FENOC

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October 28, 2005
L-05-168

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

Subject: **Beaver Valley Power Station, Unit Nos. 1 and 2**
BV-1 Docket No. 50-334, License No. DPR-66
BV-2 Docket No. 50-412, License No. NPF-73
Supplement to License Amendment Request Nos.
320 (Unit No. 1 TAC No. MC6725) and
302/173 (Unit No. 1 TAC No. MC4645/Unit No. 2 TAC No. MC4646)

This letter transmits supplements to two license amendment requests that were previously submitted by FirstEnergy Nuclear Operating Company (FENOC). The listed submittals requested amendments to the above licenses in the form of changes to the Beaver Valley Power Station (BVPS) Unit Nos. 1 and 2 Technical Specifications.

The license amendment requests being supplemented are:

1. Replacement Steam Generators, submitted by FENOC Letter L-05-069, License Amendment Request 320, dated April 13, 2005
2. Extended Power Uprate, submitted by FENOC Letter L-04-125, License Amendment Requests 302 and 173, dated October 4, 2004

The primary reason for these two supplements is because the changes to Technical Specification 3.5.1, Accumulators, proposed in the extended power uprate submittal require a revision. The revision consists of raising the minimum accumulator cover pressure to a value that is consistent with a revised small break loss of coolant accident analysis. Since the revised analysis is applicable to both the extended power uprate and replacement steam generators submittals, the revision to the Unit 1 Technical Specification is withdrawn from the extended power uprate submittal and added to the replacement steam generators submittal.

Enclosure 1 provides details of these license amendment request (LAR) supplements and an assessment of the proposed changes. The attachments to Enclosure 1 provide the proposed changes for each of the LARs being supplemented and the supporting technical justification.

Letter L-05-069 requested approval of the replacement steam generators LAR by January 2006 in order to support the installation of the BVPS Unit No. 1 replacement steam generators during the 2006 spring outage. The requested amendment shall be implemented prior to the first entry into Mode 4 during plant startup from the 1R17 refueling outage planned for the spring of 2006.

A001

Letter L-04-125 requested approval of the proposed extended power uprate (EPU) amendments by November 2005. However, as documented in the NRC acceptance review letter dated July 19, 2005, NRC approval of the EPU LAR is expected by July 19, 2006. The Unit No. 1 EPU amendment shall be implemented within 120 days following issuance of the amendment. The Unit No. 2 EPU amendment shall be implemented prior to the first entry into Mode 4 during plant startup from the 2R12 refueling outage planned for the fall of 2006.

The changes proposed in the supplements have been reviewed by the Beaver Valley Power Station review committees. The changes were determined to be safe and do not negate or negatively impact the no significant hazard considerations submitted in FENOC letters L-04-125 or L-05-069.

No new commitments are contained in this submittal. If you have questions or require additional information, please contact Mr. Gregory A. Dunn, Manager - Licensing, at 330-315-7243.

I declare under penalty of perjury that the foregoing is true and correct. Executed on October 28, 2005.

Sincerely,

Thomas S. Cosgrove

Thomas S. Cosgrove

Enclosure: FENOC Evaluation of the Proposed Supplements

c: Mr. T. G. Colburn, NRR Senior Project Manager
Mr. P. C. Cataldo, NRC Senior Resident Inspector
Mr. S. J. Collins, NRC Region I Administrator
Mr. D. A. Allard, Director BRP/DEP
Mr. L. E. Ryan (BRP/DEP)

L-05-168 Enclosure 1

FENOC Evaluation of the Proposed Supplements

BACKGROUND

The NRC review of the replacement steam generator (Reference 1) and extended power uprate (Reference 2) license amendment requests (LAR) has resulted in several requests for additional information (RAI) pertaining to small break loss of coolant accident (SBLOCA) analysis. The SBLOCA analysis supporting these two LARs was performed to be consistent with the NRC approved Westinghouse NOTRUMP-EM methodology. These RAIs requested information that incorporates adjustments to the NRC approved Westinghouse NOTRUMP-EM methodology. The requested information pertains to the impact of not applying the reactor coolant system (RCS) loop seal clearing restriction and using a smaller mesh (non-integer break) size to determine the limiting size break. A re-analysis of SBLOCA at the extended power uprate power level of 2900 MWt was performed to show that acceptable results, consistent with 10 CFR 50.46, can be obtained using the NRC approved Westinghouse NOTRUMP-EM methodology with adjustments that address the NRC concerns (loop seal clearing and non-integer break sizes). However, changes to input assumptions to support the analysis has resulted in the need to revise the minimum accumulator cover pressure in Technical Specification 3.5.1, Accumulators, for both Beaver Valley Power Station (BVPS) Units 1 and 2, to support the changes proposed in the replacement steam generator and extended power uprate LARs.

Since the re-analysis of SBLOCA requires a minimum cover pressure that is different than what is proposed in the extended power uprate (EPU) LAR, and the revised SBLOCA analysis is required to support the replacement steam generator (RSG) LAR, supplements to the RSG and EPU LARs are being submitted for NRC approval.

SUPPLEMENT DETAILS

Presently the EPU LAR proposes changes to Technical Specification 3.5.1 for both units. This EPU supplement proposes a revision to the Technical Specification 3.5.1 changes proposed in the EPU LAR. Since the revised SBLOCA analysis is applicable to both units, and required for the replacement steam generator project, the revised Technical Specification 3.5.1 markup is added to the changes proposed in the RSG LAR and withdrawn from the EPU LAR.

Presently the RSG LAR, which is applicable to Unit 1 only, does not propose any change to Technical Specification 3.5.1. Therefore, this RSG supplement incorporates the revised changes to Technical Specification 3.5.1 into the RSG LAR.

CHANGE DESCRIPTION

The existing proposed changes to Technical Specification 3.5.1 consists of three parts. The first part consists of changing the limits on accumulator water volume and cover pressure. The second part consists of replacing the word "contained" with "usable" in Surveillance Requirement 4.5.1.a.1. The third part consists of providing the accumulator volumes in percent of indicated level in addition to gallons in the volume portion of the Limiting Condition for Operation (LCO) statement.

The revision proposed by these supplements consists of raising the minimum accumulator cover pressure to 611 psig, deleting percent of indicated level in the accumulator volume portion of the LCO statement and inserting "usable" into the same portion of the LCO statement. The new minimum cover pressure is the result of the revised SBLOCA analysis. The percent of indicated level is being removed from the LCO because it needlessly complicates the LCO. This addition was originally proposed as an operator aid, but is unnecessary. Inserting "usable" into the LCO provides consistency with revised Surveillance Requirement 4.5.1.a.1.

Proposed Technical Specification Bases changes for both units were provided for information in the EPU LAR. These changes state that the Technical Specification limits for usable accumulator water volume, boron concentration and minimum cover pressure are analysis values and that the maximum cover pressure limit preserves accumulator integrity. A revision to these Bases changes, reflecting the proposed revisions to the Technical Specification, are included with these supplements. As with the Technical Specification changes, the Unit 1 Bases changes are moved to the RSG LAR and withdrawn from the EPU LAR.

Attachments A-1 and A-2 contain the markups to Technical Specification 3.5.1 for Units 1 (RSG) and 2 (EPU), respectively. Attachments B-1 and B-2 contain the markups to the applicable Technical Specification Bases section for Units 1 (RSG) and 2 (EPU), respectively. The proposed changes to the Technical Specifications and Bases have been prepared electronically. Deletions are shown with a strike-through and insertions are shown double-underlined. This presentation allows the reviewer to readily identify the information that has been deleted and added. To meet format requirements the Index, Technical Specifications and Bases pages will be revised and repaginated as necessary to reflect the changes being proposed by these supplements.

Attachment C contains a revision to Section 5.2.2 of Enclosure 2 of the RSG submittal (Reference 1) that provides the revised SBLOCA analysis. The revised analysis supports the changes proposed to the accumulator volume and the cover pressure, including the new minimum pressure value. The revisions to Section 5.2.2 of Enclosure 2 of the RSG submittal are highlighted for ease of identification.

Attachment D contains a revision to Section 5.2.2 of Enclosure 2 of the EPU submittal (Reference 2) that provides the revised SBLOCA analysis. The revised analysis supports the changes proposed to the accumulator volume and the cover pressure, including the new minimum pressure value. The revisions to Section 5.2.2 of Enclosure 2 of the EPU submittal are highlighted for ease of identification.

TECHNICAL EVALUATION

The primary reason for these LAR supplements is to change the minimum accumulator cover pressure. The change is being made to reflect the revised SBLOCA analysis that does not take credit for RCS loop seal clearing of intact loops and uses non-integer sizes to determine the limiting size break. The revised SBLOCA analysis establishes the minimum cover pressure needed to ensure that the analysis will meet the peak cladding temperature (PCT) and maximum cladding oxidation limits of 10 CFR 50.46.

Section 5.2.1 of Enclosure 2 of the RSG and EPU submittals (References 1 and 2), and the revised Section 5.2.2 of Enclosure 2 of the RSG and EPU submittals, Attachments C and D respectively, provide the technical justification for the proposed change to the accumulator limits on water volume and cover pressure. These changes are consistent with analysis inputs and provide the necessary operating margin at the current power level with the replacement steam generators and at the EPU power level. Changing the LCO statement and Surveillance Requirement 4.5.1.a.1 to address the usable volume is consistent with the analysis volume limits.

RSG SUPPLEMENTAL NO SIGNIFICANT HAZARDS CONSIDERATION

The changes to the replacement steam generator license amendment request (Reference 1) being proposed with this supplement consists of:

1. changing the limits on the accumulator water volume and cover pressure,
2. replacing the word "contained" with "usable" in the volume surveillance requirements, and
3. inserting the word "usable" in the accumulator volume portion of the Limiting Condition for Operation (LCO) statement.

The analysis supporting these changes was conducted at extended power uprate (EPU) conditions of 2900 MWt and thus bounds operation at the current power level with the replacement steam generators.

FirstEnergy Nuclear Operating Company (FENOC) has evaluated whether or not a significant hazards consideration is involved with the proposed amendments by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

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

   Response: No. The proposed supplemental changes will not involve a significant increase in the probability or consequences of an accident previously evaluated.

   A comprehensive analytical effort has been performed to incorporate the proposed supplemental changes into the design and analysis basis for Beaver Valley Power Station (BVPS) Unit No. 1. Analyses and evaluations have been performed for the nuclear steam supply systems (NSSS) and balance of plant (BOP) systems and components, including the nuclear fuel. These comprehensive analytical efforts, which include the proposed changes to the accumulator volume and pressure limits, demonstrate that BVPS Unit No. 1 meets applicable design and licensing requirements. The analyses are conservative and bounding with respect to operation with replacement steam generators at the current power level. The safety and radiological dose consequence analyses confirm that safety analysis and dose consequence analysis acceptance criteria will be satisfied with the proposed supplemental changes.

   The accumulators are not initiators of any design basis accident or event, and therefore the proposed supplemental changes will not increase the probability of any accident previously evaluated. The probability of any evaluated accident or event is independent of the supplemental changes being proposed. The proposed supplemental changes will not adversely affect accident initiators or precursors. They will not alter or prevent the accumulators from performing their intended function within the applicable acceptance limits.

The proposed supplemental changes were evaluated for their effect on accident dose consequences. The updated dose consequence analyses demonstrate compliance with the limits set forth for alternative source term (AST) applications in 10 CFR 50.67, as supplemented by Regulatory Guide 1.183 or 10 CFR 100.

Therefore, none of the proposed changes involve a significant increase in the probability of an accident previously evaluated, and the dose consequences remain within the allowable limits set forth for AST applications in 10 CFR 50.67, as supplemented by Regulatory Guide 1.183 or 10 CFR 100.

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

   Response: No. The proposed supplemental changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

   No new accident scenarios, failure mechanisms or single failures are introduced as a result of the proposed supplemental changes. All systems, structures and components previously required for the mitigation of an event remain capable of fulfilling their intended design function. The proposed supplemental changes will not have an adverse effect on the accumulators and will not challenge their performance or integrity.

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

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

   Response: No. The proposed supplemental changes will not involve a significant reduction in a margin of safety.

   The proposed supplemental changes will not alter the manner in which safety limits, limiting safety system settings or limiting conditions for operation are determined. The proposed supplemental changes to the accumulator Technical Specification limits are being made to provide adequate margin such that BVPS Unit No. 1 can be operated in a safe manner with the replacement steam generators at either the current or EPU power levels. The revised small break loss of coolant accident analysis establishes the minimum cover pressure needed to ensure that the analysis will meet the peak cladding temperature and maximum cladding oxidation limits of 10 CFR 50.46 and thereby does not involve a significant reduction in a margin of safety. These supplemental changes will not adversely impact plant safety because they will not adversely affect the ability of the accumulators to perform their function.

Therefore, the proposed supplemental changes do not involve a significant reduction in a margin of safety.

EPU SUPPLEMENTAL NO SIGNIFICANT HAZARDS CONSIDERATION

The no significant hazard considerations submitted with the EPU LAR (Reference 2) remains valid for the extended power uprate supplement changes.

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

CONCLUSIONS

The impact of all the changes to Technical Specification 3.5.1 for BVPS Unit No. 1 has been evaluated in the no significant hazard considerations provided in this supplement to the RSG LAR. Thus, the changes proposed by this RSG supplement augment the no significant hazard considerations submitted in Reference 1.

The impact of all the changes to Technical Specification 3.5.1 for both BVPS units has been evaluated in the no significant hazard considerations submitted with the EPU LAR. Thus, the change to the minimum cover pressure being proposed by this EPU supplement does not negate or negatively impact the no significant hazard considerations submitted in Reference 2.

ENCLOSURE 1 REFERENCES

1. FENOC Letter L-05-069, License Amendment Request 320, dated April 13, 2005
2. FENOC Letter L-04-125, License Amendment Requests 302 and 173, dated October 4, 2004.

ATTACHMENTS

A-1 Supplement to LAR 320 Technical Specifications (Unit 1)

A-2 Supplement to LAR 173 Technical Specifications (Unit 2)

B-1 Supplement to LAR 320 Technical Specification Bases (Unit 1)

B-2 Supplement to LAR 173 Technical Specification Bases (Unit 2)

C Revised Replacement Steam Generator Submittal (L-05-069) Enclosure 2 Section 5.2.2

D Revised Extended Power Uprate Submittal (L-04-125) Enclosure 2 Section 5.2.2

Attachment A-1

Beaver Valley Power Station, Unit No. 1
Proposed Technical Specification Changes

License Amendment Request No. 320 Supplement

The following is a list of the affected pages:

Page
3/4 5-1

3/4.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3/4.5.1 ACCUMULATORS

LIMITING CONDITION FOR OPERATION

3.5.1 Each reactor coolant system accumulator shall be OPERABLE with:

a. The isolation valve open,

b. Between ~~7664~~6681 and ~~7816~~7645 gallons of usable borated water,

c. Between 2300 and 2600 ppm of boron, and

d. A nitrogen cover-pressure of between ~~605~~611 and ~~661~~685 psig.

APPLICABILITY: MODES 1, 2 and 3.*

ACTION:

a. With one accumulator inoperable due to boron concentration not within limits, restore the inoperable accumulator to OPERABLE status within 72 hours.

b. With one accumulator inoperable for reasons other than Action a, restore the inoperable accumulator to OPERABLE status within 24 hours.

c. With either Action a or b not being completed within the specified completion time, be in at least HOT STANDBY within the next 6 hours and reduce pressurizer pressure to ≤ 1000 psig within 12 hours.

SURVEILLANCE REQUIREMENTS

4.5.1 Each accumulator shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1. Verifying the ~~contained~~usable borated water volume and nitrogen cover-pressure in the tanks are within limits, and

2. Verifying that each accumulator isolation valve is open.

* Pressurizer Pressure above 1000 psig.

Attachment A-2

Beaver Valley Power Station, Unit No. 2
Proposed Technical Specification Changes

License Amendment Request No. 173 Supplement

The following is a list of the affected pages:

Page
3/4 5-1

3/4.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

ACCUMULATORS

LIMITING CONDITION FOR OPERATION

3.5.1 Each Reactor Coolant System accumulator shall be OPERABLE with:

a. The isolation valve open,

b. Between ~~7532~~6898 gallons and ~~7802~~8019 gallons of usable borated water,

c. Between 2300 and 2600 ppm of boron, and

d. A nitrogen cover-pressure of between ~~585~~611 and ~~665~~685 psig.

APPLICABILITY: MODES 1, 2 and 3.*

ACTION:

a. With one accumulator inoperable due to boron concentration not within limits, restore the inoperable accumulator to OPERABLE status within 72 hours.

b. With one accumulator inoperable for reasons other than Action a, restore the inoperable accumulator to OPERABLE status within 24 hours.

c. With either Action a or b not being completed within the specified completion time, be in at least HOT STANDBY within the next 6 hours and reduce pressurizer pressure to ≤ 1000 psig within 12 hours.

SURVEILLANCE REQUIREMENTS

4.5.1 Each accumulator shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1. Verifying the ~~contained~~usable borated water volume and nitrogen cover-pressure in the tanks are within limits, and

2. Verifying that each accumulator isolation valve is open.

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*Pressurizer Pressure above 1000 psig.

Attachment B-1

Beaver Valley Power Station, Unit No. 1

Proposed Technical Specification Bases Changes

License Amendment Request No. 320 Supplement

The following is a list of the affected pages:

Page
B 3/4 5-1

BASES

3/4.5.1 ACCUMULATORS

The OPERABILITY of each of the RCS accumulators ensures that a sufficient volume of borated water will be immediately forced into the reactor core through each of the cold legs in the event the RCS pressure falls below the pressure of the accumulators. This initial surge of water into the core provides the initial cooling mechanism during large RCS pipe ruptures.

The limits on accumulator volume, boron concentration and pressure ensure that the assumptions used for accumulator injection in the accident analysis are met. The specified Technical Specification values for usable accumulator volume, boron concentration and minimum pressure are analysis values. The Technical Specification maximum pressure is below the maximum analysis pressure so that the relief valve is not challenged at the maximum Technical Specification pressure. The maximum nitrogen cover pressure limit prevents accumulator relief valve actuation and ultimately preserves accumulator integrity. The small break LOCA analysis is performed at the minimum nitrogen cover pressure, since sensitivity analyses have demonstrated that higher nitrogen cover pressure results in a computed peak cladding temperature benefit. The Technical Specification values for pressure and volume do not account for instrumentation uncertainty.

If the boron concentration of one accumulator is not within limits (Action a), it must be returned to within the limits within 72 hours. In this condition, ability to maintain subcriticality or minimum boron precipitation time may be reduced. The boron in the accumulators contributes to the assumption that the combined ECCS water in the partially recovered core during the early reflooding phase of a large break LOCA is sufficient to keep that portion of the core subcritical. One accumulator below the minimum boron concentration limit, however, will have no effect on available ECCS water and an insignificant effect on core subcriticality during reflood. Boiling of ECCS water in the core during reflood concentrates boron in the saturated liquid that remains in the core. In addition, current analysis techniques demonstrate that the accumulators do not discharge following a large main steam line break. Thus, 72 hours is allowed to return the boron concentration to within limits.

If one accumulator is inoperable for a reason other than boron concentration (Action b), it must be returned to OPERABLE status within 24 hours. In this condition the required contents of two accumulators cannot be assumed to reach the core during a LOCA. Due to the severity of the consequences should a LOCA occur under these conditions, the 24 hour completion time to open the valve, remove power to the valve, or restore the proper water volume or nitrogen cover pressure ensures that prompt action will be taken to return the inoperable accumulator to OPERABLE status. The completion time minimizes the potential for exposure of the plant to a LOCA under these conditions. The 24 hours allowed to restore an inoperable accumulator to OPERABLE status is justified by WCAP-15049-A, Revision 1, "Risk-Informed Evaluation of an Extension to Accumulator Completion Times", dated April 1999.

If the accumulator cannot be returned to OPERABLE status within the associated completion time (Action c), 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 HOT STANDBY within 6 hours and the reactor coolant system pressure reduced to ≤ 1000 psig within 12 hours. The

Attachment B-2

Beaver Valley Power Station, Unit No. 2

Proposed Technical Specification Bases Changes

License Amendment Request No. 173 Supplement

The following is a list of the affected pages:

Page
B 3/4 5-1

3/4.5.1 ACCUMULATORS

The OPERABILITY of each of the RCS accumulators ensures that a sufficient volume of borated water will be immediately forced into the reactor core through each of the cold legs in the event the RCS pressure falls below the pressure of the accumulators. This initial surge of water into the core provides the initial cooling mechanism during large RCS pipe ruptures.

The limits on accumulator volume, boron concentration and pressure ensure that the assumptions used for accumulator injection in the accident analysis are met. The specified Technical Specification values for usable accumulator volume, boron concentration and minimum pressure are analysis values. The Technical Specification maximum pressure is below the maximum analysis pressure so that the relief valve is not challenged at the maximum Technical Specification pressure. The maximum nitrogen cover pressure limit prevents accumulator relief valve actuation and ultimately preserves accumulator integrity. The small break LOCA analysis is performed at the minimum nitrogen cover pressure, since sensitivity analyses have demonstrated that higher nitrogen cover pressure results in a computed peak cladding temperature benefit. The Technical Specification values for pressure and volume do not account for instrumentation uncertainty.

If the boron concentration of one accumulator is not within limits (Action a), it must be returned to within the limits within 72 hours. In this condition, ability to maintain subcriticality or minimum boron precipitation time may be reduced. The boron in the accumulators contributes to the assumption that the combined ECCS water in the partially recovered core during the early reflooding phase of a large break LOCA is sufficient to keep that portion of the core subcritical. One accumulator below the minimum boron concentration limit, however, will have no effect on available ECCS water and an insignificant effect on core subcriticality during reflood. Boiling of ECCS water in the core during reflood concentrates boron in the saturated liquid that remains in the core. In addition, current analysis techniques demonstrate that the accumulators do not discharge following a large main steam line break. Thus, 72 hours is allowed to return the boron concentration to within limits.

If one accumulator is inoperable for a reason other than boron concentration (Action b), it must be returned to OPERABLE status within 24 hours. In this condition the required contents of two accumulators cannot be assumed to reach the core during a LOCA. Due to the severity of the consequences should a LOCA occur under these conditions, the 24 hour completion time to open the valve, remove power to the valve, or restore the proper water volume or nitrogen cover pressure ensures that prompt actions will be taken to return the inoperable accumulator to OPERABLE status. The completion time minimizes the potential for exposure of the plant to a LOCA under these conditions. The 24 hours allowed to restore an inoperable accumulator to OPERABLE status is justified by WCAP-15049-A, Revision

1, "Risk-Informed Evaluation of an Extension to Accumulator Completion Times", dated April 1999.

If the accumulator cannot be returned to OPERABLE status within the associated completion time (Action c), 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 HOT STANDBY within 6 hours and the reactor coolant system pressure reduced to ≤ 1000 psig within 12 hours. The

Attachment C

Revised Replacement Steam Generator Submittal (L-05-069) Enclosure 2 Section 5.2.2

5

5.1

5.2

5.2.1

5.2.2 Small Break LOCA

5.2.2.1 Introduction

This section contains information regarding the Small Break Loss-of-Coolant Accident (SBLOCA) analysis performed in support of the EPU for BVPS-1 (with Model 54F replacement steam generators) at the NSSS power level of 2910 MWt (2900 MWt reactor power). The purpose of analyzing the Small Break LOCA is to demonstrate conformance with the 10 CFR 50.46 (Reference 1) requirements for the conditions associated with the EPU. Important input assumptions, as well as analytical models and analysis methodology for the Small Break LOCA are contained in subsequent sections. Analysis results are provided in the form of tables and figures, as well as a more detailed description of the limiting transient. The analysis has shown that no design or regulatory limit related to the Small Break LOCA would be exceeded due to the EPU power and associated plant parameters.

5.2.2.2 Input Parameters and Assumptions

The important plant conditions and features for BVPS-1 are listed in Table 5.2.2-1A. Several additional considerations that are not identified in Table 5.2.2-1A are discussed below.

Figure 5.2.2-1 depicts the hot rod axial power shape modeled in the Small Break LOCA analysis. This shape was chosen because it represents a distribution with power concentrated in the upper regions of the core (the axial offset is +13%). Such a distribution is limiting for Small Break LOCA since it minimizes coolant swell while maximizing vapor superheating and fuel rod heat generation at the uncovered elevations. The chosen power shape has been conservatively scaled to a standard 2-line segment K(Z) envelope for BVPS-1 based on the peaking factors shown in Table 5.2.2-1A.

Figures 5.2.2-2 and 5.2.2-3 provide the SI flow versus pressure curves modeled in the Small Break LOCA analysis. Figure 5.2.2-2 shows the flows from one High Head Safety Injection (HHSI) pump, where the faulted loop injects to RCS pressure. Figure 5.2.2-3 shows flows from one HHSI pump and one Low Head Safety Injection (LHSI) pump, where the faulted loop injects into containment.

5.2.2.3 Description of Analyses and Evaluations

Analytical Model

The requirements for an acceptable ECCS evaluation model are presented in Appendix K of 10 CFR 50. For LOCAs due to Small Breaks, less than 1 square foot in area, the Westinghouse NOTRUMP Small Break LOCA Emergency Core Cooling System (ECCS) Evaluation Model (References 2, 3, and 4) is used. The Westinghouse NOTRUMP Small Break LOCA ECCS Evaluation Model was developed to determine the RCS response to design basis Small Break LOCAs, and to address NRC concerns expressed in NUREG-0611 (Reference 5).

The Westinghouse Small Break LOCA ECCS Evaluation Model consists of the NOTRUMP and LOCTA-IV computer codes. The NOTRUMP code is employed to calculate the transient depressurization of the Reactor Coolant System (RCS), as well as to describe the mass and energy release of the fluid flow through the break. Among the features of the NOTRUMP code are: calculation of thermal non-equilibrium in all fluid volumes, flow regime-dependent drift flux calculations with counter-current flooding limitations, mixture level tracking logic in multiple-stacked fluid nodes, regime-dependent drift flux calculations in multiple-stacked fluid nodes and regime-dependent heat transfer correlations. These features provide NOTRUMP with the capability to accurately calculate the mass and energy distribution throughout the RCS during the course of a Small Break LOCA.

The RCS model is nodalized into volumes interconnected by flow paths. The broken loop is modeled explicitly, while the intact loops are lumped together into a second loop. Transient behavior of the system is determined from the governing conservation equations of mass, energy, and momentum. The multi-node capability of the program enables explicit, detailed spatial representation of various system components which, among other capabilities, enables a calculation of the behavior of the loop seal during a Small Break LOCA. The reactor core is represented as heated control volumes with associated phase separation models to permit transient mixture height calculations.

Fuel cladding thermal analyses are performed with a version of the LOCTA-IV code (Reference 2) using the NOTRUMP calculated core pressure, fuel rod power history, uncovered core steam flow and mixture heights as boundary conditions. The LOCTA-IV code models the hot rod and the average hot assembly rod, assuming a conservative power distribution that is skewed to the top of the core. Figure 5.2.2-4 illustrates the code interface for the Small Break Model.

Analysis

The EPU Small Break LOCA analysis considered nine different break cases for BVPS-1 as indicated by the results in Table 5.2.2-6A. A break spectrum of 1.5-, 2-, 2.25-, 2.5-, 2.75-, 3-, 3.25-, 4- and 6-inch breaks was considered. For BVPS-1, the 2.75-inch break was found to be limiting for PCT and the 2.5-inch break was found to be limiting for oxidation. For BVPS-1, the 1.5-inch case was found to be non-limiting in NOTRUMP and therefore PCT information was not calculated.

The most limiting single active failure used for a Small Break LOCA is that of an emergency power train failure which results in the loss of one complete train of ECCS components. In addition, a Loss-of-Offsite Power (LOOP) is postulated to occur coincident with reactor trip. This means that credit may be taken for at most one high head safety injection (HHSI) pump. In the analysis for BVPS-1, one HHSI pump is modeled. The Small Break LOCA analysis performed for both units models the ECCS flow as being delivered to both the intact and broken loops at the RCS backpressure for breaks smaller than the cold leg HHSI nozzle (1.5-inch - 4-inch breaks) and at containment pressure for breaks greater than the cold leg HHSI nozzle (6-inch breaks). These SI flows are illustrated in Figure 5.2.2-2 and 5.2.2-3 for each scenario. Note that for the 6-inch breaks, no SI is assumed in the faulted loop because the break is postulated along the SI line. The LOOP and the failure of a diesel generator to start as the limiting single failure for Small Break LOCA is part of the NRC approved methodology and does not change as a result of the EPU conditions. The single failure assumption is extremely limiting due to the fact that one train of SI, one motor driven auxiliary feedwater (AFW) pump, and power to the reactor coolant pumps (RCPs) are all modeled to be lost. Any other active single failure would not result in a more limiting scenario since increased SI flow would improve the overall transient results.

Prior to break initiation, the plant is in a full power (100.6%) equilibrium condition, i.e., the heat generated in the core is being removed via the secondary system. Other initial plant conditions used in the analysis are given in Table 5.2.2-1A. Subsequent to the break opening, a period of reactor coolant system blowdown ensues in which the heat from fission product decay, the hot reactor internals, and the reactor vessel continues to be transferred to the RCS fluid. The heat transfer between the RCS and the secondary system may be in either direction and is a function of the relative temperatures of the primary and secondary conditions. In the case of continuous heat addition to the secondary during a period of quasi-equilibrium, an increase in the secondary system pressure results in steam relief via the steam generator safety valves.

When a Small Break LOCA occurs, depressurization of the RCS causes fluid to flow into the loops from the pressurizer resulting in a pressure and level decrease in the pressurizer. The reactor trip signal subsequently occurs when the pressurizer low-pressure reactor trip setpoint, conservatively modeled as 1935 psia, is reached. LOOP is postulated to occur coincident with reactor trip. A safety injection signal is generated when the pressurizer low-pressure safety injection setpoint, conservatively modeled as 1745 psia for BVPS-1, is reached. Safety injection flow is delayed 27 seconds after the occurrence of the low-pressure condition. This delay accounts for signal processing, diesel generator start up and emergency power bus loading consistent with the loss-of-offsite power coincident with reactor trip, as well as the pump acceleration and valve delays.

The following countermeasures limit the consequences of the accident in two ways:

1. Reactor trip and borated water injection supplement void formation in causing a rapid reduction of nuclear power to a residual level corresponding to the delayed fission and fission product decay. No credit is taken in the Small Break LOCA analysis for the boron content of the injection water. In addition, credit is taken in the Small Break LOCA analysis for the insertion of Rod Cluster Control Assemblies (RCCAs) subsequent to the reactor trip signal, considering the most reactive RCCA is stuck in the full out position. A rod drop time of 2.7 seconds was used while also considering an additional 2 seconds for the signal processing delay time. Therefore, a total delay time of 4.7 seconds from the time of reactor trip signal to full rod insertion was used in the Small Break LOCA analysis.

2. Injection of borated water provides sufficient flooding of the core to prevent excessive cladding temperatures.

During the earlier part of the Small Break transient (prior to the postulated loss-of-offsite power coincident with reactor trip), the loss of flow through the break is not sufficient to overcome the positive core flow maintained by the reactor coolant pumps. During this period, upward flow through the core is maintained. However, following the reactor coolant pump trip (due to a LOOP) and subsequent pump coastdown, a period of core uncovery occurs. Ultimately, the Small Break transient analysis is terminated when the top of the core is recovered or the core mixing level is increasing, and ECCS flow provided to the RCS exceeds the break flow rate.

The core heat transfer mechanisms associated with the Small Break transient include the break itself, the injected ECCS water, and the heat transferred from the RCS to the steam generator secondary side. Main Feedwater (MFW) is conservatively isolated in 10 seconds for BVPS-1 (consisting of a 3 second signal delay time and a 7 second main feedwater isolation valve stroke time) following the generation of the pressurizer low-pressure SI signal. Additional makeup water is also provided to the secondary using the auxiliary feedwater (AFW) system. An AFW actuation signal is derived from the pressurizer low-pressure SI signal, resulting in the delivery of AFW system flow 60 seconds after the generation of the SI signal. The heat transferred to the secondary side of the steam generator aids in the reduction of the RCS pressure.

Should the RCS depressurize to approximately 625 psia (accumulator minimum pressure), the cold leg accumulators begin to inject borated water into the reactor coolant loops.

5.2.2.4 Acceptance Criteria and Results

The acceptance criteria for the LOCA are described in 10 CFR 50.46 (Reference 1) as follows:

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

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

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

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

5. 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.

Criteria 1 through 3 are explicitly covered by the Small Break LOCA analysis at EPU conditions.

For criterion 4, the appropriate core geometry was modeled in the analysis. The results based on this geometry satisfy the Peak Clad Temperature (PCT) criterion of 10 CFR 50.46 and consequently, demonstrate that the core remains amenable to cooling.

For criterion 5, Long-Term Core Cooling (LTCC) considerations are not directly applicable to the Small Break LOCA transient analysis addressed in this section, but are assessed by Sections 5.2.3 and 5.2.4 as part of the evaluation of ECCS performance.

The acceptance criteria were established to provide a significant margin in ECCS performance following a LOCA.

In order to determine the conditions that produced the most limiting Small Break LOCA case (as determined by the highest calculated peak cladding temperature), nine break cases were examined for BVPS-1. These cases were investigated to capture the most severe postulated Small Break LOCA event. The following discussions provide insight into the analyzed conditions.

Limiting Temperature Conditions

The RCS temperature analyzed was based on a nominal vessel average temperature of 580.0°F. However, the analysis is applicable over the range of 566.2 – 580.0°F. The analysis supports a ±4°F T_{avg} uncertainty. The analysis showed that the 2.75-inch break case is the limiting PCT case for BVPS-1. The limiting case transient is discussed below.

Limiting Break Case

The results of Reference 6 demonstrate that the cold leg break location is limiting with respect to postulated cold leg, hot leg and pump suction leg break locations. The PCT results are shown in Tables 5.2.2-4A and 5.2.2-5A. Inherent in the Small Break analysis are several input parameters (see Section 5.2.2.2 and Table 5.2.2-1A), while Table 5.2.2-6A provides the key transient event times.

For the EPU Small Break LOCA analysis, the limiting PCT case for BVPS-1 was the 2.75-inch break case. A summary of the transient response for the limiting PCT case is shown in Figures 5.2.2-5A through Figure 5.2.2-15A. These figures present the response of the following parameters.

- RCS Pressure
- Core Mixture Level
- Core Exit Vapor Temperature
- Broken Loop and Intact Loop Secondary Pressure
- Break Vapor Flow Rate
- Break Liquid Flow Rate
- Broken Loop and Intact Loop Accumulator Flow Rate
- Broken Loop and Intact Loop Pumped Safety Injection Flow Rate
- Peak Clad Temperature
- Hot Spot Fluid Temperature
- Rod Film Heat Transfer Coefficient

Upon initiation of the limiting 2.75-inch break for BVPS-1, there is an initial rapid depressurization of the RCS followed by an intermediate equilibrium at approximately 1150 psia (see Figure 5.2.2-5A). The limiting 2.75-inch break depressurizes to the accumulator injection setpoint of 625 psia at approximately 1438 seconds for BVPS-1 (see Figure 5.2.2-11A). During the initial period of the Small Break transient, the effect of the break flow rate is not sufficient to overcome the flow rate maintained by the reactor coolant pumps as they coast down. As such, normal upward flow is maintained through the core and core heat is adequately removed. Following reactor trip, the removal of the heat generated as a result of fission products decay is accomplished via a two-phase mixture level covering the core. The core mixture level and peak clad temperature transient plots for the limiting break calculations are illustrated in Figures 5.2.2-6A and 5.2.2-13A, respectively. These figures show that the peak clad temperature occurs near the time when the core is most deeply uncovered and the top of the core is being cooled by steam. This time is characterized by the highest vapor superheating above the mixture level (refer to Figure 5.2.2-7A). For BVPS-1 the limiting PCT time-in-life was determined to be 8,000 MWD/MTU.

A comparison of the flow provided by the safety injection system to the intact and broken loops can be found in Figure 5.2.2-12A. The cold leg break vapor and liquid mass flow rates are provided in Figures 5.2.2-9A and 5.2.2-10A, respectively. Figures 5.2.2-14A and 5.2.2-15A provide additional information on the fluid temperature at the hot spot and hot rod surface heat transfer coefficient at the hot spot, respectively. Figure 5.2.2-8A depicts the secondary side pressure for both the intact and broken loops for the limiting PCT break case.

Total Oxidation

For the EPU Small Break LOCA analysis, the maximum local oxidation case for BVPS-1 was the 2.5-inch break case. The maximum local transient oxidation is 11.07% for BVPS-1 at 20,000 MWD/MTU. The limiting transient oxidation occurs at the burst elevation and includes both outside and post-rupture inside oxidation. Pre-existing (pre-transient) oxidation was also considered and the sum of the pre-transient and transient oxidation remains below 17% at all times in life, for all fuel resident in the core.

Additional Break Cases

Studies documented in Reference 6 have determined that the limiting PCT Small Break transient occurs for breaks of less than 10-inches in diameter in the cold leg. For BVPS-1, the limiting PCT is captured by the 1.5- 2-, 2.25-, 2.5-, 2.75-, 3-, 3.25-, 4- and 6-inch break spectrum. The beginning-of-life (BOL) results of these break spectrum cases are given in Table 5.2.2-4A. Figures 5.2.2-16A through 5.2.2-36A address the non-limiting BOL cases (2-, 2.25-, 2.5-, 3-, 3.25- 4- and 6-inch) analyzed for BVPS-1. The 1.5-inch case for BVPS-1 produced only minimal core uncovery and therefore PCT information was not calculated. Note that plots for the 1.5-inch case are not included here. The plots for each of the additional non-limiting break cases include:

1. RCS Pressure
2. Core Mixture Level
3. Peak Clad Temperature

For BVPS-1, the PCTs for each of the additional breaks considered are shown in Table 5.2.2-4A and are less than the limiting 2.75-inch break case. The PCT was not calculated for the 1.5-inch case due to the minimal core uncovery.

Transient Termination

The 10 CFR 50.46 criteria continue to be satisfied beyond the end of the calculated transient due to the presence of some or all of the following conditions:

1. The RCS pressure is gradually decreasing.
2. The net mass inventory is increasing.
3. The core mixture level is recovered, or recovering due to increasing mass inventory.
4. As the RCS inventory continues to gradually increase, the core mixture level will continue to increase and the fuel cladding temperatures will continue to decline indicating that the temperature excursion is terminated.

ZIRLO/Zirc-4 Cladding

At the time at which this analysis is implemented, no new Zirc-4 fuel is expected to be inserted into the core. All of the Zirc-4 fuel will be burned for at least one cycle, and ZIRLO™ fuel will be implemented at non-EPU conditions at least one reload cycle before the EPU is implemented. Therefore, the ZIRLO™ fuel is considered limiting with a PCT of 1895.0°F at 8,000 MWD/MTU burnup for BVPS-1. The fuel temperatures/pressures used in these calculations were based on NRC approved fuel performance code PAD 4.0 (Reference 7) which addresses all the helium release related issues. This analysis has been performed using the most limiting temperature/pressure as calculated for 17x17 non-IFBA RFA fuel. The non-IFBA fuel bounds IFBA fuel for Small Break LOCA analyses. Note that the effect of annular pellets, which are only present in the IFBA fuel, was considered in the analysis.

5.2.2.5 Conclusions

The Small Break LOCA analysis considered a break spectrum of 1.5-, 2-, 2.25-, 2.5-, 2.75-, 3-, 3.25-, 4-, and 6-inch diameters for BVPS-1. For BVPS-1, a peak cladding temperature of 1895°F was calculated at the limiting time-in-life of 8,500 MWD/MTU for the 2.75-inch case and a maximum transient oxidation of 11.07% was calculated at the limiting time-in-life of 20,000 MWD/MTU for the 2.5-inch case.

The analysis presented in this section shows that the accumulator and safety injection subsystems of the Emergency Core Cooling System, together with the heat removal capability of the steam generator, provide sufficient core heat removal capability to maintain the calculated peak cladding temperature for Small Break LOCA below the required limit of 10 CFR 50.46. Furthermore, the analysis shows that the local cladding oxidation and core wide average oxidation, including consideration of pre-existing and post-LOCA oxidation, and cladding outside and post-rupture inside oxidation, are less than the 10 CFR 50.46 (Reference 1) limits.

Table 5.2.2-7 provides a results summary for the BVPS-1 SBLOCA EPU analysis. Results include PCT, maximum local oxidation and total hydrogen generation.

The results and conclusions of the analysis performed for Small Break LOCA for the reactor power of 2900 MWt (2910 MWt NSSS power) bound and support operation at the current reactor power of 2689 MWt (2697 MWt NSSS power), thus supporting the staged implementation of EPU at BVPS-1.

5.2.2.6 References

1. "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors," 10 CFR 50.46 and Appendix K of 10 CFR 50, Federal Register, Volume 39, Number 3, January 1974, as amended in Federal Register, Volume 53, September 1988.

2. Meyer, P. E., "NOTRUMP - A Nodal Transient Small Break and General Network Code," WCAP-10079-P-A, (proprietary) and WCAP-10080-NP-A (non-proprietary), August 1985.

3. Lee, N. et al., "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," WCAP-10054-P-A (proprietary) and WCAP-10081-NP-A (non-proprietary), August 1985.

4. Thompson, C. D. et al., "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, Rev. 1 (proprietary), July 1997.

5. "Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in Westinghouse - Designed Operating Plant," NUREG-0611, January 1980.

6. Rupprecht, S. D. et al., "Westinghouse Small Break LOCA ECCS Evaluation Model Generic Study with the NOTRUMP Code," WCAP-11145-P-A (proprietary), October 1986.

7. Slagle, W. H., (ed.) et al., "Westinghouse Improved Performance Analysis and Design Model (PAD 4.0)," WCAP-15063-P-A, Revision 1, July 2000.

Table 5.2.2-1A BVPS-1 Input Parameters Used in the Small Break LOCA Analysis	
Input Parameter	**Value**
Core Rated Thermal Power-100%	2900
Calorimetric Uncertainty, %	0.6
Fuel Type	17 X 17 Robust Fuel Assembly (RFA)
Total Core Peaking Factor, F_Q	2.40
Hot Channel Enthalpy Rise Factor, $F_{\Delta H}$	1.62
Hot Assembly Average Power Factor, P_{HA}	1.42
Maximum Axial Offset, %	+13
Initial RCS Loop Flow, gpm/loop	82,840
Initial Vessel T_{avg}, °F	Max: 580.0 Min: 566.2
Initial Pressurizer Pressure (plus uncertainties), psia	2300
Reactor Coolant Pump Type	Model 93A with Weir
Pressurizer Low-Pressure Reactor Trip Setpoint, psia	1935
Reactor Trip Signal Delay Time, seconds	2.0
Rod Drop Delay Time, seconds	2.7
Auxiliary Feedwater Temperature (Maximum), °F	120
Number of AFW Pumps Available Following a LOOP	1 Motor Driven
AFW Flow (Minimum) to all 3 Steam Generators, gpm	294 (98 gpm/SG * 3) at 1107 psig
AFW Flow Delay Time (Maximum), seconds	60
AFW Actuation Signal	Pressurizer Low-Pressure Safety Injection
Isolation of Steam Line Signal	Pressurizer Low-Pressure Reactor Trip/LOOP
Steam Generator Type	Model 54F
Maximum AFW Piping Purge Volume, ft^3	168
Steam Generator Tube Plugging (Maximum), %	10
Maximum MFW Isolation Signal Delay Time, seconds	3
MFW Control Valve Isolation Ramp Time, seconds	7
MFW Isolation Signal	Pressurizer Low-Pressure Safety Injection
Steam Generator Secondary Water Mass, lbm/SG	99,930
Containment Spray Flowrate for 2 Pumps, gpm	4983 (plus 981 gpm to account for flow to the sump)
RWST Deliverable Volume (Minimum), gallons	317,000

Table 5.2.2-1A (continued) BVPS-1 Input Parameters Used in the Small Break LOCA Analysis	
Input Parameter	**Value**
SI Temp at Cold Leg Recirculation Time (Maximum), °F	190
ECCS Configuration	1 HHSI pump, faulted line injects to RCS pressure (1.5-inch – 4-inch breaks) 1 HHSI pump, 1 LHSI pump, no ECCS in the faulted loop (6-inch break)
ECCS Water Temperature (Maximum), °F	65
Pressurizer Low-Pressure Safety Injection Setpoint, psia	1745
SI Flow Delay Time, seconds	27
ECCS Flow vs. Pressure	See Tables 5.2.2-2 and 5.2.2-3
Initial Accumulator Water/Gas Temperature, °F	105
Initial Nominal Accumulator Water Volume, ft^3	957
Minimum Accumulator Pressure, psia	625

Table 5.2.2-2 Safety Injection Flows Used in the Small Break LOCA Analysis (1 HHSI pump, faulted loop injects to RCS pressure – 1.5-inch - 4-inch breaks for BVPS-1)		
RCS Pressure (psia)	**Intact Loop (lbm/sec)**	**Broken Loop (lbm/sec)**
314.7	37.59	20.28
414.7	36.63	19.79
514.7	35.56	19.17
614.7	34.45	18.61
714.7	33.42	18.06
814.7	32.34	17.50
914.7	31.25	16.88
1014.7	30.14	16.25
1114.7	29.03	15.70
1214.7	27.92	15.07
1314.7	26.67	14.45
1414.7	25.28	13.61
1514.7	23.85	12.92
1614.7	22.43	12.08
1714.7	20.97	11.39
1814.7	19.50	10.56

Table 5.2.2-3
Safety Injection Flows Used in the Small Break LOCA Analysis
(1 HHSI pump, 1 LHSI pump, no ECCS in the faulted loop because the break is postulated along the HHSI line – 6-inch breaks for BVPS-1)

RCS Pressure (psia)	BVPS-1 Intact Loop (lbm/sec)
14.7	336.34
24.7	313.32
34.7	290.45
64.7	216.34
104.7	-
109.7	-
114.7	45.12
119.7	34.73
164.7	-
214.7	34.73
314.7	32.43
414.7	30.14
514.7	27.78
614.7	25.42
714.7	22.92
814.7	20.42
914.7	17.78
1014.7	15.00
1114.7	12.22
1214.7	9.31
1314.7	6.39
1414.7	3.06
1514.7	0.0

Table 5.2.2-4A BVPS-1 SBLOCTA BOL Results								
Break Size (in)	2	2.25	2.5	2.75	3	3.25	4	6
PCT (°F)	1723.1	1804.8	1793.2	1839.2	1777.6	1617.0	1334.2	1267.9
PCT Time (s)	3160.3	2417.0	2209.8	1743.7	1386.4	1215.8	780.2	2209.2
PCT Elevation (ft)	12	12	12	12	11.75	11.5	11.25	11.5
Max. Local ZrO_2 (%)	3.15	3.6	5.04	4.28	2.87	1.19	0.14	0.13
Max. Local ZrO_2 Elev. (ft)	12	12	12	12	11.75	11.5	11.25	11.5
Core-Wide Avg. ZrO_2 (%)	0.41	0.48	0.64	0.56	0.4	0.18	0.02	0.02

Table 5.2.2-5A BVPS-1 SBLOCTA Limiting Results from the 2.5-inch (Transient Oxidation) and 2.75-inch (PCT) Time-in-Life Study		
Break Size	2.5	2.75
Time-in-Life (MWD/MTU)	20,000	8,000
PCT (°F)	1796	1895.0
PCT Time (s)	2212.9	1723.7
PCT Elevation (ft)	12	12
Hot Rod Burst Time (s)	1770.2	1721.7
Hot Rod Burst Elevation (ft)	11.75	12
Max. Local Transient ZrO_2 (%)	11.07	8.82
Max. Local Transient ZrO_2 Elev. (ft)	11.75	12
Core-Wide Avg. ZrO_2 (%)	0.62	0.52

Table 5.2.2-6A
BVPS-1 NOTRUMP Results

Event Time (sec)	1.5-inch	2-inch	2.25-inch	2.5-inch	2.75-inch	3-inch	3.25-inch	4-inch	6-inch
Break Initiation	0	0	0	0	0	0	0	0	0
Reactor Trip Signal	54.1	29.0	22.5	17.9	14.6	12.3	10.6	7.3	4.4
S-Signal	75.9	42.4	33.9	27.9	23.8	20.8	18.6	14.4	10.2
SI Flow Delivered	102.9	69.4	60.9	54.9	50.8	47.8	45.6	41.4	37.2
Loop Seal Clearing[(1)]	1840	930	725	656	484	414	358	241	62
Core Uncovery	(3)	1020	813	658	672	526	362	233	121
Accumulator Injection	N/A	4017	2378	1821	1438	1138	996	637	291
RWST Volume Delivered	3033	3025	3017	3011	3006	3001	2998	2992	N/A
PCT Time (BOL)	--	3160.3	2417.0	2209.8	1743.7	1386.4	1215.8	780.2	2209.2
Core Recovery	N/A	(2)	(2)	(2)	(2)	(2)	(2)	(2)	(2)

Notes:

(1) Loop seal clearing is defined as break vapor flow > 1 lb/s.

