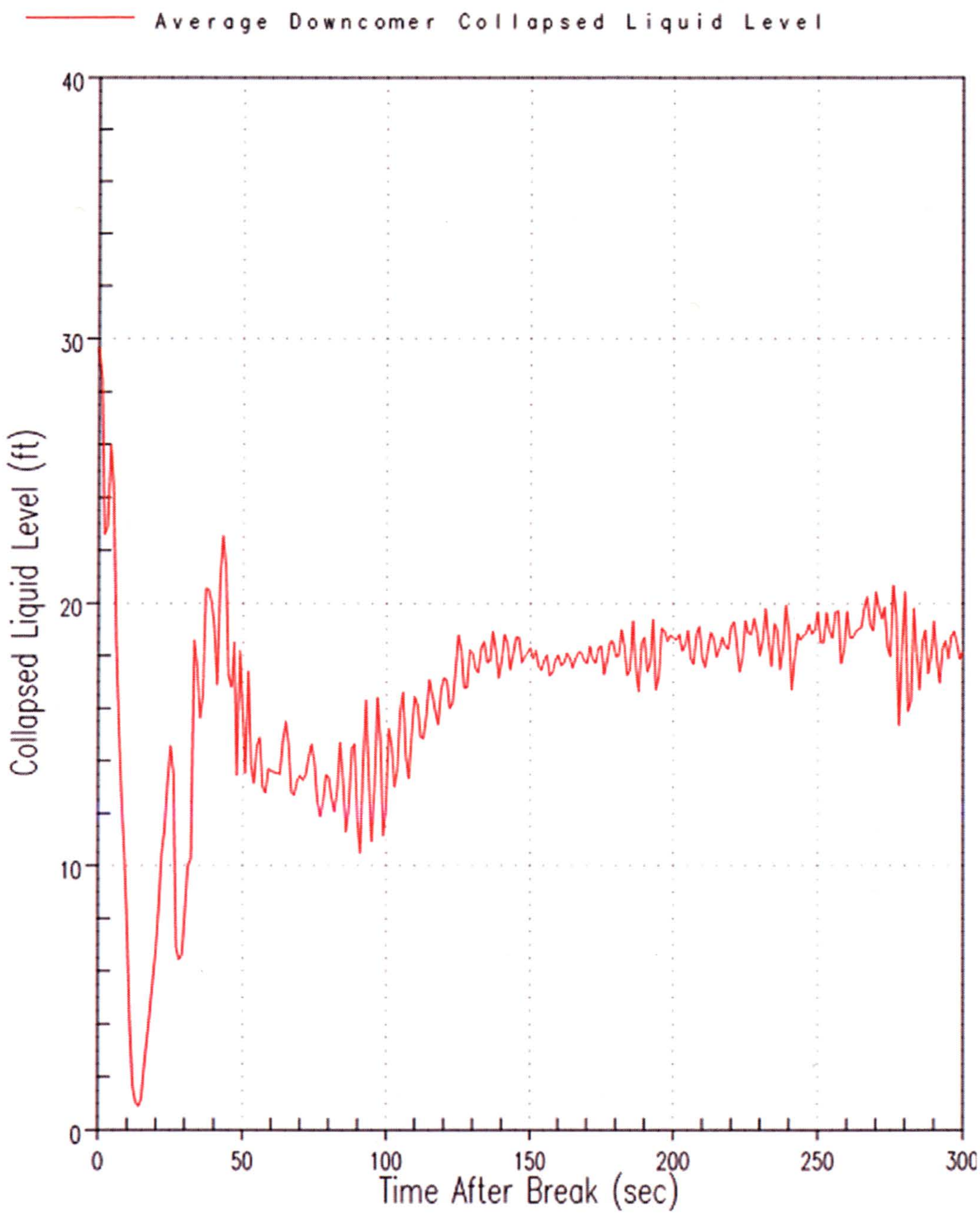

Figure 15.4.1.3-24B: Diablo Canyon Power Plant Unit 2 Average Downcomer Collapsed Liquid Level for LBLOCA (Region II)

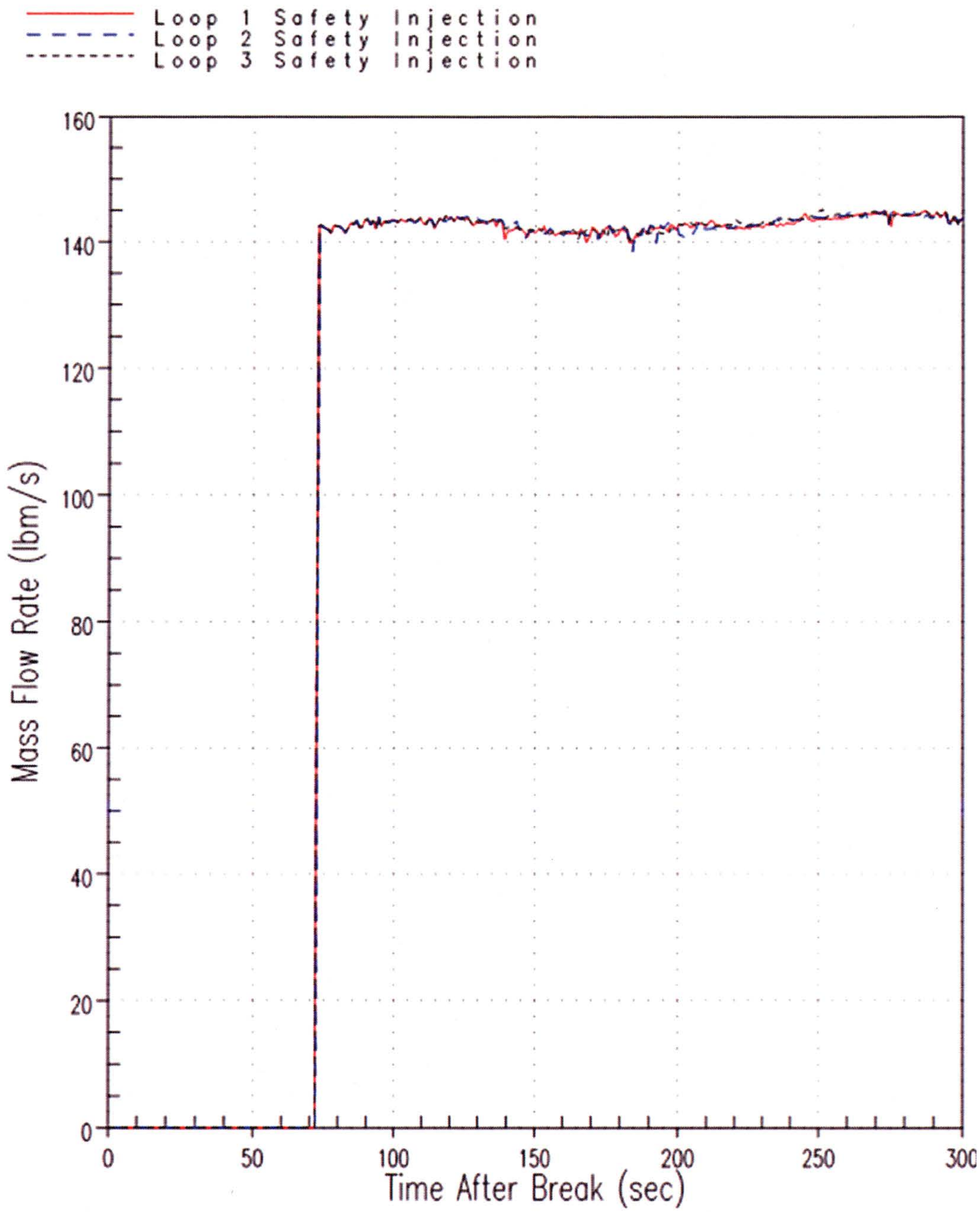

Figure 15.4.1.3-25A: Diablo Canyon Power Plant Unit 1 Total Safety Injection Flow Rate per Loop (not including Accumulator Flow) for LBLOCA (Region II)

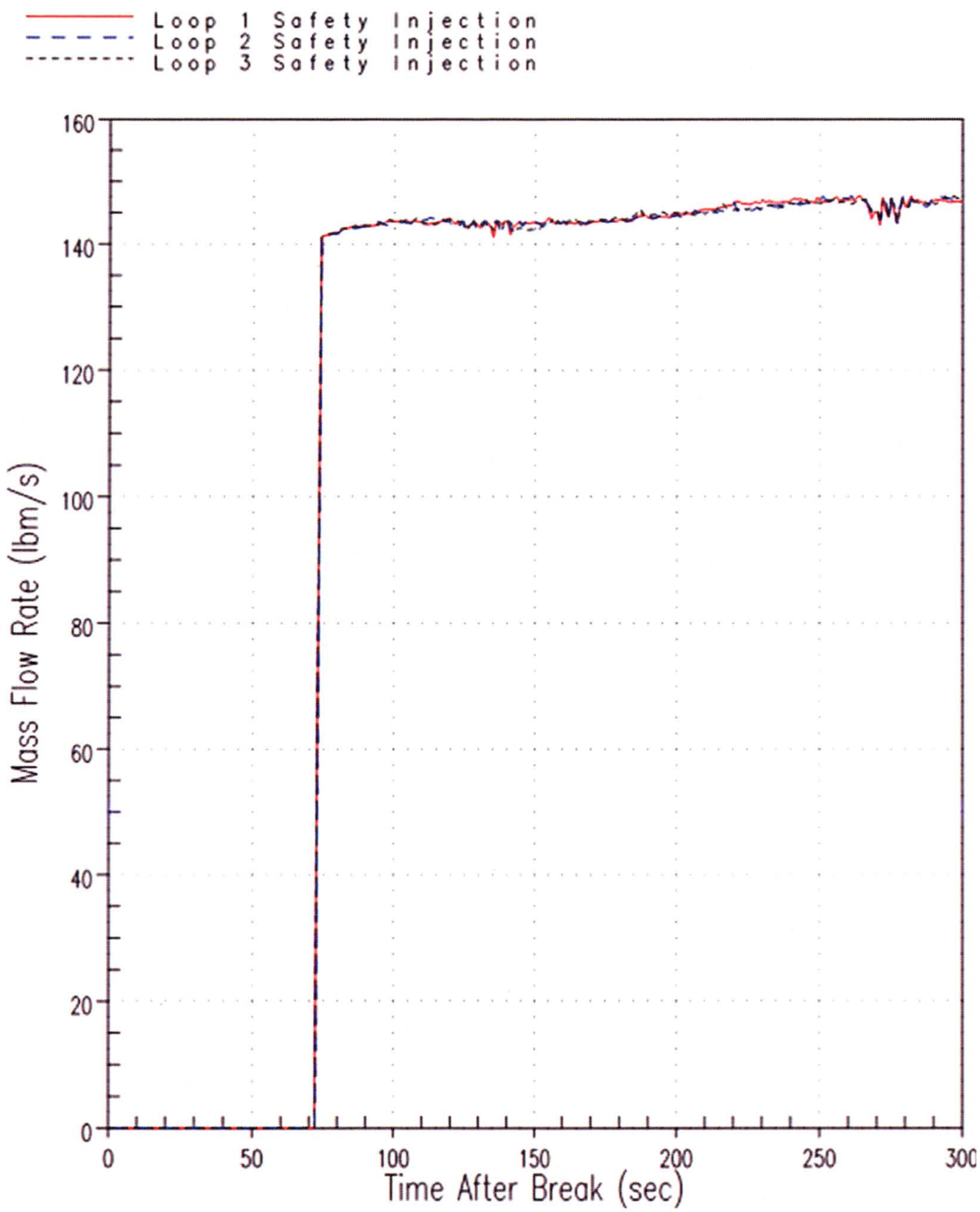

Figure 15.4.1.3-25B: Diablo Canyon Power Plant Unit 2 Total Safety Injection Flow Rate per Loop (not including Accumulator Flow) for LBLOCA (Region II)