(2) For the cases where core recovery is > TMAX, basis for transient termination can be concluded based on some or all of the following: (1) The RCS system pressure is decreasing which will increase SI flow, (2) Total RCS system mass is increasing due to SI flow exceeding break flow, and (3) Core mixture level has begun to increase and is expected to continue for the remainder of the accident.

(3) It has been judged that no core uncovery of any consequence will take place and the 1.5-inch case is non-limiting. Therefore no PCT calculations were performed.

Table 5.2.2-7 BVPS-1 SBLOCA Results Summary	
Peak Cladding Temperature (°F)	1895
Maximum Local Transient Oxidation (%)	11.07
Total Hydrogen Generation (%)	≤1%

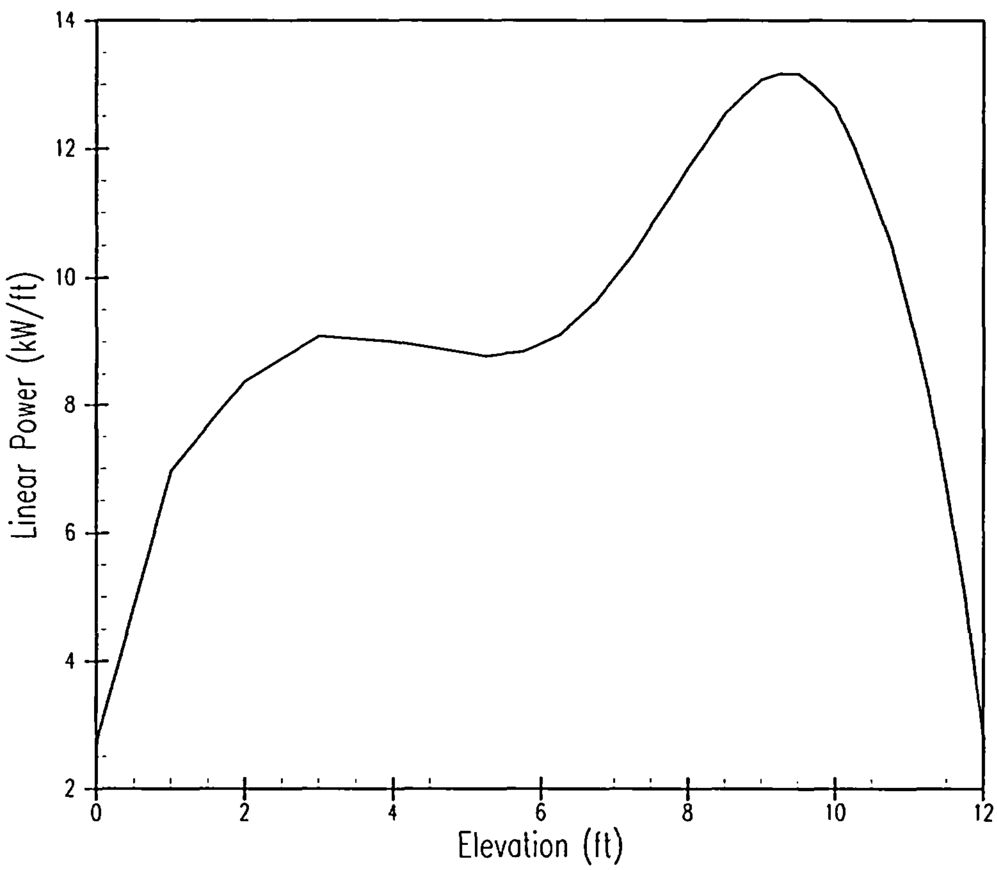

Figure 5.2.2-1
Small Break Hot Rod Power Shape

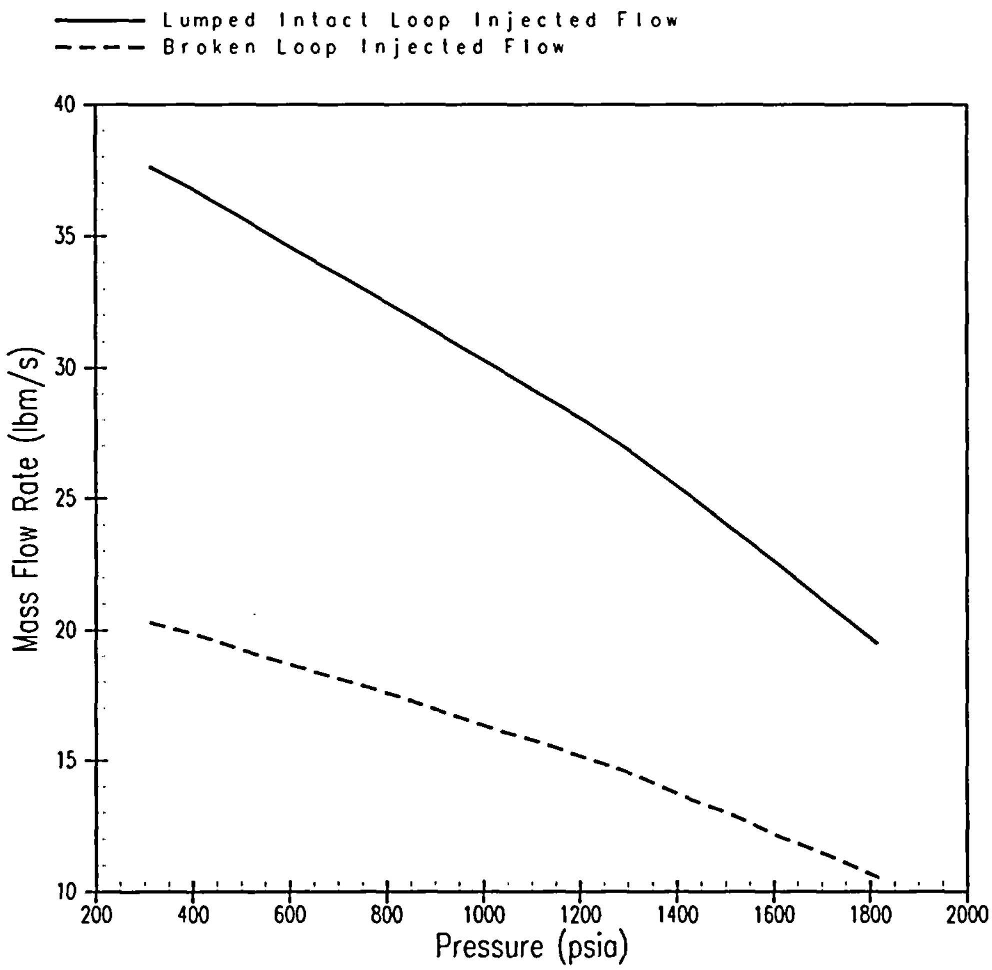

Figure 5.2.2-2
Small Break LOCA Safety Injection Flows
(1 HHSI pump, faulted loop injects to RCS pressure – 1.5-inch - 4-inch breaks for BVPS-1)

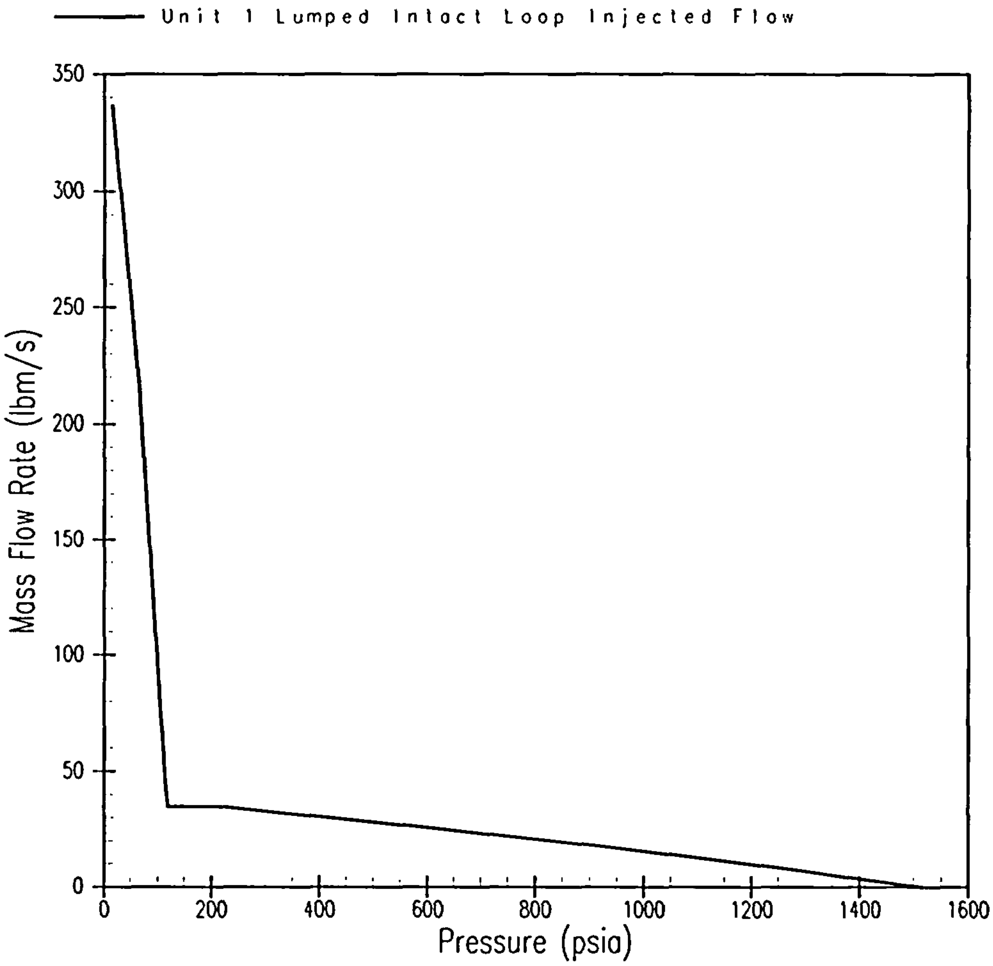

Figure 5.2.2-3
Small Break LOCA Safety Injection Flows
(1 HHSI pump, 1 LHSI pump, no ECCS in the faulted loop because the break is postulated along the HHSI line – 6-inch break for BVPS-1)

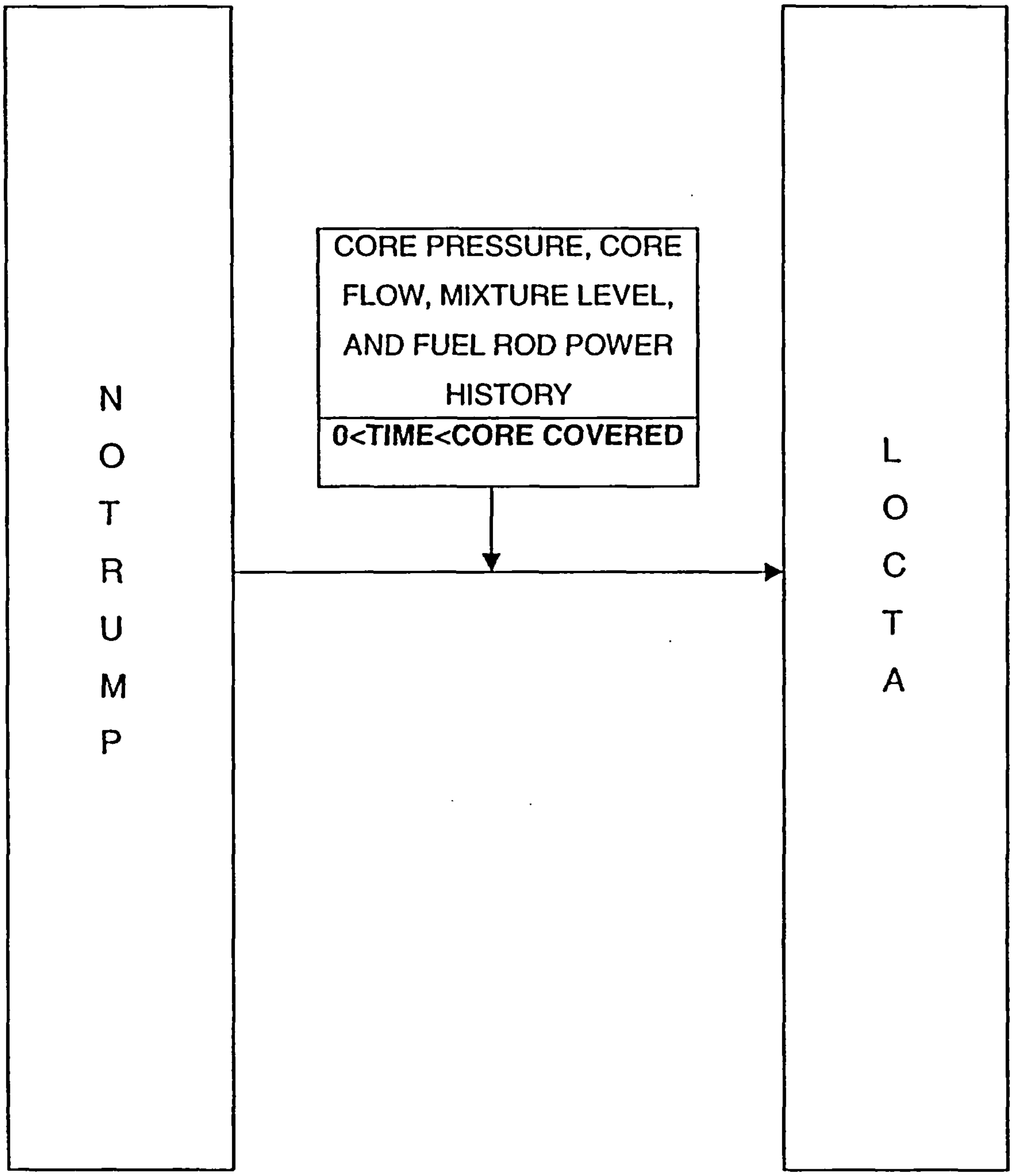

Figure 5.2.2-4
Code Interface Description
for Small Break Model

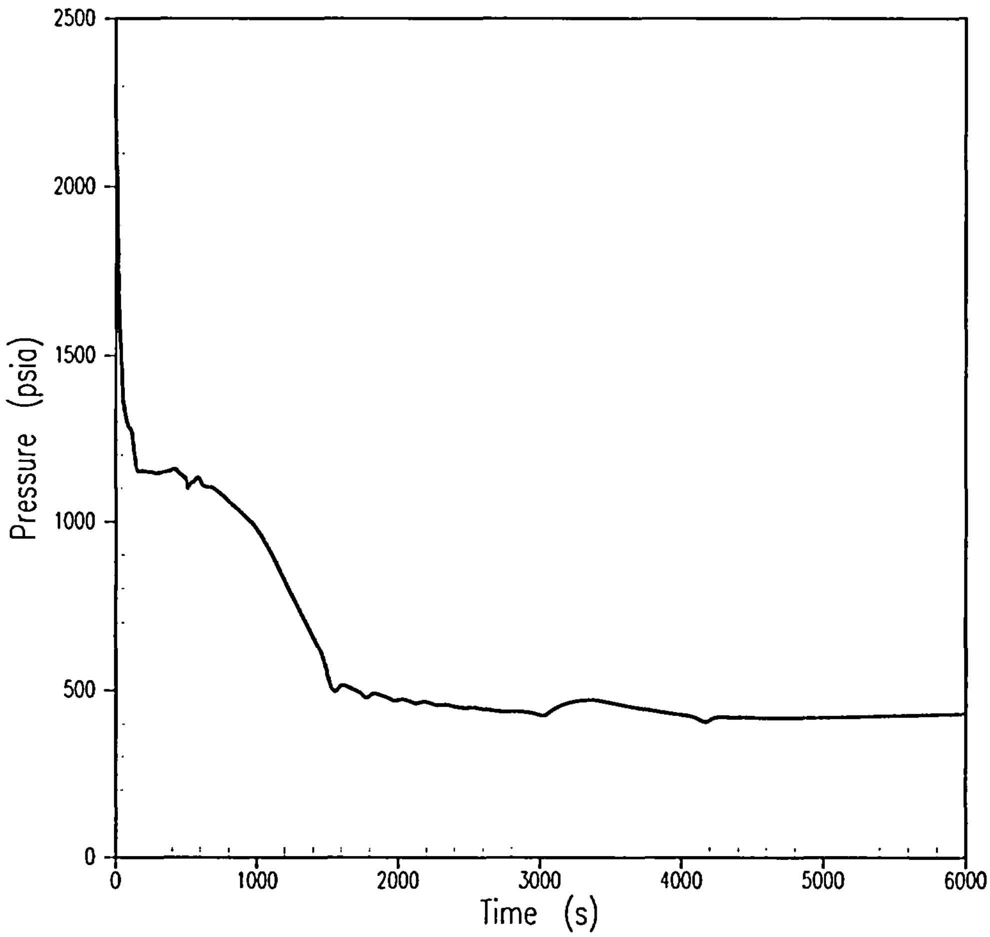

Figure 5.2.2-5A
BVPS-1 2.75-inch Break
RCS Pressure

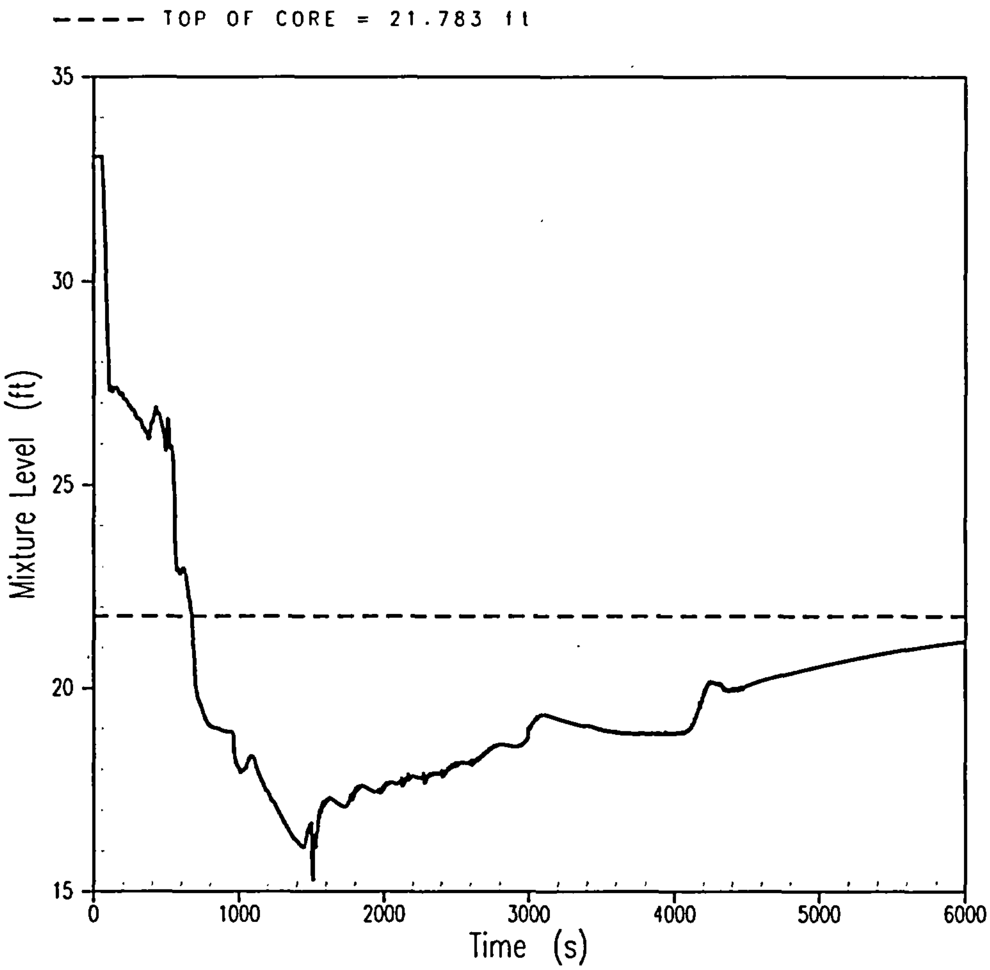

Figure 5.2.2-6A
BVPS-1 2.75-inch Break
Core Mixture Level

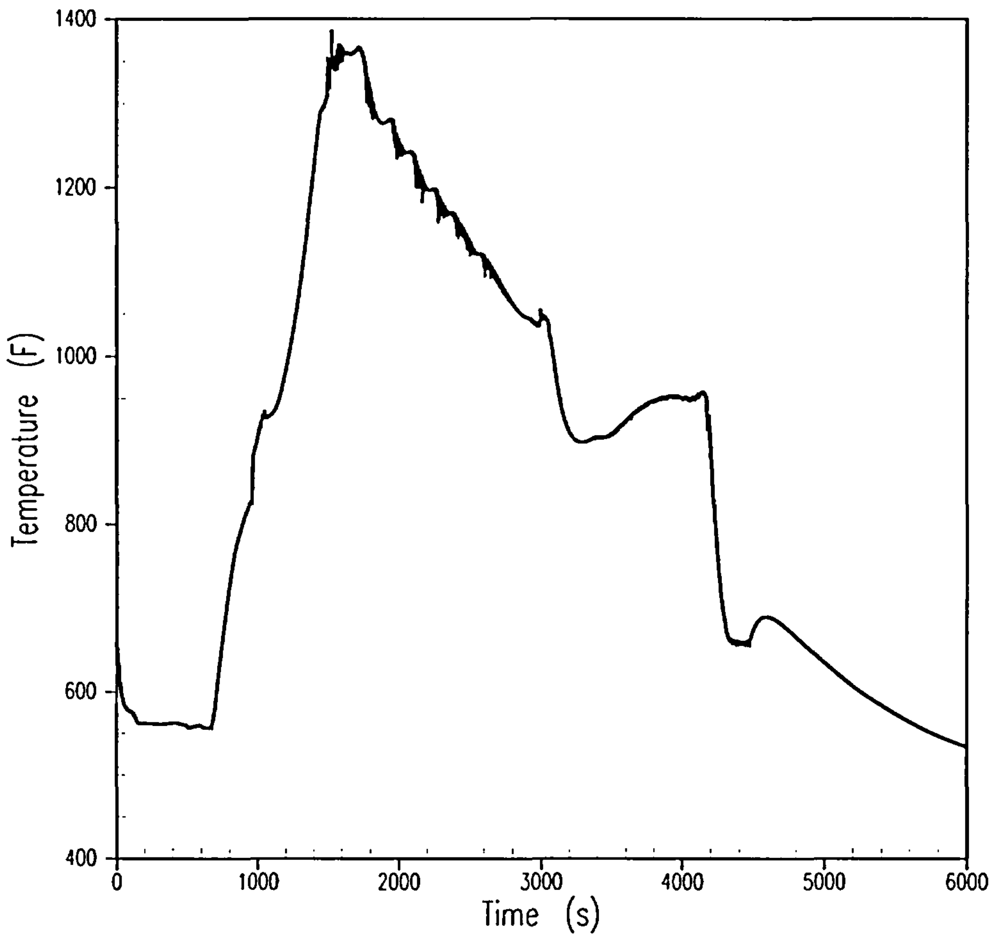

Figure 5.2.2-7A
BVPS-1 2.75-inch Break
Core Exit Vapor Temperature

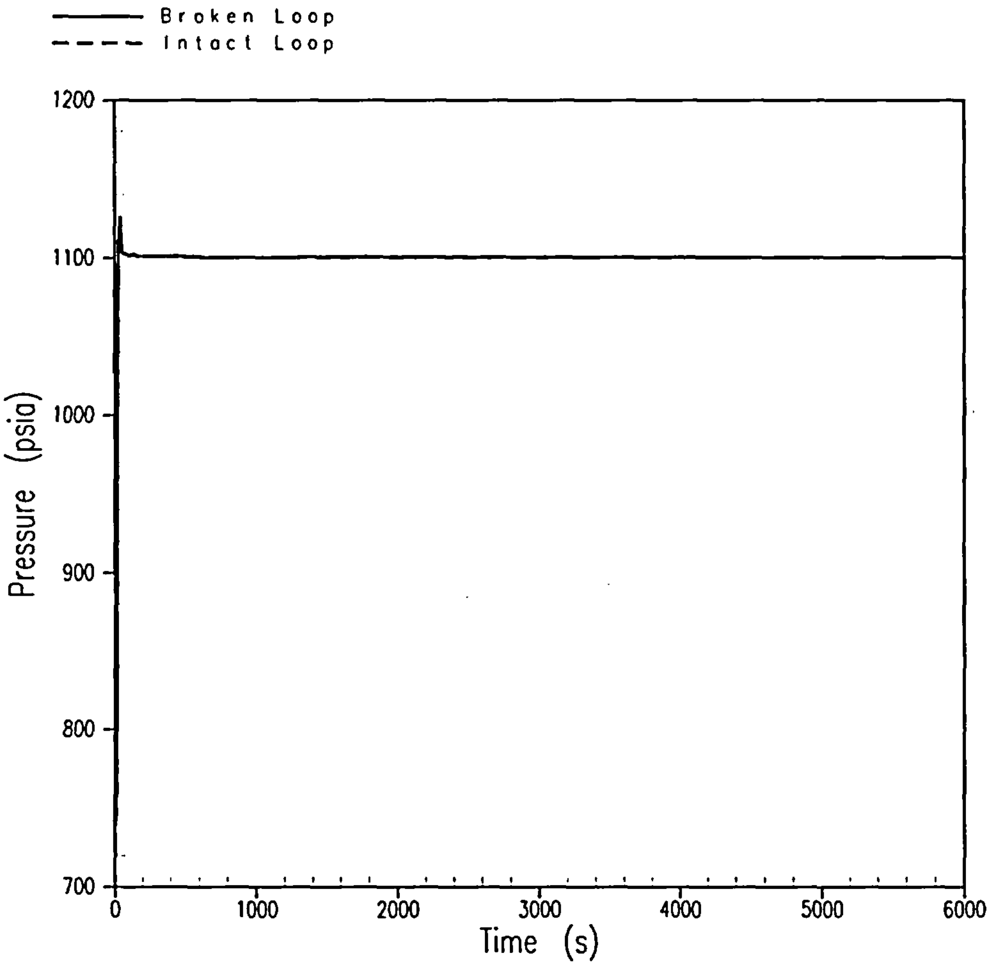

Figure 5.2.2-8A
BVPS-1 2.75-inch Break
Broken Loop and Intact Loop Secondary Pressure

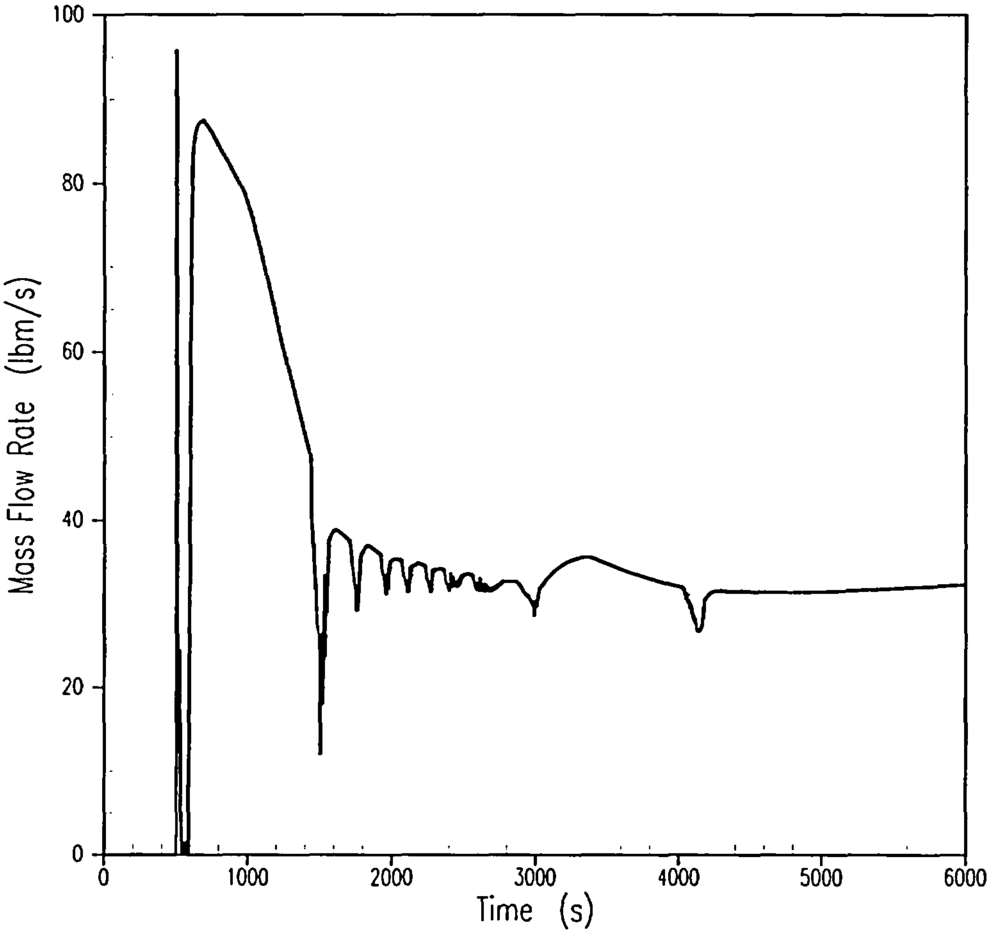

Figure 5.2.2-9A
BVPS-1 2.75-inch Break
Break Vapor Flow Rate

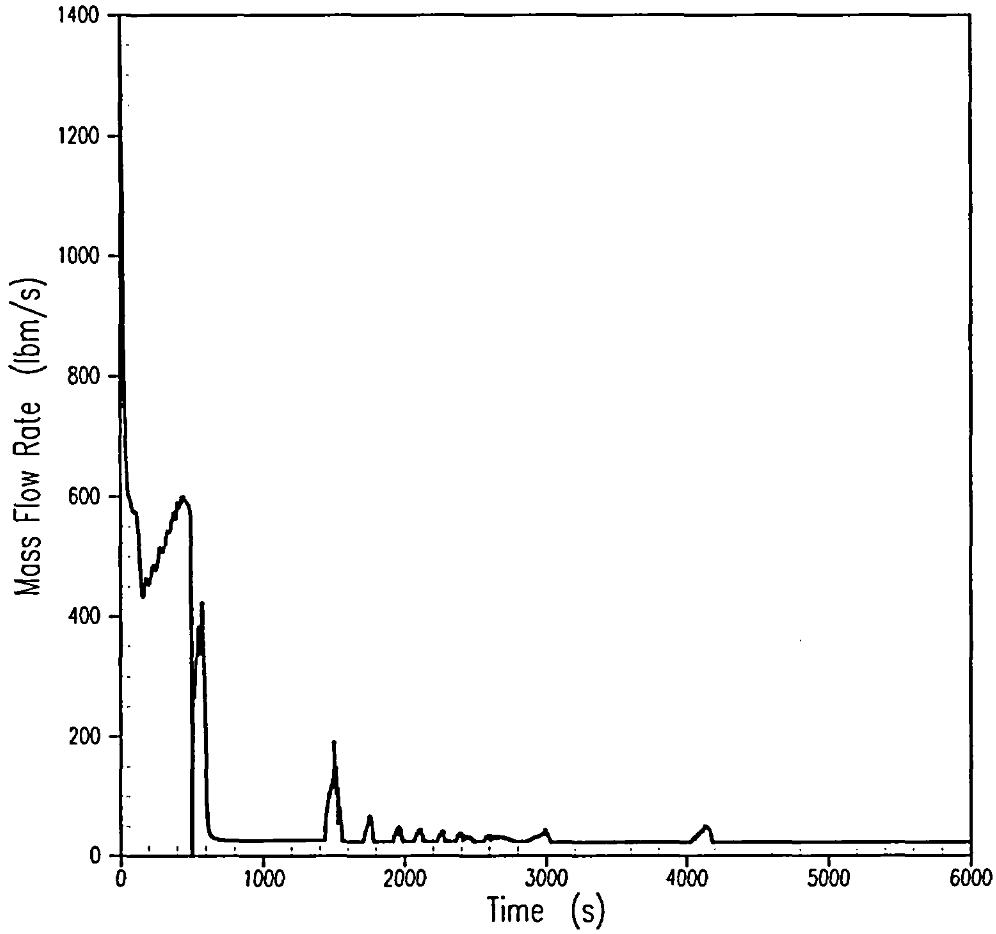

Figure 5.2.2-10A
BVPS-1 2.75-inch Break
Break Liquid Flow Rate

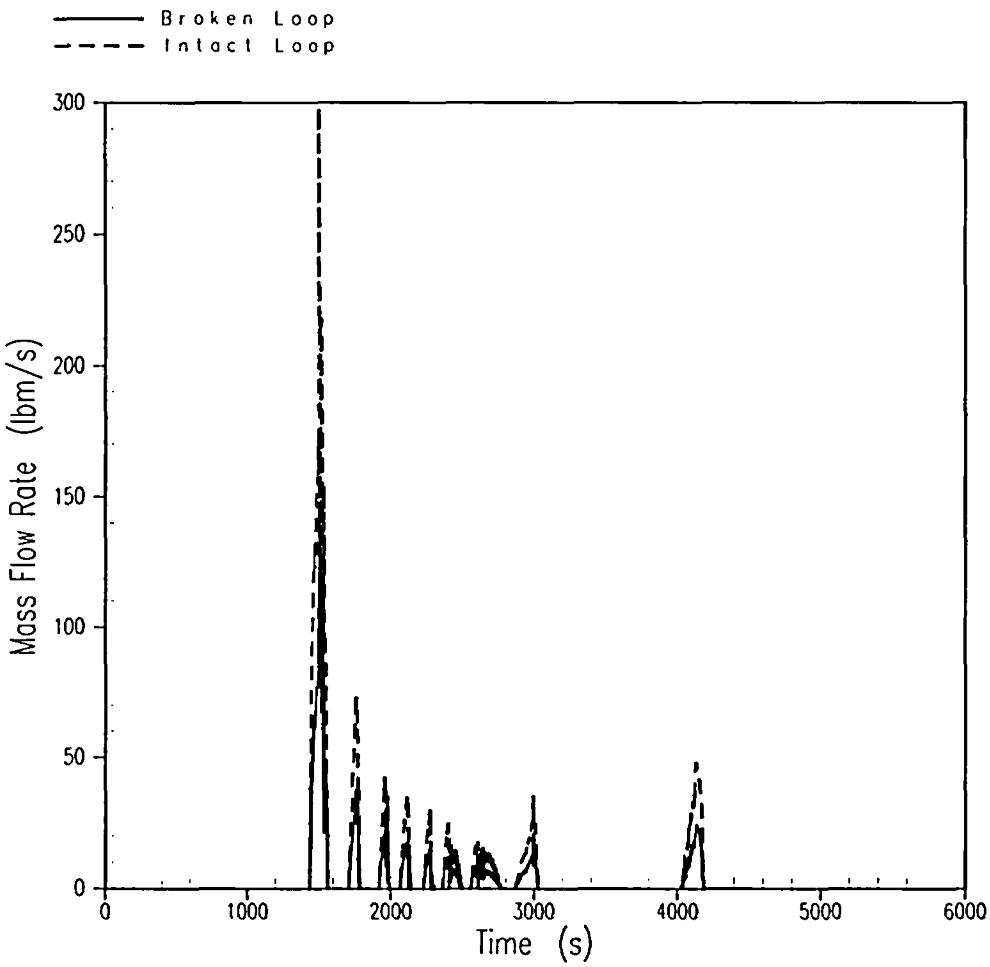

Figure 5.2.2-11A
BVPS-1 2.75-inch Break
Broken Loop and Intact Loop Accumulator Flow Rate

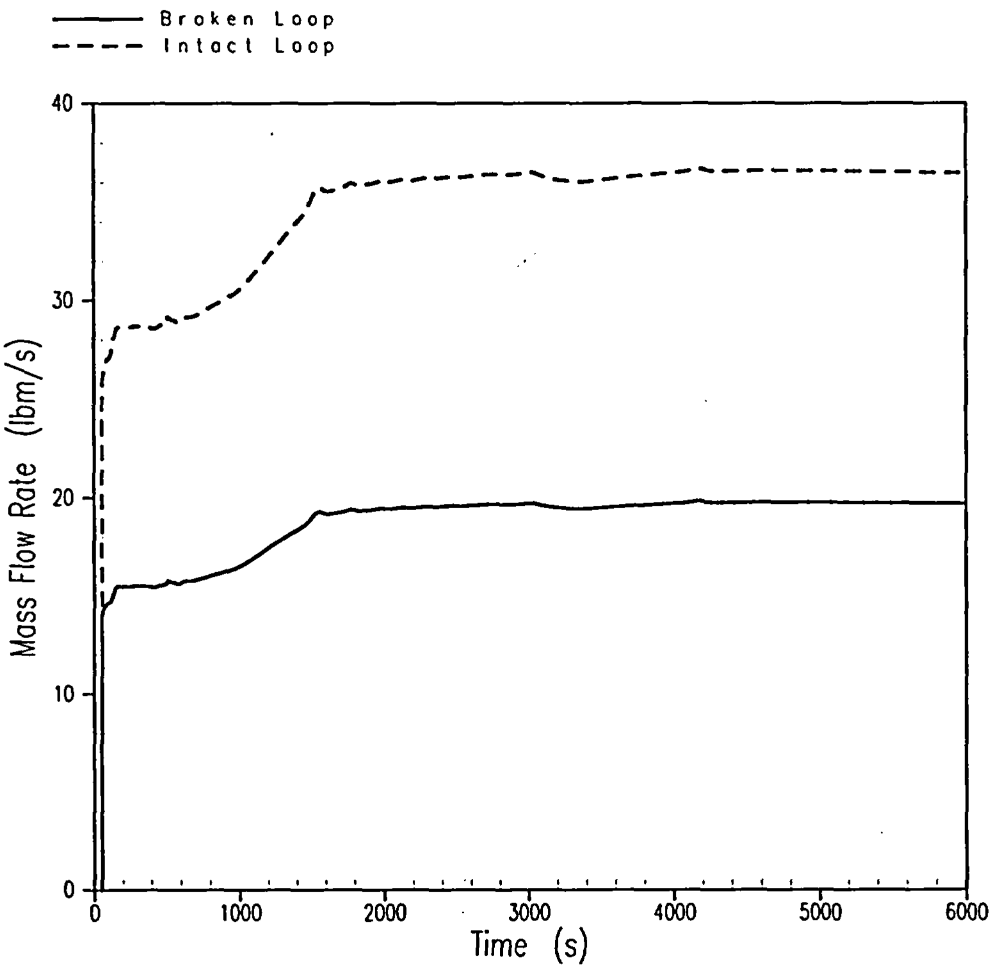

Figure 5.2.2-12A
BVPS-1 2.75-inch Break
Broken Loop and Intact Loop Pumped Safety Injection Flow Rate