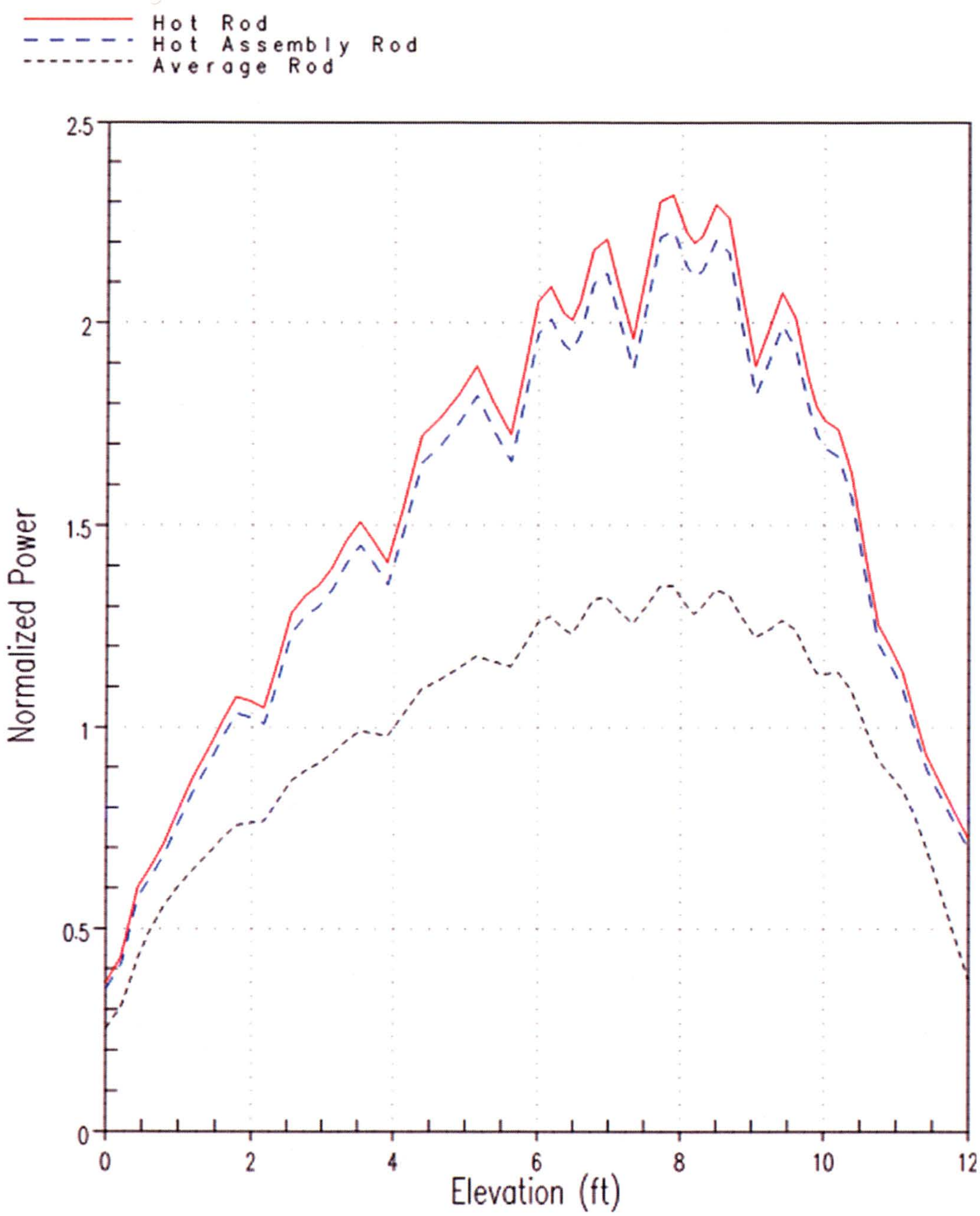

Figure 15.4.1.3-26A: Diablo Canyon Power Plant Unit 1 Normalized Core Power Shape for LBLOCA (Region II)

Note: The localized power decreases occur at grid elevations.

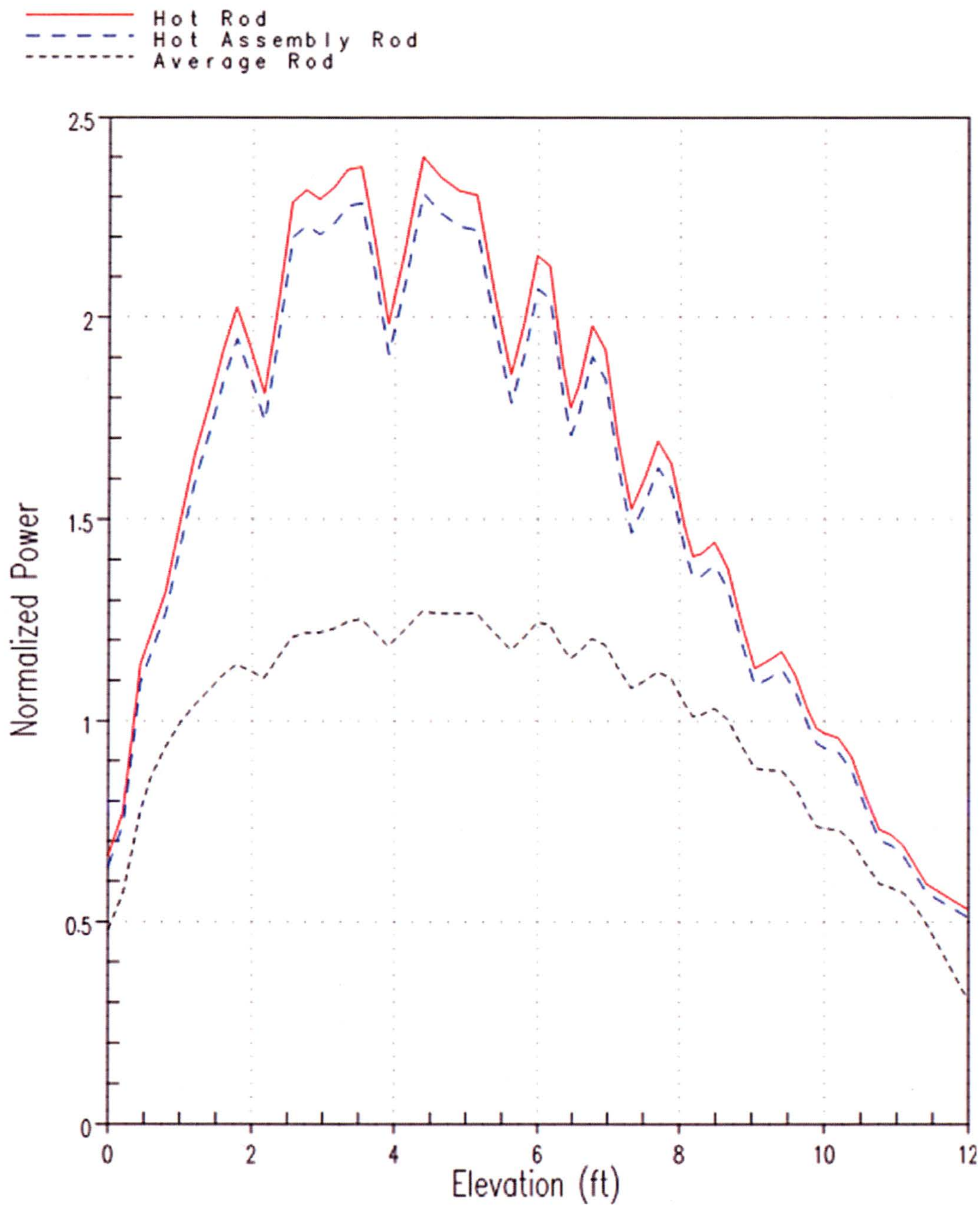

Figure 15.4.1.3-26B: Diablo Canyon Power Plant Unit 2 Normalized Core Power Shape for LBLOCA (Region II)

Note: The localized power decreases occur at grid elevations.

(3) Table 15.4-11 shows the fuel melting limited to less than the innermost 10 percent of the fuel pellet at the hot spot.

15.4.6.5.2 Maximum RCS Pressure

(1) A detailed calculation of the pressure surge for an ejection worth of one dollar reactivity insertion at BOL, hot full power, indicates that the peak pressure does not exceed that which would cause stress to exceed the faulted condition stress limits. Because the severity of the present analysis does not exceed this worst case analysis, the accident for this plant will not result in an excessive pressure rise or further damage to the RCS.

15.4.6.5.3 Radiological

(1) Section 15.5.23 concludes that offsite exposures from a RCCA ejection accident will be well below the guideline levels specified in 10 CFR Part 100, and that the occurrence of such accidents would not result in undue risk to the public. Table 15.5-52 provides a summary of offsite doses from a rod ejection accident.

15.4.7 RUPTURE OF A WASTE GAS DECAY TANK

Refer to Section 15.5.24 for the description of this event.

15.4.8 RUPTURE OF A LIQUID HOLDUP TANK

Refer to Section 15.5.25 for the description of this event.

15.4.9 RUPTURE OF VOLUME CONTROL TANK

Refer to Section 15.5.26 for the description of this event.

15.4.10 REFERENCES

1. "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 4, 1974.

2. ~~F. M. Bordelon, et al, Westinghouse ECCS Evaluation Model - Summary, WCAP-8339, July 1974.~~ Deleted.

3. Deleted in Revision 12.

41. Diablo Canyon Units 1 and 2 Replacement Steam Generator Program – NSSS Licensing Report, WCAP-16638 (Proprietary), Revision 1, January 2008.

42. Plant Manual, Volume 3A, Emergency Operating Procedures, Diablo Canyon Power Plant Units 1 and 2.

43. Deleted in Revision 12.

44. Deleted in Revision 12.

45. Deleted in Revision 12.

46. Deleted in Revision 16.

47. Deleted in Revision 18.

48. Deleted in Revision 12.

49. Deleted in Revision 12.

50. Deleted in Revision 12.

51. Deleted in Revision 12.

52. PG&E Calculation N-160, "Liquid Holdup Tank Rupture Doses," Revision 0, October 11, 1994.

53. Deleted in Revision 18.

54. Emergency Core Cooling Systems: Revisions to Acceptance Criteria, Federal Register, V53, N180, pp. 35996-36005, September 16, 1988.

55. Regulatory Guide 1.157, Best-Estimate Calculations of Emergency Core Cooling System Performance, USNRC, May 1989.

56. ~~Westinghouse~~ Code Qualification Document for Best Estimate ~~Loss-of-Coolant Accident~~ LOCA Analysis, WCAP-12945-P (Proprietary), ~~Volumes I-V.~~ Volume 1, Revision 2 and Volumes 2-5, Revision 1, March 1998.

57. Deleted in Revision 18.

58. Deleted in Revision 21.

59. Deleted in Revision 18.

60. ~~Best Estimate Analysis of the Large Break Loss of Coolant Accident for Diablo Canyon Power Plant Units 1 and 2 to Support 24-Month Fuel Cycles and Unit 1 Uprating, WCAP-14775, January 1997.~~ Deleted.

61. Containment Pressure Analysis Code (COCO), WCAP-8327 (Proprietary) and WCAP-8326 (Non-Proprietary), June 1974.

62. H. Chelemer, et al., Improved thermal Design Procedure, WCAP-8567-P-A (Proprietary) and WCAP-8568-A (Non-Proprietary), February 1989.

63. Nuclear Safety Advisory Letter NSAL-02-04, "Steam Line Break During Mode 3," October 30, 2002.

64. WCAP-11677, Pressurizer Safety Relief Valve Operation for Water Discharge During Feedwater Line Break, January 1988.

65. Deleted in Revision 21.

66. Deleted in Revision 21.

67. ~~Letter from W.R. Rice (Westinghouse Electric Company) to J. Ballard (PG&E), "Diablo Canyon Unit1, BELOCA Reanalysis Final Engineering Report," PGE-03-33, June 6, 2003.~~ Deleted.

68. SECY-83-472, Information Report from W.J. Dircks to the Commission, "Emergency Core Cooling System Analysis Methods," November 17, 1983.

69. Realistic Large-Break LOCA Evaluation Methodology Using the Automated Statistical Treatment Method (ASTRUM), WCAP-16009-P-A, (Proprietary), January 2005.

70. RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses, WCAP-14882-P-A (Proprietary), April 1999, and WCAP-15234-A (Non-Proprietary), May 1999.

71. Deleted in Revision 21.

72. PGE-10-56, "PG&E Diablo Canyon Units 1 and 2, Steam Generator Tube Rupture Margin to Overfill Analysis (CN-CRA-10-45 Rev. 0)," October 18, 2010

73. ~~WCAP-16443-P , Rev. 1, Diablo Canyon Unit 2 ASTRUM BE-LBLOCA Engineering Report, November 2005~~ Deleted.

74. Barrett, G.O., et al., Pressurizer Safety Valve Set Pressure Shift, WCAP-12910, Rev 1-A (Proprietary), May 1993.

Add Insert 1

Insert A:

	Title	Section Reference	Date Submitted to AEC/NRC
66.	J.R. Kobelak, et al, Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (FULL SPECTRUM LOCA Methodology), WCAP-16996-P-A, Revision 1, November 2016.	15.1, 15.4	11/23/2010

Insert B:

15.1.2.11 WCOBRA/TRAC-TF2

The thermal-hydraulic system computer code (WCOBRA/TRAC-TF2) was reviewed and approved for the calculation of fluid and thermal conditions in a PWR during a loss-of-coolant (LOCA) as discussed in WCAP-16996-P-A, Revision 1. This code models postulated LOCA transients through the full spectrum of break sizes. It is discussed in Section 15.4.1.2.2.2 and in the licensing topical report, Section 1.6.1, Item 66.

Insert C:

15.1.2.14 COCO

Containment pressure is calculated using the COCO code (WCAP-8327 and WCAP-8326, Reference 9 in Section 15.4.10) as discussed in Sections 15.4.1.3.2.2A and 15.4.1.3.2.2B. The COCO code is integrated into the WCOBRA/TRAC-TF2 thermal-hydraulic code as discussed in WCAP-16996-P-A, Revision 1.

	Title	Section Reference	Date Submitted To AEC/NRC
58.	D.S. Huegel, et al, RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses, WCAP-14882-P-A (Proprietary), April 1999 and WCAP-15234-A (Non-Proprietary), May 1999.	6.2D, 15.1, 15.2, 15.4	6/6/97
59.	Not used. ~~M.E. Nissley, et al, Realistic Large-Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM), WCAP-16009-P-A, January 2005.~~	~~4.3, 15.4~~	~~6/2/2003~~
60.	T. Q. Nguyen, et al, Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurized Water Reactor Cores, WCAP-11596-P-A (Proprietary),June 1988.	4.3, 15.1	11/87
61.	Y. S. Liu, et al, ANC: A Westinghouse Advanced Nodal Computer Code, WCAP-10965-P-A (Proprietary), September 1986 and WCAP-10966-A (Non-Proprietary), September 1986.	4.3, 15.1, 15.3	5/86
62.	Not used. ~~S. M. Bajorek, et al, Code Qualification Document for Best Estimate LOCA Analysis, WCAP-12945-P-A, Volume I: Models and Correlations, Revision 2, March 1998. Volume II: Heat Transfer Model Validation, Revision 1, March 1998. Volume III: Hydrodynamics, Components, and Integral Validation, Revision 1, March 1998. Volume IV: Assessment of Uncertainty, Revision 1, March 1998. Volume V: Quantification of Uncertainty, Revision 1, March 1998.~~	~~15.1, 15.4~~	~~5/97~~
63.	Not used. ~~N. Lee, et al, Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code, WCAP-10054-P-A (Proprietary) and WCAP-10081-A (Non-Proprietary), August 1985 and, Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection Into the Broken Loop and COSI Condensation Model, WCAP-10054-P-A, Addendum 2, Revision 1, July 1997 (Westinghouse Proprietary).~~	~~15.1, 15.3~~	~~11/88~~

Title	Section Reference	Date Submitted To AEC/NRC
64. ~~P. E. Meyer, NOTRUMP, A Nodal Transient Small Break And General Network Code, WCAP-10079-P-A (Proprietary) and WCAP-10080-A (Non-Proprietary), August 1985.~~	~~15.1, 15.3~~	~~11/88~~
65. J. C. Schmertz, et al, Technical Justification for Eliminating Large Primary Loop Rupture as the Structural Design Basis for the Diablo Canyon Units 1 and 2 Nuclear Power Plants, WCAP-13039 (Proprietary) and WCAP-13038 (Non-proprietary), November 1991.	3.6	3/92

Not used.

Add Insert A.

and 5.5.12 for additional information. For RCS pressure control during a steam generator tube rupture (SGTR), refer to Section 15.4.3.

(6) <u>Steam Flow Restriction</u>

Each SG has an integral flow restrictor located in the steam outlet nozzle to limit the steam blowdown from the SGs in the event of a main steam line rupture. The flow restrictor consists of seven 6.03-inch ID venturi nozzles. These flow restrictors are separate from the in-line 16-inch diameter flow restrictors in the MSS described in Section 10.3.3.13 (5). The flow restrictors are discussed in detail in Sections 5.5.4 and 15.4.2.

(7) <u>RCP Coastdown</u>

Sufficient pump rotation inertia is provided by a flywheel, in conjunction with the impeller and motor assembly, to provide adequate RCS flow during coastdown. The flywheel inertia of the four RCPs sustains reactor coolant flow for a period of time sufficient to assure the minimum heat removal needed to prevent immediate damage to the core.

The assumption of RCP coastdown in relation to the safety analyses is discussed further in Sections 5.5.1.3.2, 15.2.5, 15.2.9, 15.3.1, 15.3.4, and 15.4.2.

15.4.1,

(8) <u>Pressurizer Relief Tank</u>

The PRT and rupture disks are designed for a vacuum to prevent tank collapse if the contents cool following a discharge without nitrogen being added.

5.1.8.18 10 CFR 50.49 – Environmental Qualification of Electrical Equipment Important to Safety for Nuclear Power Plants

RCS instrumentation and control equipment required to function in a harsh environment under accident conditions is qualified to the applicable environmental conditions to ensure that they will continue to perform their PG&E Design Class I functions. Section 3.11 describes the DCPP EQ Program and the requirements for the environmental design of electrical and related mechanical equipment. The affected equipment is listed in the EQ Master List and includes junction boxes, switches, solenoid valves, valve motors, acoustic monitors, resistance temperature detectors (RTDs), differential pressure indicating switches, and pressure transmitters.

5.1.8.19 10 CFR 50.55a(f) – Inservice Testing Requirements

The PG&E Design Class I RCS components comply with the ASME Code for Operation and Maintenance of Nuclear Power Plants and are tested to the requirements of 10 CFR 50.55a(f)(4) and 10 CFR 50.55a(f)(5) to the extent practical.

containment recirculation sump water level and temperature, accumulator tank level and pressure, accumulator isolation valve position, RWST level, containment isolation valve position, and subcooling margin indication during and following an accident.

6.3.1.29 NUREG-0737 (Items I.C.1, I.D.2, II.B.2, II.F.1, II.F.2, II.K.3.30, II.K.3.31, III.D.1.1), November 1980 – Clarification of TMI Action Plan Requirements

Item I.C.1 – Guidance for the Evaluation and Development of Procedures for Transients and Accidents: NUREG-0737, Supplement 1, January 1983 provides the requirements for I.C.1 as follows:

Section 7.1(b) – Transients and accidents were reanalyzed for the purposes of preparing technical guidelines and upgrading emergency operating procedures.

Item I.D.2 – Plant Safety Parameter Display Console: NUREG-0737, Supplement 1, January 1983 provides the requirements for I.D.2 as follows:

Section 4.1(f)(v), Containment Conditions: The ECCS provides instrumentation for control room personnel to monitor containment recirculation sump water level during and following an accident. Additional monitors are provided in the technical support center (TSC) and emergency operations facility (EOF).

Item II.B.2 – Design Review of Plant Shielding and Environmental Qualification of Equipment for Space/Systems Which May Be Used in Postaccident Operations: Plant shielding provides adequate access to, and occupancy of, the switchgear rooms for the purpose of restoring power to normally de-energized ECCS valves.

Item II.F.1 – Additional Accident Monitoring Instrumentation:

Position (5) – The ECCS provides continuous instrumentation to monitor containment recirculation sump water level in the control room.