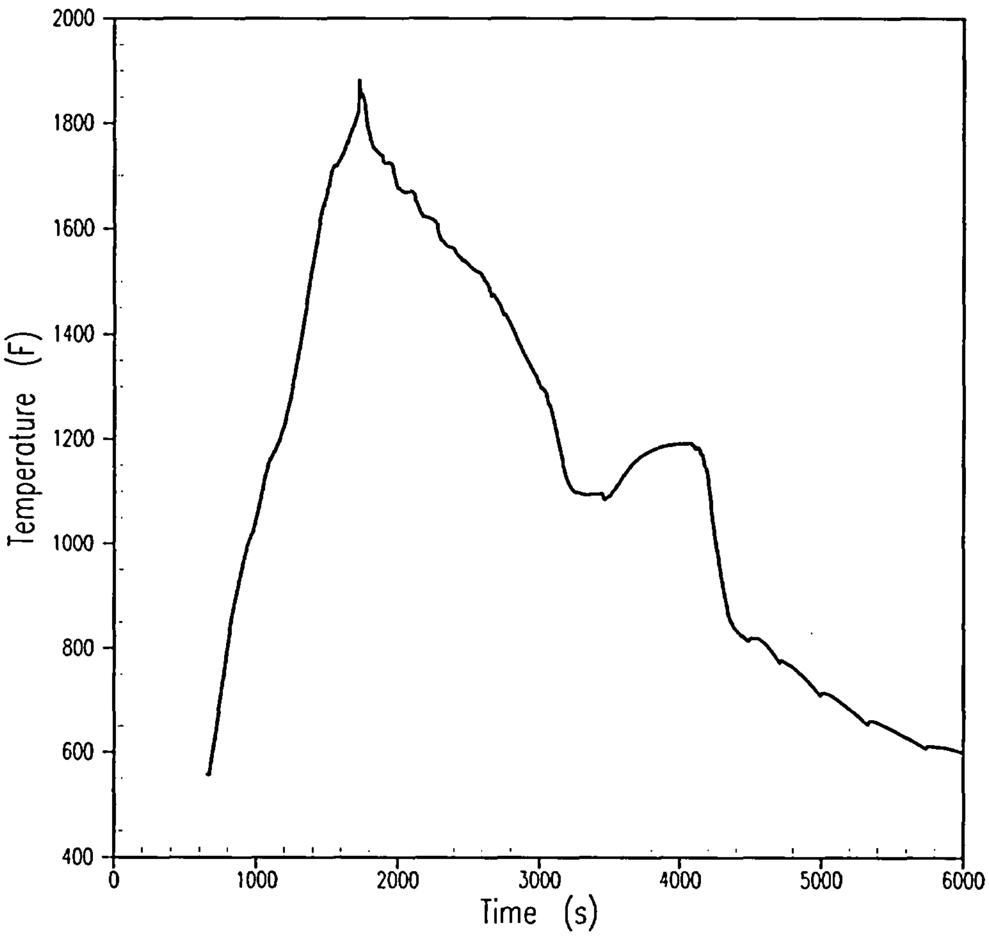

Figure 5.2.2-13A
BVPS-1 2.75-inch Break
Peak Clad Temperature

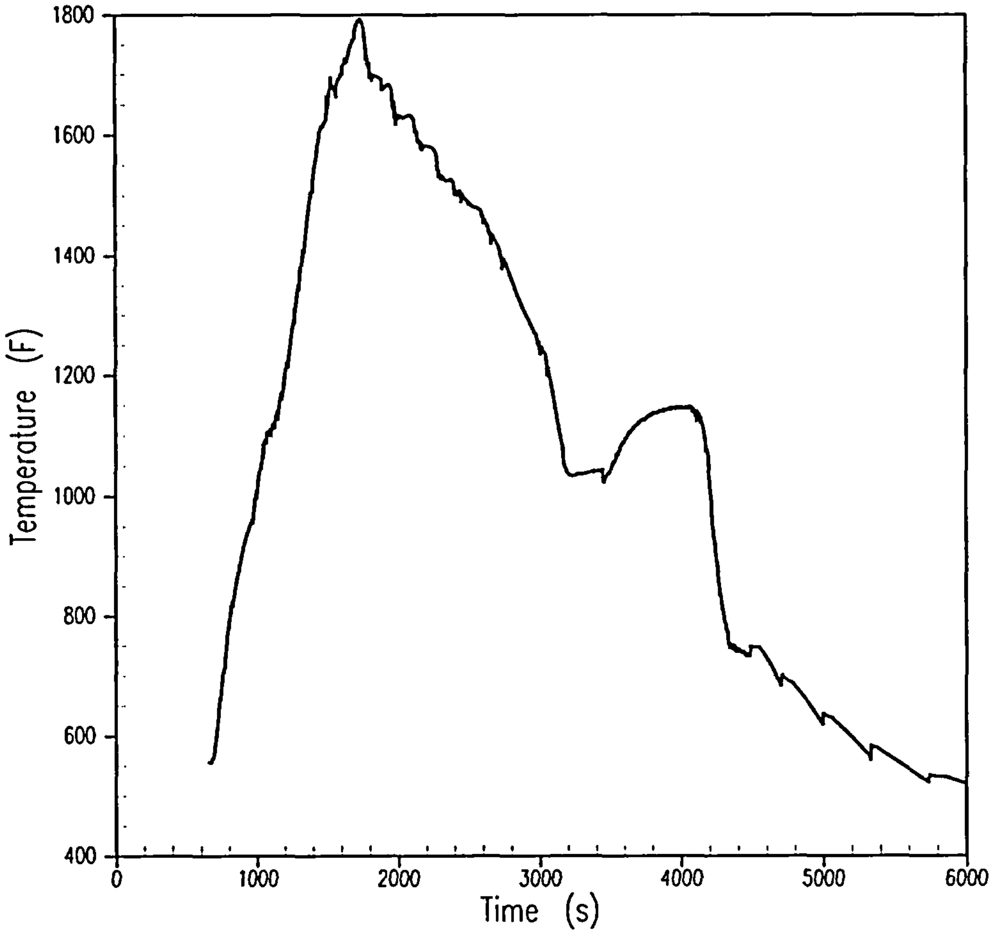

Figure 5.2.2-14A
BVPS-1 2.75-inch Break
Hot Spot Fluid Temperature

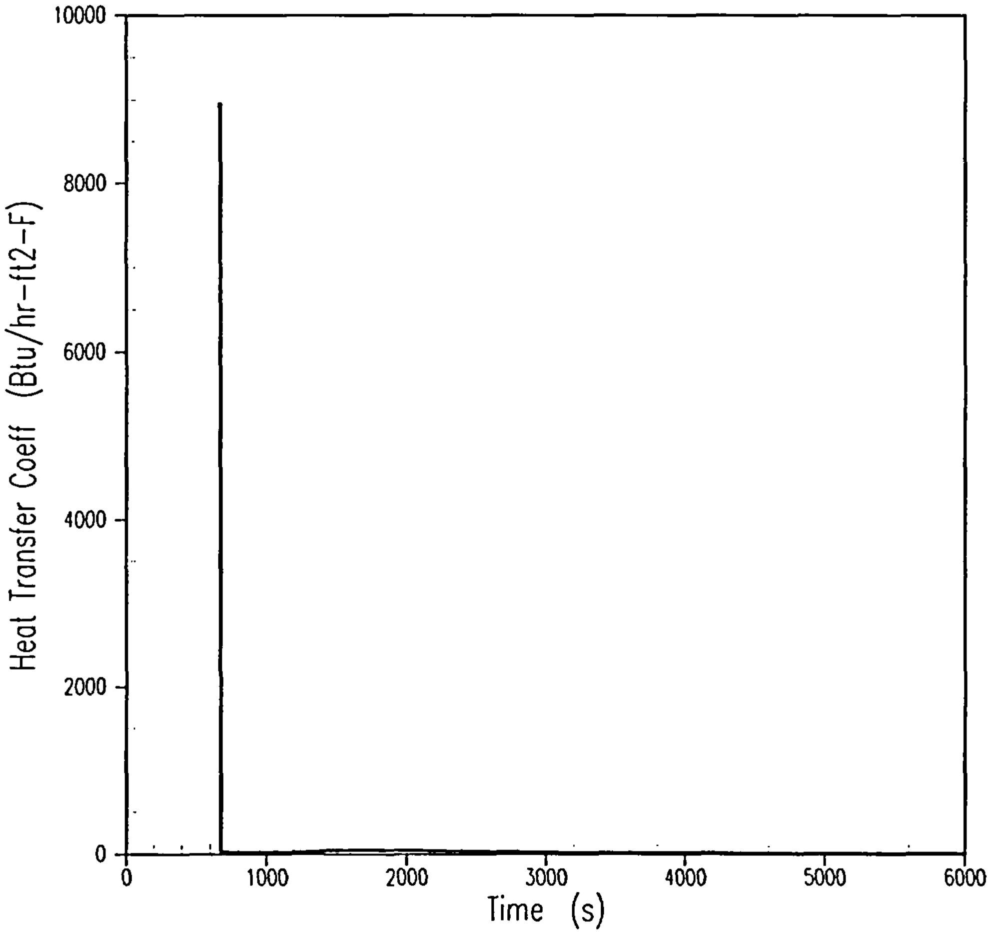

Figure 5.2.2-15A
BVPS-1 2.75-inch Break
Rod Film Heat Transfer Coefficient

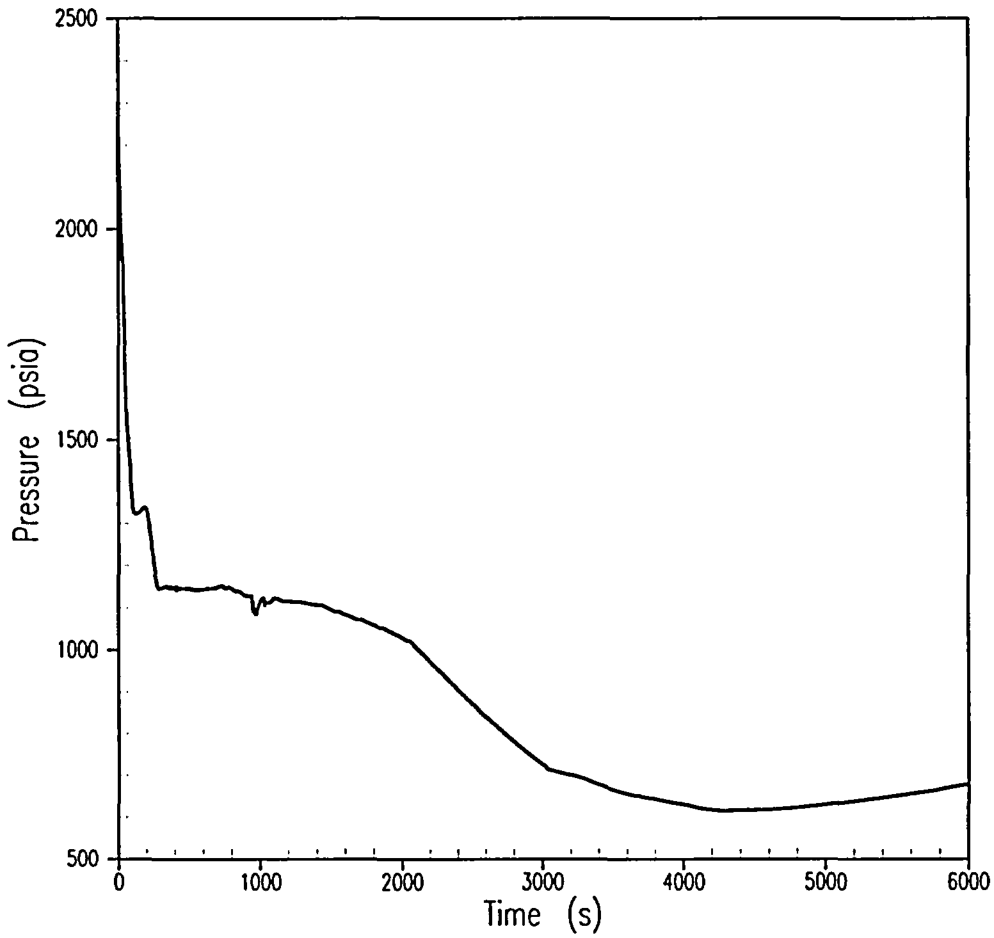

Figure 5.2.2-16A
BVPS-1 2-inch Break
RCS Pressure

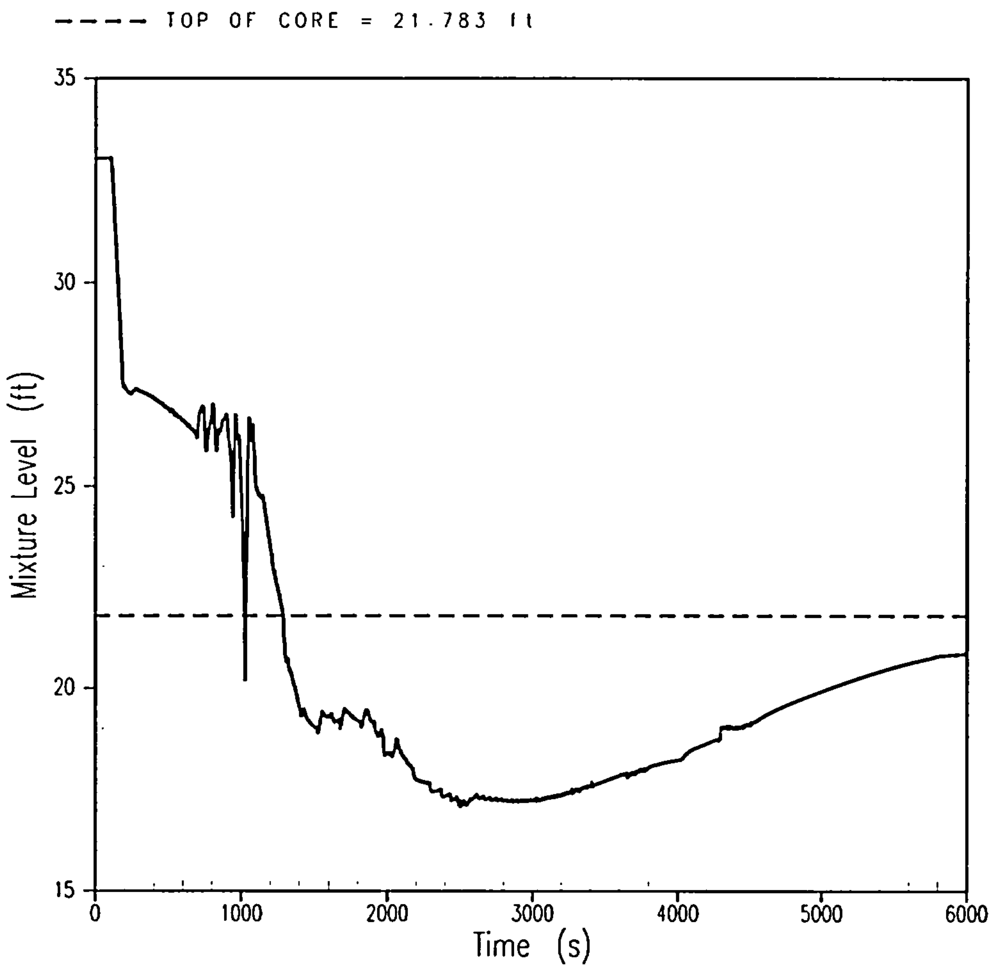

Figure 5.2.2-17A
BVPS-1 2-inch Break
Core Mixture Level

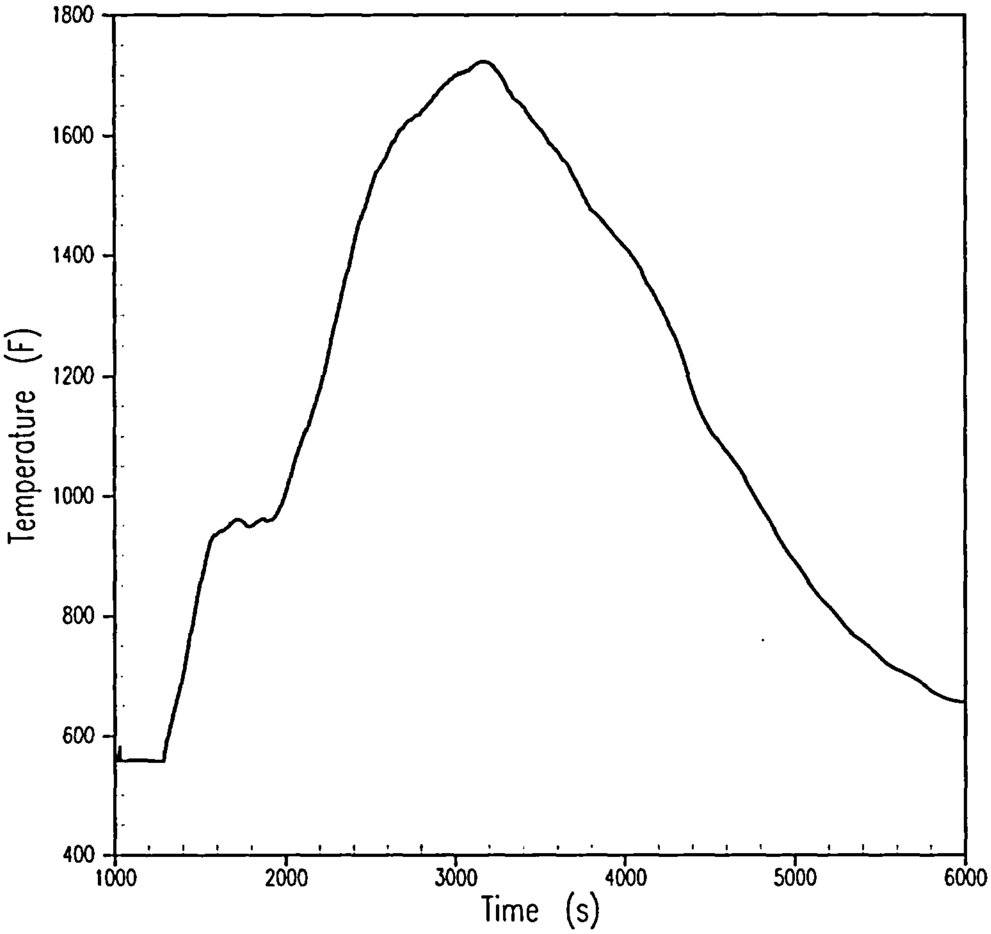

Figure 5.2.2-18A
BVPS-1 2-inch Break
Peak Clad Temperature

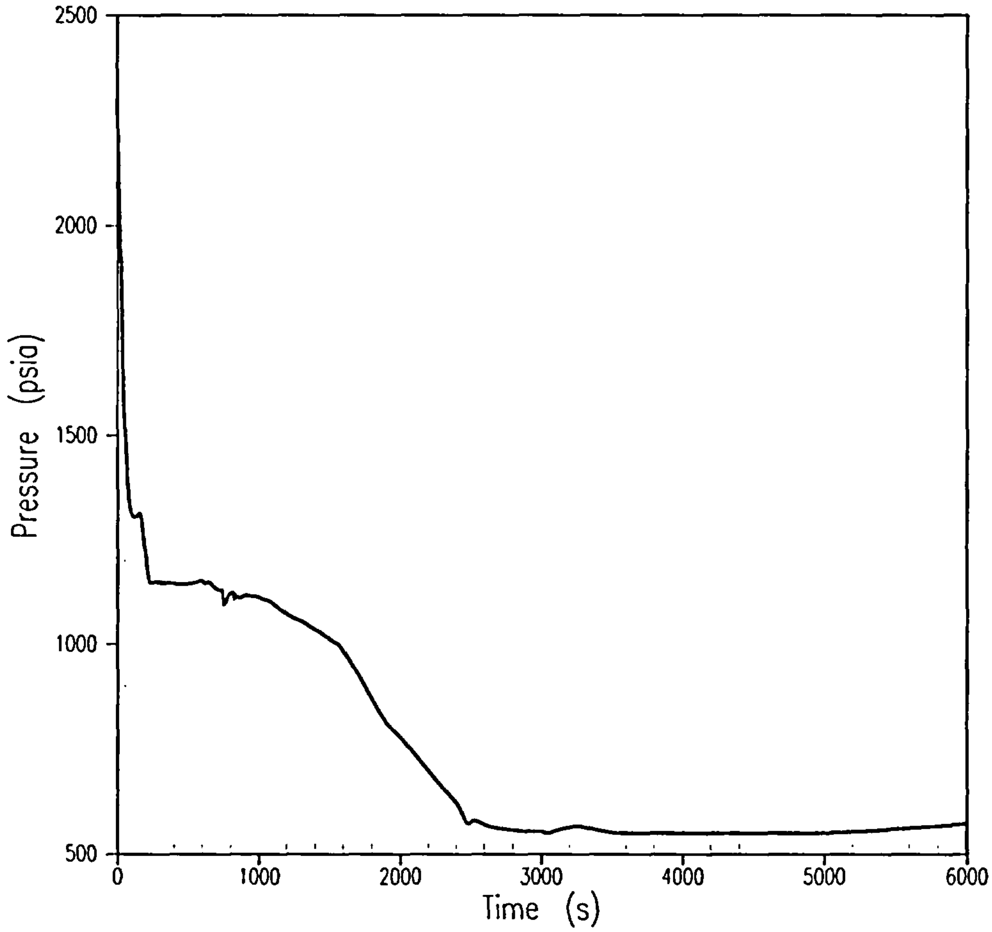

Figure 5.2.2-19A
BVPS-1 2.25-inch Break
RCS Pressure

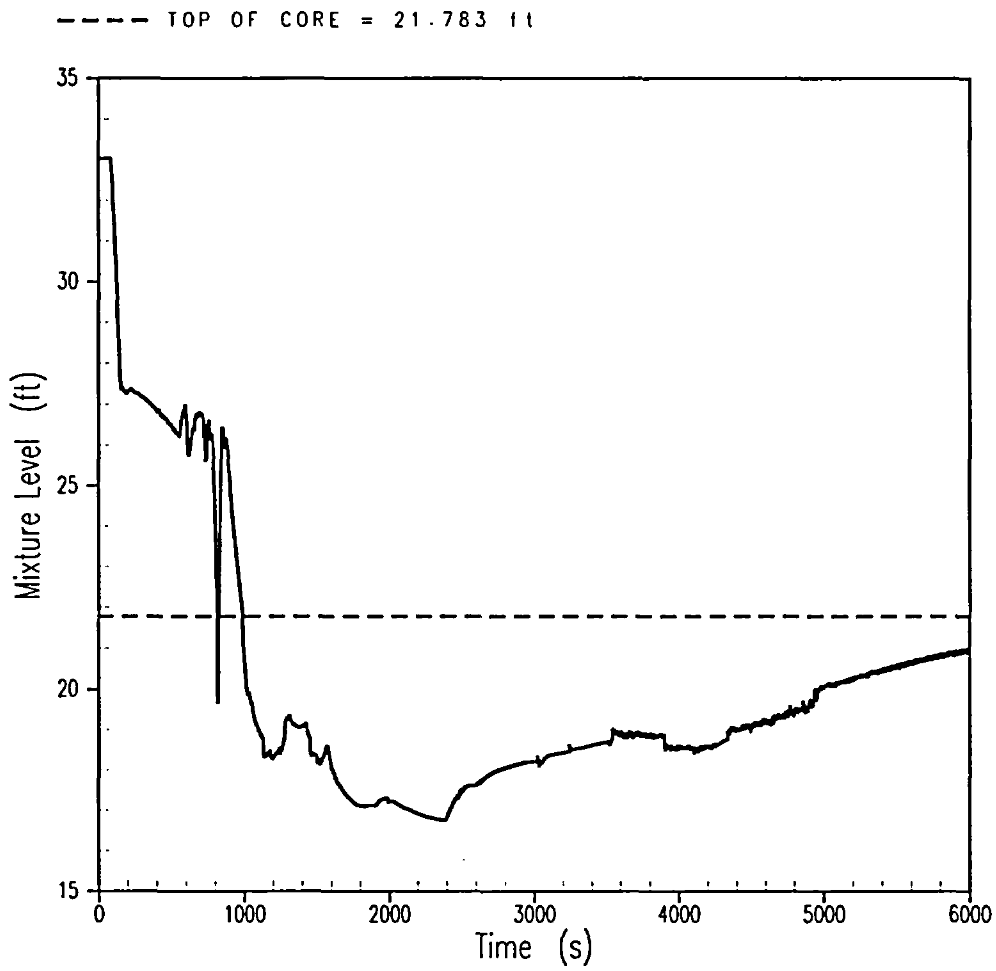

Figure 5.2.2-20A
BVPS-1 2.25-inch Break
Core Mixture Level

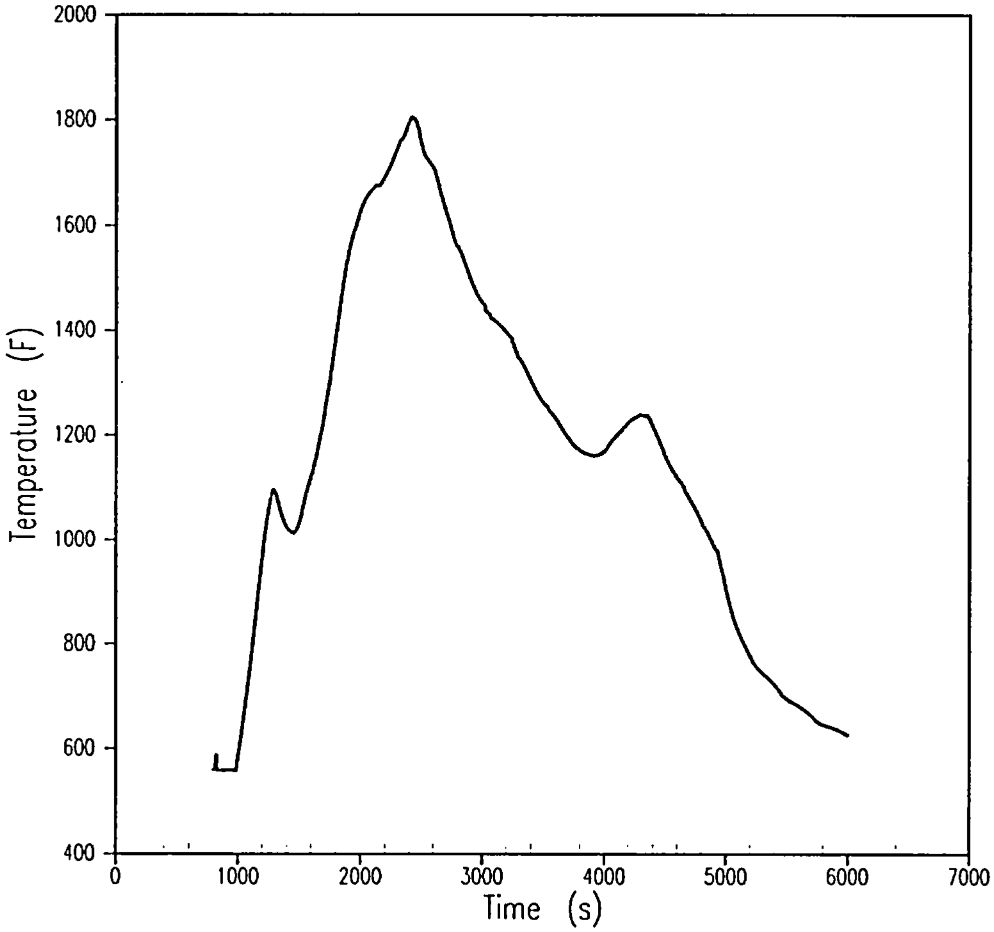

Figure 5.2.2-21A
BVPS-1 2.25-inch Break
Peak Clad Temperature

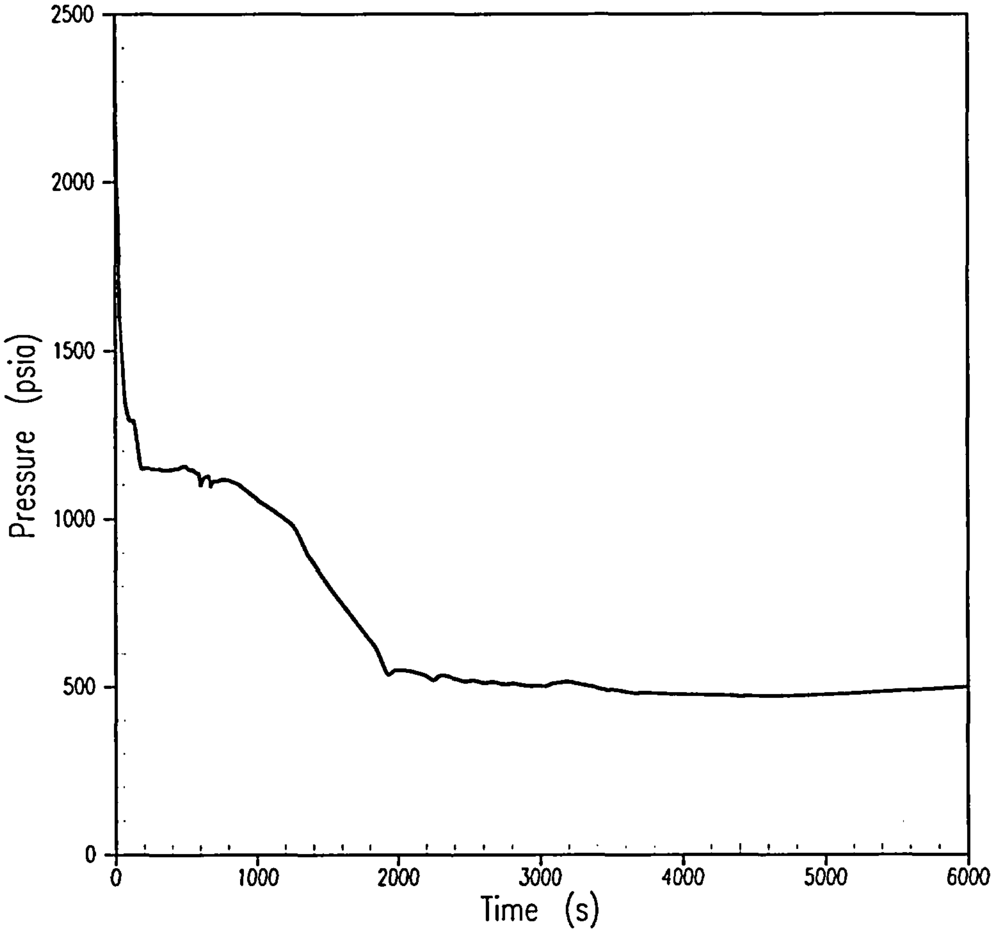

Figure 5.2.2-22A
BVPS-1 2.5-inch Break
RCS Pressure

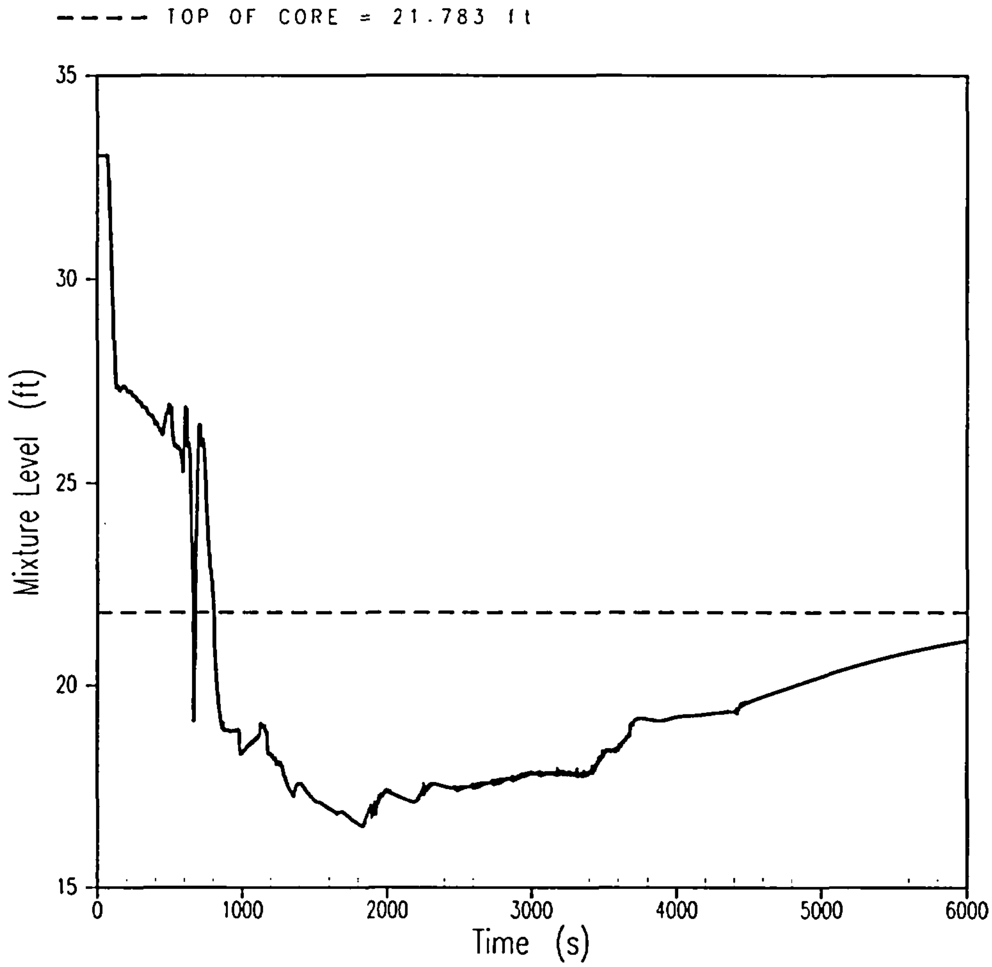

Figure 5.2.2-23A
BVPS-1 2.5-inch Break
Core Mixture Level

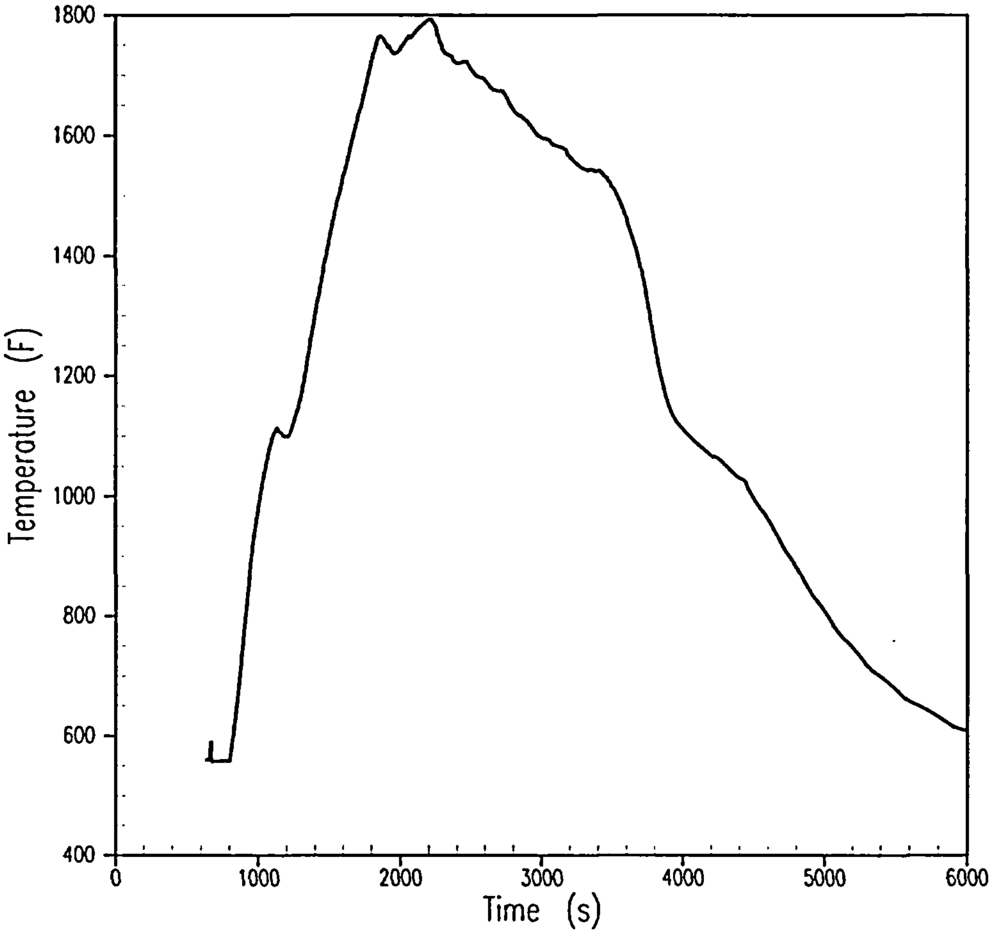

Figure 5.2.2-24A
BVPS-1 2.5-inch Break
Peak Clad Temperature

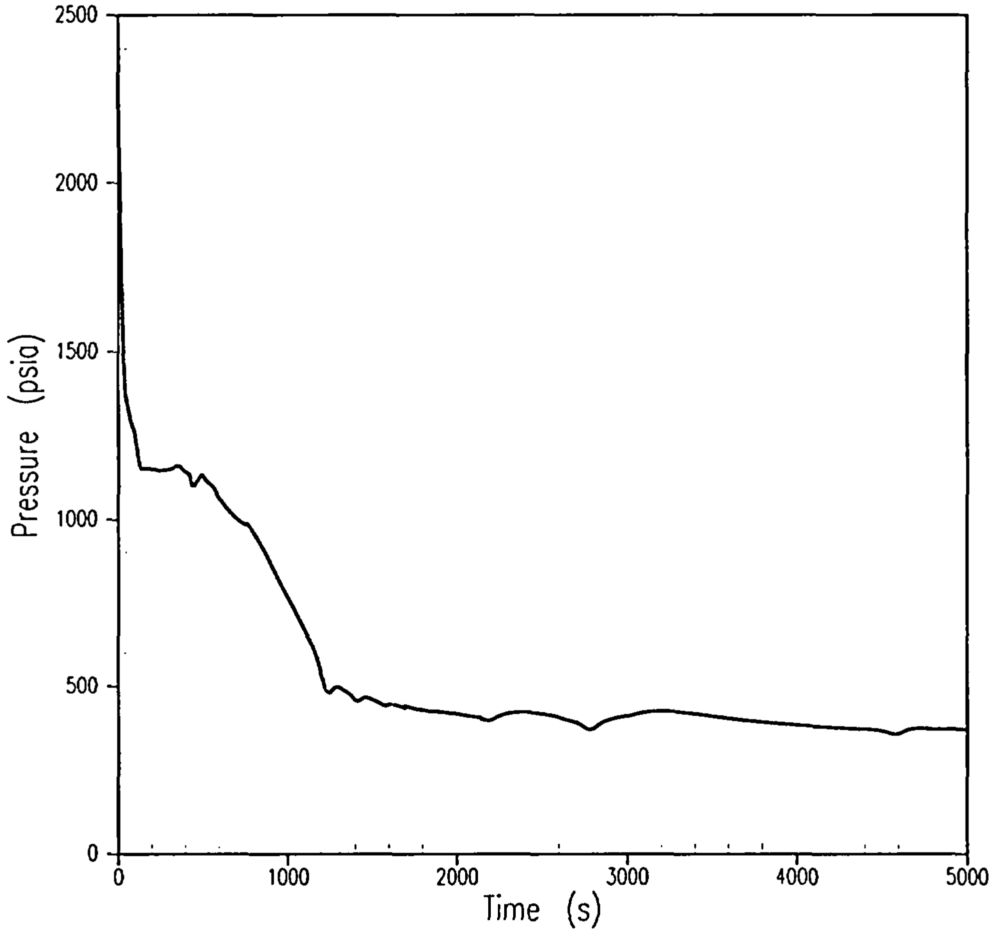

Figure 5.2.2-25A
BVPS-1 3-inch Break
RCS Pressure

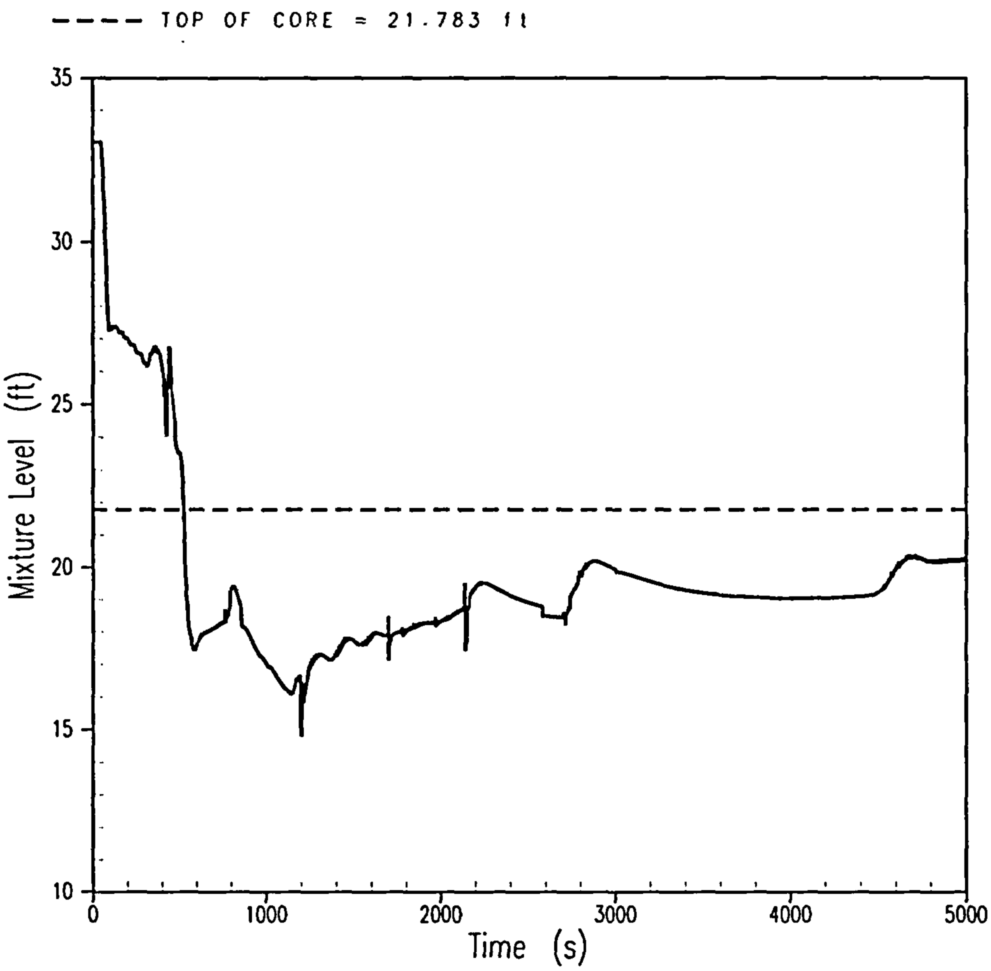

Figure 5.2.2-26A
BVPS-1 3-inch Break
Core Mixture Level

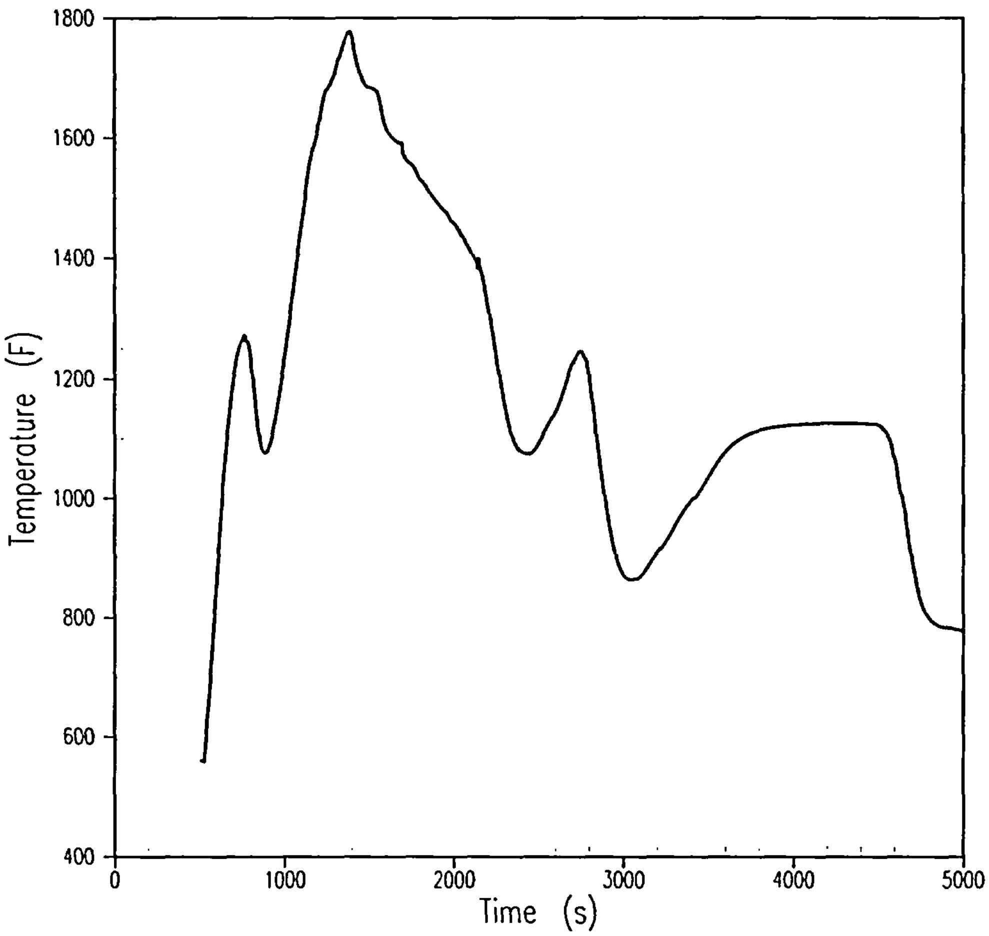

Figure 5.2.2-27A
BVPS-1 3-inch Break
Peak Clad Temperature

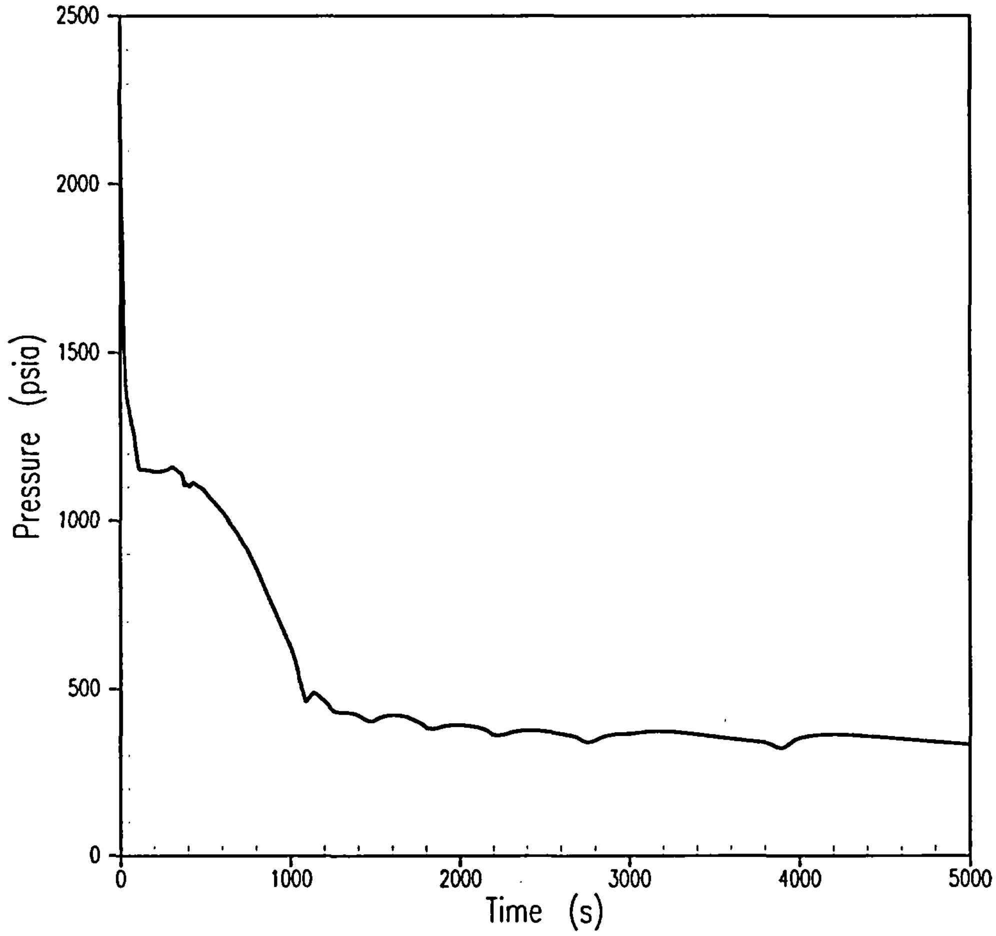

Figure 5.2.2-28A
BVPS-1 3.25-inch Break
RCS Pressure

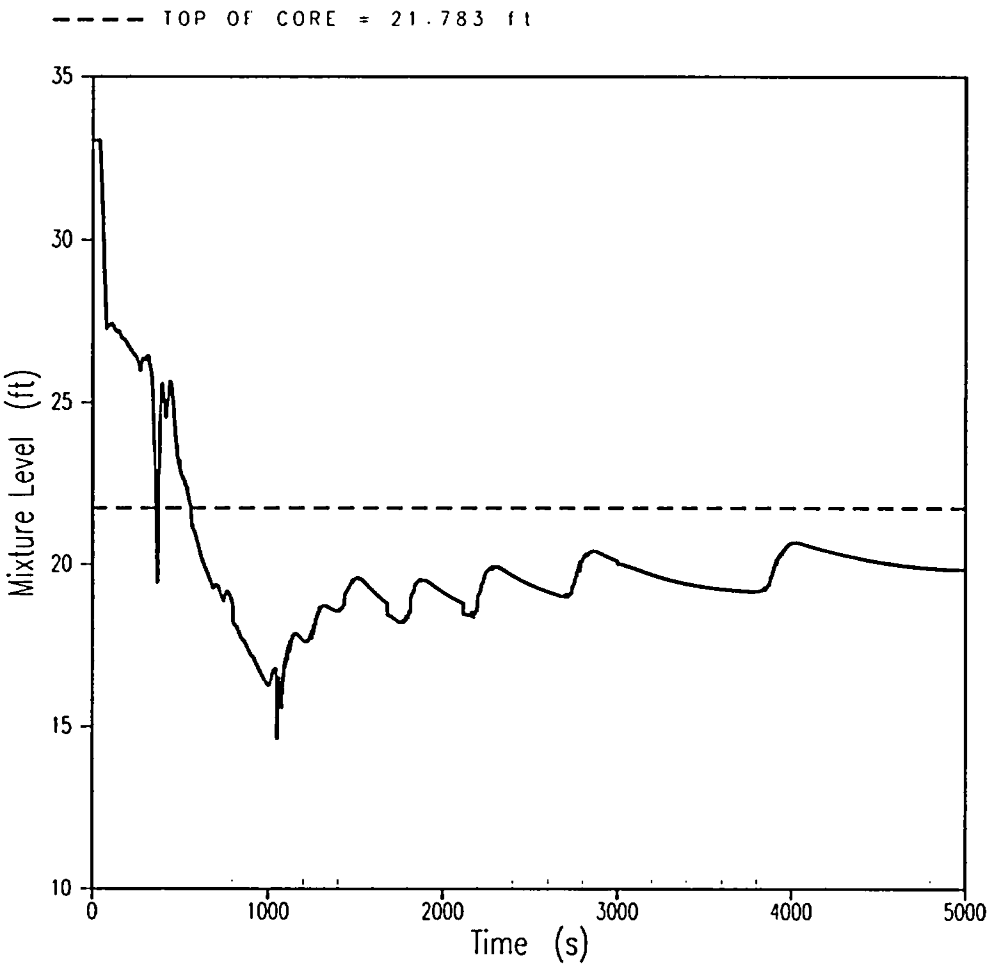

Figure 5.2.2-29A
BVPS-1 3.25-inch Break
Core Mixture Level

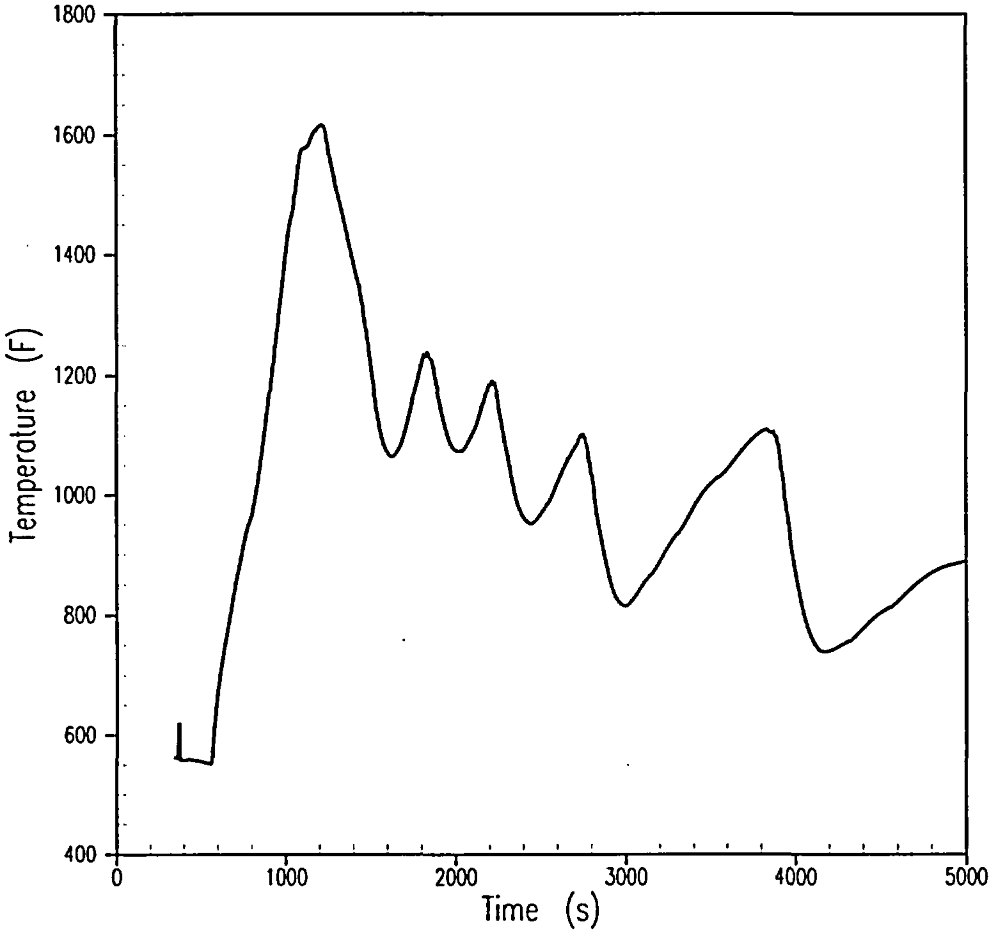

Figure 5.2.2-30A
BVPS-1 3.25-inch Break
Peak Clad Temperature

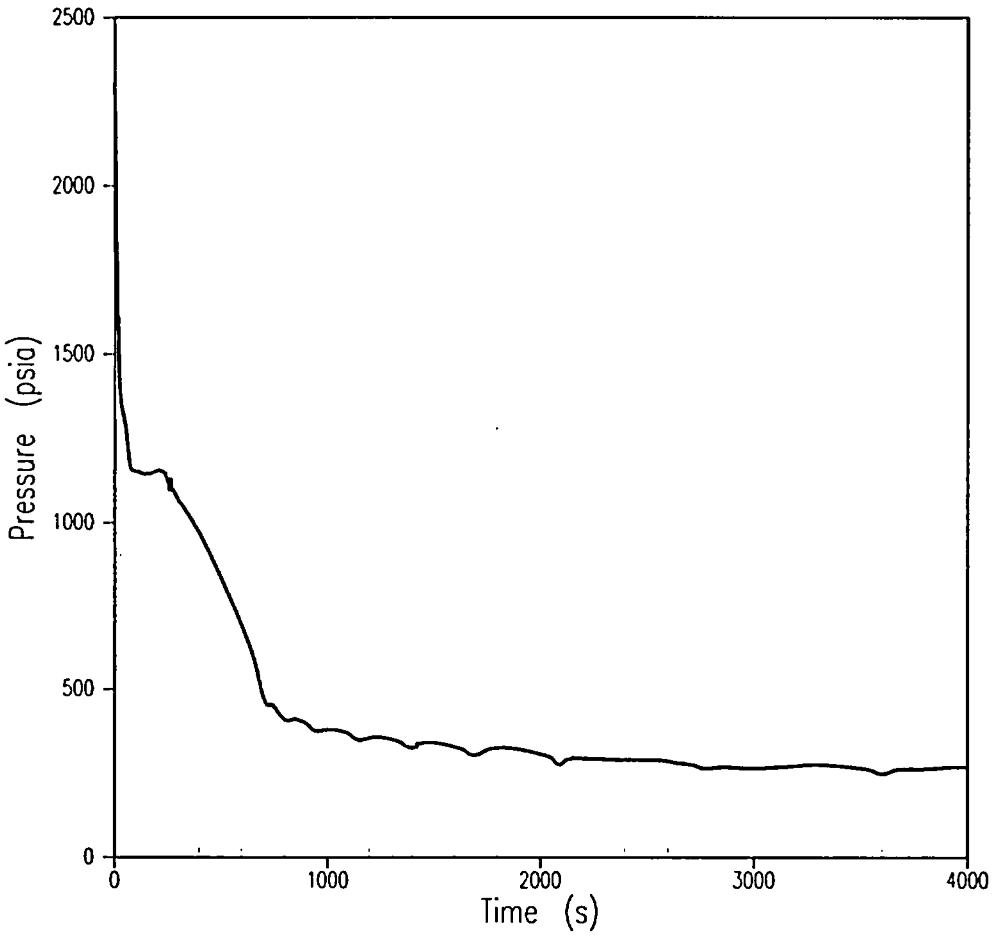

Figure 5.2.2-31A
BVPS-1 4-inch Break
RCS Pressure

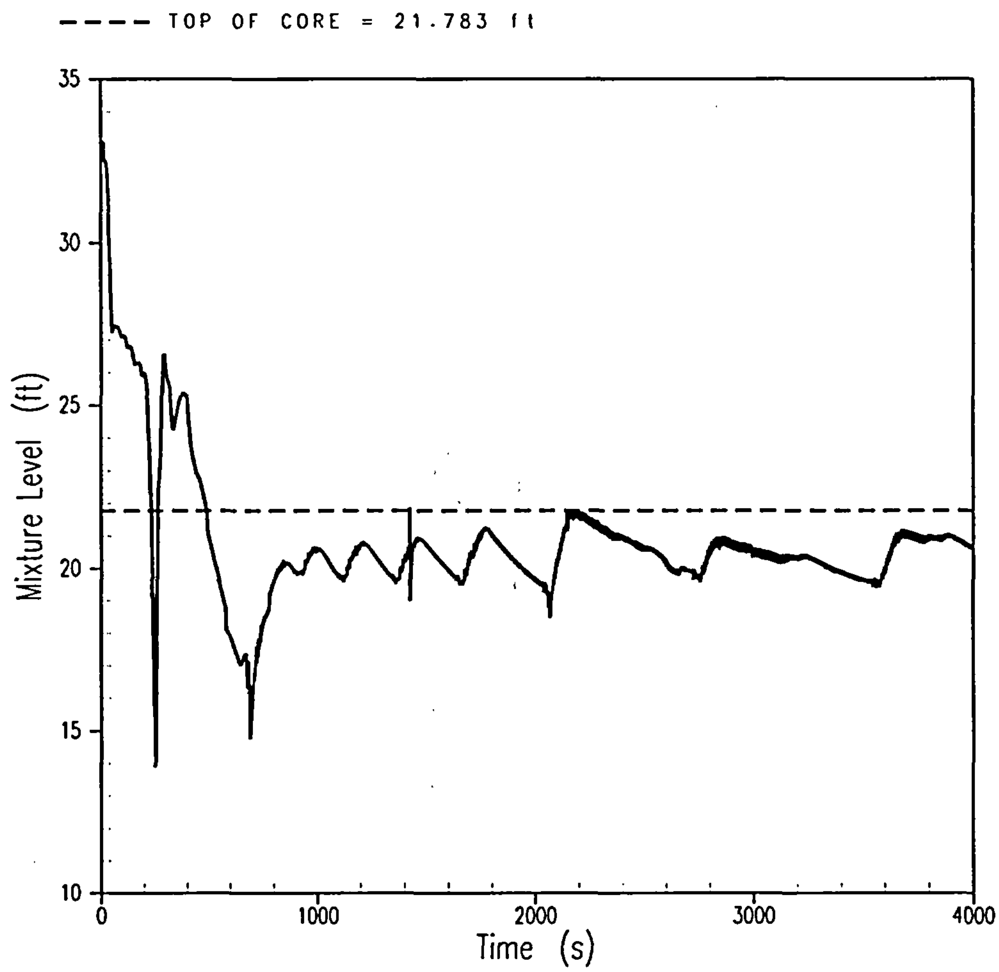

Figure 5.2.2-32A
BVPS-1 4-inch Break
Core Mixture Level

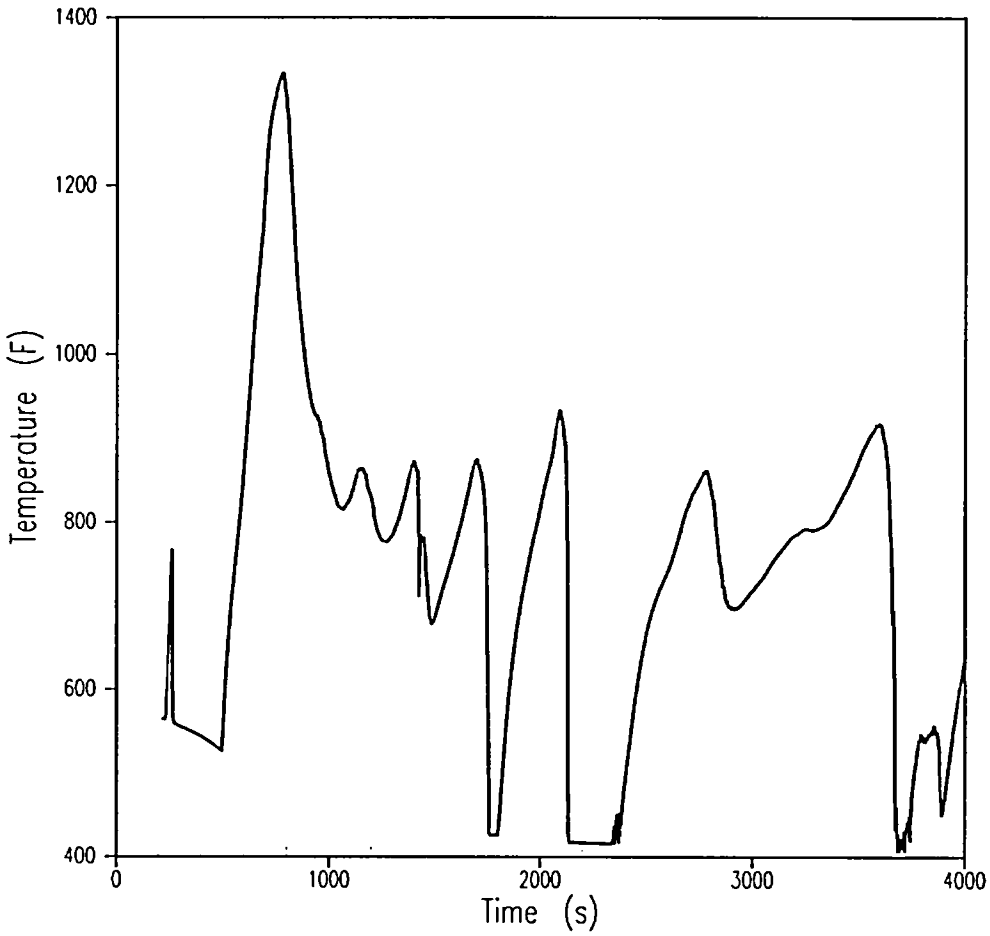

Figure 5.2.2-33A
BVPS-1 4-inch Break
Peak Clad Temperature

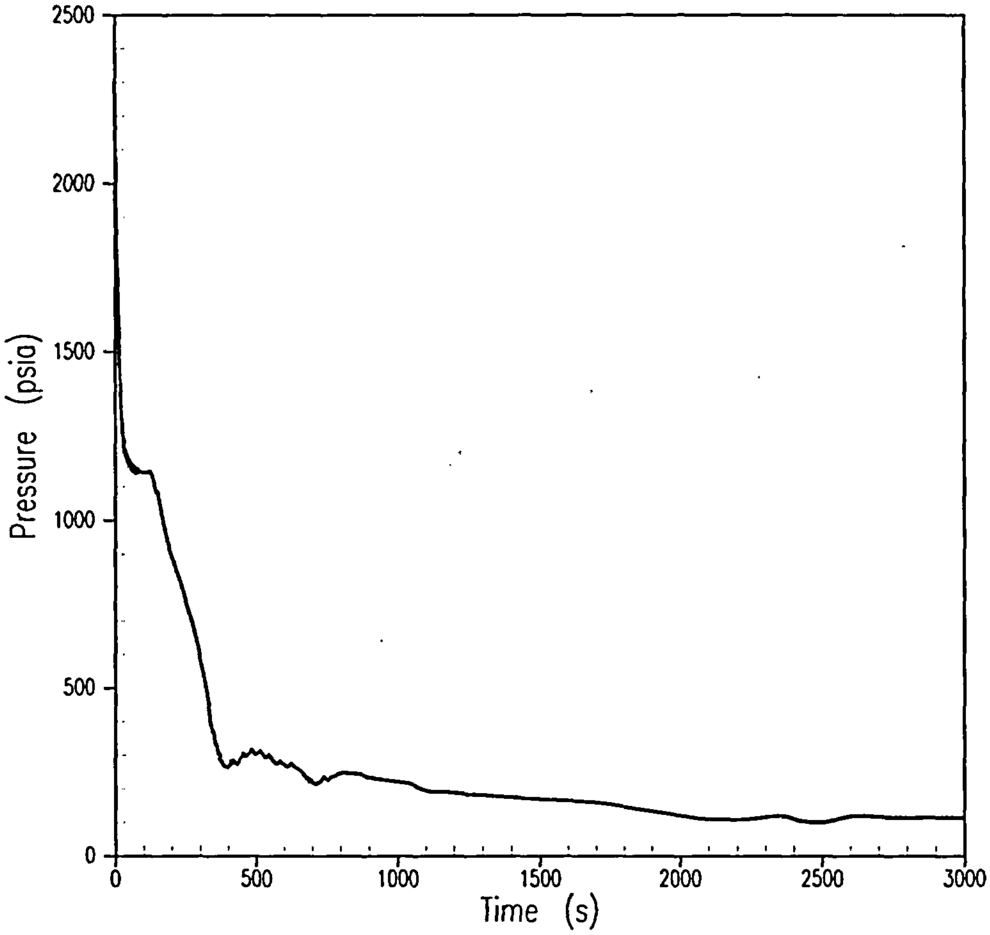

Figure 5.2.2-34A
BVPS-1 6-inch Break
RCS Pressure

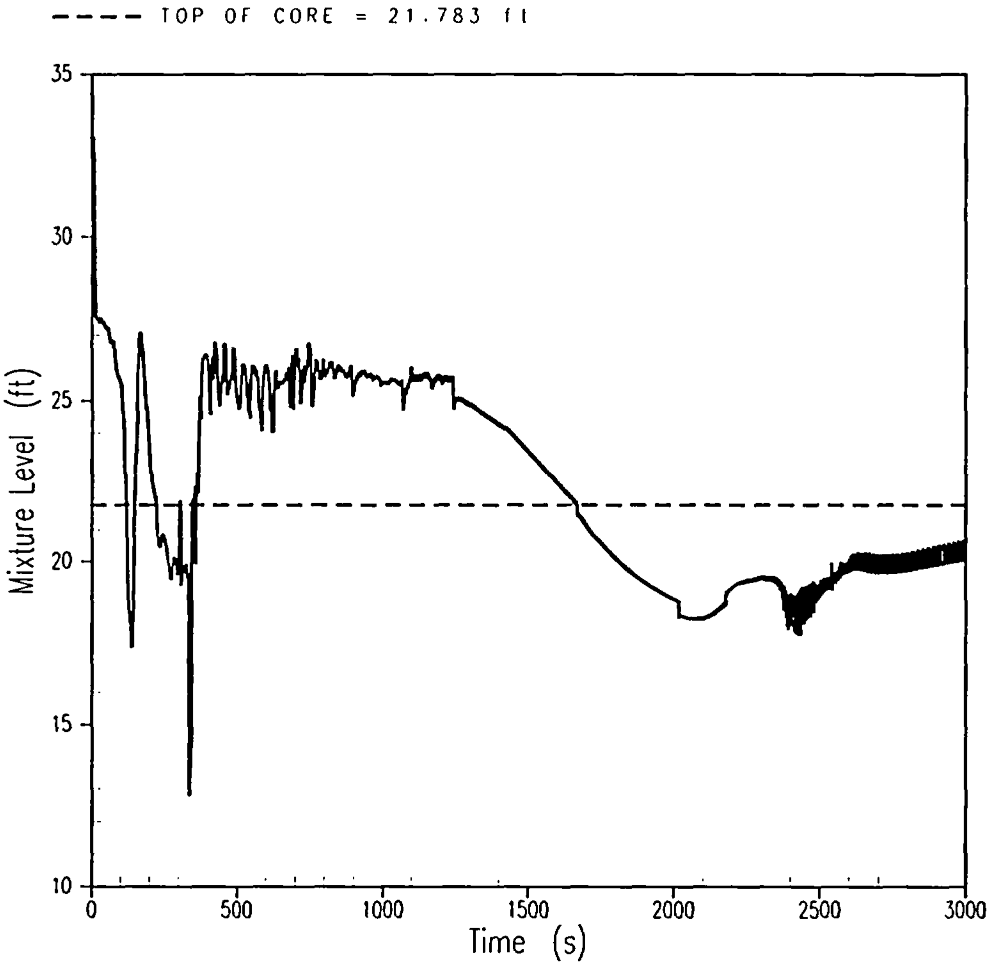

Figure 5.2.2-35A
BVPS-1 6-inch Break
Core Mixture Level

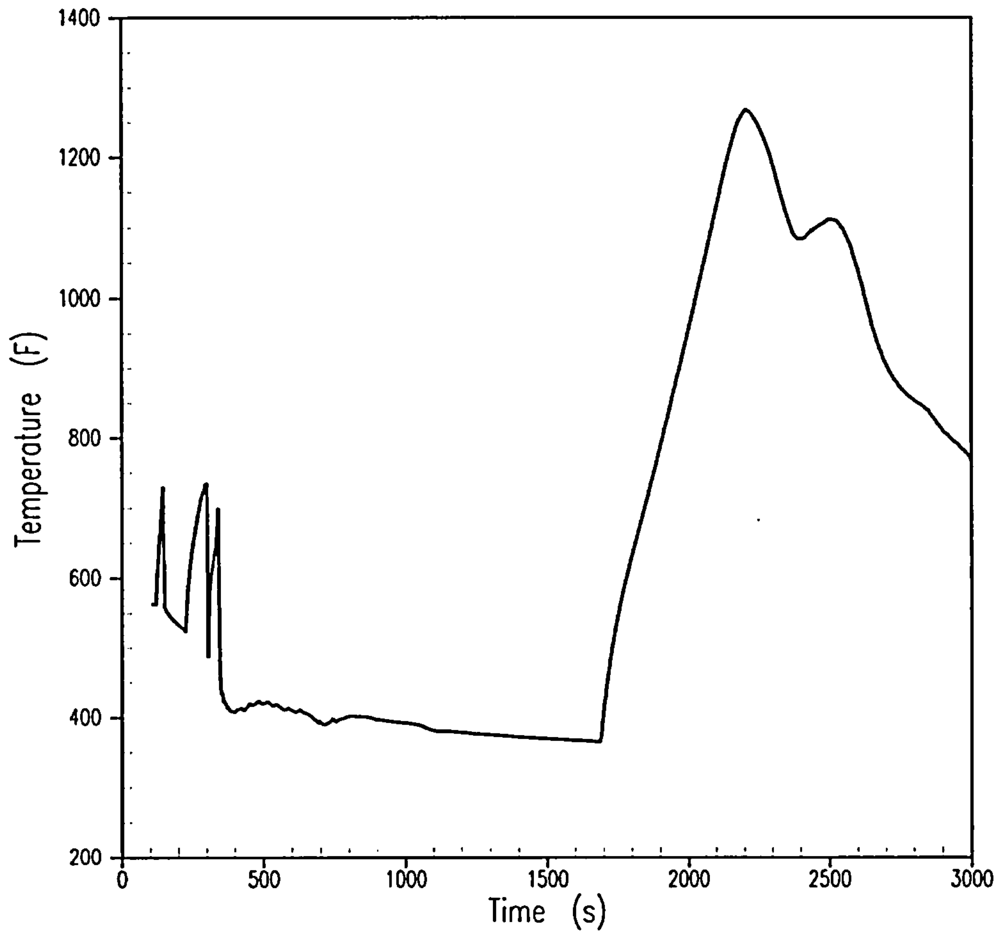

Figure 5.2.2-36A
BVPS-1 6-inch Break
Peak Clad Temperature

Attachment D

Revised Extended Power Uprate Submittal (L-04-125) Enclosure 2 Section 5.2.2

5

5.1

5.2

5.2.1

5.2.2 Small Break LOCA

5.2.2.1 Introduction

This section contains information regarding the Small Break Loss-of-Coolant Accident (SBLOCA) analyses performed in support of the EPU for BVPS-1 (with Model 54F replacement steam generators) and for BVPS-2 at the NSSS power level of 2910 MWt (2900 MWt reactor power). The purpose of analyzing the Small Break LOCA is to demonstrate conformance with the 10 CFR 50.46 (Reference 1) requirements for the conditions associated with the EPU. Important input assumptions, as well as analytical models and analysis methodology for the Small Break LOCA are contained in subsequent sections. Analysis results are provided in the form of tables and figures, as well as a more detailed description of the limiting transient. The analysis has shown that no design or regulatory limit related to the Small Break LOCA would be exceeded due to the EPU power and associated plant parameters.

5.2.2.2 Input Parameters and Assumptions

The important plant conditions and features for BVPS-1 and BVPS-2 are listed in Table 5.2.2-1A and Table 5.2.2-1B, respectively. Several additional considerations that are not identified in Table 5.2.2-1A or Table 5.2.2-1B are discussed below.

Figure 5.2.2-1 depicts the hot rod axial power shape modeled in the Small Break LOCA analyses. This shape was chosen because it represents a distribution with power concentrated in the upper regions of the core (the axial offset is +13%). Such a distribution is limiting for Small Break LOCA since it minimizes coolant swell while maximizing vapor superheating and fuel rod heat generation at the uncovered elevations. The chosen power shape has been conservatively scaled to a standard 2-line segment K(Z) envelope for BVPS-1 and BVPS-2 based on the peaking factors shown in Table 5.2.2-1A or 5.2.2-1B.

Figures 5.2.2-2 and 5.2.2-3 provide the SI flow versus pressure curves modeled in the Small Break LOCA analyses. Figure 5.2.2-2 shows the flows from one High Head Safety Injection (HHSI) pump, where the faulted loop injects to RCS pressure. Figure 5.2.2-3 shows flows from one HHSI pump and one Low Head Safety Injection (LHSI) pump, where the faulted loop injects into containment.

5.2.2.3 Description of Analyses and Evaluations

Analytical Model

The requirements for an acceptable ECCS evaluation model are presented in Appendix K of 10 CFR 50. For LOCAs due to Small Breaks, less than 1 square foot in area, the Westinghouse NOTRUMP Small

Break LOCA Emergency Core Cooling System (ECCS) Evaluation Model (References 2, 3, and 4) is used. The Westinghouse NOTRUMP Small Break LOCA ECCS Evaluation Model was developed to determine the RCS response to design basis Small Break LOCAs, and to address NRC concerns expressed in NUREG-0611 (Reference 5).

The Westinghouse Small Break LOCA ECCS Evaluation Model consists of the NOTRUMP and LOCTA-IV computer codes. The NOTRUMP code is employed to calculate the transient depressurization of the Reactor Coolant System (RCS), as well as to describe the mass and energy release of the fluid flow through the break. Among the features of the NOTRUMP code are: calculation of thermal non-equilibrium in all fluid volumes, flow regime-dependent drift flux calculations with counter-current flooding limitations, mixture level tracking logic in multiple-stacked fluid nodes, regime-dependent drift flux calculations in multiple-stacked fluid nodes and regime-dependent heat transfer correlations. These features provide NOTRUMP with the capability to accurately calculate the mass and energy distribution throughout the RCS during the course of a Small Break LOCA.

The RCS model is nodalized into volumes interconnected by flow paths. The broken loop is modeled explicitly, while the intact loops are lumped together into a second loop. Transient behavior of the system is determined from the governing conservation equations of mass, energy, and momentum. The multi-node capability of the program enables explicit, detailed spatial representation of various system components which, among other capabilities, enables a calculation of the behavior of the loop seal during a Small Break LOCA. The reactor core is represented as heated control volumes with associated phase separation models to permit transient mixture height calculations.

Fuel cladding thermal analyses are performed with a version of the LOCTA-IV code (Reference 2) using the NOTRUMP calculated core pressure, fuel rod power history, uncovered core steam flow and mixture heights as boundary conditions. The LOCTA-IV code models the hot rod and the average hot assembly rod, assuming a conservative power distribution that is skewed to the top of the core. Figure 5.2.2-4 illustrates the code interface for the Small Break Model.

Analysis

The EPU Small Break LOCA analyses considered nine different break cases each for BVPS-1 and BVPS-2 as indicated by the results in Tables 5.2.2-6A and 5.2.2-6B, respectively. A break spectrum of 1.5-, 2-, 2.25-, 2.5-, 2.75-, 3-, 3.25-, 4- and 6-inch breaks was considered. For BVPS-1, the 2.75-inch break was found to be limiting for PCT and the 2.5-inch break was found to be limiting for oxidation. For BVPS-2, the 3-inch case was found to be limiting for PCT and the 2.5-inch break case was found to be limiting for oxidation. For both units, the 1.5-inch case was found to be non-limiting in NOTRUMP and therefore PCT information was not calculated.

The most limiting single active failure used for a Small Break LOCA is that of an emergency power train failure which results in the loss of one complete train of ECCS components. In addition, a Loss-of-Offsite Power (LOOP) is postulated to occur coincident with reactor trip. This means that credit may be taken for at most one high head safety injection (HHSI) pump. In the analyses for BVPS-1 and BVPS-2, one HHSI pump is modeled. The Small Break LOCA analysis performed for both units models the ECCS flow as being delivered to both the intact and broken loops at the RCS backpressure for breaks smaller than the cold leg HHSI nozzle (1.5-inch - 4-inch breaks) and at containment pressure for breaks greater than the cold leg HHSI nozzle (6-inch breaks). These SI flows are illustrated in Figure 5.2.2-2 and 5.2.2-3 for each scenario. Note that for the 6-inch breaks, no SI is assumed in the faulted loop because the break is postulated along the SI line. The LOOP and the failure of a diesel generator to start as the limiting single failure for Small Break LOCA is part of the NRC approved methodology and does not change as a result of the EPU conditions. The single failure assumption is extremely limiting due to the fact that one train of SI, one motor driven auxiliary feedwater (AFW) pump, and power to the reactor coolant pumps (RCPs) are all modeled to be lost. Any other active single failure would not result in a more limiting scenario since increased SI flow would improve the overall transient results.

Prior to break initiation, the plant is in a full power (100.6%) equilibrium condition, i.e., the heat generated in the core is being removed via the secondary system. Other initial plant conditions used in the analysis are given in Table 5.2.2-1A or Table 5.2.2-1B. Subsequent to the break opening, a period of reactor coolant system blowdown ensues in which the heat from fission product decay, the hot reactor internals, and the reactor vessel continues to be transferred to the RCS fluid. The heat transfer between the RCS and the secondary system may be in either direction and is a function of the relative temperatures of the primary and secondary conditions. In the case of continuous heat addition to the secondary during a period of quasi-equilibrium, an increase in the secondary system pressure results in steam relief via the steam generator safety valves.

When a Small Break LOCA occurs, depressurization of the RCS causes fluid to flow into the loops from the pressurizer resulting in a pressure and level decrease in the pressurizer. The reactor trip signal subsequently occurs when the pressurizer low-pressure reactor trip setpoint, conservatively modeled as 1935 psia, is reached. LOOP is postulated to occur coincident with reactor trip. A safety injection signal is generated when the pressurizer low-pressure safety injection setpoint, conservatively modeled as 1745 psia for BVPS-1 and 1760 psia for BVPS-2, is reached. Safety injection flow is delayed 27 seconds after the occurrence of the low-pressure condition. This delay accounts for signal processing, diesel generator start up and emergency power bus loading consistent with the loss-of-offsite power coincident with reactor trip, as well as the pump acceleration and valve delays.

The following countermeasures limit the consequences of the accident in two ways:

1. Reactor trip and borated water injection supplement void formation in causing a rapid reduction of nuclear power to a residual level corresponding to the delayed fission and fission product decay. No credit is taken in the Small Break LOCA analysis for the boron content of the injection water. In addition, credit is taken in the Small Break LOCA analysis for the insertion of Rod Cluster Control Assemblies (RCCAs) subsequent to the reactor trip signal, considering the most reactive RCCA is stuck in the full out position. A rod drop time of 2.7 seconds was used while also considering an additional 2 seconds for the signal processing delay time. Therefore, a total delay time of 4.7 seconds from the time of reactor trip signal to full rod insertion was used in the Small Break LOCA analysis.

2. Injection of borated water provides sufficient flooding of the core to prevent excessive cladding temperatures.

During the earlier part of the Small Break transient (prior to the postulated loss-of-offsite power coincident with reactor trip), the loss of flow through the break is not sufficient to overcome the positive core flow maintained by the reactor coolant pumps. During this period, upward flow through the core is maintained. However, following the reactor coolant pump trip (due to a LOOP) and subsequent pump coastdown, a period of core uncovery occurs. Ultimately, the Small Break transient analysis is terminated when the top of the core is recovered or the core mixing level is increasing, and ECCS flow provided to the RCS exceeds the break flow rate.