Item II.F.2 – Instrumentation for Detection of Inadequate Core Cooling: ECCS instrumentation provides an unambiguous indication of inadequate core cooling by indicating the existence of inadequate core cooling caused by various phenomena and does not erroneously indicate inadequate core cooling due to the presence of an unrelated phenomenon. The instrumentation includes reactor water level indication and provides an advance warning of the approach to inadequate core cooling. The instrumentation covers the full range from normal operation to complete core uncovery.

methodology

Item II.K.3.30 – Revised Small-Break Loss-Of-Coolant-Accident Methods to Show Compliance with 10 CFR Part 50, Appendix K: The analysis method for small-break loss-of-coolant accidents (SBLOCAs) ~~was revised for compliance with 10 CFR Part 50, Appendix K.~~ The ~~revision~~ accounts for comparisons with experimental data, including data from the loss of fluid test and semiscale test facilities.

, rig-of-safety assessment,

6.3-6

is a realistic evaluation model (EM) in compliance with 10 CFR Part 50.46(a)(1)(i). An Appendix K method is no longer used for the SBLOCA analysis, but the use of a realistic method has been approved for use in SBLOCA analyses.

redundant signals from each isolation valve. This interlock prevents opening the sump isolation valve when the RWST isolation valves are open and thus prevents dumping the RWST contents into the containment recirculation sump.

To preclude spurious movement of specific MOVs that could result in a loss of ECCS function, electric power is removed from certain valves during normal operation. These valves are listed in Table 6.3-12 and further discussion is provided in Section 6.3.3.42.

6.3.2.6 Schematic Piping and Instrumentation Diagrams

Piping schematic diagrams of the ECCS are shown in Figures 3.2-8, 3.2-9, and 3.2-10.

6.3.2.7 ECCS Flow Diagrams

Alignment of the major ECCS components during the injection and recirculation phases is shown in Figures 6.3-4 and 6.3-5, respectively. Tables 15.4.1.3-2A, 15.4.1.3-2B, 15.4.1.3-3A, 15.4.1.3-3B, ~~15.3-2~~ and 15.3-3 summarize the calculated times at which the major components perform their safety-related functions for various accident conditions (tabulated in Table 15.1-2) that require ECCS operations.

6.3.2.8 Applicable Codes and Classifications

The codes and standards to which the individual ECCS components are designed are listed in Table 6.3-2.

6.3.2.9 Materials

Table 6.3-3 lists the materials used in ECCS components.

6.3.2.9.1 Material Specifications and Compatibility

Materials employed for ECCS components are given in Table 6.3-3. Materials are selected to meet the applicable material code requirements of Table 6.3-2 and the following additional requirements:

(1) All parts of components in contact with borated water are fabricated of or clad with austenitic stainless steel or similar corrosion-resistant material.

(2) All parts of components in contact (internal) with sump solution during recirculation are fabricated of austenitic stainless steel or similar corrosion resistant material.

(3) Valve seating surfaces are hard-faced with Stellite No. 6 or similar to prevent galling and to reduce wear.

(4) Valve stem materials were selected for their corrosion resistance, high tensile properties, and resistance to surface scoring by the packing.

6.3.2.12 Accumulator Availability

Accumulator availability requirements during power operation, hot standby, and startup conditions are detailed in the Technical Specifications (Reference 10).

6.3.2.13 Dependence on Other Systems

Other systems that operate in conjunction with the ECCS are as follows:

(1) The CCW system (refer to Section 9.2.2) cools the RHR heat exchangers during the recirculation mode of operation. It also supplies cooling water to CCP1 and CCP2, the SI pumps, and the RHR pumps during the injection and recirculation modes of operation.

(2) The auxiliary salt water (ASW) system (refer to Section 9.2.1) provides cooling water to the CCW heat exchangers.

(3) The Preferred Power Supply (230-kV and 500-kV), Onsite Distribution System (120-Vac, 480-V, and 4.16-kV), and Standby Power Supply provide normal and emergency power sources for the ECCS (refer to Sections 8.2, and 8.3.1.1.3 through 8.3.1.1.6).

(4) The engineered safety features actuation system (ESFAS) (refer to Section 7.3) generates the initiation signal for emergency core cooling.

(5) The auxiliary feedwater (AFW) system (refer to Section 6.5) supplies feedwater to the steam generators.

(6) The auxiliary building ventilation system (ABVS) (refer to Section 9.4.2) removes heat from the pump compartments and provides for radioactivity contamination control should some leakage occur in a compartment.

6.3.2.14 Lag Times

15.4.1.3-2A, 15.4.1.3-2B,
15.4.1.3-3A, 15.4.1.3-3B,

The sequence and time-delays for actuation of ECCS components for the injection and recirculation phases of emergency core cooling are given in Table 6.3-7. Alignment of the major ECCS components during the injection and recirculation phases is shown in Figures 6.3-4 and 6.3-5, respectively. Tables ~~15.3-2~~ and 15.3-3 summarize the calculated times at which the major components perform the safety-related functions for those various accident conditions (tabulated in Table 15.1-2) that require the ECCS.

The minimum active components will be capable of delivering full rated flow within a specified time interval after process parameters reach the setpoints for the "S" signal. Response of the system is automatic with appropriate allowances for delays in actuation of circuitry and active components. The active portions of the system are actuated by the "S" signal. In analyses of system performance, delays in reaching the programmed

trip points and in actuation of components are established on the basis that only Standby Power Supply is available. The starting sequence following a loss of offsite power is discussed in detail in Chapter 8.

The ECCS is operational after an elapsed time not greater than 25 seconds, including the time to bring the RHR pumps up to full speed.

The starting times for components of the ECCS are consistent with the delay times used in the LOCA analyses for large and small breaks.

In the LOCA analysis presented in Sections 15.3 and 15.4, no credit is assumed for partial flow prior to the establishment of full flow and no credit is assumed for the availability of the Preferred Power Supply.

For smaller LOCAs, there is some additional delay before the process variables reach their respective programmed trip setpoints since this is a function of the severity of the transient imposed by the accident. This is allowed for in the analyses of the range of LOCAs (refer to Tables ~~15.3-1, 15.4.1-1A, and 15.4.1-1B~~). 15.4.1.2-1A and 15.4.1.2-1B

Accumulator injection occurs immediately when RCS pressure has decreased below the operating pressure of the accumulator.

6.3.3 Safety Evaluation

6.3.3.1 General Design Criterion 2, 1967 – Performance Standards

With the exception of the RWST, all ECCS components are housed within the PG&E Design Class I auxiliary and containment buildings. These buildings, or applicable portions thereof, are designed to withstand the effects of winds and tornadoes (refer to Section 3.3), floods and tsunamis (refer to Section 3.4), external missiles (refer to Section 3.5), earthquakes (refer to Section 3.7), and other natural phenomena, to protect ECCS SSCs, ensuring their safety-related design functions will be performed.

The RWST, as discussed in Section 3.8, is a PG&E Design Class I outdoor water storage tank which is designed to withstand the effects of floods and tsunamis (refer to Section 3.4), external missiles (refer to Section 3.5), earthquakes (refer to Section 3.7), and other natural phenomena. The ability of the RWST to withstand the effects of winds and tornadoes is addressed in Section 3.3. The loss of the RWST, which is not contained within a building and is exposed directly to potential wind and tornado loads, has been evaluated. Loss of this equipment does not compromise the capability of shutting down the plant safely (refer to Section 3.3.2.3). Leakage from the refueling water storage tanks due to tornado or missile-induced damage will not result in the flooding of PG&E Design Class I equipment in the auxiliary building since essentially watertight cover plates are installed over the pipe entranceway from each tank into the auxiliary building. The water will drain away from the building via the plant yard drainage system.

6.3.3.6.2.1 Large Pipe Break Analysis

The large pipe break analysis is used to evaluate the initial core thermal transient for a spectrum of pipe ruptures from a break size greater than 1.0 square foot up to the double-ended rupture of the largest pipe in the RCS.

The injection flow from active components is required to control the cladding temperature subsequent to accumulator injection, complete reactor vessel refill, and eventually return the core to a subcooled state. The results indicate that the maximum cladding temperature attained at any point in the core is such that the limits on core behavior as specified in Section 15.4.1 are met.

to ensure that the small breaks with a likelihood of the most severe core uncovery are captured within the analysis.

6.3.3.6.2.2 Small Pipe Break Analysis

The small pipe break analysis is used to evaluate the initial core thermal transient for a spectrum of pipe ruptures~~, which bounds breaks corresponding to the smallest break size, typically a 3/8 inch diameter opening (0.11 square inch), up to and including a break size of 1.0 square foot. For a break opening 3/8 inch or smaller, the makeup flow rate from either CCP1 or CCP2 is adequate to allow time for an orderly plant shutdown without automatic ECCS actuation.~~

The results of the small pipe break analysis indicate that the limits on core behavior are adequately met, as shown in Section 15.3.1

and Section 15.4.1.

6.3.3.6.2.3 Recirculation Cooling

Core cooling during recirculation can be maintained by the flow from one RHR pump if RCS pressure is low. If RCS pressure remains high, either CCP1 or CCP2 and one SI pump operating in series with one RHR pump provide the added head and flow needed to maintain adequate cooling.

Heat removal from the recirculated sump water is accomplished via operation of one or both of the RHR heat exchangers.

6.3.3.6.2.4 Required Operating Status of ECCS Components

Normal operating status of ECCS components is given in Table 6.3-6.

ECCS components are available whenever the coolant energy is high and the reactor is critical. During low temperature physics tests, there is a negligible amount of stored energy and low decay heat in the coolant; therefore, an accident comparable in severity to accidents occurring at operating conditions is not possible in low temperature physics tests, and ECCS components are not required.

The design philosophy with respect to active components in the high-head/low-head injection system is to provide backup equipment so that maintenance is possible during operation without impairment of the system safety function (refer to Section 6.3.3.5). Routine servicing and maintenance of equipment of this type that is not required more frequently than on an outage basis would generally be scheduled for periods of refueling and maintenance outages. The Technical Specifications (Reference 10) discuss in detail the applicable limiting conditions for maintenance during operations.

6.3.3.10 General Design Criterion 42, 1967 – Engineered Safety Features Components Capability

Instrumentation, motors, and penetrations located inside the containment are selected to meet the most adverse accident conditions to which they may be subjected. These items are either protected from containment accident conditions or are designed to withstand, without failure, exposure to the worst combination of temperature, pressure, and humidity expected during the required operational period.

The ECCS pipes serving each loop are anchored at the missile barrier (i.e., the crane wall) in each loop area to restrict potential accident damage to the portion of piping beyond this point. The anchorage is designed to withstand, without failure, the thrust force exerted by any branch line severed from the reactor coolant pipe and discharging fluid to the atmosphere, and to withstand a bending moment equivalent to that producing failure of the piping under the action of a free-end discharge to atmosphere or motion of the broken reactor coolant pipe to which the ECCS pipes are connected. This prevents possible failure at any point upstream from the support point including the branch line connection into the piping header.

6.3.3.11 General Design Criterion 43, 1967 – Accident Aggravation Prevention

15.4.1.3.2

As discussed in Section ~~15.4.1.4~~, the introduction of ECCS supplied borated cooling water into the core does not result in a net positive reactivity addition.

When water in the RWST at its minimum boron concentration is mixed with the contents of the RCS, the resulting boron concentration ensures that the reactor will remain subcritical in the cold condition with all control rods, except the most reactive RCCA, inserted into the core.

The boron concentration of the accumulator and the RWST is below the solubility limit of boric acid at the respective temperatures.

Thermal stresses on the RCS are discussed in Section 5.2.

pump reliability is achieved by elevating the drive motors above the compartment overflow drain so that the pump motors would not be flooded. Gross leakage from the RHR system can be accommodated in the pump compartments, each of which has a capacity of 9450 gallons.

The RHR heat exchangers and pumps can also be isolated, in the event of gross leakage, through appropriate isolation valves. The isolation valves are operated manually by means of remote valve reach-rod operators located in a shielded valve gallery. Radiation levels in the vicinity of the recirculation loop are discussed in Chapter 12.

Recirculation loop component leakage is detected by means of a radiation monitor that samples the air in the ventilation exhaust ducts from each compartment. Supplemental radiation monitoring is provided in the plant vent (refer to Section 9.4.2.3.6). Alarms in the control room alert the operator when the activity exceeds a preset level, and the capability exists to detect small leaks within a short period of time. Operation of the sump pumps is a less sensitive indication of leakage. Recirculation loop components that are potential sources of leaks are described in Table 5.5-11. The table lists conservative estimates of the maximum leakage expected from each leak source during normal operation. However, the design basis for sizing auxiliary building sump pumps that will be required to dispose of this leakage employs a conservative value of 35 gpm, as described above.

The consequences of a leak through an RHR heat exchanger to the CCW system are discussed in Section 9.2.

6.3.3.22 10 CFR 50.46 – Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Plants

methodology

The ECCS calculated cooling performance is capable of demonstrating a high level of probability that the limits set forth in 10 CFR 50.46 are met. Sections 15.3.1 and 15.4.1 provide discussion of SBLOCA and LBLOCA analyses, respectively, including the approved evaluation ~~methodologies~~, range of coolant rupture and leak sizes evaluated, other properties sufficient to provide assurance that the most severe postulated LOCAs are calculated, and demonstration that the limits set forth in 10 CFR 50.46 are met.

6.3.3.23 10 CFR 50.49 – Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants

The ECCS SSCs required to function in harsh environments under accident conditions are qualified to the applicable environmental conditions to ensure that they will continue to perform their safety functions. Section 3.11 describes the DCPP EQ Program and the requirements for the environmental design of electrical and related mechanical equipment. The affected equipment includes flow and pressure transmitters, valves, and switches, and are listed on the EQ Master List.

Westinghouse designed 3-loop and 4-loop plants with cold leg injection systems.

Figures 7.5-1, and 7.5-1B). Instrument ranges and accuracies are provided in Tables 7.5-2 and 7.5-4.

Item II.F.2 – Instrumentation for Detection of Inadequate Core Cooling: The instrumentation for detection of inadequate core cooling includes the subcooled margin monitors, core exit thermocouple system, and RVLIS. Refer to Section 7.5.2.2 for further discussion.

FSLOCA evaluation model (EM)

Item II.K.3.30 – Revised Small-Break Loss-Of-Coolant-Accident Methods to Show Compliance with 10 CFR Part 50, Appendix K: The methodology for calculating SBLOCAs (i.e., the ~~NOTRUMP model~~) ~~was submitted generically by the WOG (now PWROG), of which PG&E is a participating member. The NOTRUMP model~~ was approved by the NRC Staff for use on ~~DCPP Unit 1 and Unit 2~~. Refer to Section 15.3.1 for further discussion.

generically

on September 12, 2017

and Section 15.4.1.

Item II.K.3.31 – Plant-Specific Calculations to Show Compliance with 10 CFR Part 50.46: Plant specific calculations were performed for DCPP using an NRC approved model ~~(NOTRUMP)~~ for evaluating SBLOCAs. These calculations determined the acceptance criteria of 10 CFR 50.46 are met for DCPP. Refer to Section 15.3.1 ~~for further discussion~~.

FSLOCA EM

and Section 15.4.1.

Item III.D.1.1 – Integrity of Systems Outside Containment Likely to Contain Radioactive Material for Pressurized-Water Reactors and Boiling-Water Reactors: Pressure containing portions of the ECCS are tested periodically to check for leakage. This testing includes the portions of the system that would circulate radioactive water from the containment recirculation sump. The requirements for a leakage reduction program from reactor coolant sources outside containment are included in the Technical Specifications (Reference 10). Inservice valve leakage requirements are specified in the IST Program. Refer to Section 6.2.4 for additional information on the CIS and Section 6.3.3.14 for additional discussion on ECCS leakage testing.

6.3.3.30 Generic Letter 89-10, June 1989 – Safety-Related Motor-Operated Valve Testing and Surveillance

PG&E Design Class I MOVs are designed to function with a pressure differential across the valve disk determined in accordance with Generic Letter 89-10, June 1989. ECCS MOVs, except the accumulator isolation valves, are subject to the requirements of Generic Letter 89-10, June 1989, and associated Generic Letter 96-05, September 1996, "Periodic Verification of Design-Basis Capability of Safety-Related Motor-Operated Valves," and meet the requirements of the DCPP MOV Program Plan. The accumulator isolation valves have no active or credited safety function; therefore, they are not included in DCPP's formal MOV Program Plan.

aligned to the faulted steam generator is lost through the break. The AFW flow is asymmetrically split between two of the three unaffected steam generators. The analysis demonstrates that the reactor coolant remains subcooled, assuring that the core remains covered with water and no bulk boiling occurs in the hot leg.

For the analysis in Section 15.4.2.4, the limiting single failure is the failure of a PG&E Design Class I pressurizer PORV. It is assumed the motor-driven AFW pump that is initially aligned to two intact SGs delivers 390 gpm total AFW flow to two of the three intact SGs at 1 minute after the trip. It is also assumed that operator actions are taken within 10 minutes to isolate the faulted steam generator. For this reason, the other motor-driven AFW pump, which is initially aligned to both the third intact SG and the faulted SG, is assumed to deliver an additional 195 gpm of AFW flow to the third intact SG at 10 minutes after the trip. Similarly, the turbine-driven AFW pump, which is initially aligned to all four SGs, is assumed to deliver an additional total 585 gpm of AFW flow to all three intact SGs at 10 minutes after the trip. The analysis demonstrates that water relief through the pressurizer safety valves is precluded.

Rupture of a Main Steam Pipe Inside Containment

Because the result of the steam line break transient is an initial RCS cooldown, the AFW system does not have a requirement to remove heat in the short term. However, addition of AFW to the faulted steam generator will increase the secondary mass available for release to the containment thus maximizing the peak containment pressure following a steam line break inside containment. This transient is performed at four power levels for several break sizes. AFW is assumed to be initiated at the time the SI setpoint is reached. The AFW flowrate to the faulted SG is maximized based on flow from both motor-driven AFW pumps and the turbine-driven AFW pump where runout protection is not credited. Table 6.5-2 summarizes the assumptions used in this analysis. At 10 minutes after the break, it is assumed that the operator has isolated the AFW system from the faulted steam generator, which subsequently blows down to ambient pressure. This assumption for operator action is also used for temperature profile development for main steam line breaks outside containment. Refer to Section 6.2D.3 for further discussion on a main steam line break inside containment.

Small Break Loss of Coolant Accident

A SBLOCA is described in UFSAR Section 15.3.1. The AFW system plays a minor role in response to an SBLOCA. This is due to the primary means of RCS heat removal occurring through the break. However, once the SG secondary sides' are isolated, the boiling process essentially ceases unless the safety valves are lifting. This means, excluding the safety valves, the only heat removal mechanism through the SGs will be through sensible heat gain of the AFW mass addition. Therefore, the significance of AFW flow with respect to the SBLOCA transient is considered small. For this transient, only 390 gpm of AFW flow is assumed to be provided by one motor-driven AFW pump divided equally to four SGs (97.5 gpm per SG). ~~Normally AFW flow provided by one motor-driven AFW pump would be asymmetrically split between two SGs; however, since the NOTRUMP model used to analyze a SBLOCA event cannot explicitly model~~

~~this condition, a lumped loop model was used.~~ Refer to Section 15.3.1 for a discussion on the SBLOCA event.

Natural Circulation Cooldown

Maximum and minimum flow requirements from the previously discussed transients meet the flow requirements of plant cooldown. This operation, however, defines the basis for tank size, based on the required cooldown duration, maximum decay heat input, and maximum stored heat in the system. The AFW system partially cools the RCS to the point where the RHR system may complete the cooldown . Table 6.5-2 shows the assumptions used to determine the cooldown heat capacity of the AFW system.

The minimum CST volume alone is sufficient to perform the plant cooldown described in Section 6.5.2.2.5 and to address a NRC Generic Letter 81-21, May 1981, postulated worst-case natural circulation cooldown.

Due to Unit 2 being converted to a T_{cold} reactor vessel head design, the natural circulation cooldown rates, and subsequent water volume requirement, between the two Units is different. With the Unit 2 T_{cold} upper head design, a cooldown rate of up to 50°F per hour can be used. With Unit 1 being a T_{hot} upper head design, a reduced cooldown rate of 25°F per hour is required to maintain sub-cooling in the reactor vessel upper head region. The natural circulation cooldown analysis has shown that with the two reactor vessel head designs, the worst case conditions for each Unit occurred when Unit 2 was held in hot standby for two hours followed by a four hour cooldown at a rate of 50°F per hour. The worst-case natural circulation cooldown for Unit 1 was determined to be when Unit 1 was held in hot standby for one hour followed by an eight hour cooldown at 25 °F per hour.

For a worst-case natural circulation cooldown, 196,881 gallons for Unit 1 and 163,058 gallons for Unit 2 are required to cooldown the RCS to 350°F (Mode 4, RHR entry temperature conditions). An additional volume of 3,119 gallons for Unit 1 and 2,942 gallons for Unit 2 are reserved for allowed leakage through internal plenums at CST connections and for margin. The inventory of the CST at the minimum Technical Specification usable volumes of 200,000 gallons for Unit 1 and 166,000 gallons for Unit 2 envelops this total required amount. The usable reserved inventory in both CSTs was increased from 164,678 gallons to 224,860 gallons (Reference 3). In addition to cooling the RCS down to RHR entry conditions, the AFW system allows the plant to remain in hot standby (Mode 3) for an extended period of time if desired by operation and/or required for safe shutdown. Holding Unit 1 in hot standby for 8 hours requires 140,584 gallons. Holding Unit 2 in hot standby for 8 hours requires 140,703 gallons.

The Technical Specifications do not permit the RCS to be heated above 350°F without at least 200,000 gallons for Unit 1 and 166,000 gallons for Unit 2 of usable water in the CST.