The core heat transfer mechanisms associated with the Small Break transient include the break itself, the injected ECCS water, and the heat transferred from the RCS to the steam generator secondary side. Main Feedwater (MFW) is conservatively isolated in 10 seconds for BVPS-1 (consisting of a 3 second signal delay time and a 7 second main feedwater isolation valve stroke time) and 7 seconds for BVPS-2 (consisting of a 2 second signal delay time and a 5 second main feedwater isolation valve stroke time) following the generation of the pressurizer low-pressure SI signal. Additional makeup water is also provided to the secondary using the auxiliary feedwater (AFW) system. An AFW actuation signal is derived from the pressurizer low-pressure SI signal, resulting in the delivery of AFW system flow 60 seconds after the generation of the SI signal. The heat transferred to the secondary side of the steam generator aids in the reduction of the RCS pressure.

Should the RCS depressurize to approximately 625 psia (accumulator minimum pressure), the cold leg accumulators begin to inject borated water into the reactor coolant loops.

5.2.2.4 Acceptance Criteria and Results

The acceptance criteria for the LOCA are described in 10 CFR 50.46 (Reference 1) as follows:

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

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

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

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

5. 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.

Criteria 1 through 3 are explicitly covered by the Small Break LOCA analysis at EPU conditions.

For criterion 4, the appropriate core geometry was modeled in the analysis. The results based on this geometry satisfy the Peak Clad Temperature (PCT) criterion of 10 CFR 50.46 and consequently, demonstrate that the core remains amenable to cooling.

For criterion 5, Long-Term Core Cooling (LTCC) considerations are not directly applicable to the Small Break LOCA transient analysis addressed in this section, but are assessed by Sections 5.2.3 and 5.2.4 as part of the evaluation of ECCS performance.

The acceptance criteria were established to provide a significant margin in ECCS performance following a LOCA.

In order to determine the conditions that produced the most limiting Small Break LOCA case (as determined by the highest calculated peak cladding temperature), nine break cases were examined for BVPS-1 and BVPS-2. These cases were investigated to capture the most severe postulated Small Break LOCA event. The following discussions provide insight into the analyzed conditions.

Limiting Temperature Conditions

The RCS temperature analyzed was based on a nominal vessel average temperature of 580.0°F. However, the analysis is applicable over the range of 566.2 – 580.0°F. The analysis supports a ±4°F T_{avg} uncertainty. The analysis showed that the 2.75-inch break case is the limiting PCT case for BVPS-1 and the 3-inch break case is the limiting PCT case for BVPS-2. The limiting case transients are discussed below.

Limiting Break Case

The results of Reference 6 demonstrate that the cold leg break location is limiting with respect to postulated cold leg, hot leg and pump suction leg break locations. The PCT results are shown in Table 5.2.2-4A and 5.2.2-5A or Table 5.2.2-4B and Table 5.2.2-5B. Inherent in the Small Break analysis are several input parameters (see Section 5.2.2.2 and Table 5.2.2-1A or Table 5.2.2-1B), while Table 5.2.2-6A or Table 5.2.2-6B provide the key transient event times.

For the EPU Small Break LOCA analysis, the limiting PCT case for BVPS-1 was the 2.75-inch break case and the limiting PCT case for BVPS-2 was the 3-inch break case. A summary of the transient response for the limiting PCT case is shown in Figures 5.2.2-5A or Figure 5.2.2-5B through Figure 5.2.2-15A or Figure 5.2.2-15B. These figures present the response of the following parameters.

- RCS Pressure
- Core Mixture Level
- Core Exit Vapor Temperature
- Broken Loop and Intact Loop Secondary Pressure
- Break Vapor Flow Rate
- Break Liquid Flow Rate
- Broken Loop and Intact Loop Accumulator Flow Rate
- Broken Loop and Intact Loop Pumped Safety Injection Flow Rate
- Peak Clad Temperature
- Hot Spot Fluid Temperature
- Rod Film Heat Transfer Coefficient

Upon initiation of the limiting 2.75-inch break for BVPS-1 and the limiting 3-inch break for BVPS-2, there is an initial rapid depressurization of the RCS followed by an intermediate equilibrium at approximately 1150 psia (see Figure 5.2.2-5A or Figure 5.2.2-5B). The limiting 2.75-inch break depressurizes to the accumulator injection setpoint of 625 psia at approximately 1438 seconds for BVPS-1 (see Figure 5.2.2-11A). The limiting 3-inch break depressurizes to the accumulator injection setpoint of 625 psia at approximately 1082 seconds for BVPS-2 (see Figure 5.2.2-11B). During the initial period of the Small Break transient, the effect of the break flow rate is not sufficient to overcome the flow rate maintained by the reactor coolant pumps as they coast down. As such, normal upward flow is maintained through the core and core heat is adequately removed. Following reactor trip, the removal of the heat generated as a result of fission products decay is accomplished via a two-phase mixture level covering the core. The core mixture level and peak clad temperature transient plots for the limiting break calculations are illustrated in Figures 5.2.2-6A or 5.2.2-6B and 5.2.2-13A or 5.2.2-13B, respectively. These figures show that the peak clad temperature occurs near the time when the core is most deeply uncovered and the top of the core is being cooled by steam. This time is characterized by the highest vapor superheating above the mixture level (refer to Figure 5.2.2-7A or 5.2.2-7B). For BVPS-1 the limiting PCT time-in-life was determined to be 8,000 MWD/MTU. For BVPS-2 the limiting PCT time-in-life was determined to be 6,500 MWD/MTU.

A comparison of the flow provided by the safety injection system to the intact and broken loops can be found in Figure 5.2.2-12A or 5.2.2-12B. The cold leg break vapor and liquid mass flow rates are provided in Figures 5.2.2-9A or 5.2.2-9B and 5.2.2-10A or 5.2.2-10B, respectively. Figures 5.2.2-14A or 5.2.2-14B and 5.2.2-15A or 5.2.2-15B provide additional information on the fluid temperature at the hot spot and hot rod surface heat transfer coefficient at the hot spot, respectively. Figure 5.2.2-8A or 5.2.2-8B depicts the secondary side pressure for both the intact and broken loops for the limiting PCT break case.

Total Oxidation

For the EPU Small Break LOCA analysis, the maximum local oxidation case for BVPS-1 and BVPS-2 was the 2.5-inch break case. The maximum local transient oxidation is 11.07% for BVPS-1 at 20,000 MWD/MTU and 13.42% for BVPS-2 at 15,000 MWD/MTU. The limiting transient oxidation occurs at the burst elevation and includes both outside and post-rupture inside oxidation. Pre-existing (pre-transient) oxidation was also considered and the sum of the pre-transient and transient oxidation remains below 17% at all times in life, for all fuel resident in the core.

Additional Break Cases

Studies documented in Reference 6 have determined that the limiting PCT Small Break transient occurs for breaks of less than 10-inches in diameter in the cold leg. For BVPS-1 and BVPS-2, the limiting PCT is captured by the 1.5- 2-, 2.25-, 2.5-, 2.75-, 3-, 3.25-, 4- and 6-inch break spectrum. The beginning-of-life (BOL) results of these break spectrum cases are given in Table 5.2.2-4A or Table 5.2.2-4B. Figures 5.2.2-16A through 5.2.2-36A address the non-limiting BOL cases (2-, 2.25-, 2.5-, 3-, 3.25-, 4- and 6-inch) analyzed for BVPS-1. Figures 5.2.2-16B through 5.2.2-36B address the non-limiting BOL cases (2-, 2.25-, 2.5-, 2.75-, 3.25-, 4- and 6-inch) analyzed for BVPS-2. The 1.5-inch cases for BVPS-1 and BVPS-2 produced only minimal core uncovery and therefore PCT information was not calculated. Note that plots for the 1.5-inch cases are not included here. The plots for each of the additional non-limiting break cases include:

1. RCS Pressure
2. Core Mixture Level
3. Peak Clad Temperature

For BVPS-1, the PCTs for each of the additional breaks considered are shown in Table 5.2.2-4A and PCTs are less than the limiting 2.75-inch break case. For BVPS-2, the PCTs for the additional breaks are shown in Table 5.2.2-4B and are less than the limiting 3-inch break case. The PCT was not calculated for either BVPS-1 or BVPS-2 for the 1.5-inch case due to the minimal core uncovery.

Transient Termination

The 10 CFR 50.46 criteria continue to be satisfied beyond the end of the calculated transient due to the presence of some or all of the following conditions:

1. The RCS pressure is gradually decreasing.
2. The net mass inventory is increasing.
3. The core mixture level is recovered, or recovering due to increasing mass inventory.
4. As the RCS inventory continues to gradually increase, the core mixture level will continue to increase and the fuel cladding temperatures will continue to decline indicating that the temperature excursion is terminated.

ZIRLO/Zirc-4 Cladding

At the time at which this analysis is implemented, no new Zirc-4 fuel is expected to be inserted into the core. All of the Zirc-4 fuel will be burned for at least one cycle, and ZIRLO™ fuel will be implemented at non-EPU conditions at least one reload cycle before the EPU is implemented. Therefore, the ZIRLO™ fuel is considered limiting with a PCT of 1895.0°F at 8,000 MWD/MTU burnup for BVPS-1 and with a PCT of 1917.1°F at 6,500 MWD/MTU) for BVPS-2. The fuel temperatures/pressures used in these calculations were based on NRC approved fuel performance code PAD 4.0 (Reference 7) which addresses all the helium release related issues. This analysis has been performed using the most limiting temperature/pressure as calculated for 17x17 non-IFBA RFA fuel. The non-IFBA fuel bounds IFBA fuel for Small Break LOCA analyses. Note that the effect of annular pellets, which are only present in the IFBA fuel, was considered in the analyses.

5.2.2.5 Conclusions

The Small Break LOCA analyses considered a break spectrum of 1.5-, 2-, 2.25-, 2.5-, 2.75-, 3-, 3.25-, 4- and 6-inch diameters for BVPS-1 and BVPS-2. For BVPS-1, a peak cladding temperature of 1895°F was calculated at the limiting time-in-life of 8500 MWD/MTU for the 2.75-inch case and a maximum transient oxidation of 11.07% was calculated at the limiting time-in-life of 20,000 MWD/MTU for the 2.5-inch case. For BVPS-2, a peak cladding temperature of 1917°F was calculated at the limiting time-in-life of 6500 MWD/MTU for the 3-inch case and a maximum transient oxidation of 13.42% was calculated at the limiting time-in-life of 15,000 MWD/MTU for the 2.5-inch case.

The analyses presented in this section show that the accumulator and safety injection subsystems of the Emergency Core Cooling System, together with the heat removal capability of the steam generator, provide sufficient core heat removal capability to maintain the calculated peak cladding temperatures for Small Break LOCA below the required limit of 10 CFR 50.46. Furthermore, the analyses show that the local cladding oxidation and core wide average oxidation, including consideration of pre-existing and post-LOCA oxidation, and cladding outside and post-rupture inside oxidation, are less than the 10 CFR 50.46 (Reference 1) limits.

Table 5.2.2-7 provides a results summary for the BVPS-1 and BVPS-2 SBLOCA EPU analyses. Results include PCT, maximum local oxidation and total hydrogen generation.

The results and conclusions of the analyses performed for Small Break LOCA for the reactor power of 2900 MWt (2910 MWt NSSS power) bound and support operation at the current reactor power of 2689 MWt (2697 MWt NSSS power), thus supporting the staged implementation of EPU at BVPS-1 and BVPS-2.

5.2.2.6 References

1. "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors," 10 CFR 50.46 and Appendix K of 10 CFR 50, Federal Register, Volume 39, Number 3, January 1974, as amended in Federal Register, Volume 53, September 1988.

2. Meyer, P. E., "NOTRUMP - A Nodal Transient Small Break and General Network Code," WCAP-10079-P-A, (proprietary) and WCAP-10080-NP-A (non-proprietary), August 1985.

3. Lee, N. et al., "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," WCAP-10054-P-A (proprietary) and WCAP-10081-NP-A (non-proprietary), August 1985.

4. Thompson, C. D. et al., "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, Rev. 1 (proprietary), July 1997.

5. "Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in Westinghouse - Designed Operating Plant," NUREG-0611, January 1980.

6. Rupprecht, S. D. et al., "Westinghouse Small Break LOCA ECCS Evaluation Model Generic Study with the NOTRUMP Code," WCAP-11145-P-A (proprietary), October 1986.

7. Slagle, W. H., (ed.) et al., "Westinghouse Improved Performance Analysis and Design Model (PAD 4.0)," WCAP-15063-P-A, Revision 1, July 2000.

Table 5.2.2-1A BVPS-1 Input Parameters Used in the Small Break LOCA Analysis	
Input Parameter	**Value**
Core Rated Thermal Power-100%	2900
Calorimetric Uncertainty, %	0.6
Fuel Type	17 X 17 Robust Fuel Assembly (RFA)
Total Core Peaking Factor, F_Q	2.40
Hot Channel Enthalpy Rise Factor, $F_{\Delta H}$	1.62
Hot Assembly Average Power Factor, P_{HA}	1.42
Maximum Axial Offset, %	+13
Initial RCS Loop Flow, gpm/loop	82,840
Initial Vessel T_{avg}, °F	Max: 580.0 Min: 566.2
Initial Pressurizer Pressure (plus uncertainties), psia	2300
Reactor Coolant Pump Type	Model 93A with Weir
Pressurizer Low-Pressure Reactor Trip Setpoint, psia	1935
Reactor Trip Signal Delay Time, seconds	2.0
Rod Drop Delay Time, seconds	2.7
Auxiliary Feedwater Temperature (Maximum), °F	120
Number of AFW Pumps Available Following a LOOP	1 Motor Driven
AFW Flow (Minimum) to all 3 Steam Generators, gpm	294 (98 gpm/SG * 3) at 1107 psig
AFW Flow Delay Time (Maximum), seconds	60
AFW Actuation Signal	Pressurizer Low-Pressure Safety Injection
Steam Generator Type	Model 54F
Maximum AFW Piping Purge Volume, ft^3	168
Steam Generator Tube Plugging (Maximum), %	10
Maximum MFW Isolation Signal Delay Time, seconds	3
MFW Control Valve Isolation Ramp Time, seconds	7
MFW Isolation Signal	Pressurizer Low-Pressure Safety Injection
Isolation of Steam Line Signal	Pressurizer Low-Pressure Reactor Trip/LOOP
Steam Generator Secondary Water Mass, lbm/SG	99,930
Containment Spray Flowrate for 2 Pumps, gpm	4983 (plus 981 gpm to account for flow to the sump)
RWST Deliverable Volume (Minimum), gallons	317,000

Table 5.2.2-1A (continued) BVPS-1 Input Parameters Used in the Small Break LOCA Analysis	
Input Parameter	**Value**
SI Temp at Cold Leg Recirculation Time (Maximum), °F	190
ECCS Configuration	1 HHSI pump, faulted line injects to RCS pressure (1.5-inch – 4-inch breaks) 1 HHSI pump, 1 LHSI pump, no ECCS in the faulted loop (6-inch break)
ECCS Water Temperature (Maximum), °F	65
Pressurizer Low-Pressure Safety Injection Setpoint, psia	1745
SI Flow Delay Time, seconds	27
ECCS Flow vs. Pressure	See Tables 5.2.2-2 and 5.2.2-3
Initial Accumulator Water/Gas Temperature, °F	105
Initial Nominal Accumulator Water Volume, ft^3	957
Minimum Accumulator Pressure, psia	625

Table 5.2.2-1B BVPS-2 Input Parameters Used in the Small Break LOCA Analysis	
Input Parameter	**Value**
Core Rated Thermal Power-100%	2900
Calorimetric Uncertainty, %	0.6
Fuel Type	17 X 17 Robust Fuel Assembly (RFA)
Total Core Peaking Factor, F_Q	2.40
Hot Channel Enthalpy Rise Factor, $F_{\Delta H}$	1.62
Hot Assembly Average Power Factor, P_{HA}	1.42
Maximum Axial Offset, %	+13
Initial RCS Loop Flow, gpm/loop	82,840
Initial Vessel T_{avg}, °F	Max: 580.0 Min: 566.2
Initial Pressurizer Pressure (plus uncertainties), psia	2300
Reactor Coolant Pump Type	Model 93A with Weir
Pressurizer Low-Pressure Reactor Trip Setpoint, psia	1935
Reactor Trip Signal Delay Time, seconds	2.0
Rod Drop Delay Time, seconds	2.7
Auxiliary Feedwater Temperature (Maximum), °F	120
Number of AFW Pumps Available Following a LOOP	1 Motor Driven
AFW Flow (Minimum) to all 3 Steam Generators, gpm	294 (98 gpm/SG *3) at 1107 psig
AFW Flow Delay Time (Maximum), seconds	60
AFW Actuation Signal	Pressurizer Low-Pressure Safety Injection
Steam Generator Type	Model 51M
Maximum AFW Piping Purge Volume, ft^3	125.7
Steam Generator Tube Plugging (Maximum), %	22
Maximum MFW Isolation Signal Delay Time, seconds	2
MFW Control Valve Isolation Ramp Time, seconds	5
MFW Isolation Signal	Pressurizer Low-Pressure Safety Injection
Isolation of Steam Line Signal	Pressurizer Low-Pressure Reactor Trip/LOOP
Steam Generator Secondary Water Mass, lbm/SG	99,500
Containment Spray Flowrate for 2 Pumps, gpm	4450
RWST Deliverable Volume (Minimum), gallons	403,000

Table 5.2.2-1B (continued) BVPS-2 Input Parameters Used in the Small Break LOCA Analysis	
Input Parameter	**Value**
SI Temp at Cold Leg Recirculation Time (Maximum), °F	212
ECCS Configuration	1 HHSI pump, faulted line injects to RCS pressure (1.5-inch – 4-inch breaks) 1 HHSI pump, 1 LHSI pump, no ECCS in the faulted loop (6-inch break)
ECCS Water Temperature (Maximum), °F	65
Pressurizer Low-Pressure Safety Injection Setpoint, psia	1760
SI Flow Delay Time, seconds	27
ECCS Flow vs. Pressure	See Tables 5.2.2-2 and 5.2.2-3
Initial Accumulator Water/Gas Temperature, °F	105
Initial Nominal Accumulator Water Volume, ft^3	997
Minimum Accumulator Pressure, psia	625

Table 5.2.2-2 Safety Injection Flows Used in the Small Break LOCA Analysis (1 HHSI pump, faulted loop injects to RCS pressure – 1.5-inch - 4-inch breaks for BVPS-1 and BVPS-2)		
RCS Pressure (psia)	**Intact Loop (lbm/sec)**	**Broken Loop (lbm/sec)**
314.7	37.59	20.28
414.7	36.63	19.79
514.7	35.56	19.17
614.7	34.45	18.61
714.7	33.42	18.06
814.7	32.34	17.50
914.7	31.25	16.88
1014.7	30.14	16.25
1114.7	29.03	15.70
1214.7	27.92	15.07
1314.7	26.67	14.45
1414.7	25.28	13.61
1514.7	23.85	12.92
1614.7	22.43	12.08
1714.7	20.97	11.39
1814.7	19.50	10.56

Table 5.2.2-3
Safety Injection Flows Used in the Small Break LOCA Analysis
(1 HHSI pump, 1 LHSI pump, no ECCS in the faulted loop because the break is postulated along the HHSI line – 6-inch breaks for BVPS-1 and BVPS-2)

RCS Pressure (psia)	BVPS-1 Intact Loop (lbm/sec)	BVPS-2 Intact Loop (lbm/sec)
14.7	336.34	376.26
24.7	313.32	353.74
34.7	290.45	330.07
64.7	216.34	250.07
104.7	-	60.64
109.7	-	34.35
114.7	45.12	34.35
119.7	34.73	-
164.7	-	34.35
214.7	34.73	34.35
314.7	32.43	32.23
414.7	30.14	30.14
514.7	27.78	27.99
614.7	25.42	25.7
714.7	22.92	23.41
814.7	20.42	20.97
914.7	17.78	18.61
1014.7	15.00	16.11
1114.7	12.22	13.47
1214.7	9.31	10.83
1314.7	6.39	7.92
1414.7	3.06	5.0
1514.7	0.0	-

Table 5.2.2-4A BVPS-1 SBLOCTA BOL Results								
Break Size (in)	2	2.25	2.5	2.75	3	3.25	4	6
PCT (°F)	1723.1	1804.8	1793.2	1839.2	1777.6	1617.0	1334.2	1267.9
PCT Time (s)	3160.3	2417.0	2209.8	1743.7	1386.4	1215.8	780.2	2209.2
PCT Elevation (ft)	12	12	12	12	11.75	11.5	11.25	11.5
Max. Local ZrO_2 (%)	3.15	3.6	5.04	4.28	2.87	1.19	0.14	0.13
Max. Local ZrO_2 Elev. (ft)	12	12	12	12	11.75	11.5	11.25	11.5
Core-Wide Avg. ZrO_2 (%)	0.41	0.48	0.64	0.56	0.4	0.18	0.02	0.02

Table 5.2.2-5A BVPS-1 SBLOCTA Limiting Results from the 2.5-inch (Transient Oxidation) and 2.75-inch (PCT) Time-in-Life Study		
Break Size	2.5	2.75
Time-in-Life (MWD/MTU)	20,000	8,000
PCT (°F)	1796	1895.0
PCT Time (s)	2212.9	1723.7
PCT Elevation (ft)	12	12
Hot Rod Burst Time (s)	1770.2	1721.7
Hot Rod Burst Elevation (ft)	11.75	12
Max. Local Transient ZrO_2 (%)	11.07	8.82
Max. Local Transient ZrO_2 Elev. (ft)	11.75	12
Core-Wide Avg. ZrO_2 (%)	0.62	0.52

Table 5.2.2-4B BVPS-2 SBLOCTA BOL Results								
Break Size (in)	2	2.25	2.5	2.75	3	3.25	4	6
PCT (°F)	1752.9	1846.2	1838.8	1829.3	1852.6	1711.8	1455.7	899.9
PCT Time (s)	3220.2	2317.6	2118.9	1680.8	1329.4	1151.1	723.5	3274.6
PCT Elevation (ft)	12	12	12	12	12	11.75	11.5	11.25
Max. Local ZrO_2 (%)	3.83	4.3	5.95	4.74	3.74	1.9	0.37	0.01
Max. Local ZrO_2 Elev. (ft)	12	12	12	12	11.75	11.75	11.25	11.25
Core-Wide Avg. ZrO_2 (%)	0.48	0.56	0.74	0.62	0.52	0.29	0.06	0.00

Table 5.2.2-5B BVPS-2 SBLOCTA Limiting Results from the 2.5-inch (Transient Oxidation) and 3-inch (PCT) Time-in-Life Study		
Break Size	2.5	3
Time-in-Life (MWD/MTU)	15,000	6,500
PCT (°F)	1845.1	1917.1
PCT Time (s)	2118.9	1316.6
PCT Elevation (ft)	12	12
Hot Rod Burst Time (s)	1725.2	1314.4
Hot Rod Burst Elevation (ft)	11.75	12
Max. Local Transient ZrO_2 (%)	13.42	7.79
Max. Local Transient ZrO_2 Elev. (ft)	11.75	12
Core-Wide Avg. ZrO_2 (%)	0.77	0.49

Table 5.2.2-6A
BVPS-1 NOTRUMP Results

Event Time (sec)	1.5-inch	2-inch	2.25-inch	2.5-inch	2.75-inch	3-inch	3.25-inch	4-inch	6-inch
Break Initiation	0	0	0	0	0	0	0	0	0
Reactor Trip Signal	54.1	29.0	22.5	17.9	14.6	12.3	10.6	7.3	4.4
S-Signal	75.9	42.4	33.9	27.9	23.8	20.8	18.6	14.4	10.2
SI Flow Delivered	102.9	69.4	60.9	54.9	50.8	47.8	45.6	41.4	37.2
Loop Seal Clearing(1)	1840	930	725	656	484	414	358	241	62
Core Uncovery	(3)	1020	813	658	672	526	362	233	121
Accumulator Injection	N/A	4017	2378	1821	1438	1138	996	637	291
RWST Volume Delivered	3033	3025	3017	3011	3006	3001	2998	2992	N/A
PCT Time (BOL)	–	3160.3	2417.0	2209.8	1743.7	1386.4	1215.8	780.2	2209.2
Core Recovery	N/A	(2)	(2)	(2)	(2)	(2)	(2)	(2)	(2)

Notes:

(1) Loop seal clearing is defined as break vapor flow ≥ 1 lb/s.

(2) For the cases where core recovery is > TMAX, basis for transient termination can be concluded based on some or all of the following: (1) The RCS system pressure is decreasing which will increase SI flow, (2) Total RCS system mass is increasing due to SI flow exceeding break flow, and (3) Core mixture level has begun to increase and is expected to continue for the remainder of the accident.

(3) It has been judged that no core uncovery of any consequence will take place and the 1.5-inch case is non-limiting. Therefore no PCT calculations were performed.

Table 5.2.2-6B
BVPS-2 NOTRUMP Results

Event Time (sec)	1.5-inch	2-inch	2.25-inch	2.5-inch	2.75-inch	3-inch	3.25-inch	4-inch	6-inch
Break Initiation	0	0	0	0	0	0	0	0	0
Reactor Trip Signal	57.7	30.6	23.8	18.9	15.3	12.6	10.9	7.5	4.4
S-Signal	74.7	42.1	33.8	28.1	24.1	20.9	18.9	15.3	11.1
SI Flow Delivered	101.7	69.1	60.8	55.1	51.1	47.9	45.9	42.3	38.1
Loop Seal Clearing(1)	1887	910	720	549	452	372	384	207	110
Core Uncovery	(3)	1169	882	689	577	611	560	280	252
Accumulator Injection	N/A	3494	2278	1714	1340	1082	915	557	261
RWST Volume Delivered	5076.7	5039.8	5025.5	5014.3	5004.7	4998.0	4991.8	N/A	N/A
PCT Time	--	3220.2	2317.6	2118.9	1680.8	1329.4	1151.1	723.5	3274.6
Core Recovery	N/A	5687	5599	(2)	(2)	(2)	(2)	(2)	3564

Notes:

(1) Loop seal clearing is defined as break vapor flow > 1 lb/s.

(2) For the cases where core recovery is > TMAX, basis for transient termination can be concluded based on some or all of the following: (1) The RCS system pressure is decreasing which will increase SI flow, (2) Total RCS system mass is increasing due to SI flow exceeding break flow, and (3) Core mixture level has begun to increase and is expected to continue for the remainder of the accident.

(3) It has been judged that no core uncovery of any consequence will take place and the 1.5-inch case is non-limiting. Therefore no PCT calculations were performed.

Table 5.2.2-7 BVPS-1 and BVPS-2 SBLOCA Results Summary		
	BVPS-1	BVPS-2
Peak Cladding Temperature (°F)	1895	1917
Maximum Local Transient Oxidation (%)	11.07	13.42
Total Hydrogen Generation (%)	<1%	<1%

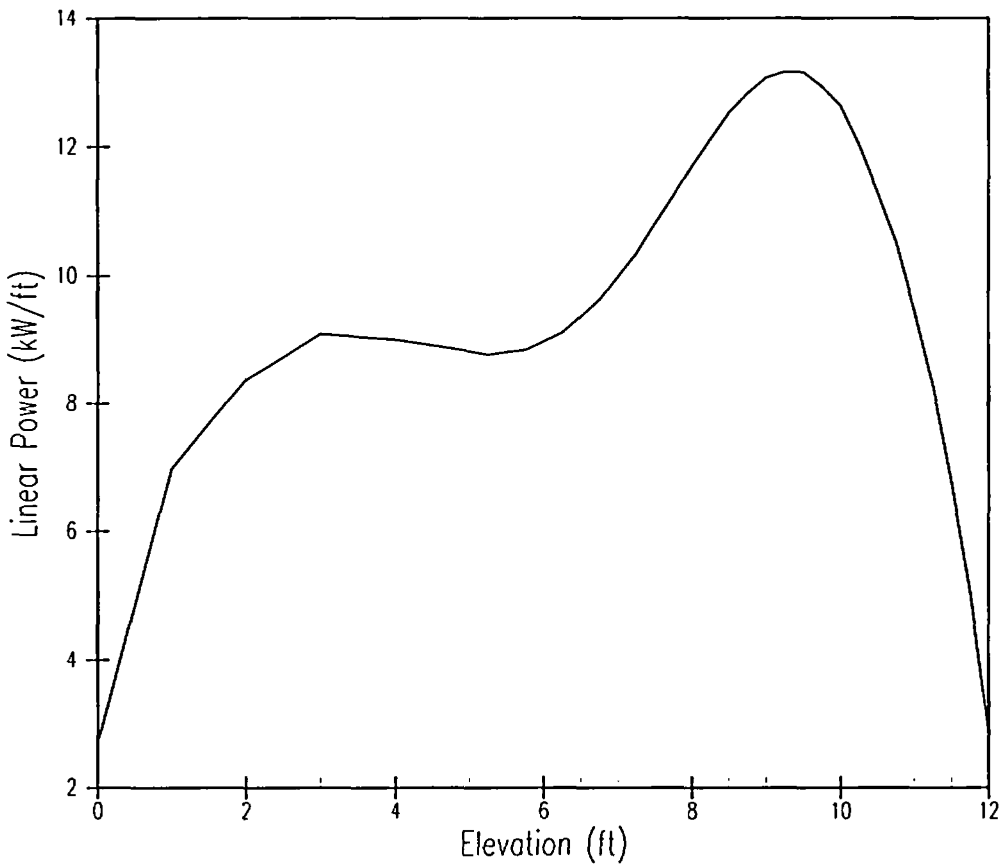

Figure 5.2.2-1
Small Break Hot Rod Power Shape

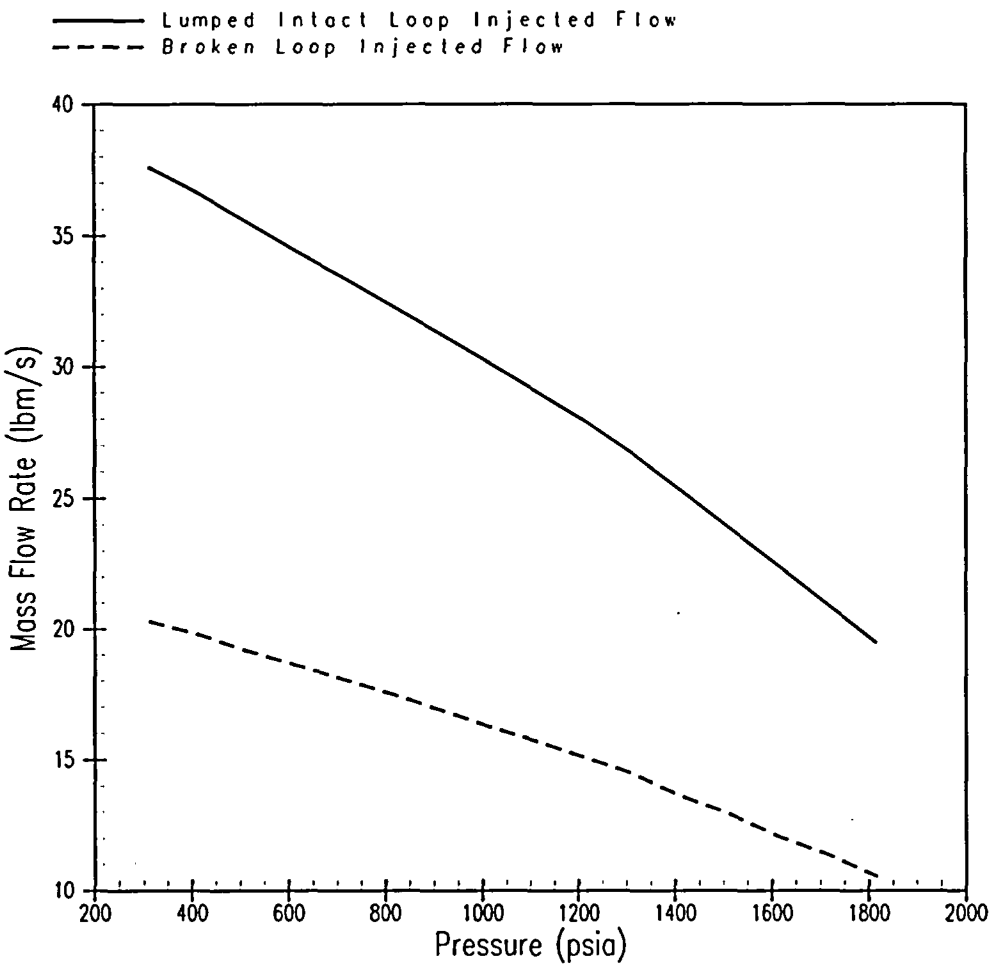

Figure 5.2.2-2
Small Break LOCA Safety Injection Flows
(1 HHSI pump, faulted loop injects to RCS pressure – 1.5-inch - 4-inch breaks for BVPS-1 and BVPS-2)

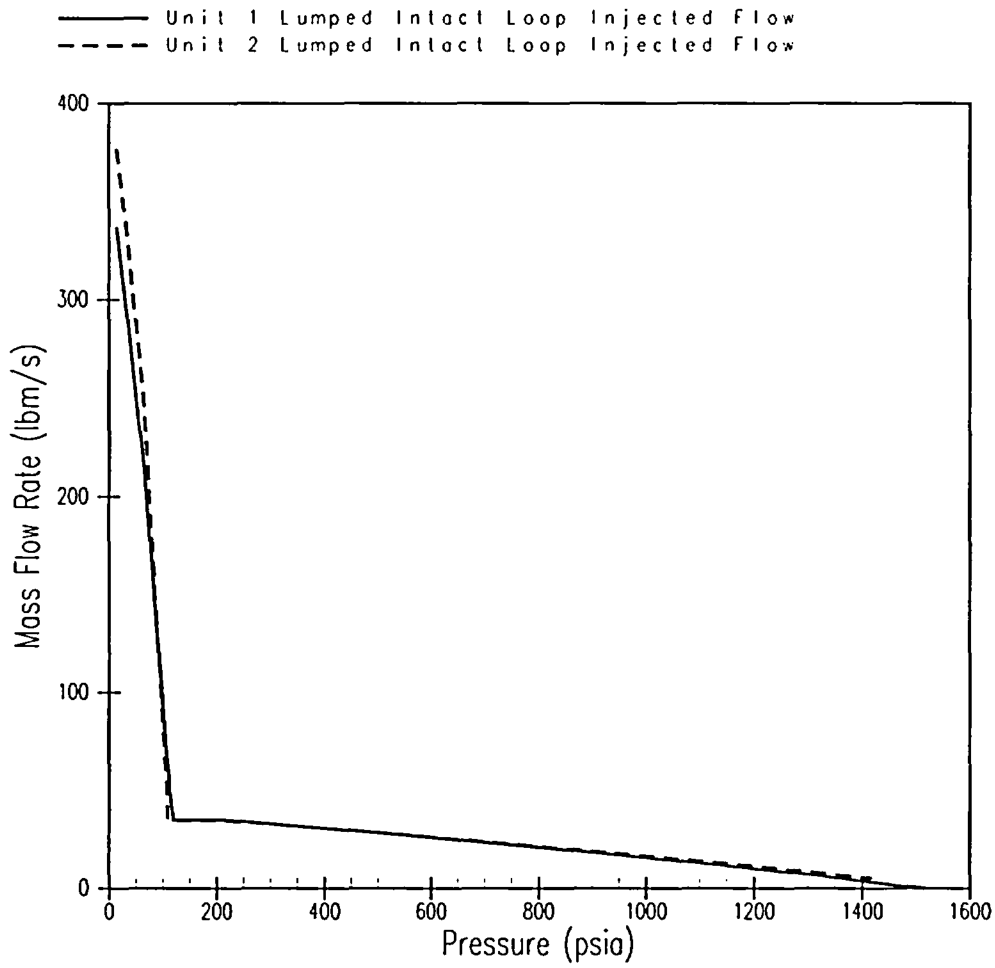

Figure 5.2.2-3
Small Break LOCA Safety Injection Flows
(1 HHSI pump, 1 LHSI pump, no ECCS in the faulted loop because the break is postulated along the HHSI line – 6-inch breaks for BVPS-1 and BVPS-2)

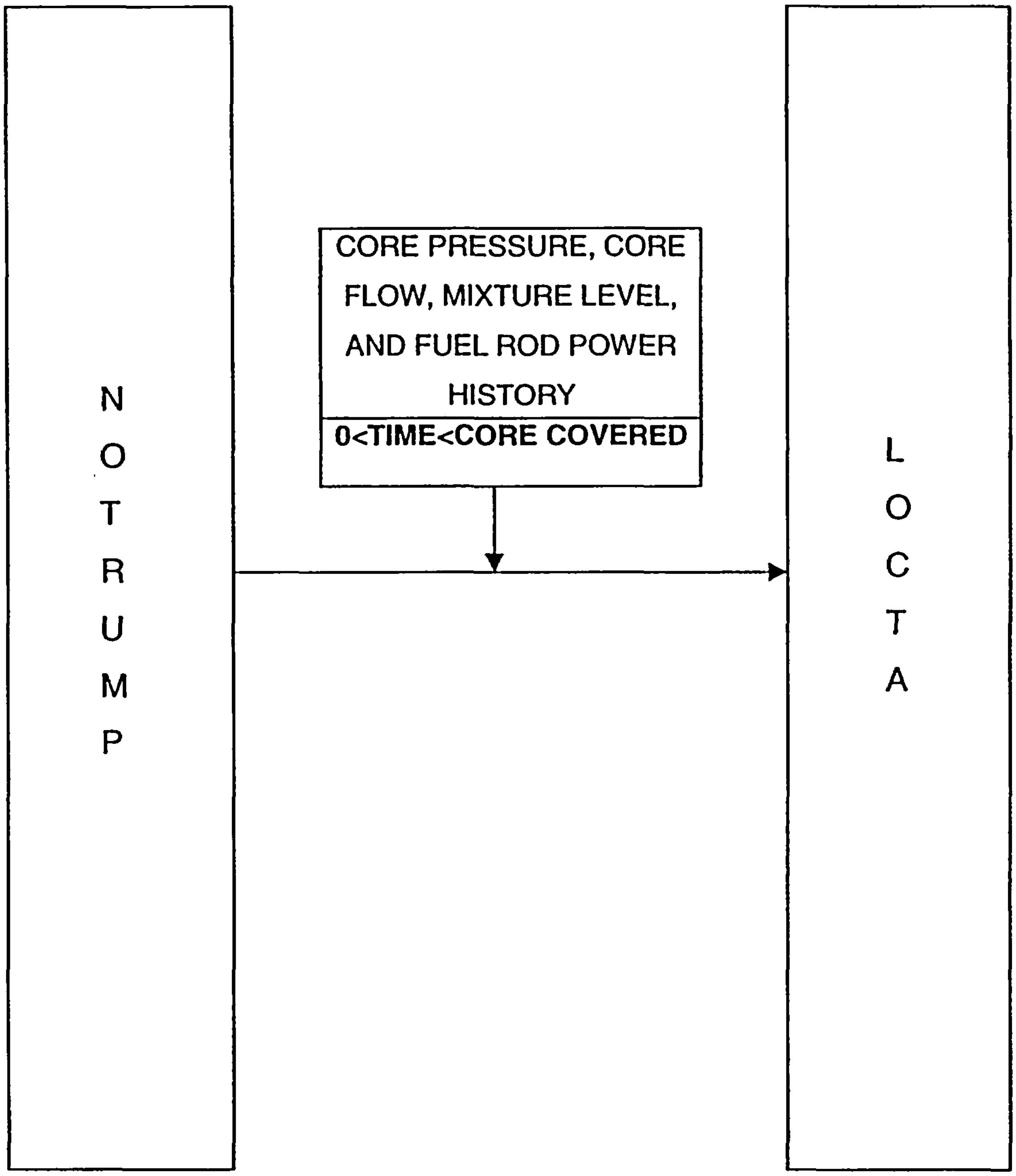

Figure 5.2.2-4
Code Interface Description
for Small Break Model

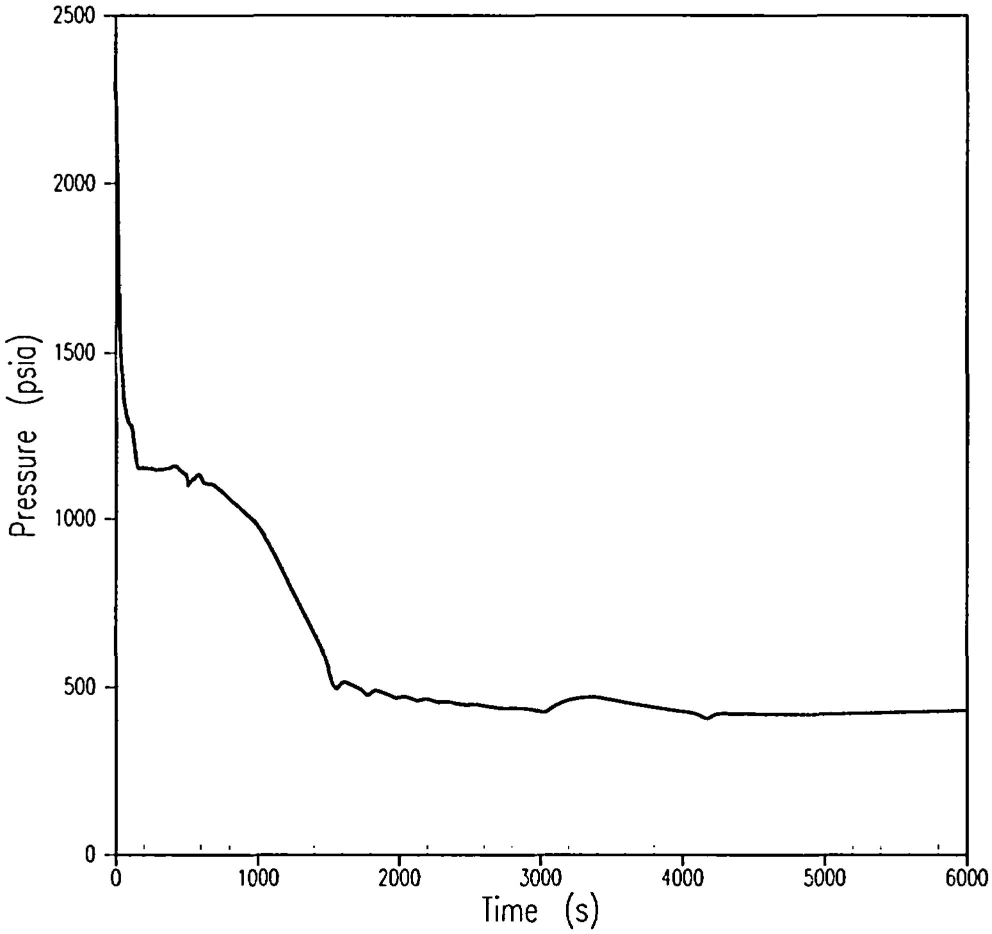

Figure 5.2.2-5A
BVPS-1 2.75-inch Break
RCS Pressure

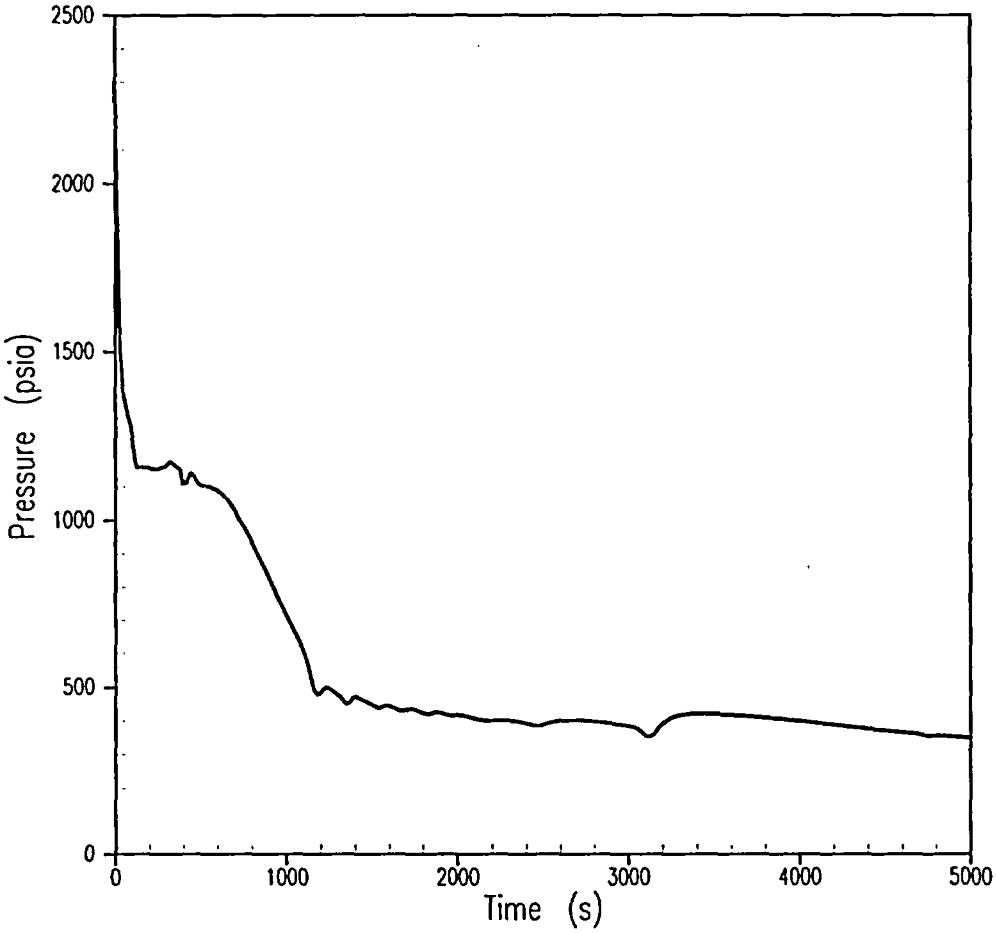

Figure 5.2.2-5B
BVPS-2 3-inch Break
RCS Pressure

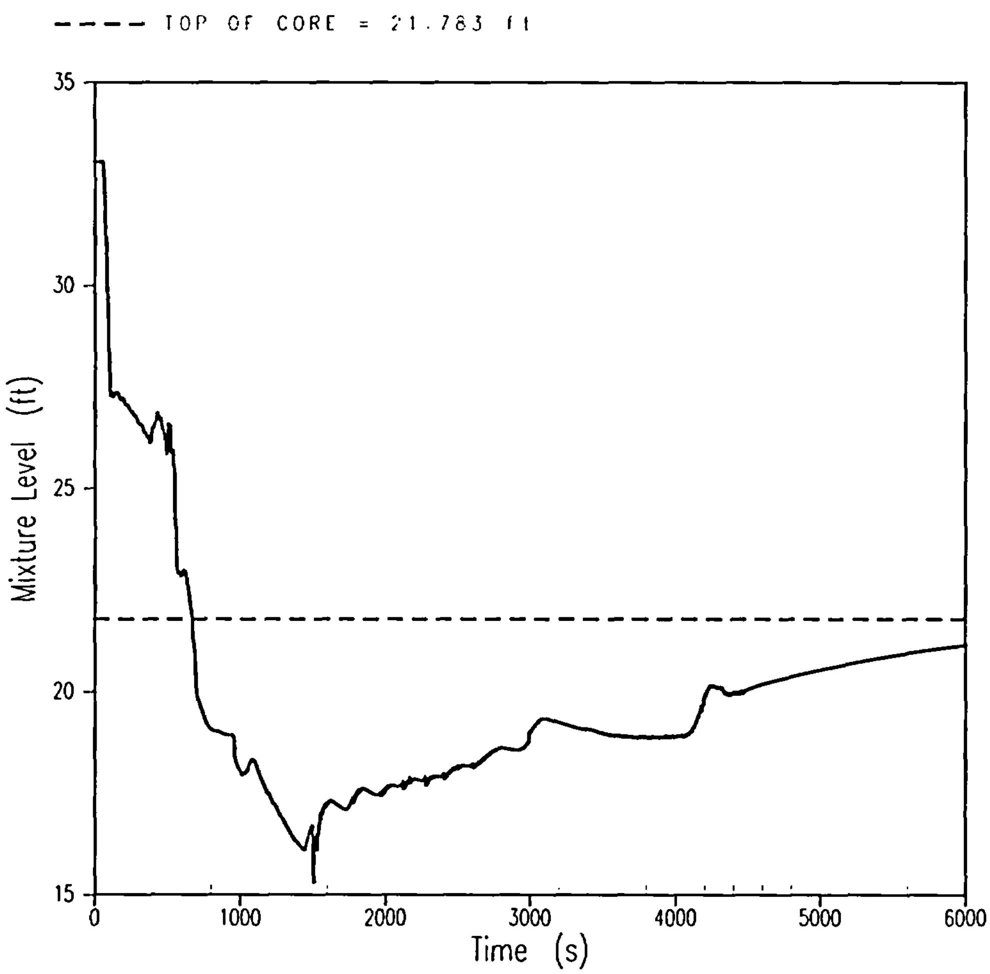

Figure 5.2.2-6A
BVPS-1 2.75-inch Break
Core Mixture Level

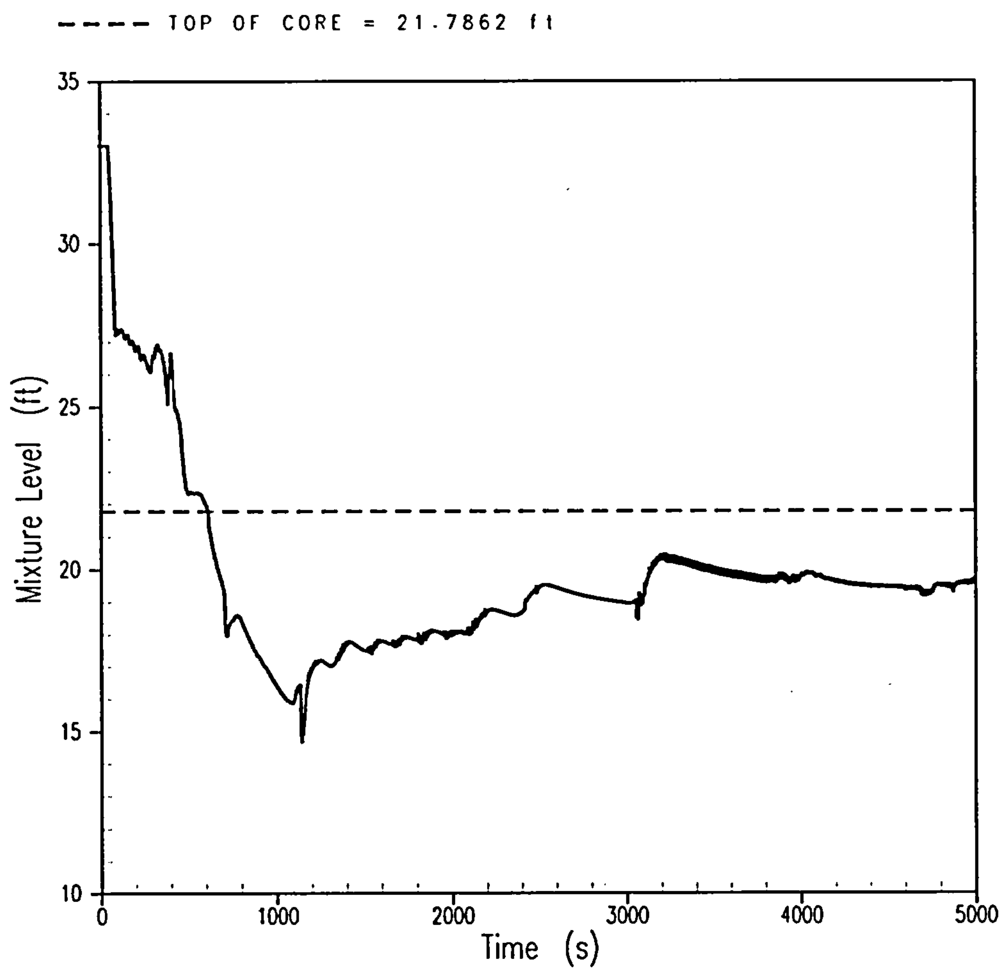

Figure 5.2.2-6B
BVPS-2 3-inch Break
Core Mixture Level

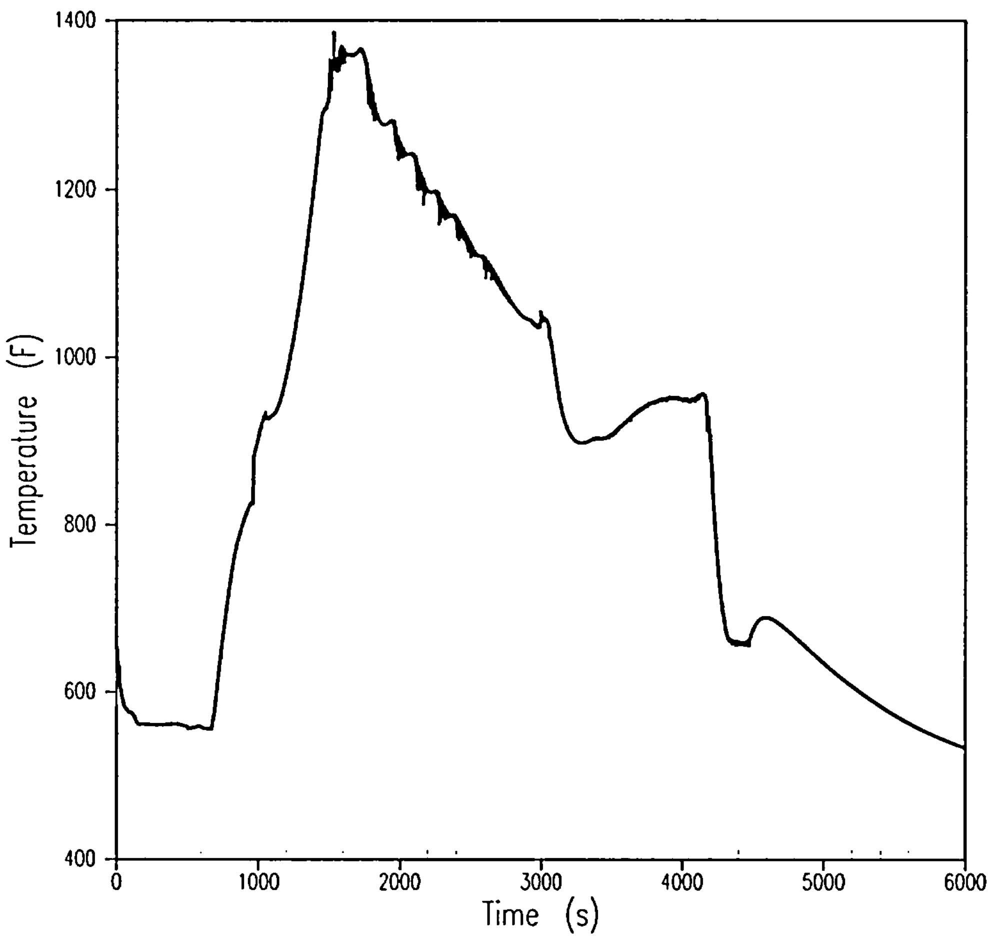

Figure 5.2.2-7A
BVPS-1 2.75-inch Break
Core Exit Vapor Temperature

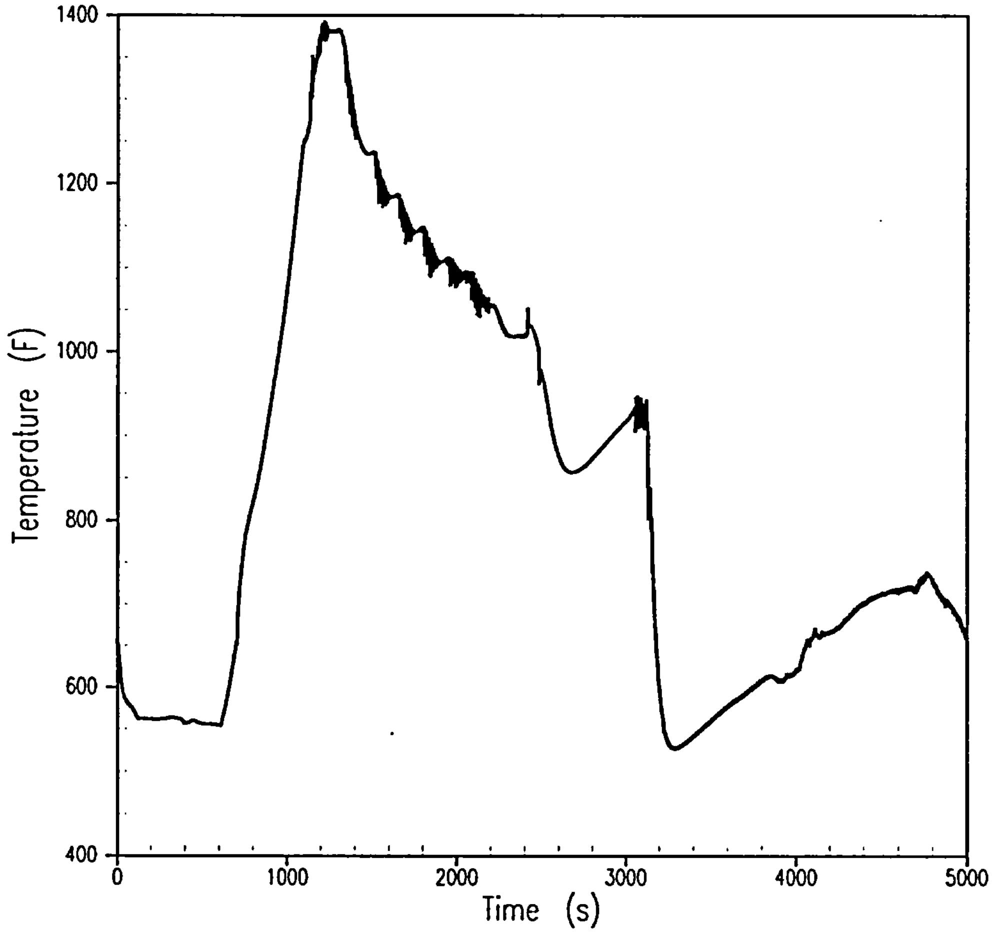

Figure 5.2.2-7B
BVPS-2 3-inch Break
Core Exit Vapor Temperature

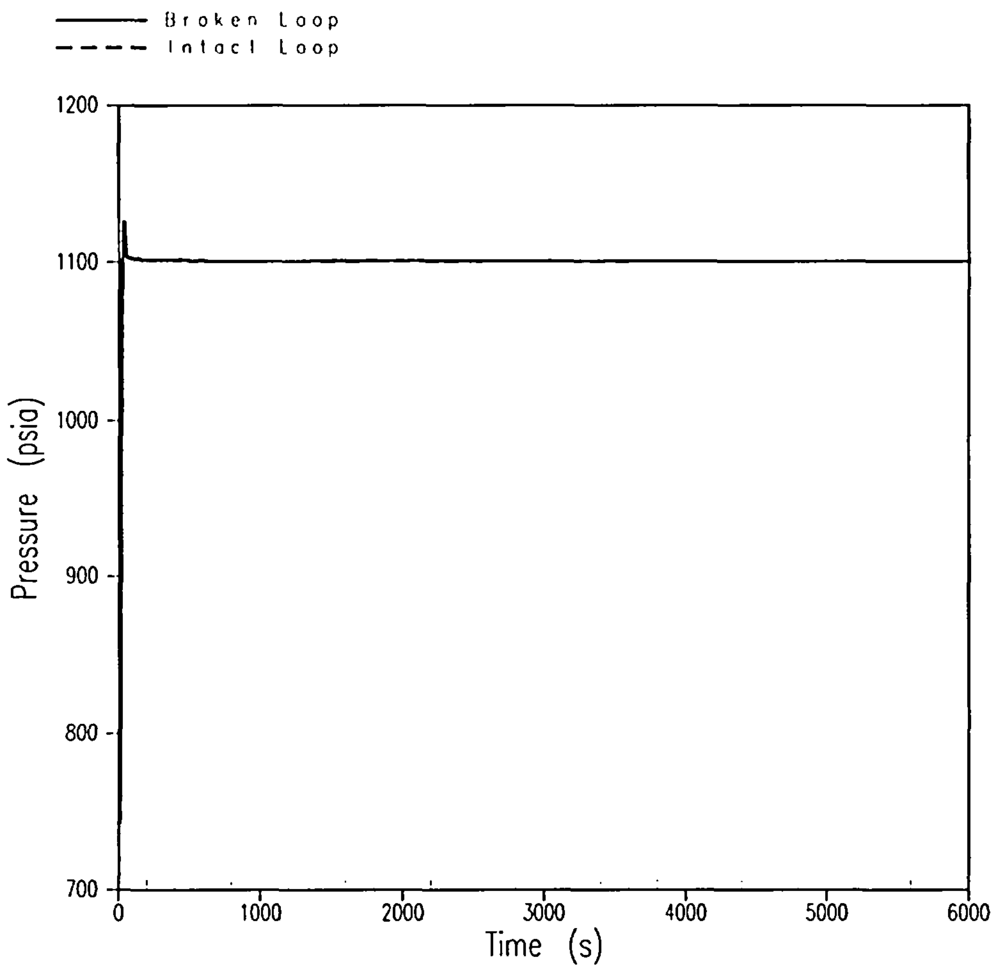

Figure 5.2.2-8A
BVPS-1 2.75-inch Break
Broken Loop and Intact Loop Secondary Pressure