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TABLE 6.3-1 Sheet 2 of 3

<u>Residual Heat Exchangers</u>	(Design parameters for these heat exchangers are given in Table 5.5-10)	
<u>Refueling Water Storage Tank</u>	Number (per unit)	1
	Total available tank volume (includes only usable volume)(b), gal	450,000
	Minimum Technical Specifications required volume (includes usable and unusable volume), gal	455,300
	Accident analysis minimum volume ~~(assumed)~~ to low level, gal	~~350,000~~ 250,000
	Boron Concentration, ppm	2300-2500
	Design Pressure, psig	Atmospheric
	Operating Pressure, psig	Atmospheric
	Design Temperature, °F	100
	Material	Austenitic stainless steel with reinforced concrete shroud
<u>Valves</u>		
(1) All Motor-Operated Valves That Must Function on Safety Injection ("S") Signal		
(a) Up to and including 8 inches (excluding SI-8805 A&B, CVCS-8107 and CVCS-8108)	Maximum opening or closing time, sec	10
(b) CVCS-8107 and CVCS 8108	Maximum opening and closing time, sec	14

TABLE 6.5-1

CRITERIA FOR AUXILIARY FEEDWATER SYSTEM DESIGN BASIS CONDITIONS

Page 2 of 2

(a) Ref: ANS N18.2 (This information provided for those transients analyzed in Chapter 15.)
(b) A better-estimate analysis has also been performed to demonstrate that the pressurizer does not fill with a single motor-driven auxiliary feedwater pump feeding 2 SGs a total of 390 gpm.
(c) Refer to Section 15.5 for 10 CFR 100 acceptance criteria for accident analysis dose consequences.
(d) An AFW flowrate of 97.5 gpm per SG is assumed in ~~NOTRUMP~~ the FSLOCA evaluation model (EM) based on 390 gpm divided evenly among 4 SGs ~~since NOTRUMP cannot explicitly model asymmetric flow~~.

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TABLE 6.5-2

Core power is 101.7% of 3411 MW_t, with a ±0.3% calorimetric uncertainty

Sheet 1 of 2

SUMMARY OF ASSUMPTIONS
AFW SYSTEM DESIGN VERIFICATION

Transient	Loss of Normal Feedwater (Loss of Offsite Power)	Natural Circulation Cooldown	Major Rupture of a Main Feedwater Pipe	Major Steam Line Break(b) (Containment)	Small Break Loss of Coolant Accident
a. Max NSSS power	102% of 3425 MW_t	102% of 3411MW_t	102% of 3425 MW_t	102% of 3425 MW_t	~~102% of 3411 MW_t~~
b. Time delay from event to Rx trip	(Refer to Table 15.2-1)	2 sec	(Refer to Table 15.4-8)	Variable	~~4.7 sec~~ Variable
c. AFW system actuation signal/time delay for AFW system flow	Low-low SG level 1 minute	Low-low SG Level 1 minute	Low-low SG level (Refer to Table 15.4-8)	Assumed immediately 0 sec (no delay)	Low pressurizer pressure SI signal / 60 sec
d. SG water level at time of reactor trip.	Low-low SG level 8 % narrow range span (NRS)	Same as LOOP	Low-low SG level 0% NRS	N/A	N/A
e. Decay heat	Figure 15.1-7	Figure 15.1-7	Figure 15.1-7	Figure 15.1-7	Figure 15.1-~~7~~ 6
f. AFW pump design pressure	1102 psig	1112 psia	1102 psig	N/A	~~1130 psig~~ N/A
g. Min. No. of SGs that must receive AFW flow	4 of 4	Same as LONF/LOOP	2 of 4 (Section 15.4.2.2) 3 of 4 (Section 15.4.2.4)	N/A	4 of 4(c)
h. Maximum AFW temperature	100 °F	100 °F	100 °F	100 °F	~~100 °F~~ 80°F (nominal)

DCPP UNITS 1 & 2 FSAR UPDATE

TABLE 6.5-2

Sheet 2 of 2

Transient	Loss of Normal Feedwater (Loss of Offsite Power)	Natural Circulation Cooldown	Major Rupture of a Main Feedwater Pipe	Major Steam Line Break[b] (Containment)	Small Break Loss of Coolant Accident
i. Operator action	None	N/A	10 minutes to isolate the faulted SG	10 minutes to isolate the faulted SG	None
j. AFW purge volume/ temperature	113 ft^3per loop/435° F	113 ft^3 per loop/435 °F	113 ft^3per loop/435° F (Section 15.4.2.2), 425°F (Section 15.4.2.4)	0.0ft^3/ based on power	N/A
k. Normal blowdown	None assumed	None assumed	None assumed	None assumed	None assumed
l. Sensible heat	Table 6.5-3	Table 6.5-3	Refer to cooldown	N/A	~~Refer to cooldown~~ N/A
m. Time at standby/time to cooldown to RHR	2 hr/4 hr with offsite power available (without offsite power available refer to Natural Circulation Cooldown)	Unit 1 – 1 hr/8 hr @ 25 °F Unit 2 – 2 hr/4 hr @ 50 °F	N/A	N/A	N/A
n. AFW flowrate	600 gpm (total) constant (minimum requirement	Variable based on maintaining SG level at lower NR level tap at SG backpressure	For Section 15.4.2.2, 390 gpm (total) constant (after 10 minutes)[a] (minimum requirement) For Section 15.4.2.4, refer to Table 15.4-8	569 gpm to 1588 gpm varying due to faulted SG pressure changes	390 gpm to 4 SGs[c]

(a) Minimum flow of 175.5 / 214.5 gpm to each of the two steam generators receiving AFW flow.
(b) A rupture of a main steam pipe inside containment does not impose any performance related requirements on the AFW system. For the accident analysis, AFW flowrates were maximized to increase the mass and energy contributions from the AFW system (Refer to Section 6.2D.3).
(c) 390 gpm to four SGs was assumed to be provided by one motor-driven AFW pump. The ~~approved NOTRUMP model cannot model asymmetric flow, therefore the~~ 390 gpm is assumed to be distributed equally among the four SGs

(c) Activity in the reactor coolant

1. Fission products

2. Corrosion products

3. Tritium

(d) Operation with steam generator leaks up to the maximum allowed by the Technical Specifications

(3) Normal Operational transients

Normal design transients which do not result in a reactor trip are listed below. Refer to Section 5.2.2.1.5.1 for additional details on these transients.

(a) Plant heatup and cooldown

(b) Step load changes (up to plus or minus 10 percent between 15 percent load and full load)

(c) Ramp load changes (up to 5 percent per minute between 15 percent load and full load)

(d) Turbine load reduction up to and including a 50 percent load rejection from full power

(e) Steady state fluctuations of the reactor coolant average temperature, for purposes of design, is assumed to increase or decrease at a maximum rate of 6°F in 1 minute.

15.1.2 COMPUTER CODES UTILIZED

Summaries of some of the principal computer codes used in transient analyses are given below. Other codes, in particular, very specialized codes in which the modeling has been developed to simulate one given accident, ~~such as the NOTRUMP code used in the analysis of the RCS small pipe rupture (Section 15.3.1),~~ and which consequently have a direct bearing on the analysis of the accident itself, are summarized in their respective accident analyses sections. The codes used in the analyses of each transient event are listed in Table 15.1-4.

15.1.2.1 FACTRAN

FACTRAN calculates the transient temperature distribution in a cross section of a metalclad UO_2 fuel rod (refer to Figure 15.1-8) and the transient heat flux at the surface

RETRAN-02W is further described in the licensing topical report, Section 1.6.1, Item 58.

~~15.1.2.9 NOTRUMP~~

~~The NOTRUMP computer code is a state-of-the-art, one-dimensional general network code consisting of a number of advanced features. Among these features is the calculation of thermal nonequilibrium in all fluid volumes, flow regime-dependent drift flux calculations with counter current flow limitations, mixture level tracking logic in multiple-stacked fluid nodes, and regime-dependent heat transfer correlations. Additional features of the code are condensation heat transfer model applied in the steam generator region, loop seal model, core reflux model, flow regime mapping, etc. NOTRUMP is used to model the thermal-hydraulic behavior of the system and thereby obtain time-dependent values of various core region parameters, such as system pressure, temperature, fluid levels and flow rates, etc.~~

~~Small-Break LOCA (SBLOCA) analysis performed using the NOTRUMP code is further described in Section 15.3 and in the licensing topical reports, Section 1.6.1, Items 63 and 64.~~

Sections 15.1.2.9 and 15.1.2.10 are deleted.

~~15.1.2.10 SBLOCTA (LOCTA-IV)~~

~~The NOTRUMP topical report WCAP-10054-P-A makes reference to the LOCTA-IV code (WCAP-8301) and provides modifications to the LOCTA-IV code for use in small break LOCA analyses (i.e., Small Break LOCTA). Further modifications for an annular fuel pellet model were submitted and approved by the NRC in WCAP-14710-P-A, which states, "the revised model has been installed in the SBLOCTA code, which is one of a series of codes descended from the original LOCTA-IV code, and is specific to analyzing small-break LOCA transients." So, SBLOCTA is the actual computer code name, with base references of WCAP-8301 and WCAP-10054-P-A.~~
~~Small-Break LOCA analysis performed using the LOCTA-IV code is further described in Section 15.3 and listed as Reference 4 in that section.~~

15.1.2.11 ~~WCOBRA/TRAC~~ Add Insert B

~~The thermal-hydraulic computer code (WCOBRA/TRAC, Version Mod 7A, Revision 1) that was reviewed and approved for the calculation of fluid and thermal conditions in the PWR during a large break LOCA in WCAP-12945-P-A, Volumes I through V is described in Section 15.4.1.3 and in the licensing topical report, Section 1.6.1, Item 62.~~

~~15.1.2.12 HOTSPOT~~

~~The use of HOTSPOT along with WCOBRA/TRAC to examine Unit 2 uncertainty using the ASTRUM methodology is discussed in Section 15.4.1.7B.~~

Sections 15.1.2.12 and 15.1.2.13 are deleted.

~~15.1.2.13 MONTECF~~

~~Unit 2 uncertainty evaluation calculations using the ASTRUM methodology was performed by applying a direct, random Monte Carlo sampling to generate the input for the WCOBRA/TRAC and HOTSPOT computer codes as discussed in Section 15.4.1.7B.~~

15.1.2.14 COCO

Add Insert C

~~Containment pressure is calculated using the COCO code (WCAP-8327 and WCAP-8326) as discussed in Section 15.4.1.3 and listed as Reference 61 in that section.~~

15.1.3 OPTIMIZATION OF CONTROL SYSTEMS

Prior to initial startup, a setpoint study (Reference 2) was performed in order to simulate performance of the reactor control and protection systems. Emphasis was placed on the development of a control system that will automatically maintain prescribed conditions in the plant even under the most conservative set of reactivity parameters with respect to both system stability and transient performance.

For each mode of plant operation, a group of optimum controller setpoints was determined. In areas where the resultant setpoints were different, compromises based on the optimum overall performance were made and verified. A consistent set of control system parameters was derived satisfying plant operational requirements throughout the core life and for power levels between 15 and 100 percent. The study contained an analysis of the following control systems: rod cluster assembly control, steam dump, steam generator level, pressurizer pressure, and pressurizer level.