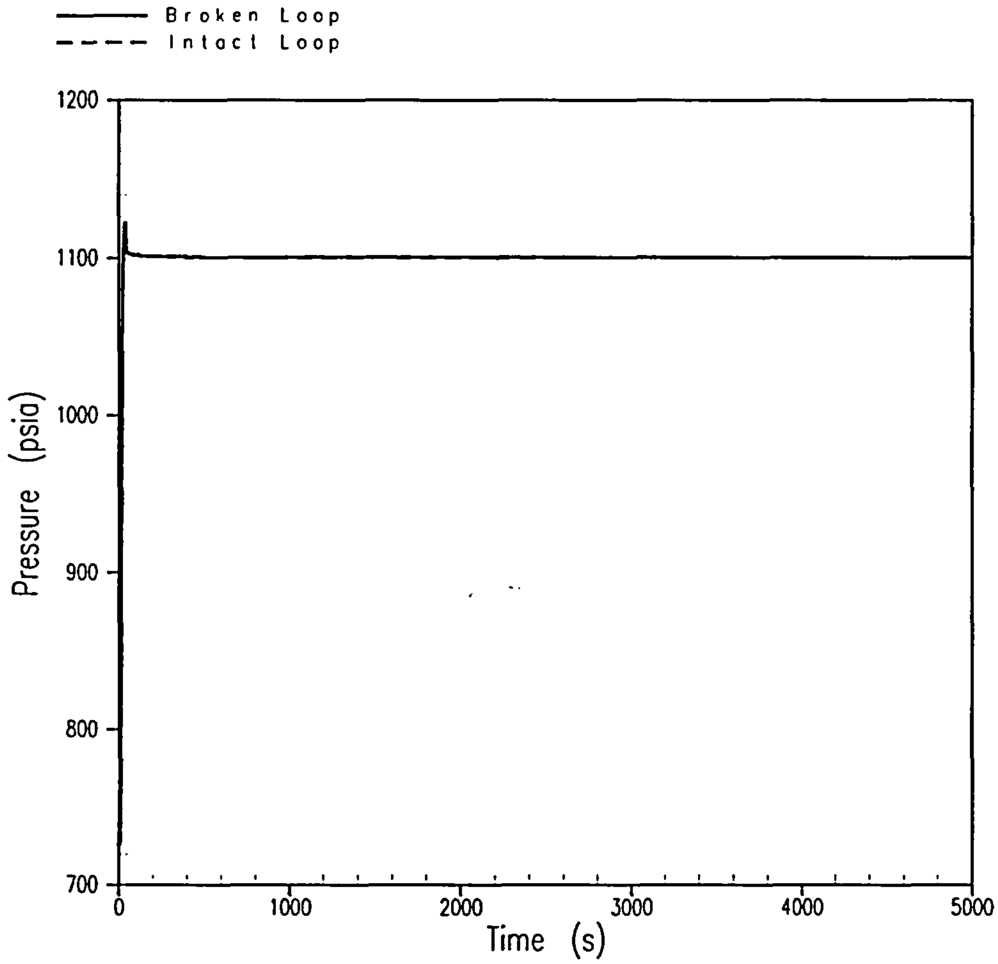

Figure 5.2.2-8B
BVPS-2 3-inch Break
Broken Loop and Intact Loop Secondary Pressure

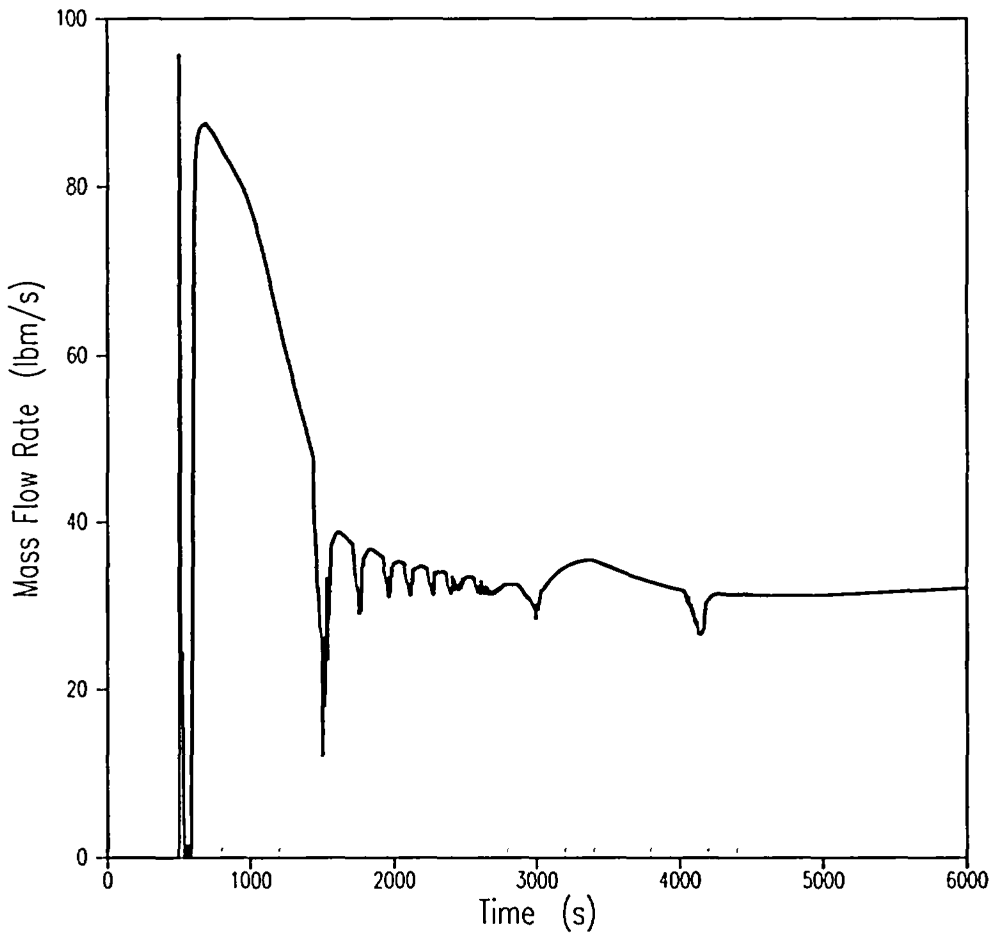

Figure 5.2.2-9A
BVPS-1 2.75-inch Break
Break Vapor Flow Rate

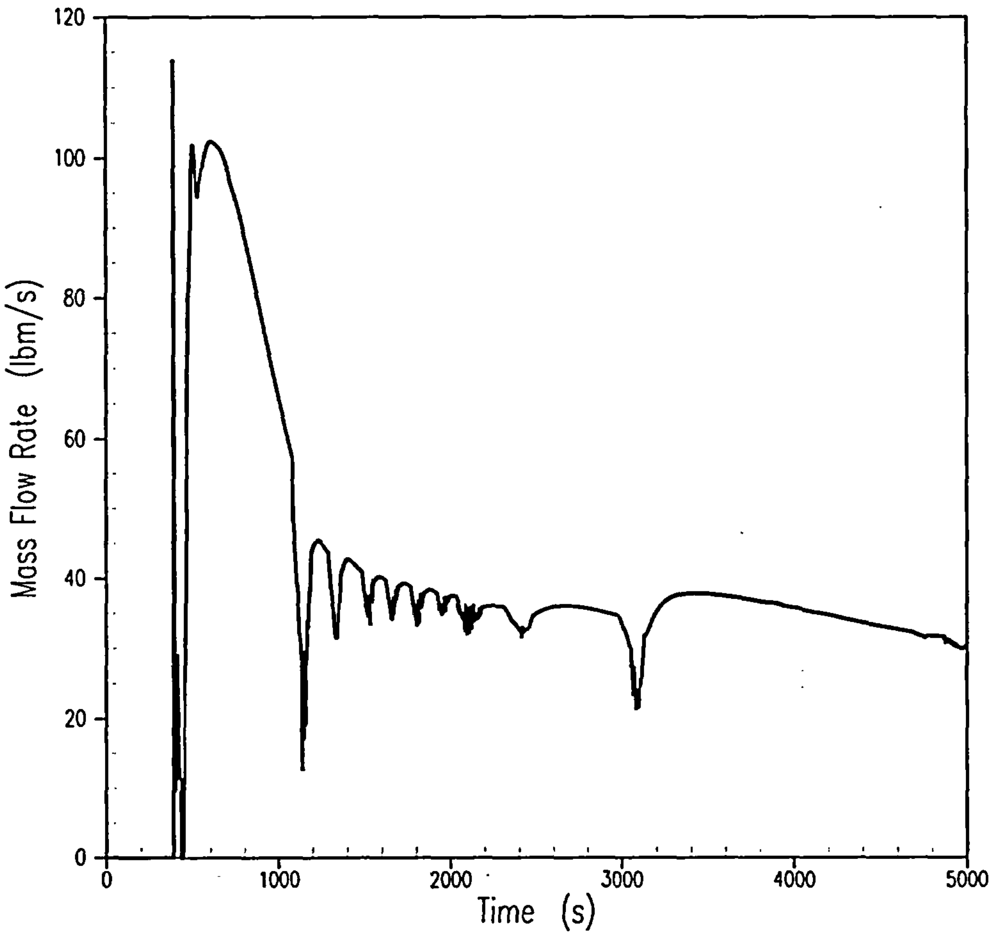

Figure 5.2.2-9B
BVPS-2 3-inch Break
Break Vapor Flow Rate

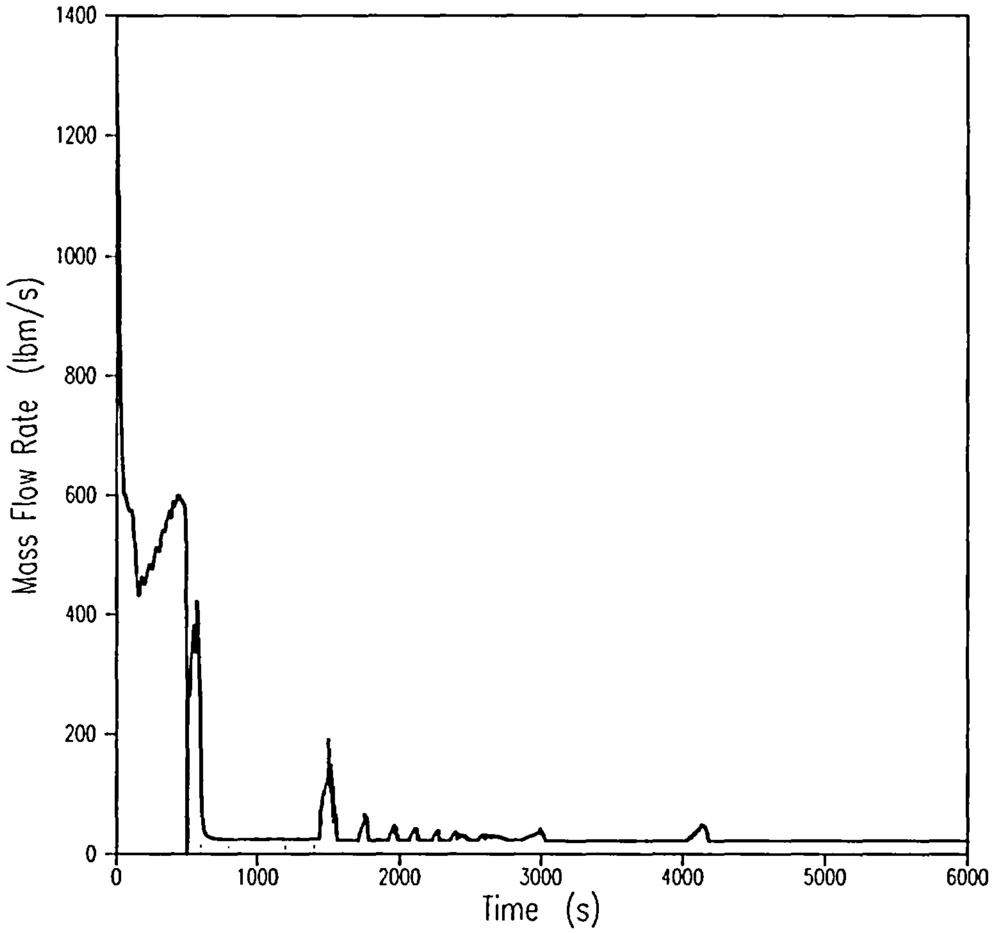

Figure 5.2.2-10A
BVPS-1 2.75-inch Break
Break Liquid Flow Rate

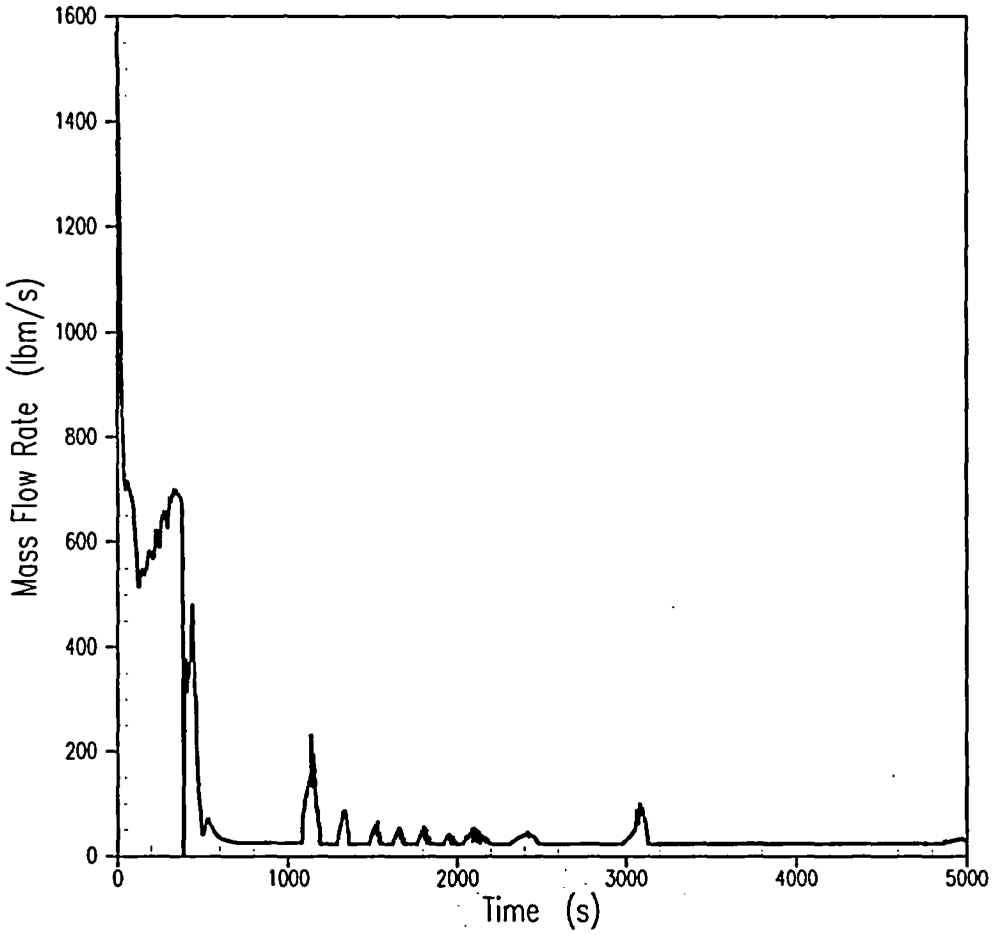

Figure 5.2.2-10B
BVPS-2 3-inch Break
Break Liquid Flow Rate

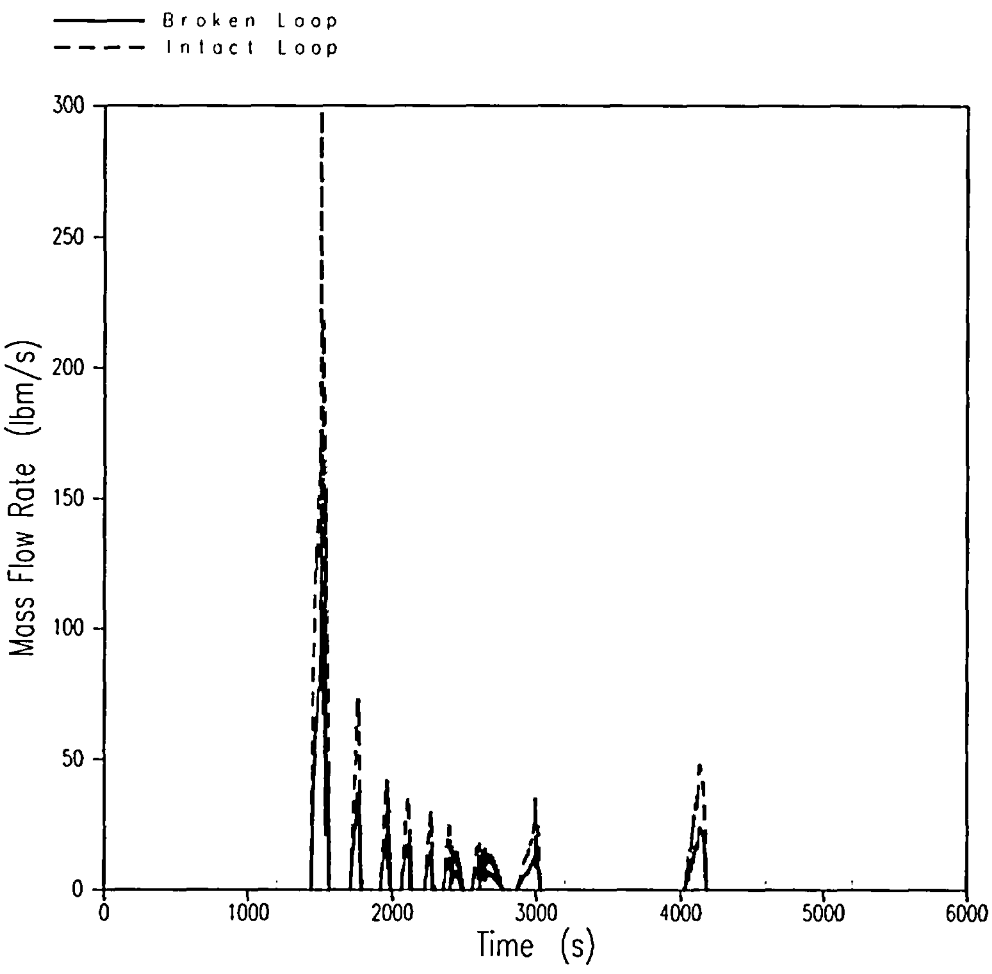

Figure 5.2.2-11A
BVPS-1 2.75-inch Break
Broken Loop and Intact Loop Accumulator Flow Rate

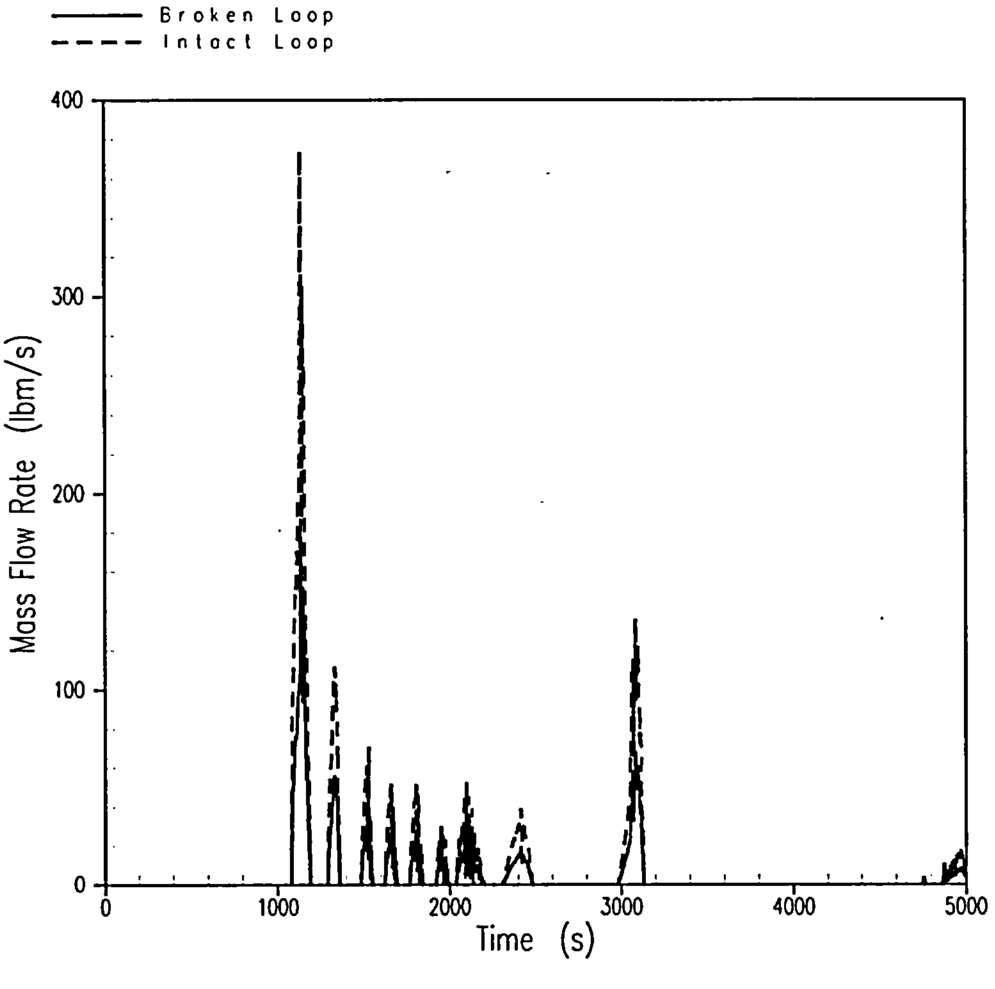

Figure 5.2.2-11B
BVPS-2 3-inch Break
Broken Loop and Intact Loop Accumulator Flow Rate

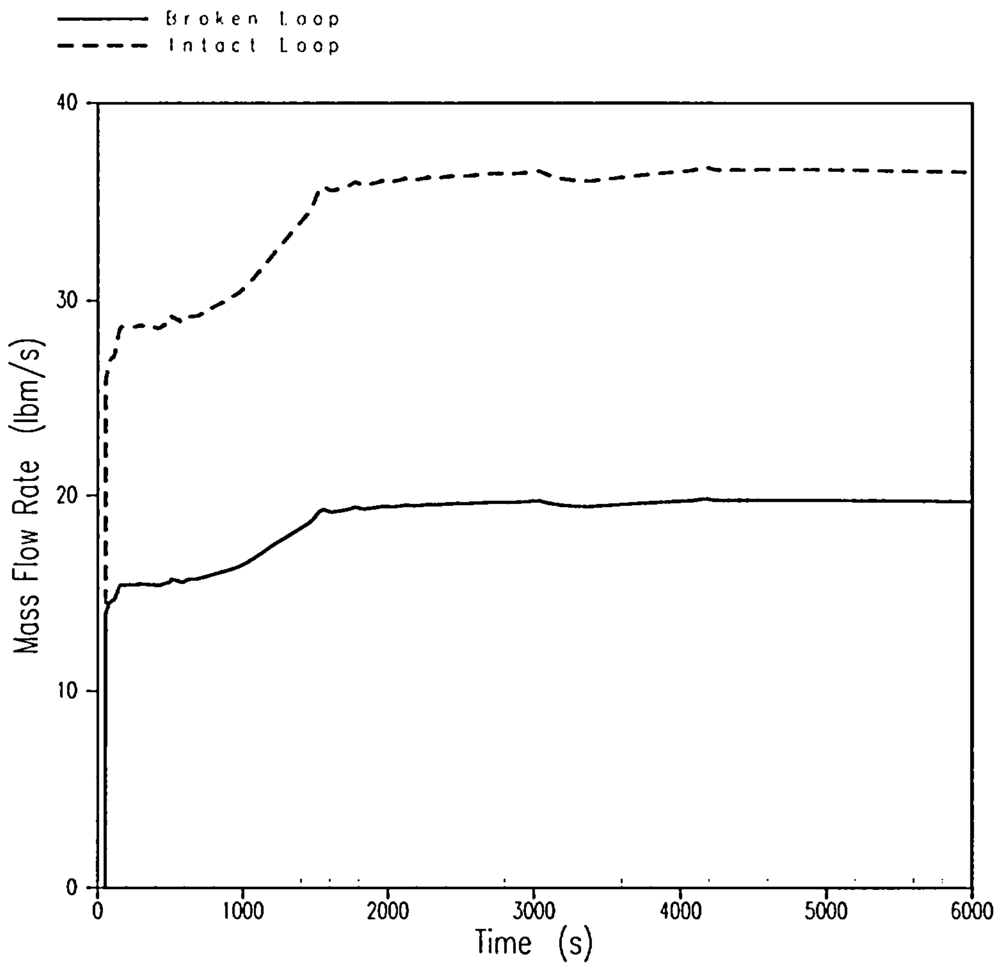

Figure 5.2.2-12A
BVPS-1 2.75-inch Break
Broken Loop and Intact Loop Pumped Safety Injection Flow Rate

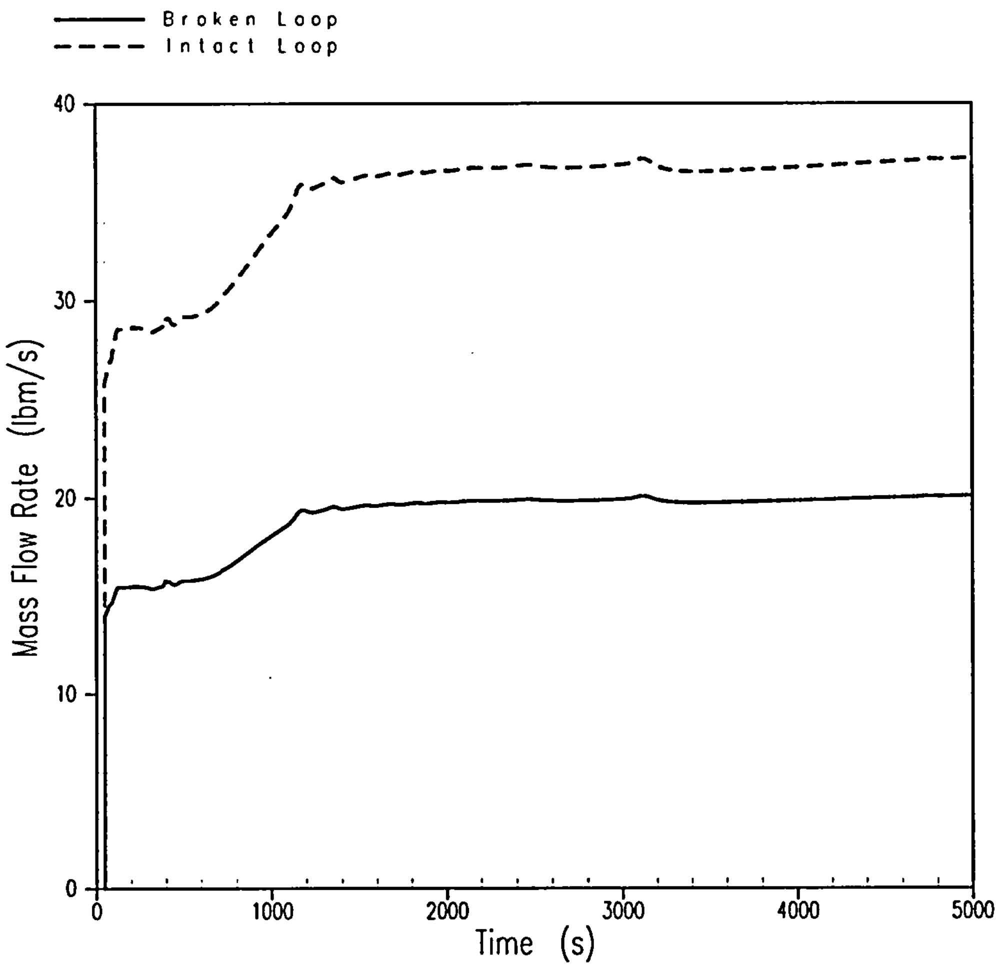

Figure 5.2.2-12B
BVPS-2 3-inch Break
Broken Loop and Intact Loop Pumped Safety Injection Flow Rate