Since initial startup, setpoints and control system components have been maintained to optimize performance. Plant operability margin-to-trip analyses are performed on the NSSS control systems for DCPP Units 1 and 2. The purpose of these analyses is to demonstrate that the margin to relevant reactor trip and Engineered Safety Features Actuation System (ESFAS) setpoints is adequate. The NSSS control systems setpoints and time constants are analyzed to provide stable plant response during and following the operational (Condition I) transients:

:

- 50 percent load rejection from 100 percent power
- 10 percent step-load decrease from 100 percent power
- 10 percent step-load increase from 90 percent power
- Turbine trip without reactor trip from permissive P-9 setpoint

When changes are made, the accident analyses are reviewed and revised as necessary. The impact of maintaining pressurizer level between 22% and 35% during a shutdown to mode 3 and when power is ≤20%, has been evaluated as acceptable since it was determined that there is no adverse impact on any accident analyses (Reference 31). The impact of maintaining pressurizer level greater than or equal to

15.1.10.1 Fission Product Decay

~~The heat generation rates from radioactive decay of fission products that have been assumed in the small break LOCA (SBLOCA) accident analyses are equal to 1.2 times the values for infinite operating time in the 1971 Draft ANS-5 Standard. (Reference 30)~~

The decay heat curve used for the ~~Best Estimate large break~~ LOCA ~~(LBLOCA) analysis~~ analyses is based on the 1979 ANS decay heat ~~curve~~ standard as described in Section ~~8~~ 9 of Reference ~~23~~ 32. ~~This curve with the 20 percent factor included is shown in Figure 15.1-6.~~ The 1979 ANS decay heat curve (Reference 11) is used for the non-LOCA analyses. Figure 15.1-7 presents this curve as a function of time after shutdown for non-LOCA analyses.

15.1.10.2 Decay of U-238 Capture Products

Betas and gammas from the decay of U-239 (23.5-minute half-life) and Np-239 (2.35-day half-life) contribute significantly to the heat generation after shutdown. The cross sections for production of these isotopes and their decay schemes are relatively well known. For long irradiation times their contribution can be written as:

$$P_1/P_0 = \frac{(E_{\gamma 1} + E_{\beta 1})\, c\,(1+\alpha)}{200\,\text{MeV}}\, e^{-\lambda_1 t} \text{ watts/watt} \qquad (15.1\text{-}1)$$

$$P_2/P_0 = \frac{(E_{\gamma 2} + E_{\beta 2})\, c\,(1+\alpha)}{200\,\text{MeV}} \left[\frac{\lambda_2}{\lambda_1 - \lambda_2} \left(e^{-\lambda_2 t} - e^{-\lambda_1 t}\right) + e^{-\lambda_2 t} \right] \text{ watts/watt} \qquad (15.1\text{-}2)$$

where:

P_1/P_0 is the energy from U-239 decay
P_2/P_0 is the energy from Np-239 decay
t is the time after shutdown (seconds)
$c(1+\alpha)$ is the ratio of U-238 captures to total fissions = 0.6 (1 + 0.2)
λ_1 = the decay constant of U-239 = 4.91×10^{-4} per second
λ_2 = the decay constant of Np-239 decay = 3.41×10^{-6} per second
$E_{\gamma 1}$ = total γ-ray energy from U-239 decay = 0.06 MeV
$E_{\gamma 2}$ = total γ-ray energy from Np-239 decay = 0.30 MeV
$E_{\beta 1}$ = total β-ray energy from U-239 decay = 1/3[a] x 1.18 MeV
$E_{\beta 2}$ = total β-ray energy from Np-239 decay = 1/3[a] x 0.43 MeV

[a] Two-thirds of the potential β-energy is assumed to escape by the accompanying neutrinos.

~~For the SBLOCA, based on conservative modeling of the ratio of U-238 captures to total fissions, heavy element decay heat is calculated without applying further uncertainty~~

analyses

32

9

actinides

~~correction (Reference 24).~~ For the ~~Best Estimate~~ LOCA ~~analysis~~, the heat from the radioactive decay of U-239 and Np-239 is calculated as described in Section ~~8~~ of Reference ~~23~~. The decay of other ~~isotopes, produced by neutron reactions other than fission,~~ is neglected. For the non-LOCA analysis, the decay of U-238 capture products is included as an integral part of the 1979 decay heat curve presented as Figure 15.1-7.

, as justified in Section 9.9.1 of Reference 32. The effect of neutron capture in fission products is included, as discussed in Reference 11.

15.1.10.3 Residual Fissions

The time dependence of residual fission power after shutdown depends on core properties throughout a transient under consideration. Core average conditions are more conservative for the calculation of reactivity and power level than actual local conditions as they would exist in hot areas of the core. Thus, unless otherwise stated in the text, static power shapes have been assumed in the analysis and these are factored by the time behavior of core average fission power calculated by a point kinetics model calculation with six delayed neutron groups.

~~For the purpose of illustration, only one delayed neutron group calculation, with a constant shutdown reactivity of -4 percent Δk is shown in Figure 15.1-6.~~

15.1.10.4 Distribution of Decay Heat Following Loss-of-Coolant Accident

~~During an SBLOCA the core is rapidly shut down by void formation or RCCA insertion, or both, and long-term shutdown is assured by the borated ECCS water. A large fraction of the heat generation to be considered comes from fission product decay gamma rays. This heat is not distributed in the same manner as steady state fission power. Local peaking effects that are important for the neutron dependent part of the heat generation do not apply to the gamma ray source contribution. The steady state factor of 97.4 percent that represents the fraction of heat generated within the cladding and pellet drops to 95 percent for the hot rod in a LOCA.~~
~~For example, 1/2 second after the rupture about 30 percent of the heat generated in the fuel rods is from gamma ray absorption. The gamma power shape is less peaked than the steady state fission power shape, reducing the energy deposited in the hot rod at the expense of adjacent colder rods. A conservative estimate of this effect is a reduction of 10 percent of the gamma ray contribution or 3 percent of the total. Since the water density is considerably reduced at this time, an average of 98 percent of the available heat is deposited in the fuel rods, the remaining 2 percent being absorbed by water, thimbles, sleeves, and grids. The net effect is a factor of 0.95, rather than 0.974, to be applied to the heat production in the hot rod.~~

analyses

For the ~~Best Estimate~~ LOCA ~~analysis~~, the energy deposition modeling is performed as described in ~~Section 8 of Reference 23~~.

For the LOCA analyses, the residual fission energy is calculated as described in Section 9 of Reference 32.

Section 9 of Reference 32.

18. PG&E Nuclear Plant, Diablo Canyon Units 1 and 2, Pressurizer Pressure Control System Uncertainty Safety Assessment, Westinghouse Letter PGE-93-659, November 18, 1993.

19. S. Miranda, et al., Steam Generator Low Water Level Protection System Modifications to Reduce Feedwater Related Trips, WCAP-11325-P-A, Rev. 1, February 1988.

20. Deleted in Revision 13.

21. RETRAN-02 -- A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems, Volume 1: Theory and Numerics, (Revision 5), EPRI NP-1850-CCM-A, March 1992.

22. Westinghouse Improved Thermal Design Procedure Instrument Uncertainty Methodology, Diablo Canyon Units 1 and 2, 24-Month Fuel Cycle Evaluation, WCAP-11594, Revision 2, January 1997.

23. ~~S. M. Bajorek, et al., Code Qualification Document for Best Estimate LOCA Analysis, Volume I: Models and Correlations, WCAP-12945-P-A, Volume I, Revision 2, March 1998.~~ Not used.

24. ~~NUREG-0800, Standard Review Plan, Branch Technical Position ASB 9-2, Residual Decay Energy for Light-Water Reactors for Long Term Cooling, July 1981.~~ Not used.

25. Deleted in Revision 22.

26. Deleted in Revision 22.

27. Deleted in Revision 22.

28. Westinghouse Letter PGE-10-53, "Transmittal of LBIE to Address the Increase in Pressurizer Level in Modes 3, 4, & 5, September 23, 2010.

29. Diablo Canyon Units 1 and 2 Tavg and Tfeed Ranges Program NSSS Engineering Report, WCAP-16985-P, Revision 2 (Proprietary), April 2009.

30. ~~Proposed American Nuclear Society Standard—"Decay Energy Release Rates Following Shutdown of Uranium-Fueled Thermal Reactors." Approved by Subcommittee ANS-5, ANS Standards Committee, October 1971.~~ Not used.

31. Westinghouse Letter PGE-12-25, "Pressurizer Level Increase up to 35% Span at 20% Power to Mode 3 Revised Final Engineering Report", March 7, 2012.

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

15.3 CONDITION III - INFREQUENT FAULTS

By definition, Condition III occurrences are faults that may occur very infrequently during the life of the plant. They will be accompanied with the failure of only a small fraction of the fuel rods although sufficient fuel damage might occur to preclude resumption of operation for a considerable outage time. The release of radioactivity will not be sufficient to interrupt or restrict public use of those areas beyond the exclusion radius. A Condition III fault will not, by itself, generate a Condition IV fault or result in a consequential loss of function of the reactor coolant system (RCS) or containment barriers. For the purposes of this report the following faults have been grouped into this category:

(1) Loss of reactor coolant, from small ruptured pipes or from cracks in large pipes, that actuates the emergency core cooling system (ECCS).

(2) Minor secondary system pipe breaks.

(3) Inadvertent loading of a fuel assembly into an improper position.

(4) Complete loss of forced reactor coolant flow.

(5) Single rod cluster control assembly (RCCA) withdrawal at full power.

Each of these infrequent faults is analyzed in this section. In general, each analysis includes acceptance criteria, an identification of causes and description of the accident, an analysis of effects and consequences, a presentation of results, and relevant conclusions.

The time sequences of events during ~~four~~ a Condition III fault~~s~~ of type (1) above, small-break loss-of-coolant accident (SBLOCA), are shown in ~~Table 15.3-1.~~ Tables 15.4.1.3-2A and 15.4.1.3-2B.

15.3.1 LOSS OF REACTOR COOLANT FROM SMALL RUPTURED PIPES OR FROM CRACKS IN LARGE PIPES THAT ACTUATES EMERGENCY CORE COOLING SYSTEM

15.3.1.1 Acceptance Criteria

15.3.1.1.1 10 CFR Part 50, Section 50.46, Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors

(1) *Peak cladding temperature.* The calculated maximum fuel element cladding temperature shall not exceed 2200°F.

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

results of the analyses

The ~~large break~~ LOCA ~~analysis contained in Section 15.4.1 has been revised to incorporate separate Best Estimate LOCA analyses for Units 1 and 2. The general discussion of the Best Estimate LOCA transient in Sections 15.4.1.2, 15.4.1.3, and 15.4.1.4 are applicable to Units 1 and 2. However, the statistical treatment methodologies are slightly different for Units 1 and 2. Statistical treatment methodologies for Units 1 and 2~~ are discussed in Sections ~~15.4.1.4A and 15.4.1.4B respectively~~.