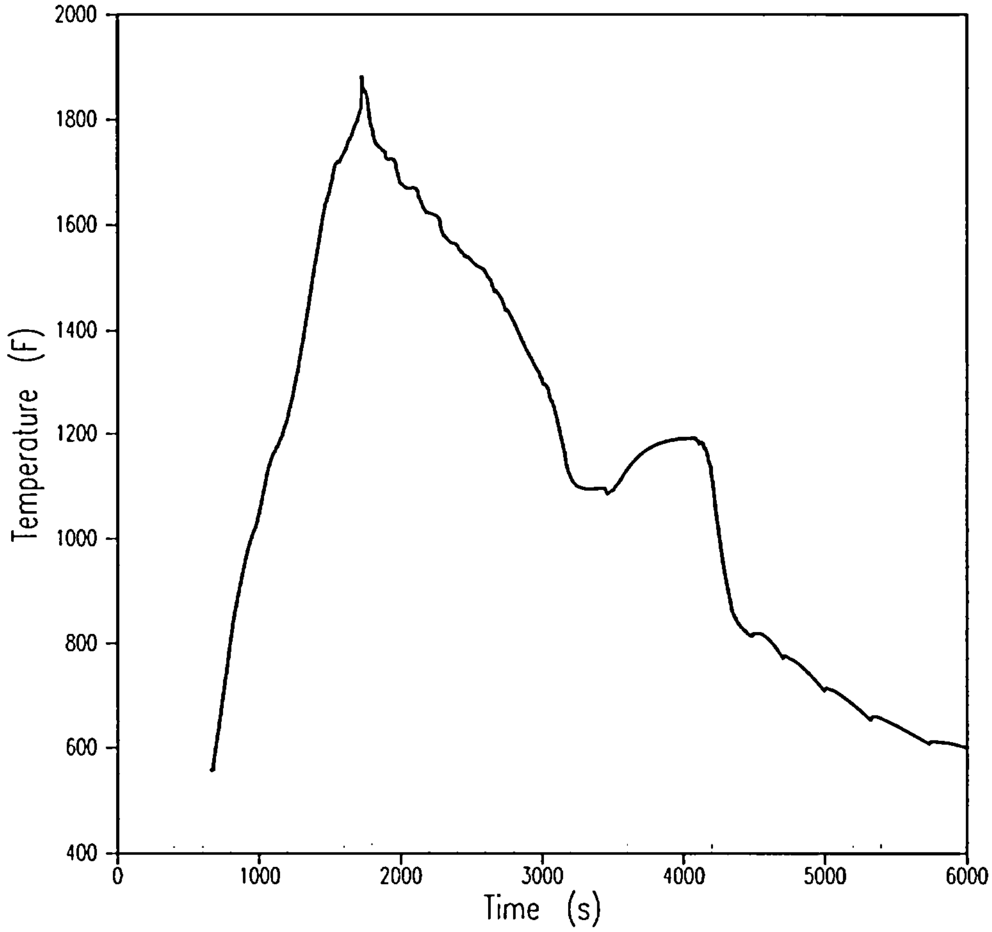

Figure 5.2.2-13A
BVPS-1 2.75-inch Break
Peak Clad Temperature

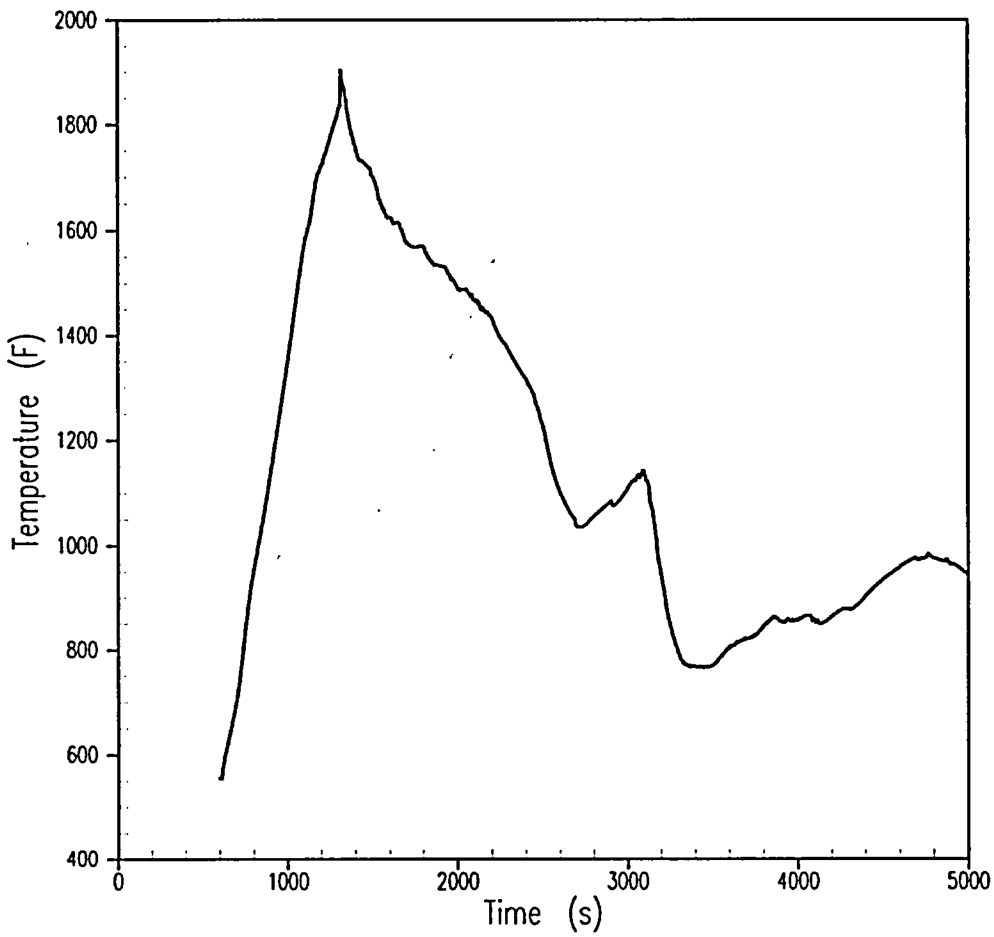

Figure 5.2.2-13B
BVPS-2 3-inch Break
Peak Clad Temperature

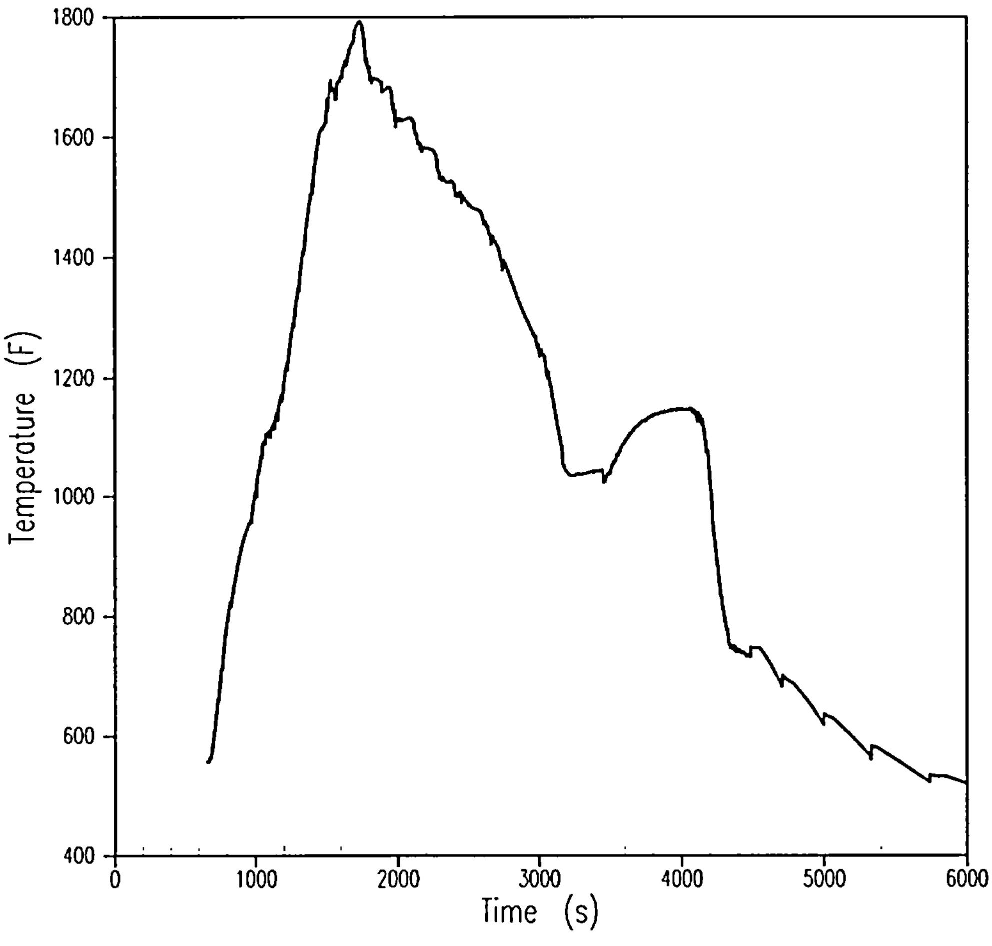

Figure 5.2.2-14A
BVPS-1 2.75-inch Break
Hot Spot Fluid Temperature

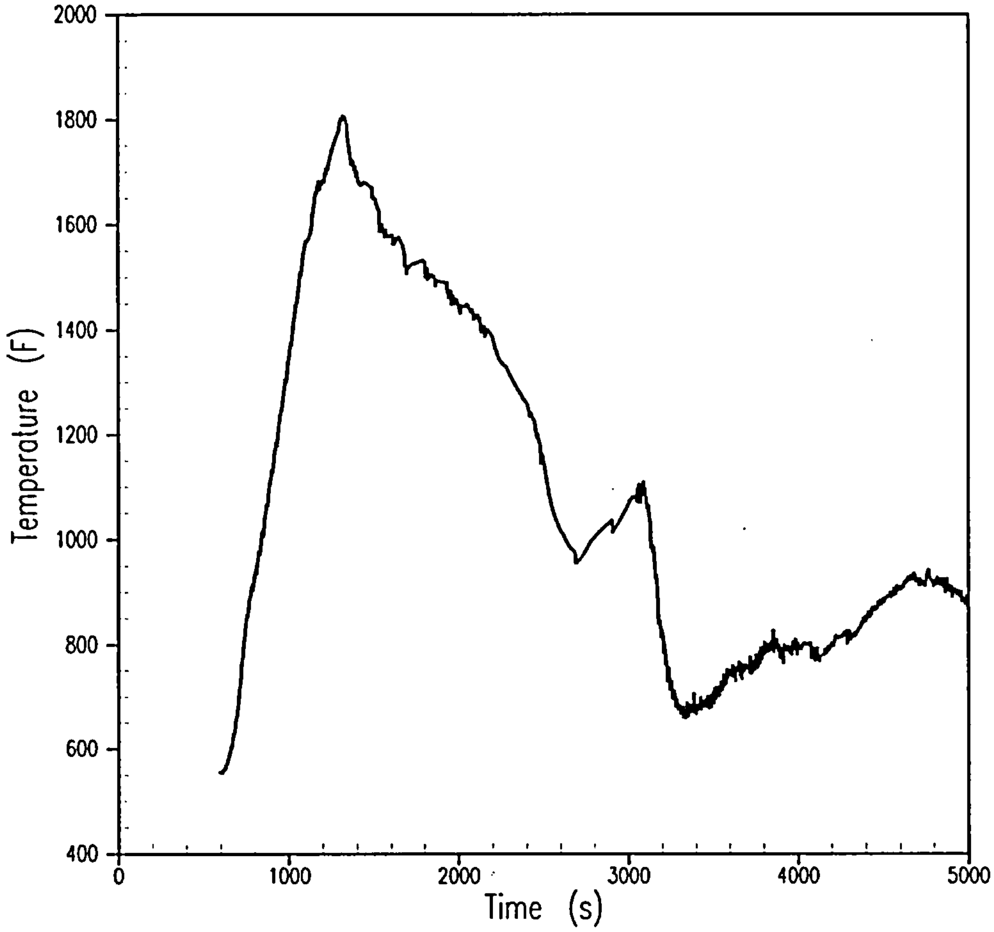

Figure 5.2.2-14B
BVPS-2 3-inch Break
Hot Spot Fluid Temperature

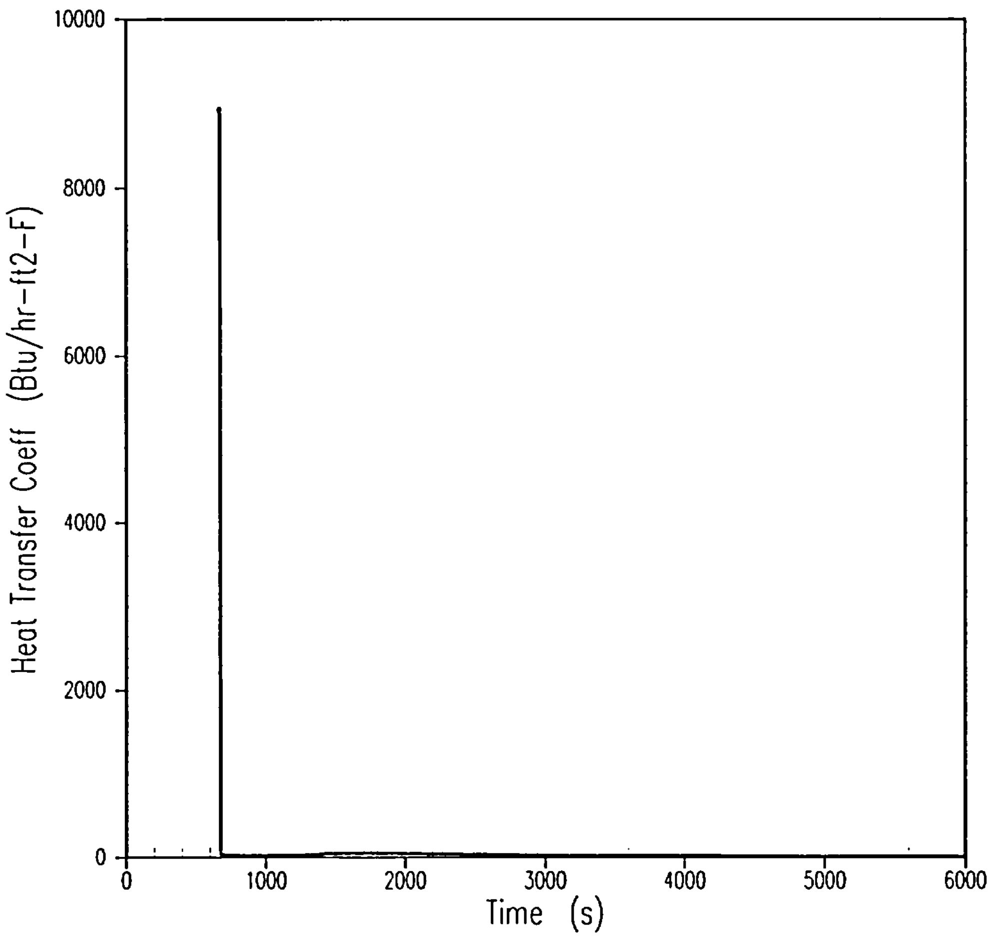

Figure 5.2.2-15A
BVPS-1 2.75-inch Break
Rod Film Heat Transfer Coefficient

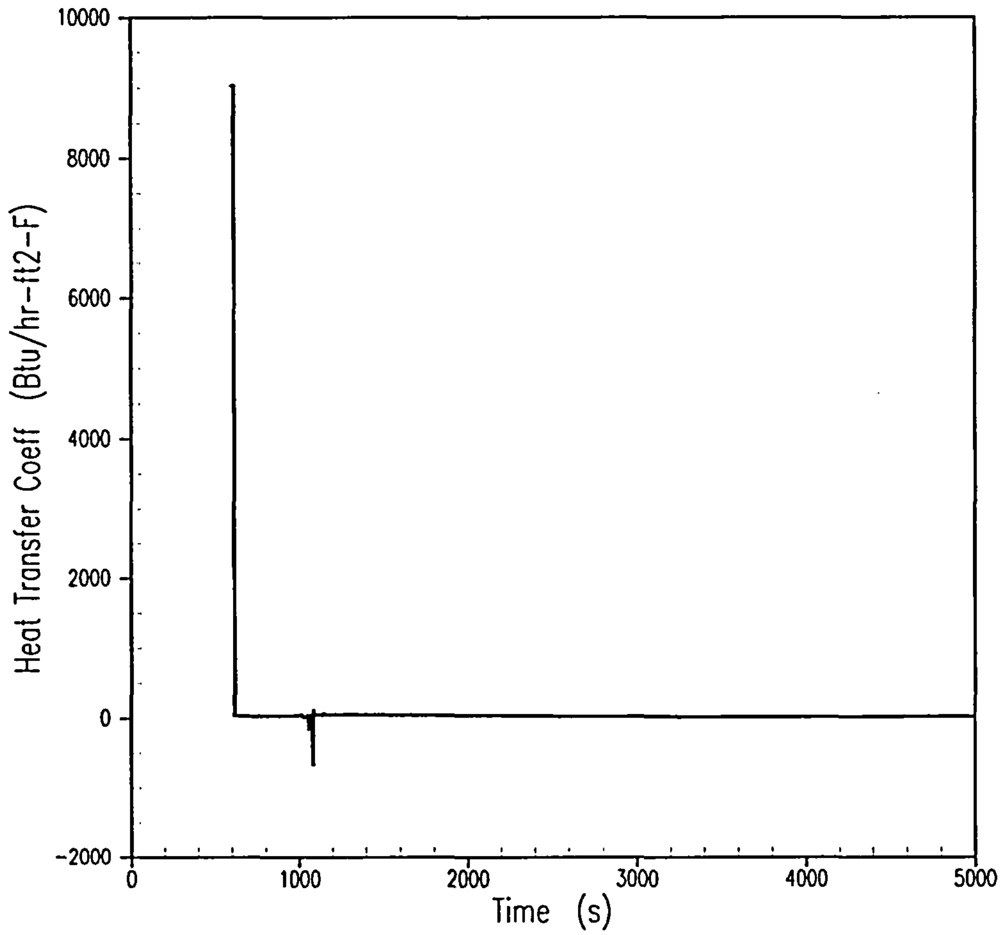

Figure 5.2.2-15B
BVPS-2 3-inch Break
Rod Film Heat Transfer Coefficient

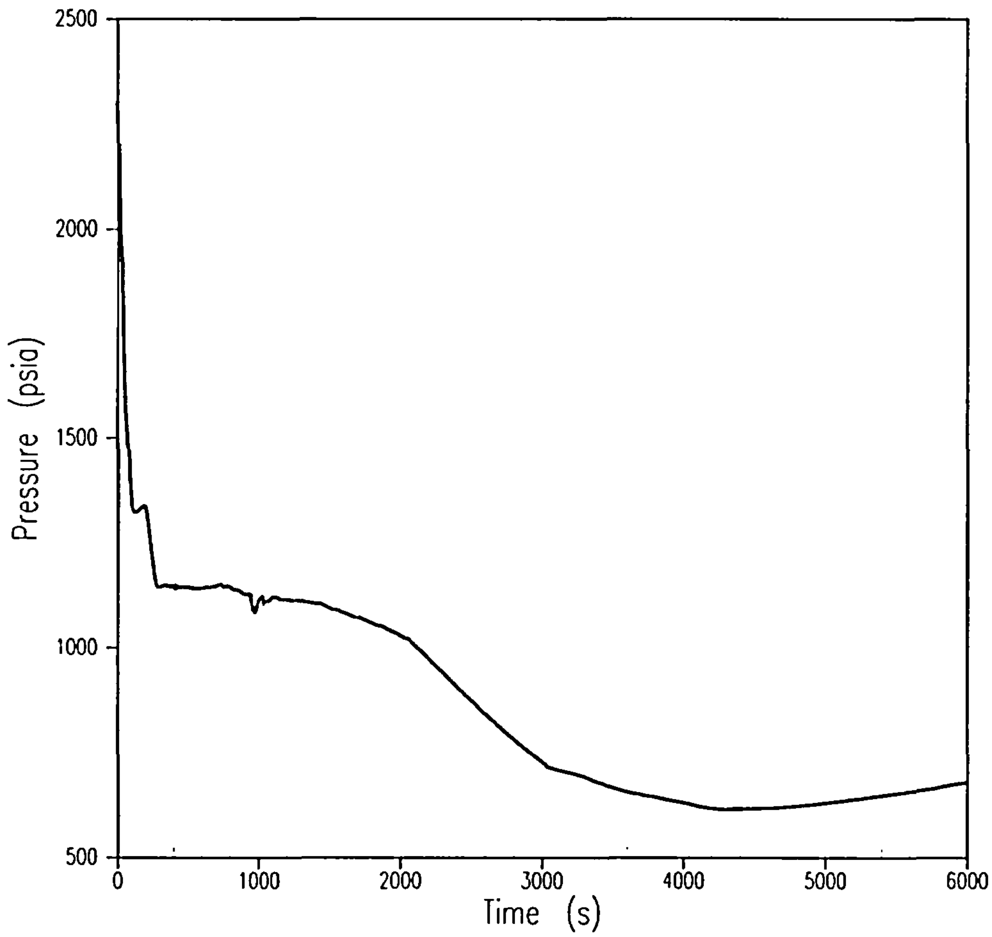

Figure 5.2.2-16A
BVPS-1 2-inch Break
RCS Pressure

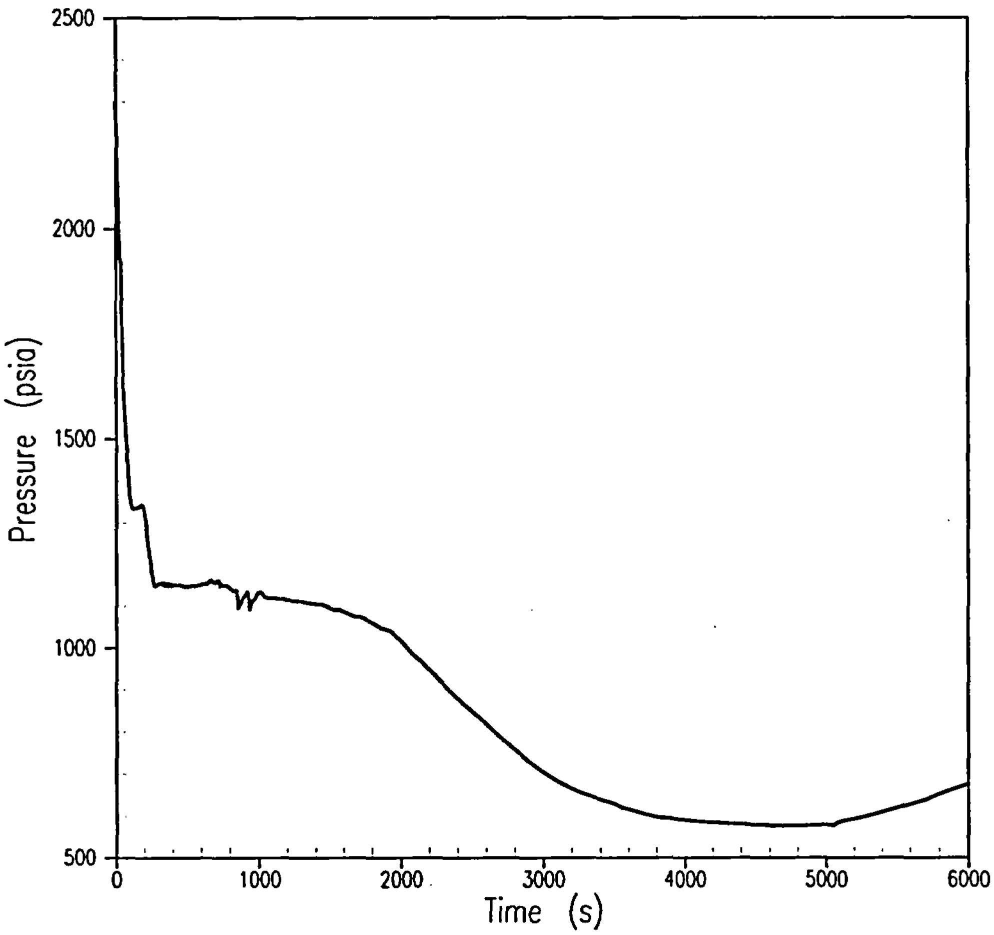

Figure 5.2.2-16B
BVPS-2 2-inch Break
RCS Pressure

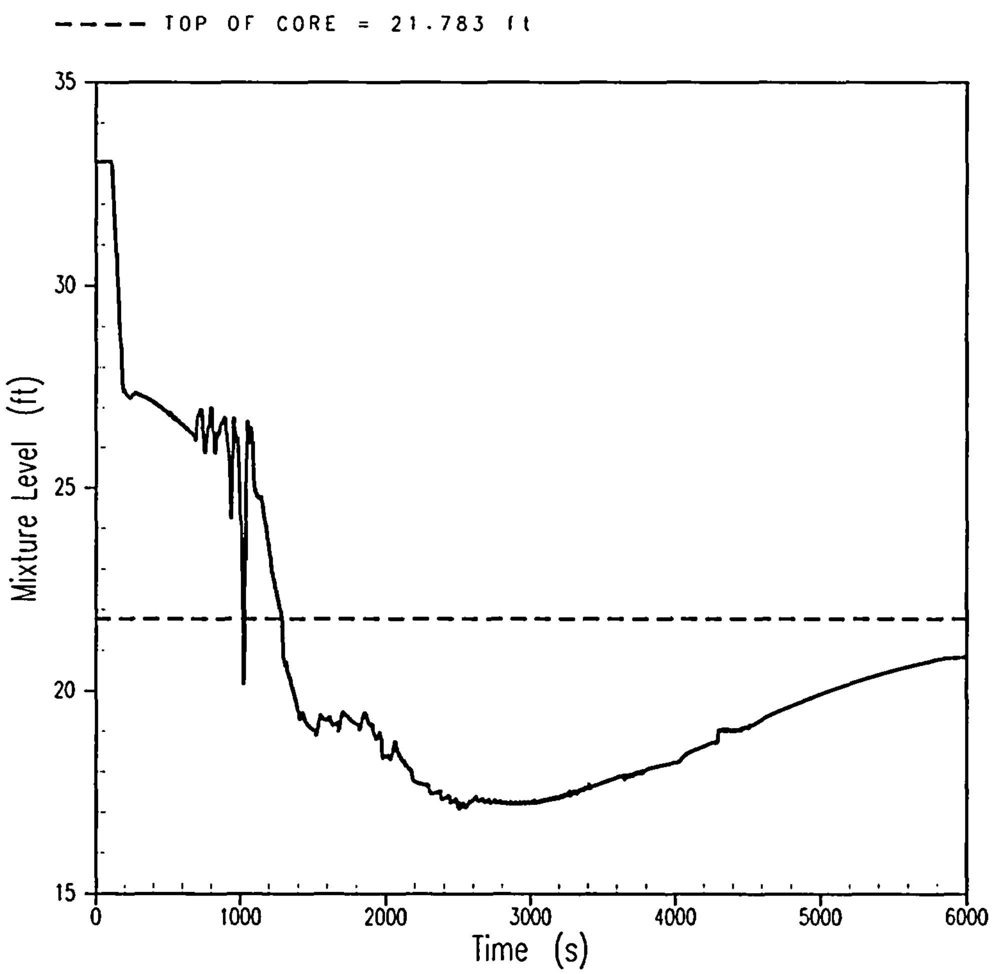

Figure 5.2.2-17A
BVPS-1 2-inch Break
Core Mixture Level

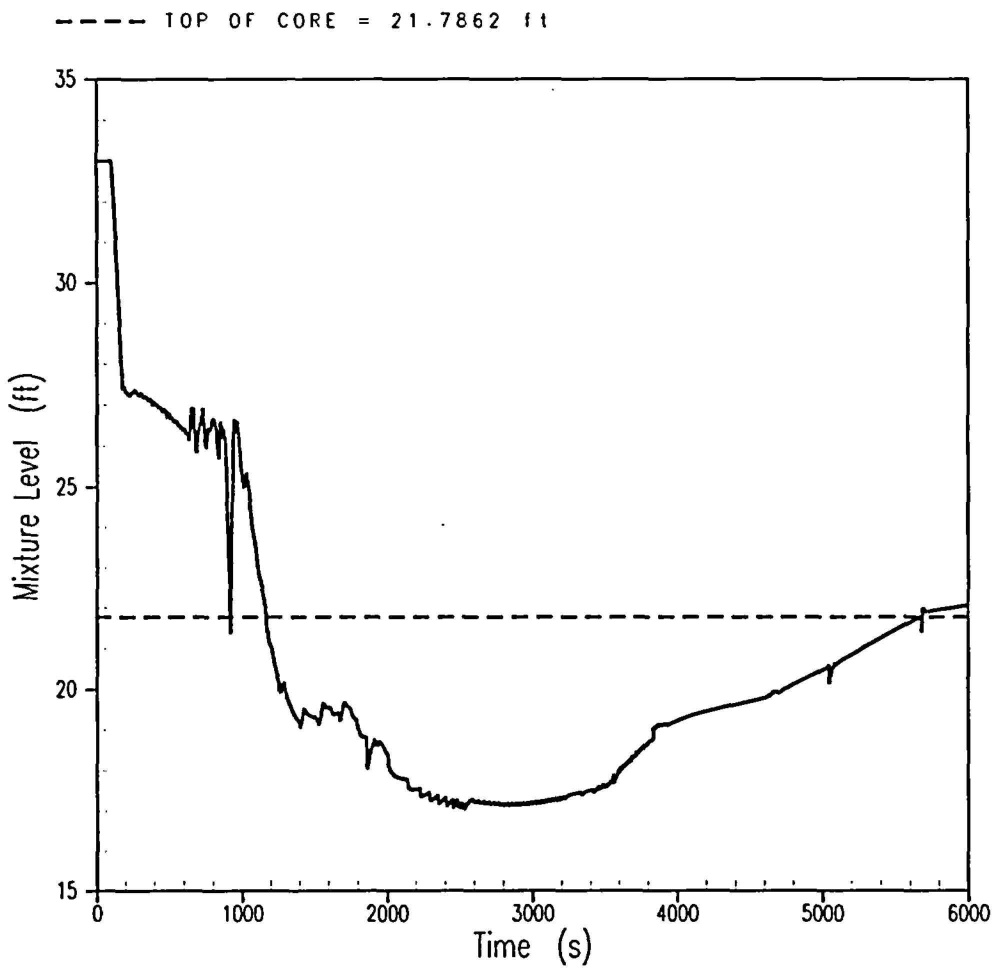

Figure 5.2.2-17B
BVPS-2 2-inch Break
Core Mixture Level

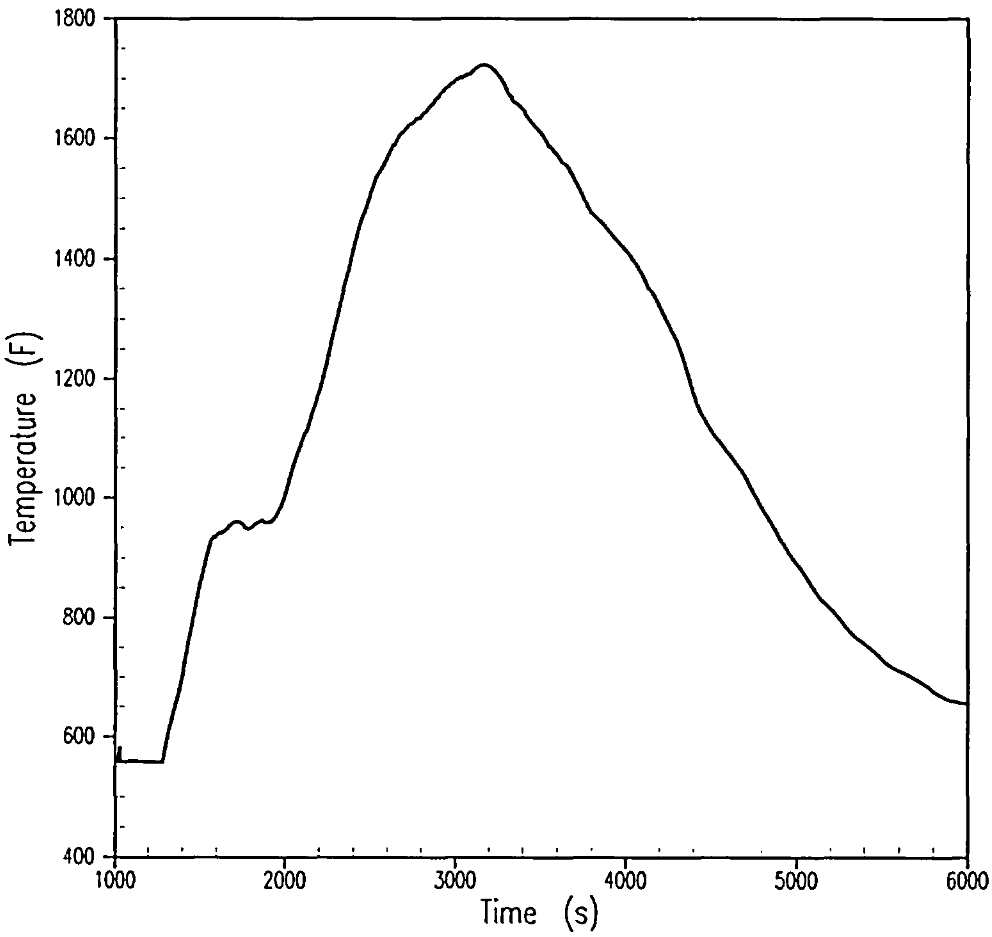

Figure 5.2.2-18A
BVPS-1 2-inch Break
Peak Clad Temperature

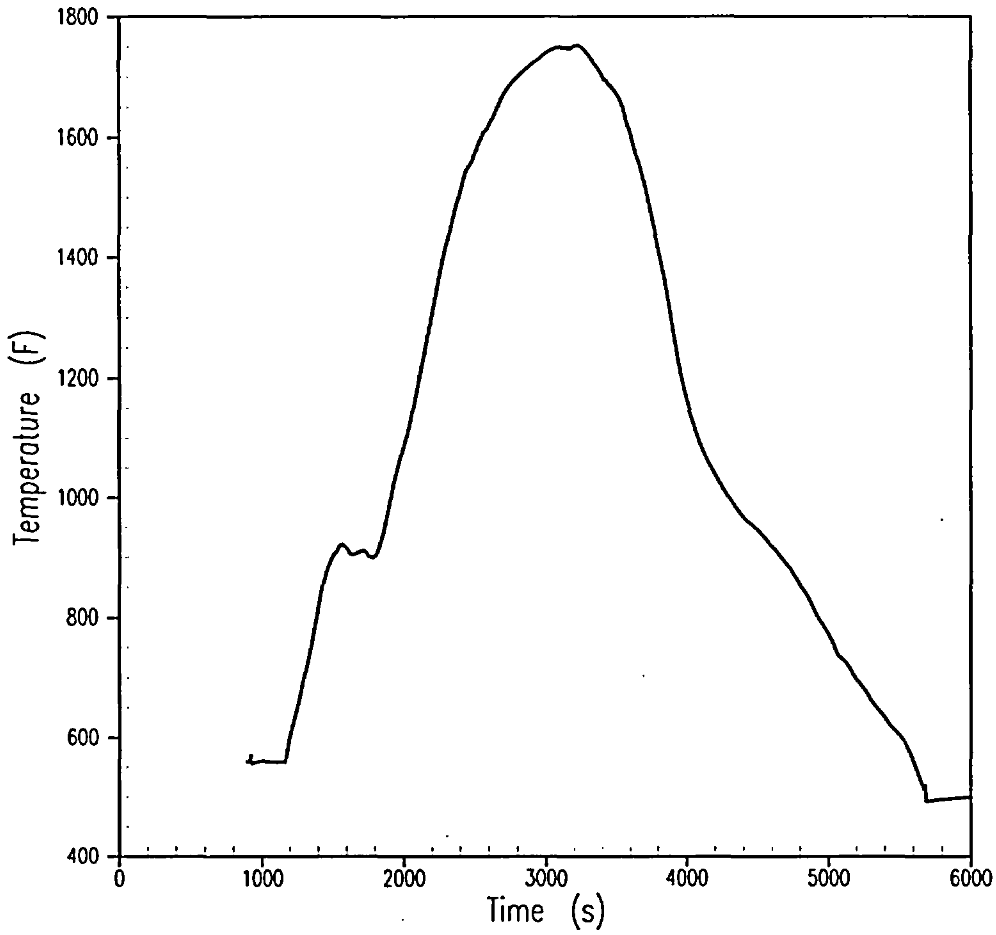

Figure 5.2.2-18B
BVPS-2 2-inch Break
Peak Clad Temperature

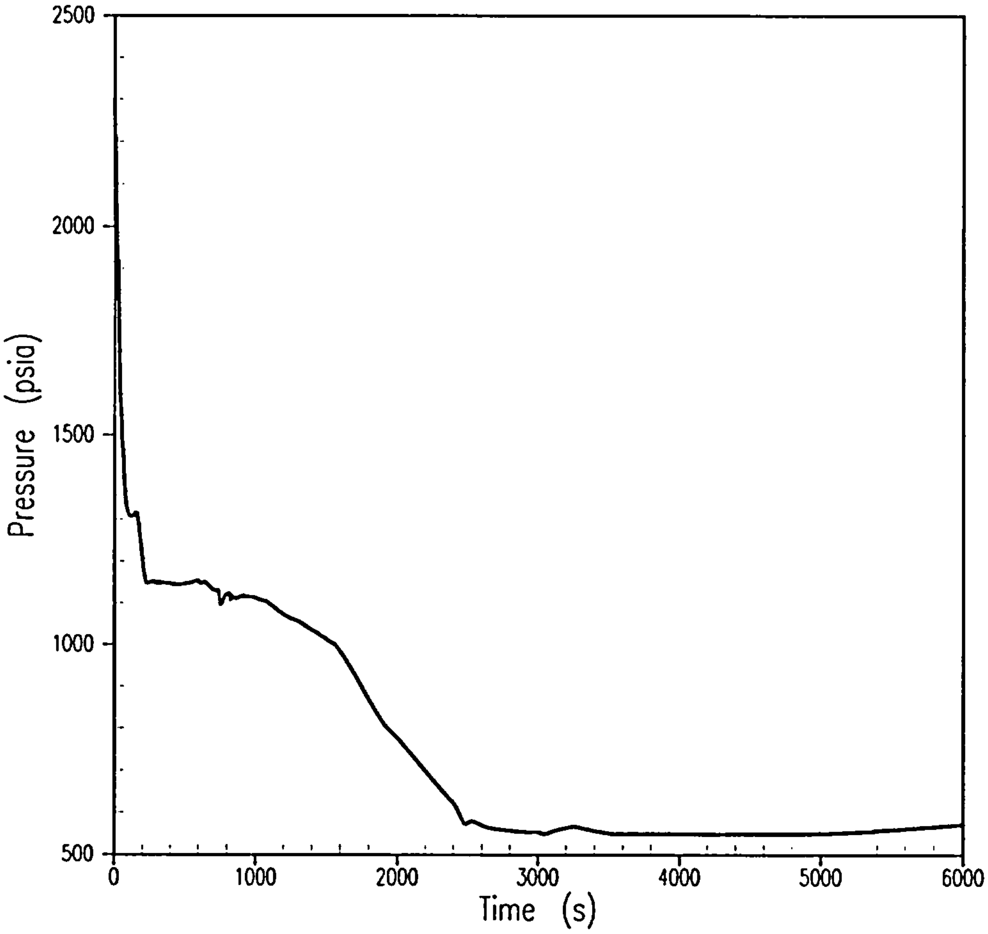

Figure 5.2.2-19A
BVPS-1 2.25-inch Break
RCS Pressure

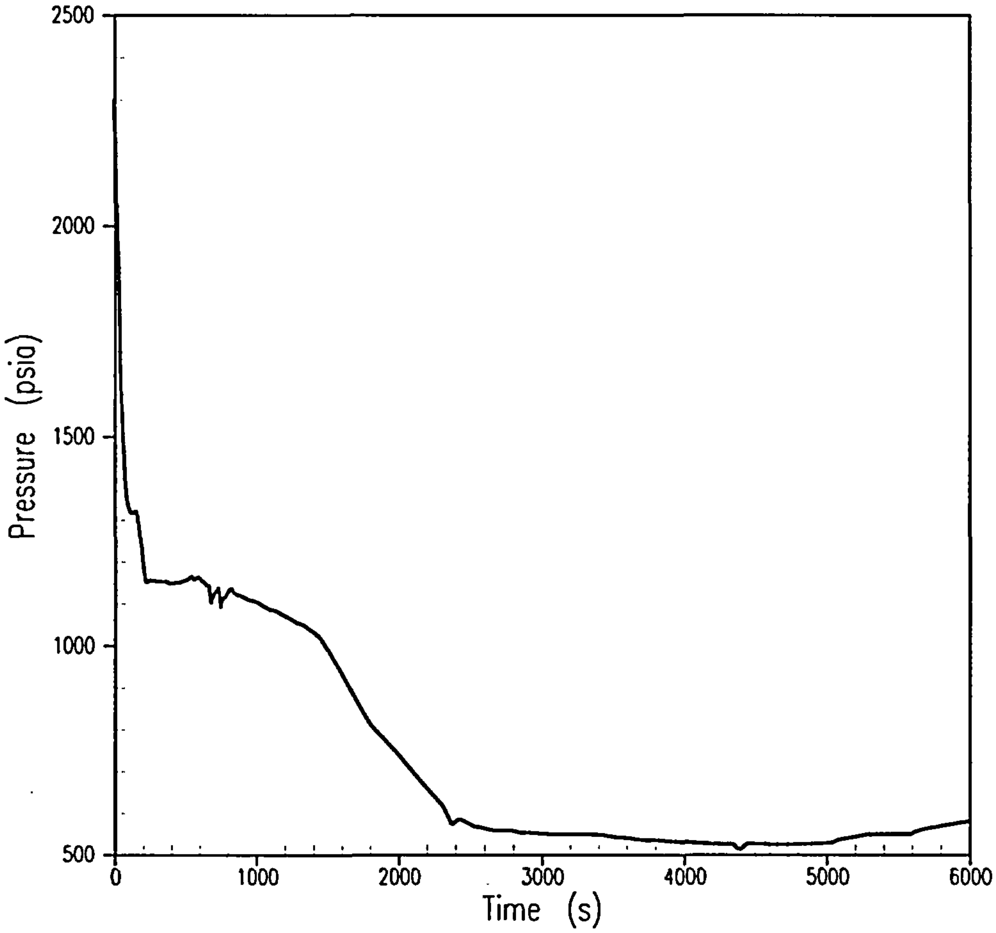

Figure 5.2.2-19B
BVPS-2 2.25-inch Break
RCS Pressure

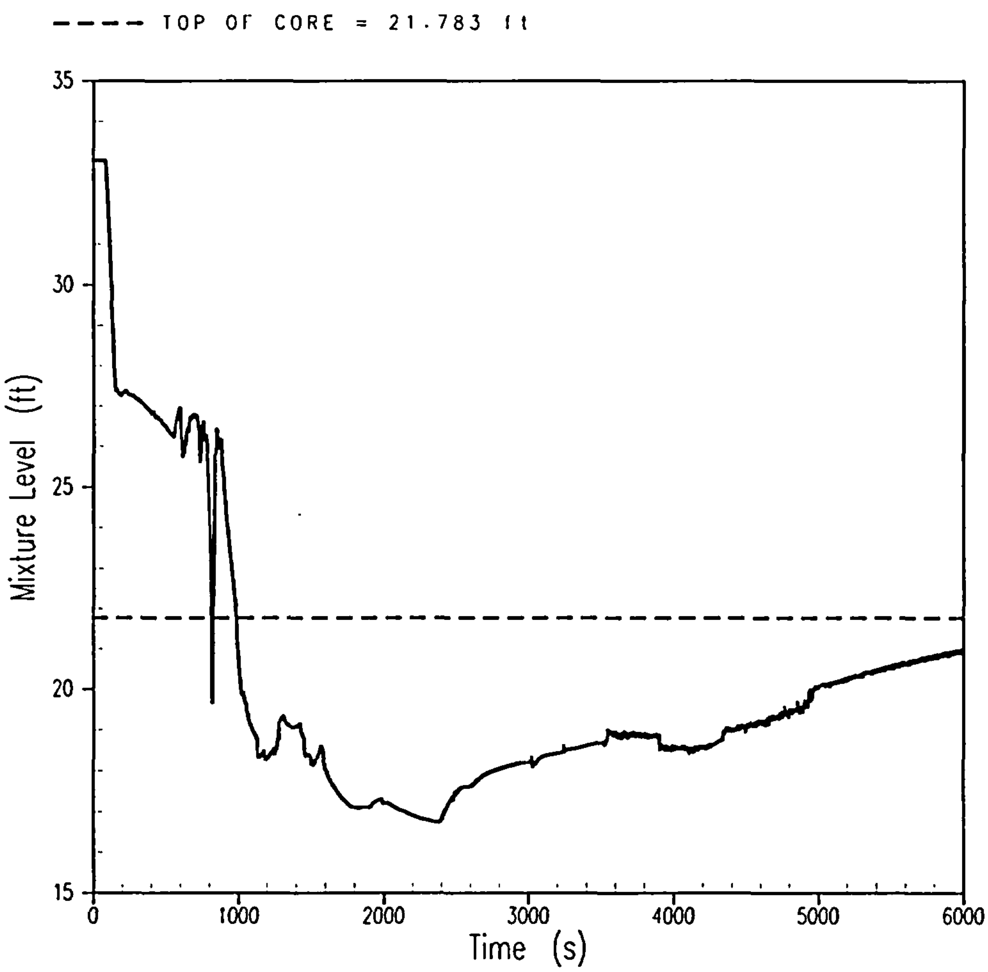

Figure 5.2.2-20A
BVPS-1 2.25-inch Break
Core Mixture Level

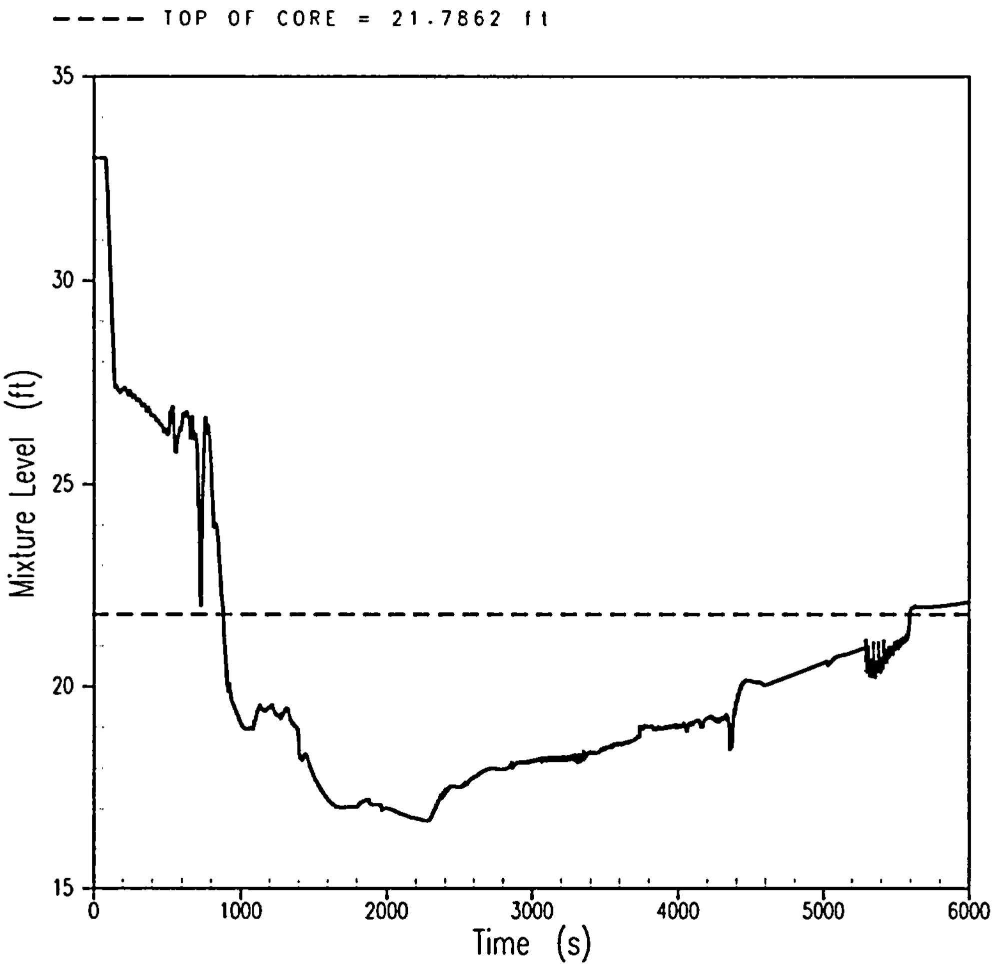

Figure 5.2.2-20B
BVPS-2 2.25-inch Break
Core Mixture Level

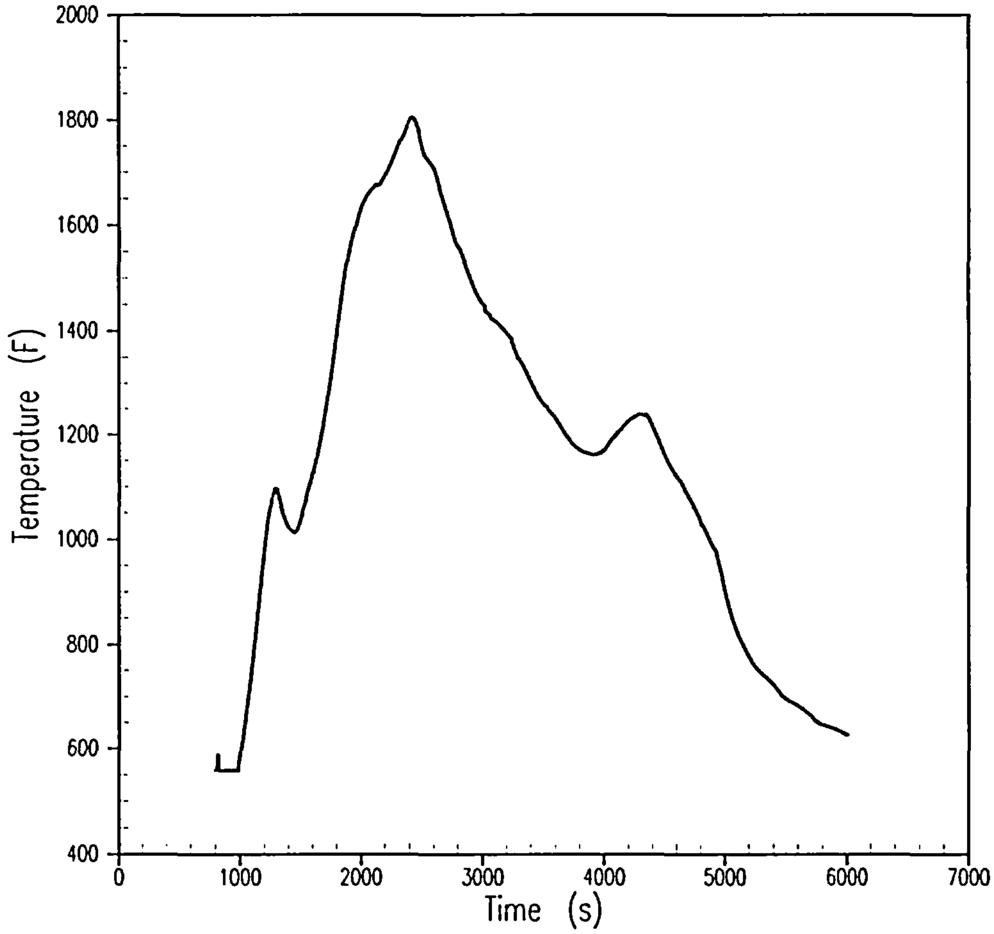

Figure 5.2.2-21A
BVPS-1 2.25-inch Break
Peak Clad Temperature

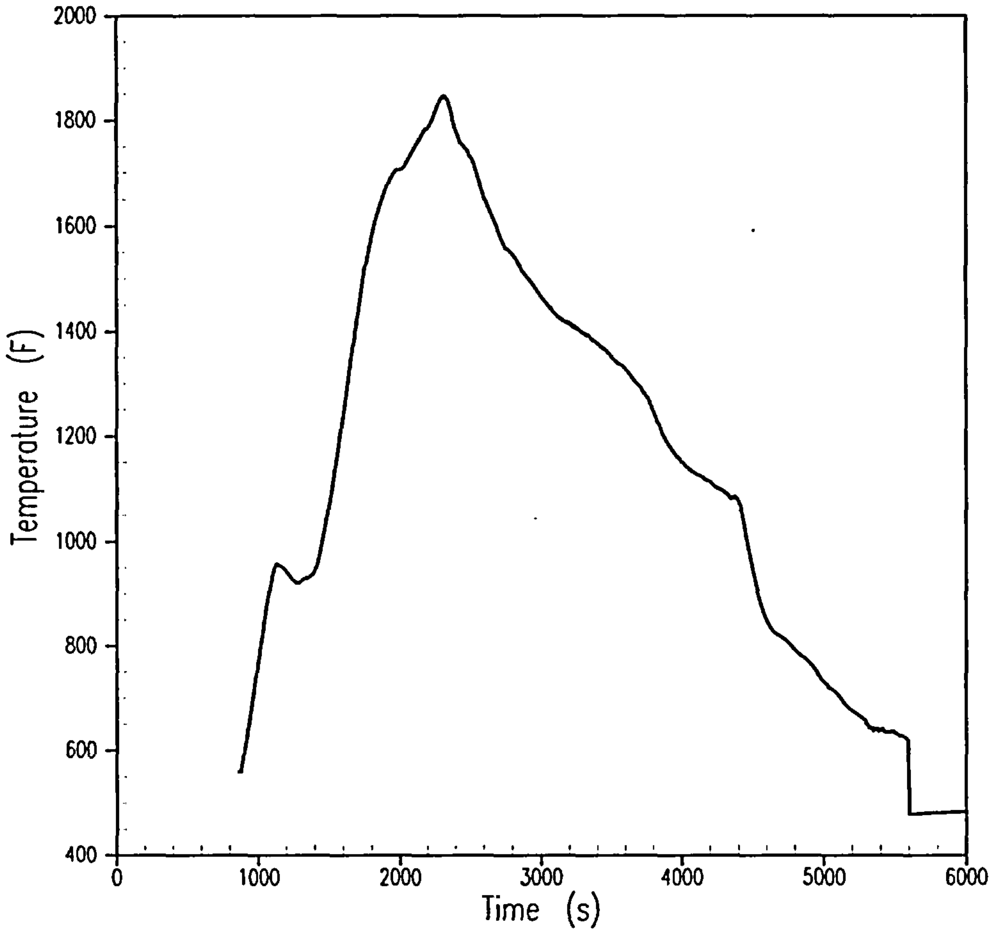

Figure 5.2.2-21B
BVPS-2 2.25-inch Break