15.4.1 MAJOR REACTOR COOLANT SYSTEM PIPE RUPTURES (LOCA)

15.4.1.3.1.2A, 15.4.1.3.1.2B, 15.4.1.3.2.2A, and 15.4.1.3.2.B

15.4.1.1 Acceptance Criteria

15.4.1.1.1 10 CFR Part 50, Section 50.46, Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors

It must be demonstrated that there is a high level of probability that the limits set forth in 10 CFR 50.46 are met. The acceptance criteria are listed below:

(1) *Peak cladding temperature*. The calculated maximum fuel element cladding temperature shall not exceed 2200 °F.

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

(3) *Maximum hydrogen generation.* The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

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

(5) *Long-term cooling*. After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

15.4.1.1.2 Radiological Criteria

(1) The resulting potential exposures to individual members of the public and to the general population shall be lower than the applicable guidelines and limits specified in 10 CFR Part 100.

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TABLE 15.1-4

Sheet 2 of 4

Events	Computer Codes Utilized	Assumed Reactivity Coefficients: Moderator Temp.[a], pcm/°F[d]	Assumed Reactivity Coefficients: Moderator Density[a], Δk/gm/cc	Assumed Reactivity Coefficients: Doppler[b]	Initial NSSS Thermal Power Output Assumed[c], MWt
CONDITION II (Cont'd)					
Loss of normal feedwater	RETRAN-02W	0	-	Upper	3,425
Loss of offsite power to the plant auxiliaries	RETRAN-02W	0	-	Upper	3,425
Excessive heat removal due to feedwater system malfunctions	RETRAN-02W	-	0.43	Lower	3,425
Excessive load increase	LOFTRAN	-	0 and 0.43	Lower and Upper	3,423
Accidental depressurization of the reactor coolant system	LOFTRAN	+7	-	Lower	3,425
Inadvertent operation of ECCS during power operation - DNBR	LOFTRAN	+5	0.43	Lower and Upper	3,423
Inadvertent operation of ECCS during power Operation – Pressurizer Overfill	RETRAN-02	-	-	-	3,425
CONDITION III					
Loss of reactor coolant from small ruptured pipes or from cracks in large pipe which actuate emergency core cooling	~~NOTRUMP~~ ~~SBLOCTA~~ WCOBRA/TRAC-TF2	~~-~~ Function of moderator density.	~~-~~	~~-~~ Function of fuel temp.	~~3,479~~
Inadvertent loading of a fuel assembly into an improper position	PHOENIX-P, ANC	-	-	-	3,483
Complete loss of forced reactor coolant flow	LOFTRAN, THINC, FACTRAN	0	-	Upper	3,425

Core power is 101.7% of 3411 MW_t, with a ±0.3% calorimetric uncertainty

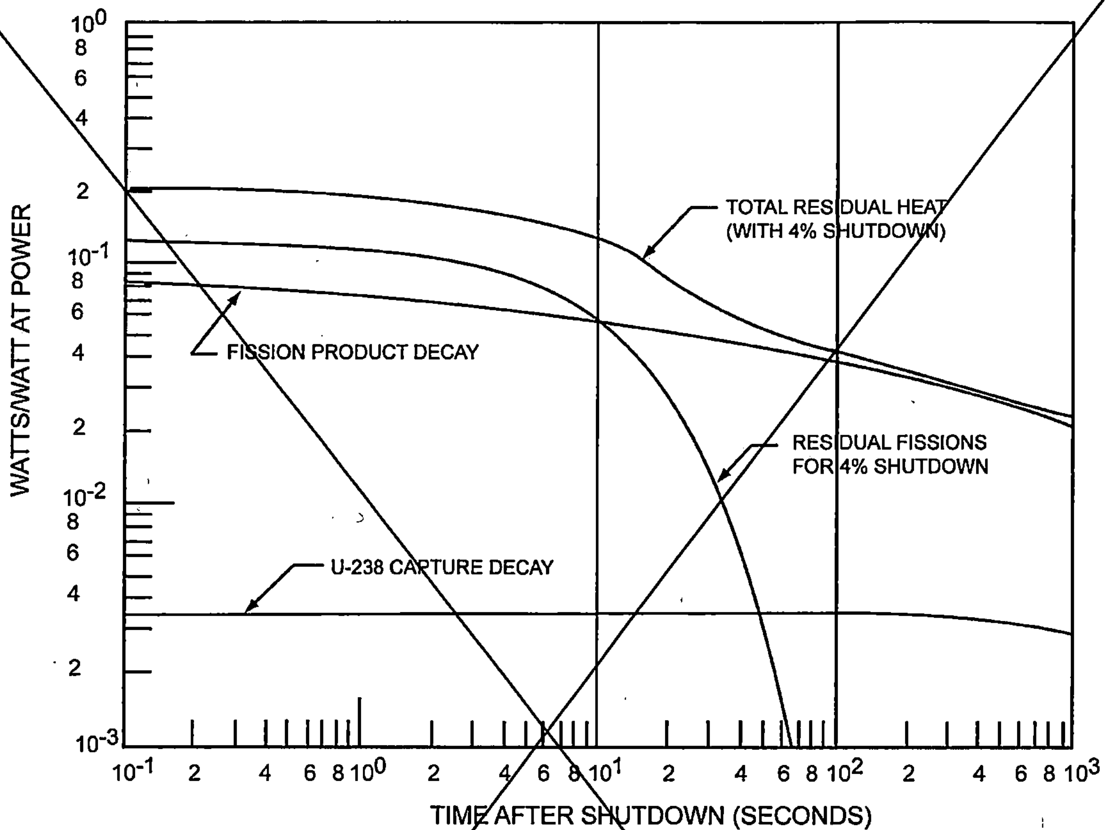
DCPP UNITS 1 & 2 FSAR UPDATE

TABLE 15.1-4

Sheet 3 of 4

Events	Computer Codes Utilized	Assumed Reactivity Coefficients: Moderator Temp[(a)], pcm/°F[(d)]	Moderator Density[(a)], Δk/gm/cc	Doppler[(b)]	Initial NSSS Thermal Power Output Assumed[(c)], MWt
CONDITION III (Cont'd)					
Single RCCA withdrawal at full power	ANC, THINC, PHOENIX-P	-	-	-	3,425[(e)]
CONDITION IV					
Major rupture of pipes containing reactor coolant up to and including double-ended rupture of the largest pipe in the reactor coolant system (loss-of-coolant accident)	~~WCOBRA/TRAC~~ ~~HOTSPOT~~ ~~MONTECF~~ WCOBRA/TRAC-TF2	Function of moderator density. ~~See Sec. 15.4.1~~	~~0~~	Function of fuel temp.	~~3,479~~ Core power is 101.7% of 3411 MW_t, with a ±0.3% calorimetric uncertainty
Major secondary system pipe rupture up to and including double-ended rupture (rupture of a steam pipe)	RETRAN-02W, ANC,THINC	-	Function of moderator density. See Figure 15.4.2-2.	See Figure 15.4.2-1	0.0 (Subcritical)
Major rupture of a main feedwater pipe	RETRAN-02W		0.0	Lower	3,425
Rupture of a main steam line at power	RETRAN-02W, ANC, THINC-IV		0.43	Lower	3,425
Waste gas decay tank rupture	-	-	-	-	3,577
Steam generator tube rupture	RETRAN-02W	-	0.0	Lower and Upper	3,425

Delete Figure 15.1-6



FSAR UPDATE
UNITS 1 AND 2 DIABLO CANYON SITE
FIGURE 15.1-6 RESIDUAL DECAY HEAT (~~BEST ESTIMATE LBLOCA~~ 1979 ANS DECAY HEAT)

Westinghouse Authorization Letter, CAW 18-4798, with Accompanying Affidavit; Proprietary Information Notice; and Copyright Notice

Enclosure
Attachment 6
PG&E Letter DCL-18-100

Compliance with the Limitations and Conditions in the Revised NRC Final Safety Evaluation (FSE) for WCAP-16996-A, Rev. 1 (Non-Proprietary)

Compliance with the Limitations and Conditions in the Revised NRC Final Safety Evaluation (FSE) for WCAP-16996-A, Rev. 1 (Non-Proprietary)

The revised Nuclear Regulatory Commission (NRC) final safety evaluation (FSE) for WCAP-16996-A, Rev. 1 (Reference 1) contains 15 Limitations and Conditions that must be met in order to implement the NRC approved FULL SPECTRUM loss-of-coolant accident (FSLOCA) evaluation methodology (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 Diablo Canyon Power Plant (DCPP) Units 1 and 2 with the FSLOCA EM are only being used to demonstrate compliance with 10 CFR 50.46 (b)(1) through (b)(4).

Limitation and Condition Number 2

Summary

The FSLOCA EM is approved for the analysis of Westinghouse-designed 3-loop and 4-loop pressurized water reactors (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

DCPP Units 1 and 2 are Westinghouse-designed 4-loop PWRs with cold-side injection, so they are within the NRC approved methodology. The analyses for DCPP Units 1 and 2 utilize the NRC approved FSLOCA methodology, except for the changes discussed in Section 3.2 which were previously transmitted to the NRC pursuant to 10 CFR 50.46 in LTR-NRC-18-30 (Reference 2).

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 modelled). 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 DCPP Unit 1 and 2 analyses was performed consistent with the NRC approved methodology. Appropriate design parameters and conditions were modelled, as were the engineered safety features which can reduce the containment pressure. A minimum initial temperature associated with normal full-power operating conditions was modelled, 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 DCPP Units 1 and 2 analyses. The analysis simulations were all executed for less 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 1 for DCPP Unit 1 and Table 2 for DCPP 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 DCPP Units 1 and 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 DCPP Unit 1 and 2 analyses with the FSLOCA EM. The analyzed fuel pellet average temperatures bound the maximum values calculated in accordance with Section 7.5.1 of WCAP-17642-A, Revision 1 (Reference 3), and the analyzed rod internal pressures were calculated in accordance with Section 7.5.2 of WCAP-17642-A, Revision 1.

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 DCPP 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 DCPP 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

DCPP Units 1 and 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 LTR-NRC-18-50 (Reference 4).

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 reactor coolant system (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

DCPP Units 1 and 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 LTR-NRC-18-50.

The minimum sampled break area for the DCPP Units 1 and 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 analyses 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 DCPP Unit 1 and 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 analyses 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 DCPP Units 1 and 2 in Table 3.

<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.

Compliance

A bounding plant-specific dynamic pressure loss from the steam generator secondary-side to the main steam safety valves was modelled in the DCPP Unit 1 and 2 analyses.

Limitation and Condition Number 13

Summary

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

Compliance

The [ ]a,c in the analyses for DCPP Units 1 and 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 DCPP Unit 1 and 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 percent.

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 DCPP Units 1 and 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, and the LOOP configuration is the limiting configuration (see Tables 4 and 5).

The [ ]^{a,c}