Peak Clad Temperature

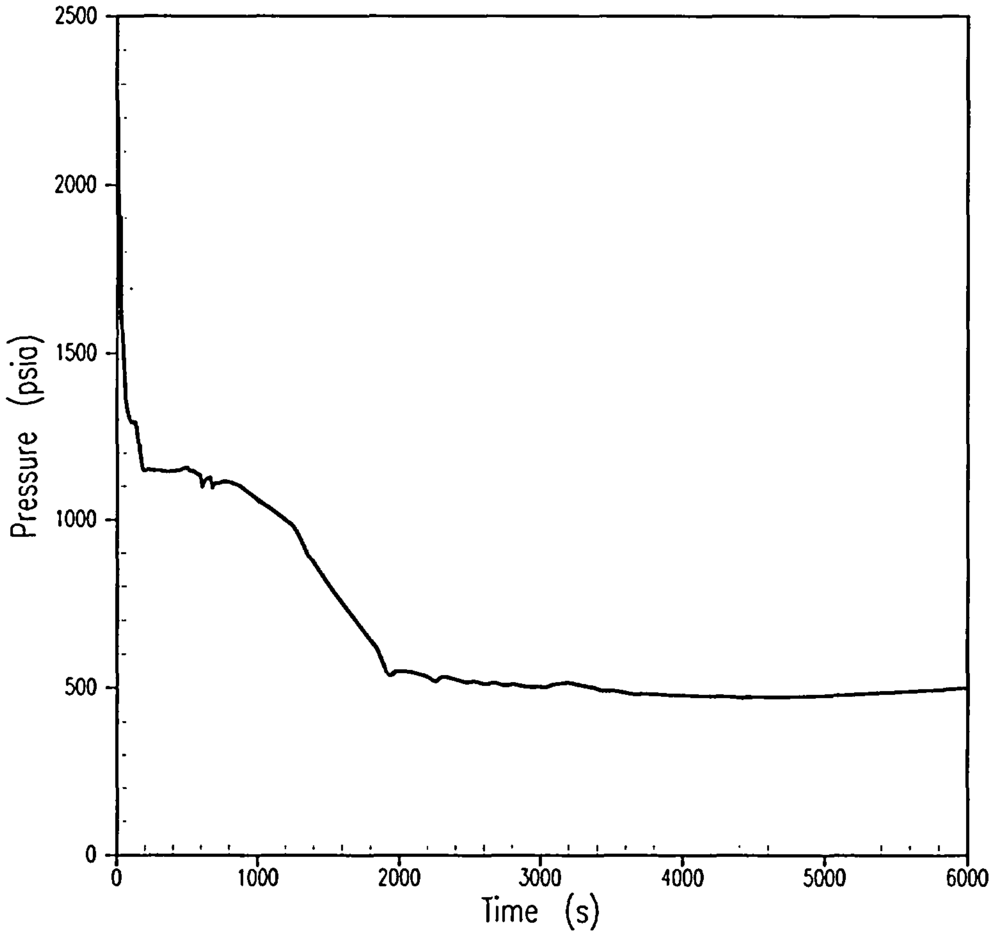

Figure 5.2.2-22A
BVPS-1 2.5-inch Break
RCS Pressure

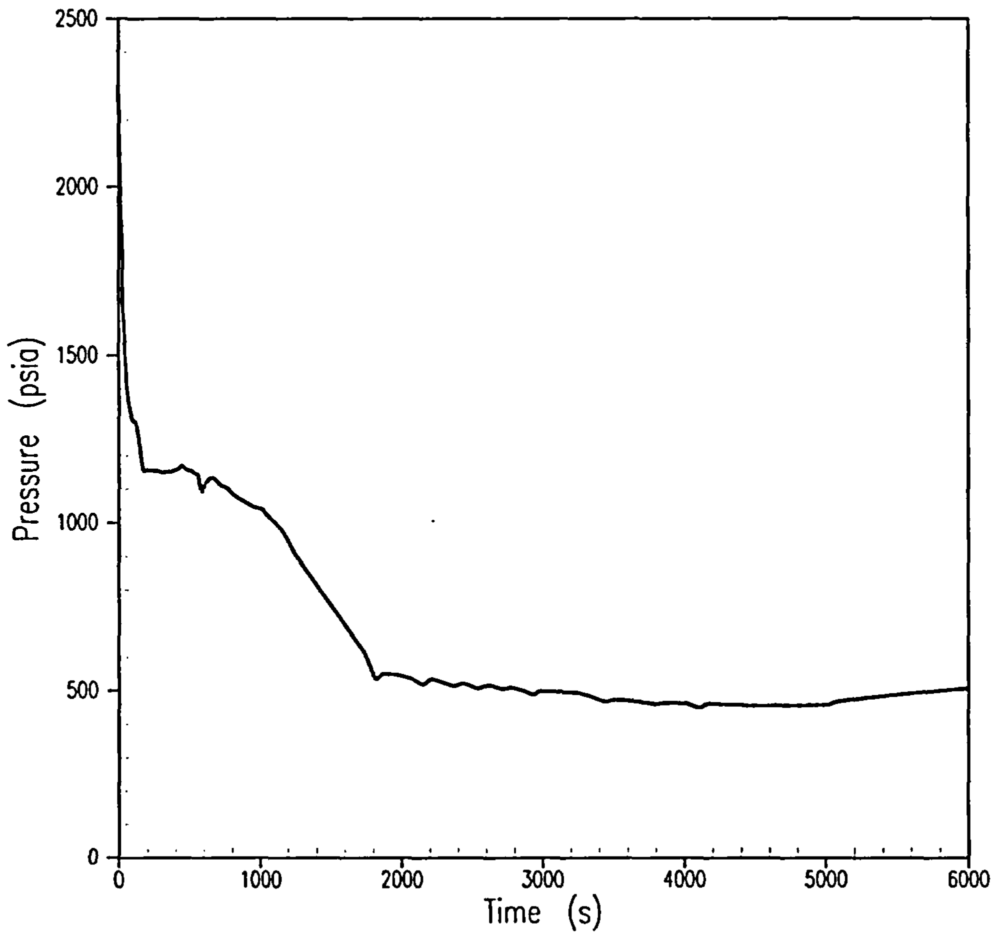

Figure 5.2.2-22B
BVPS-2 2.5-inch Break
RCS Pressure

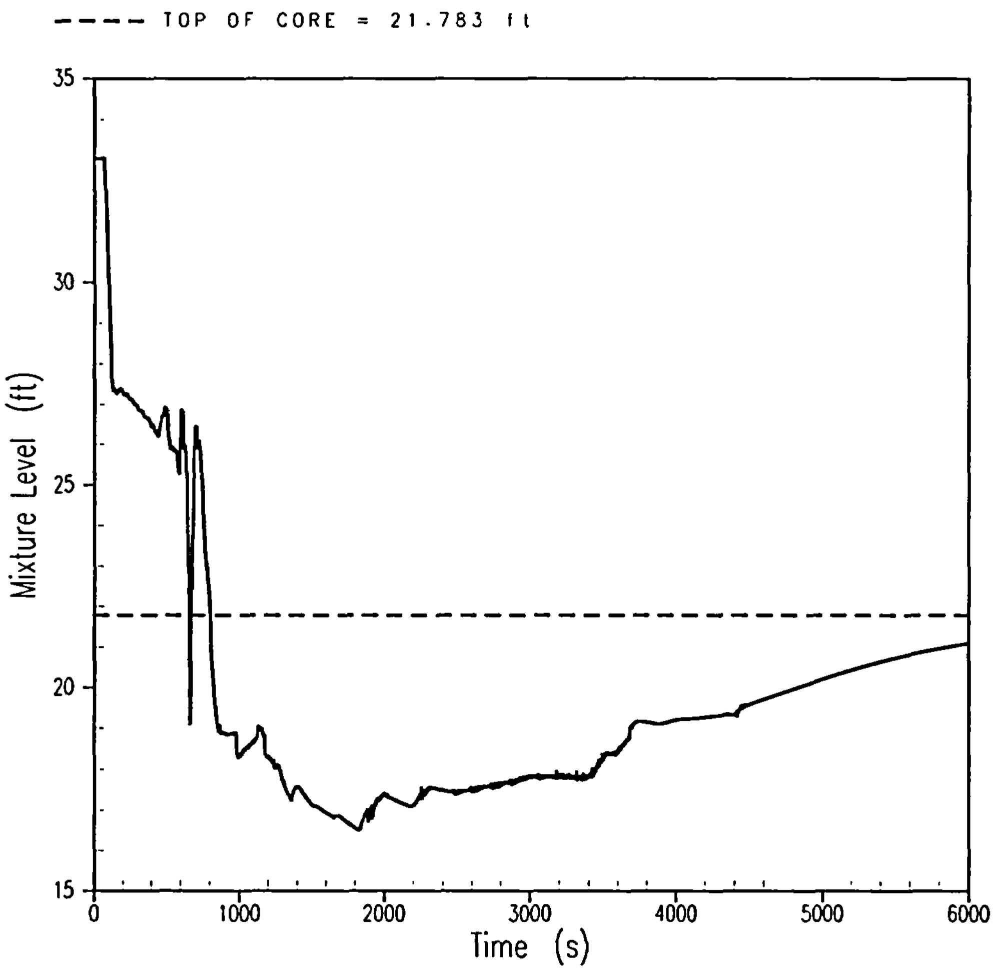

Figure 5.2.2-23A
BVPS-1 2.5-inch Break
Core Mixture Level

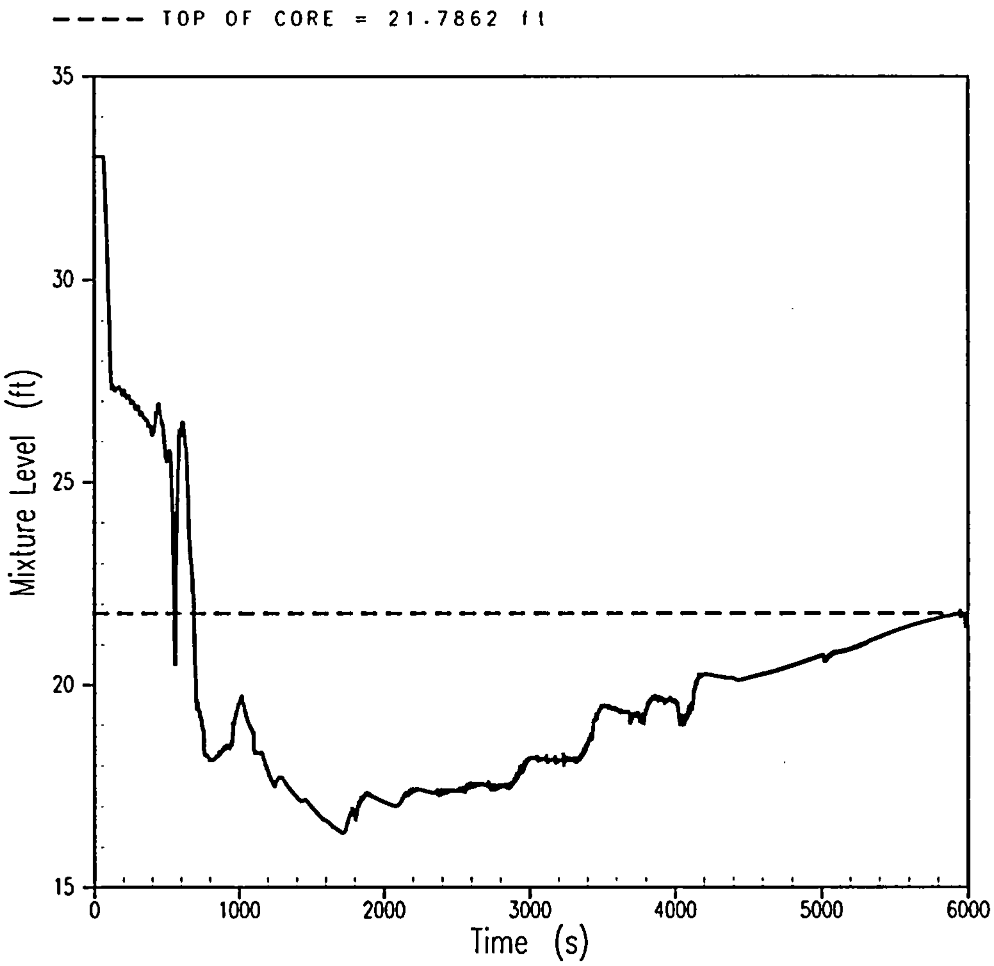

Figure 5.2.2-23B
BVPS-2 2.5-inch Break
Core Mixture Level

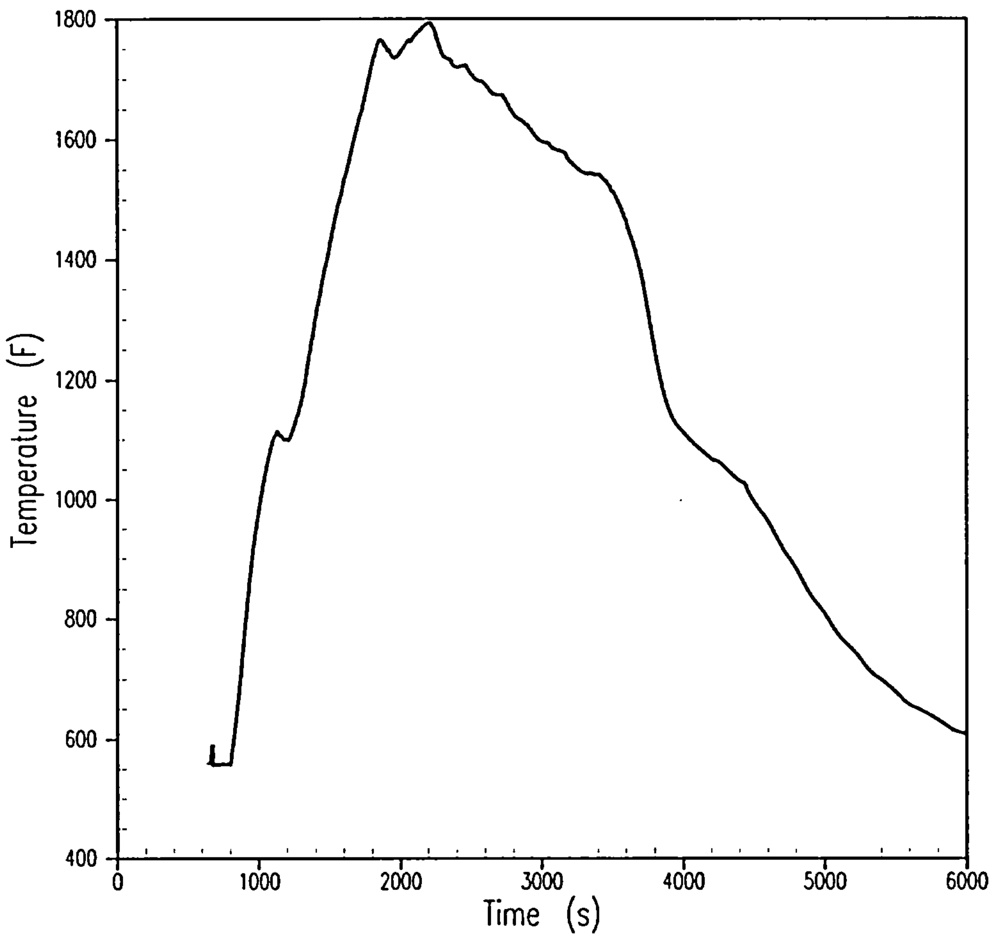

Figure 5.2.2-24A
BVPS-1 2.5-inch Break
Peak Clad Temperature

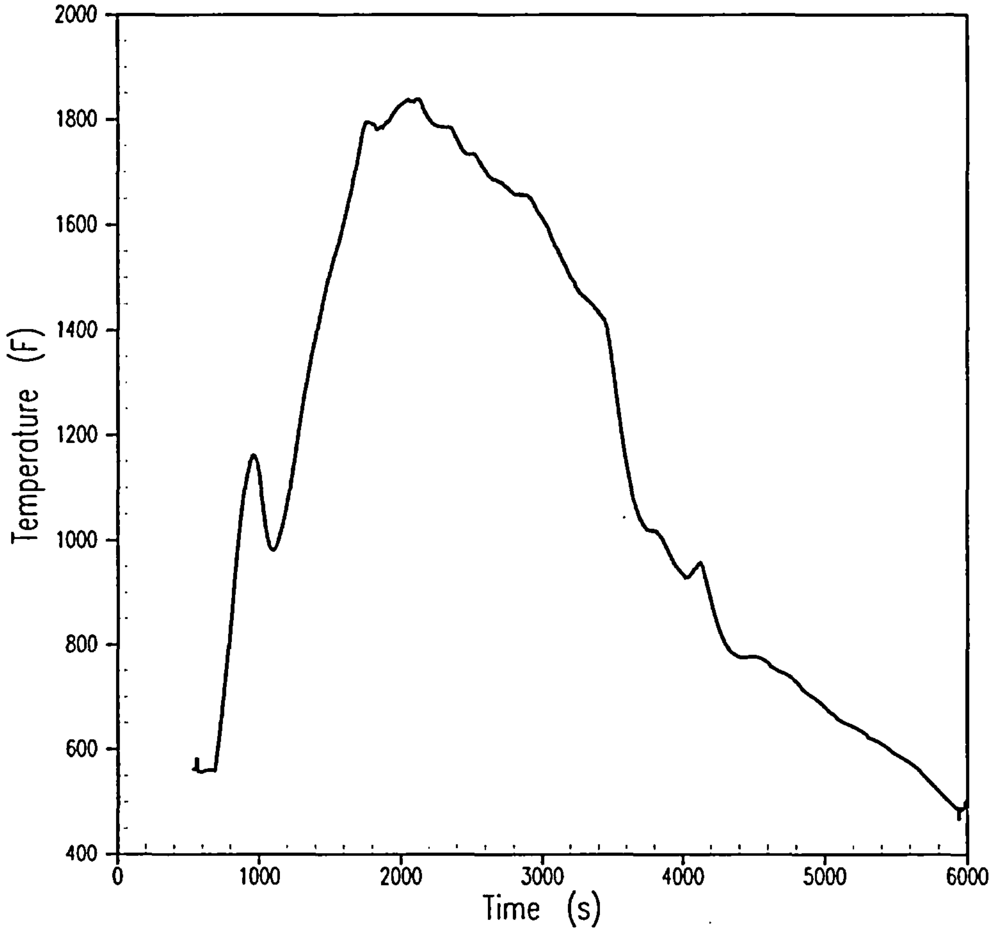

Figure 5.2.2-24B
BVPS-2 2.5-inch Break
Peak Clad Temperature

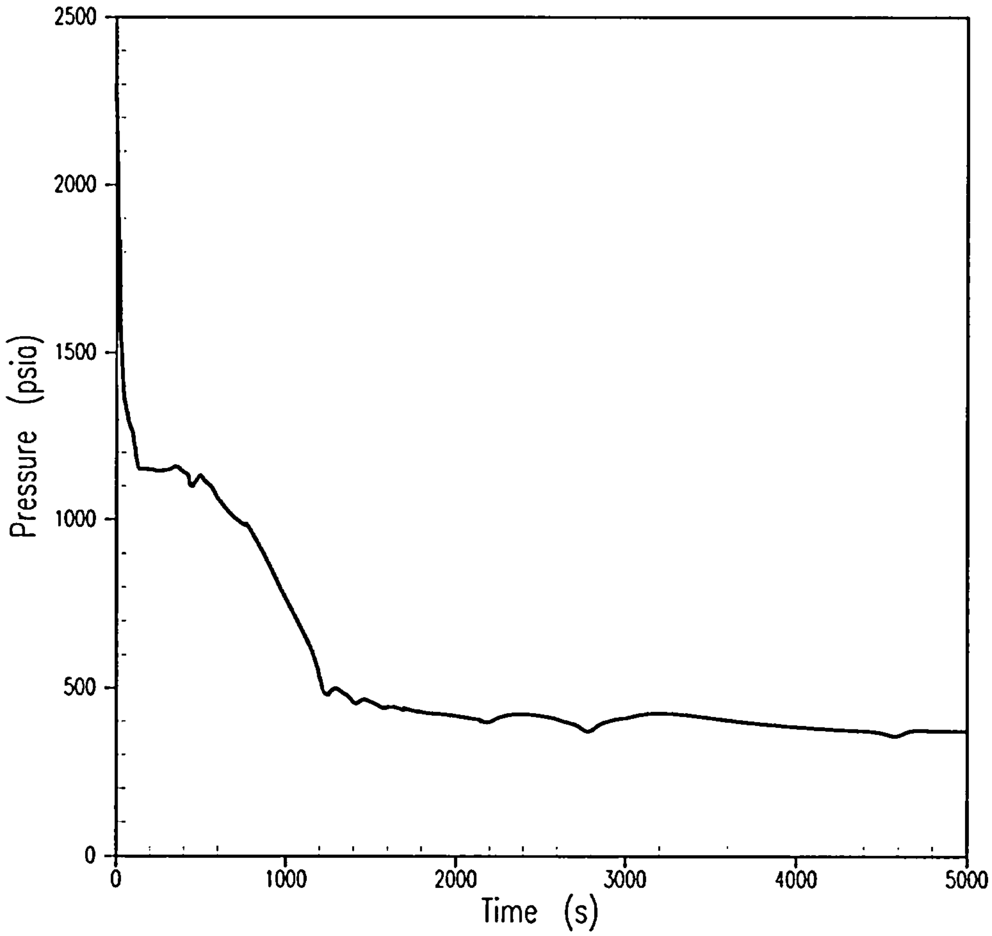

Figure 5.2.2-25A
BVPS-1 3-inch Break
RCS Pressure

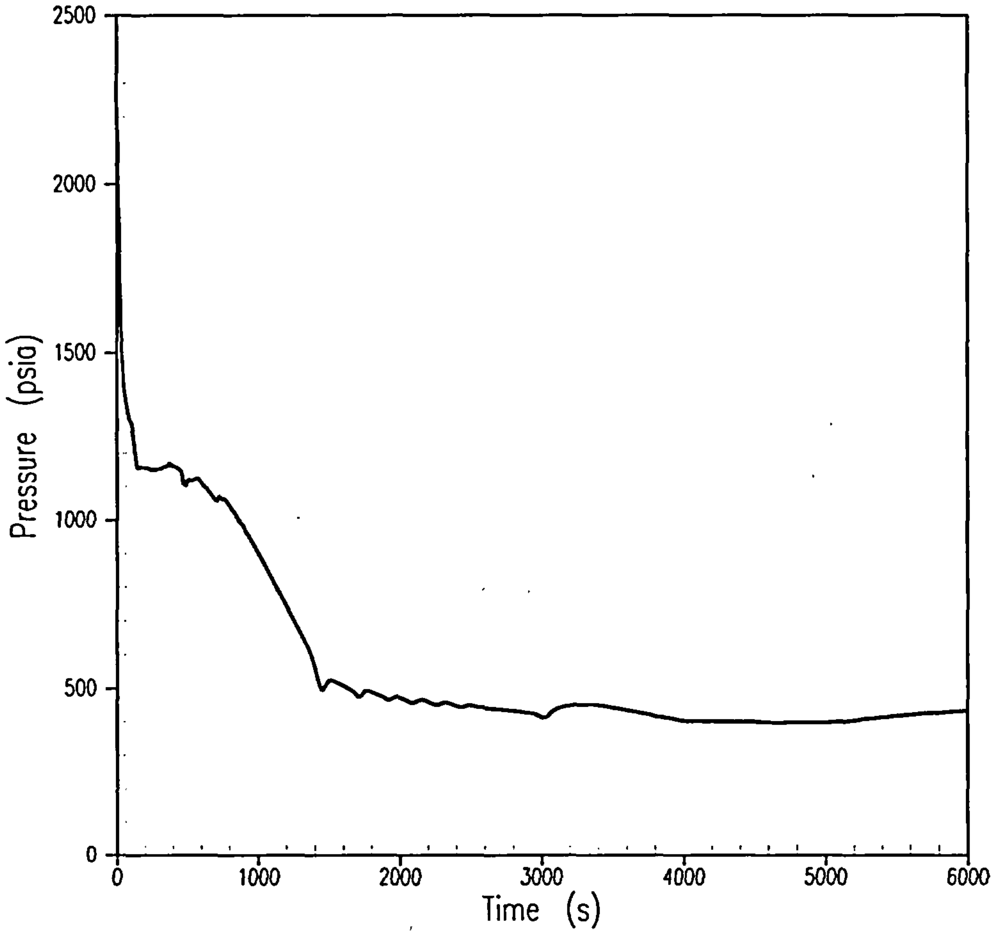

Figure 5.2.2-25B
BVPS-2 2.75-inch Break
RCS Pressure

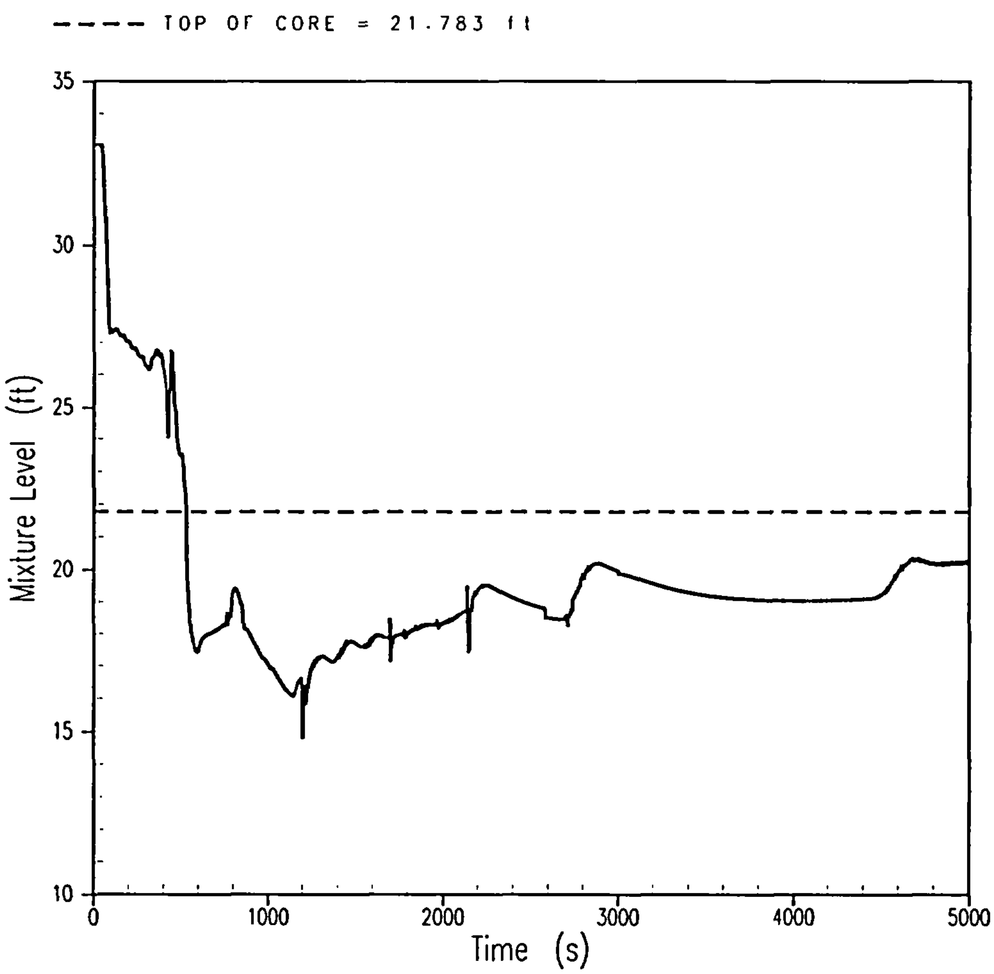

Figure 5.2.2-26A
BVPS-1 3-inch Break
Core Mixture Level

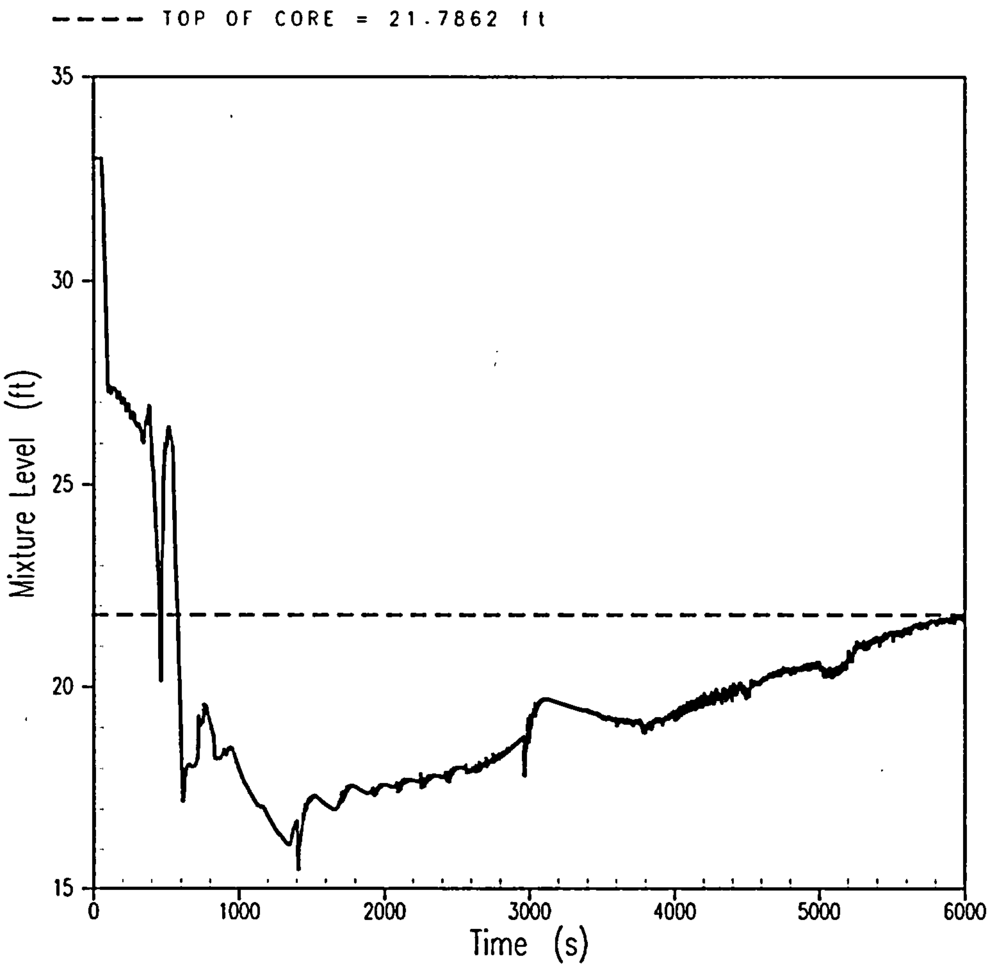

Figure 5.2.2-26B
BVPS-2 2.75-inch Break
Core Mixture Level

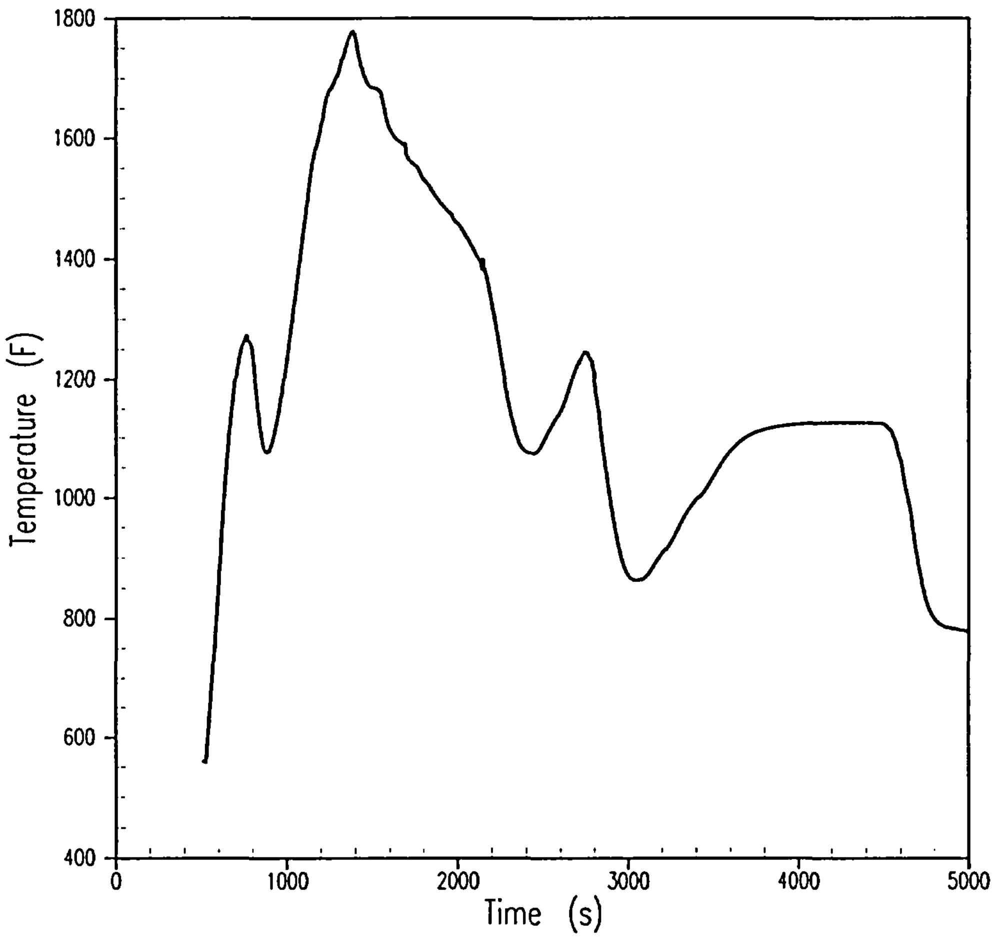

Figure 5.2.2-27A
BVPS-1 3-inch Break
Peak Clad Temperature

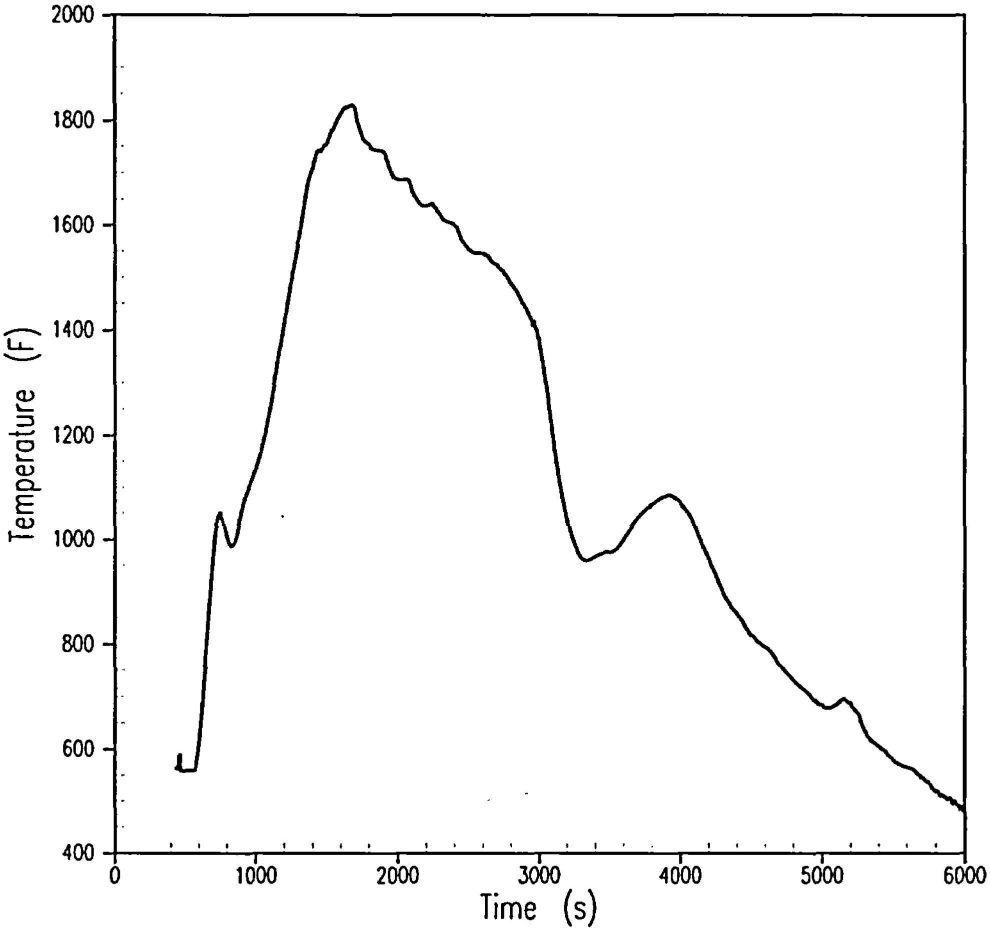

Figure 5.2.2-27B
BVPS-2 2.75-inch Break
Peak Clad Temperature

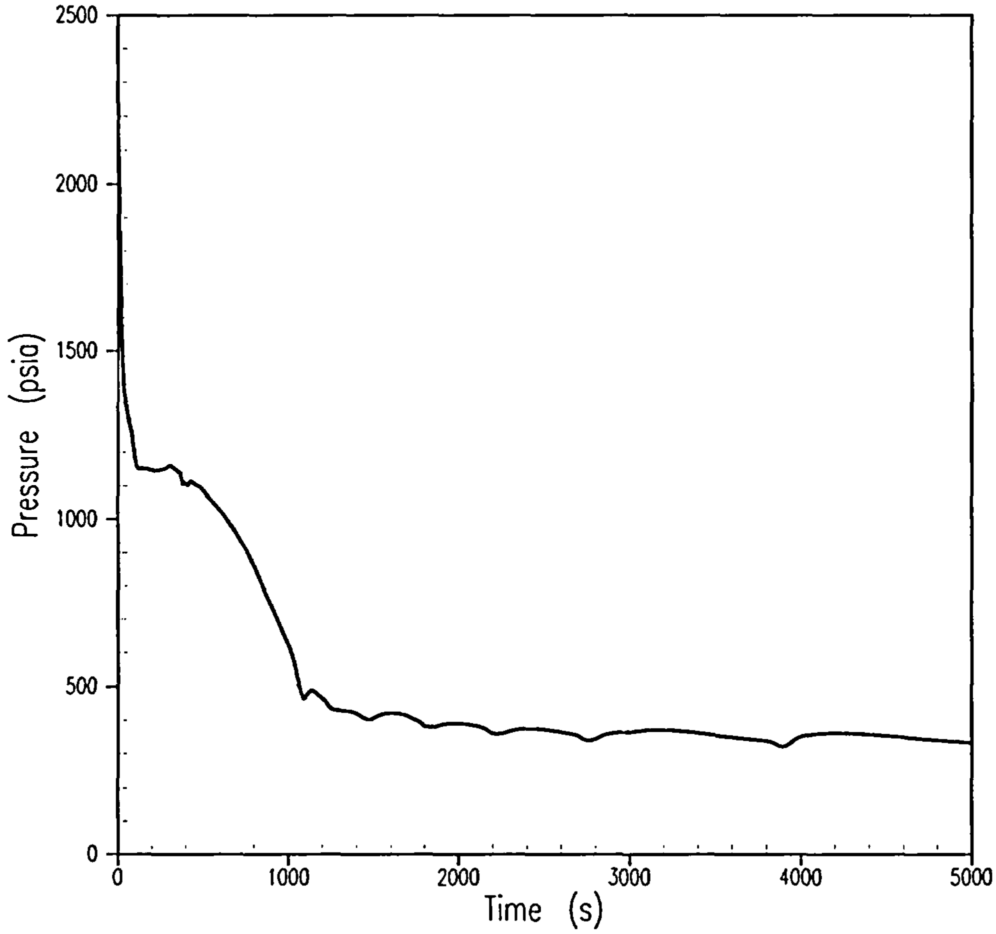

Figure 5.2.2-28A
BVPS-1 3.25-inch Break
RCS Pressure

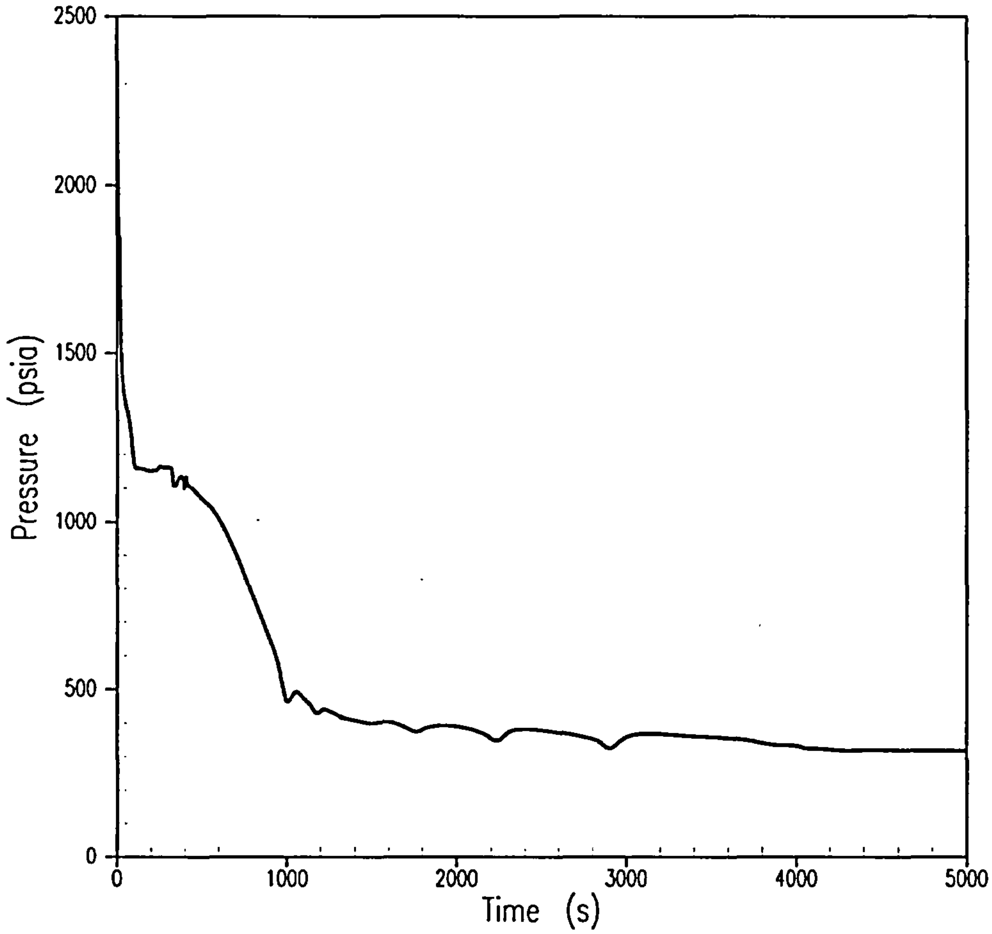

Figure 5.2.2-28B
BVPS-2 3.25-inch Break
RCS Pressure

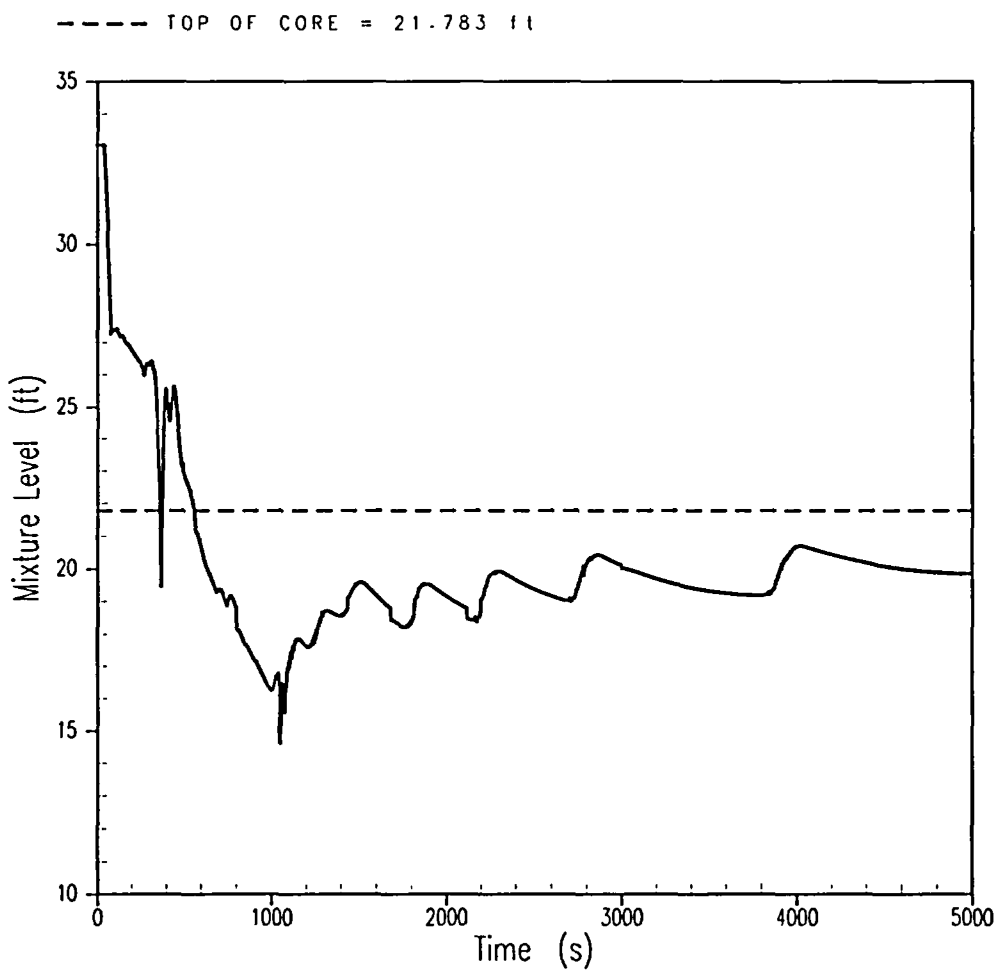

Figure 5.2.2-29A
BVPS-1 3.25-inch Break
Core Mixture Level

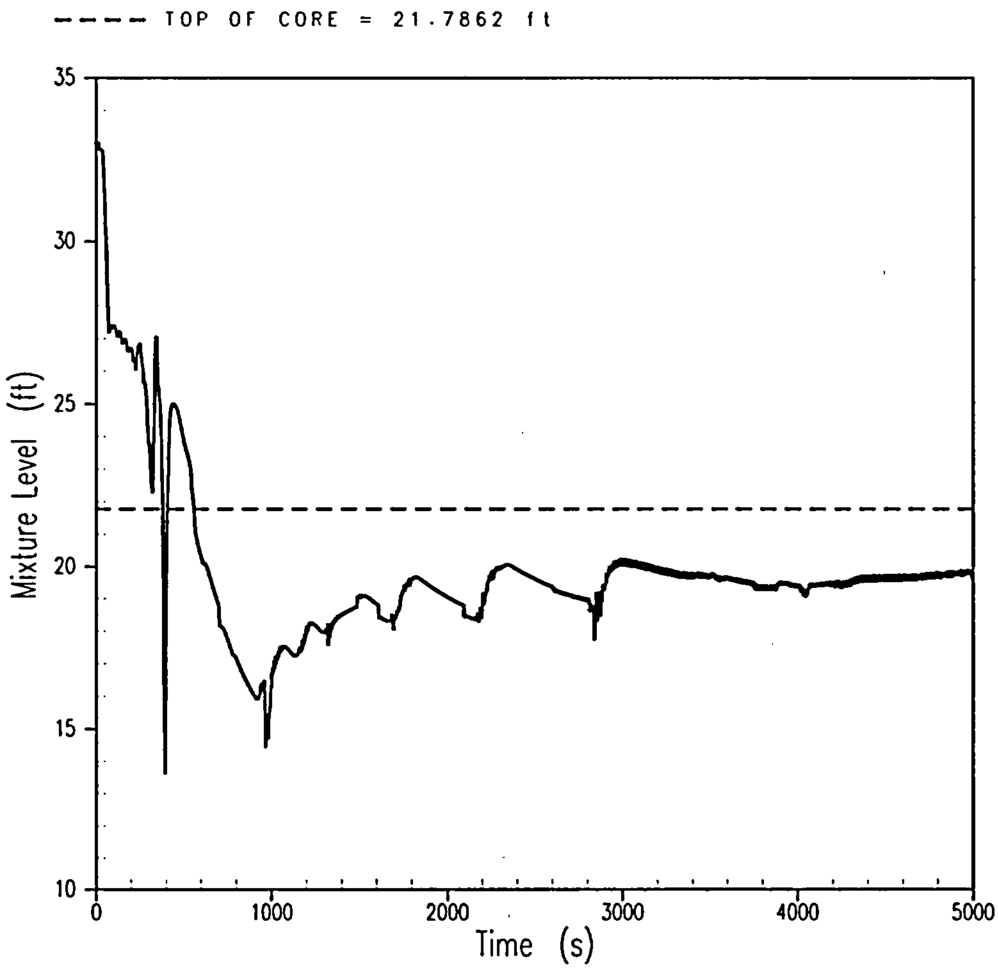

Figure 5.2.2-29B
BVPS-2 3.25-inch Break
Core Mixture Level

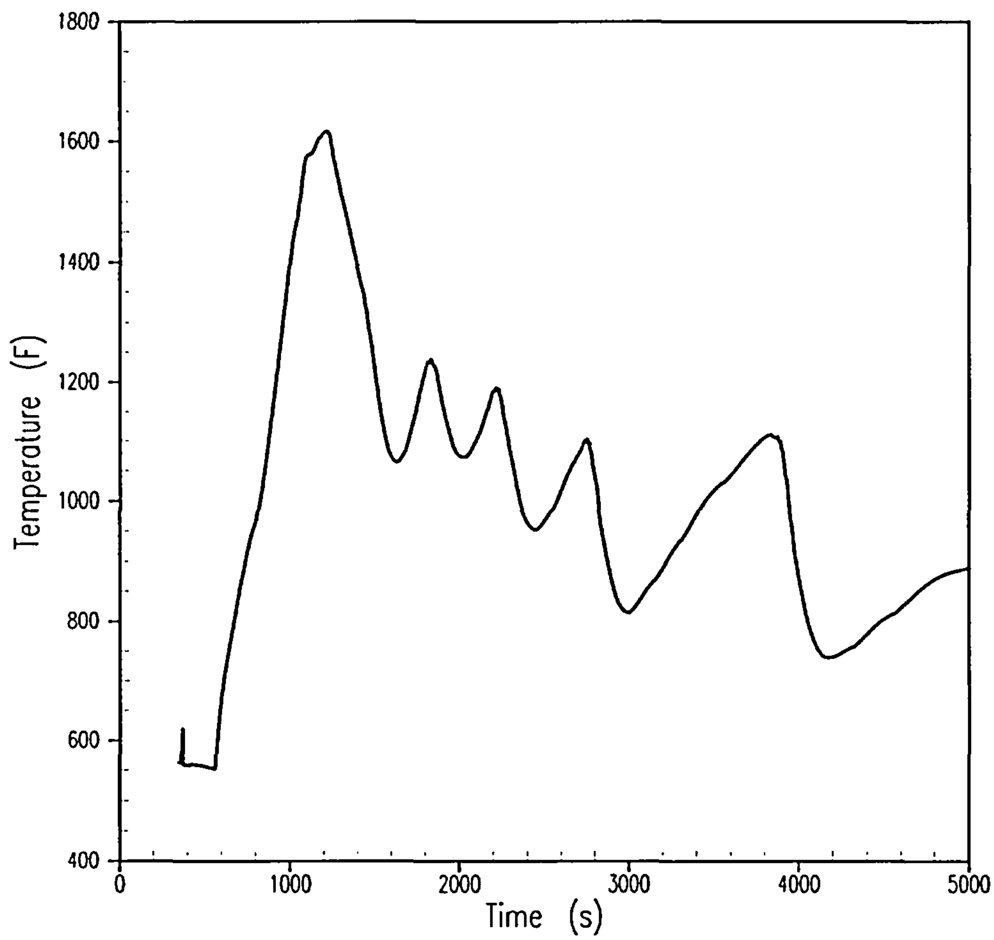

Figure 5.2.2-30A
BVPS-1 3.25-inch Break
Peak Clad Temperature

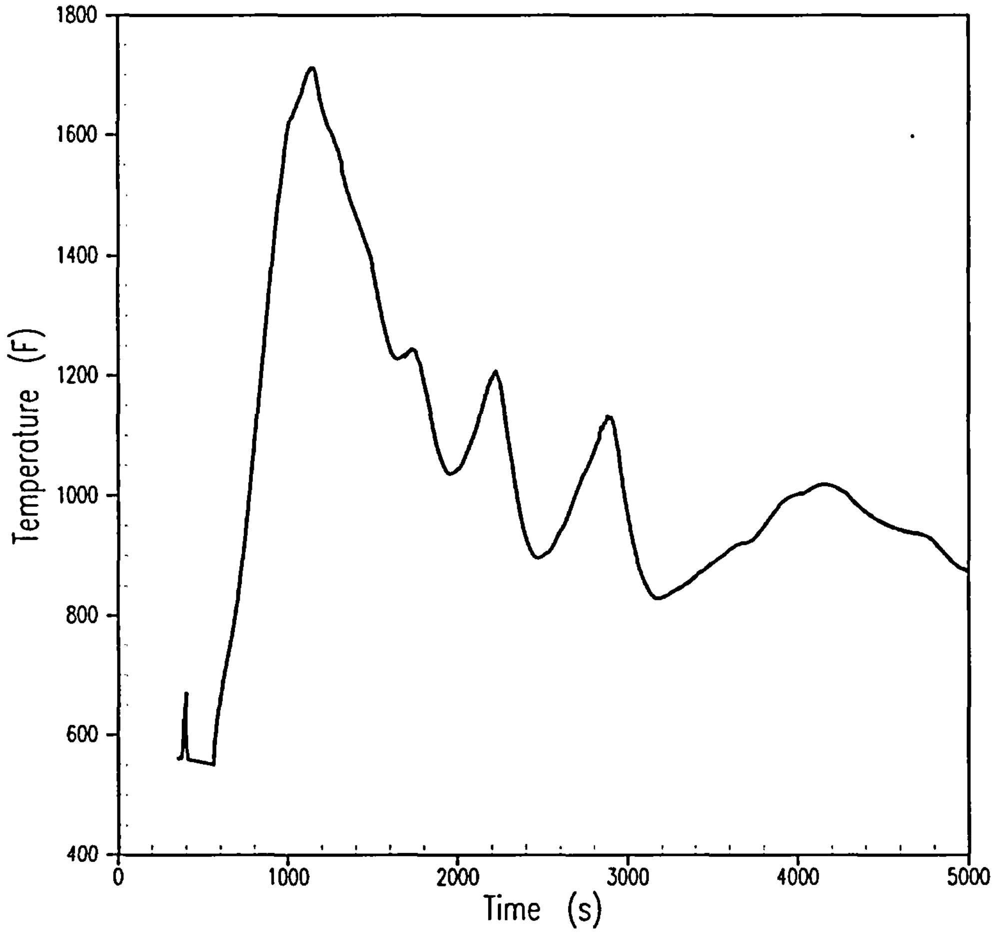

Figure 5.2.2-30B
BVPS-2 3.25-inch Break
Peak Clad Temperature

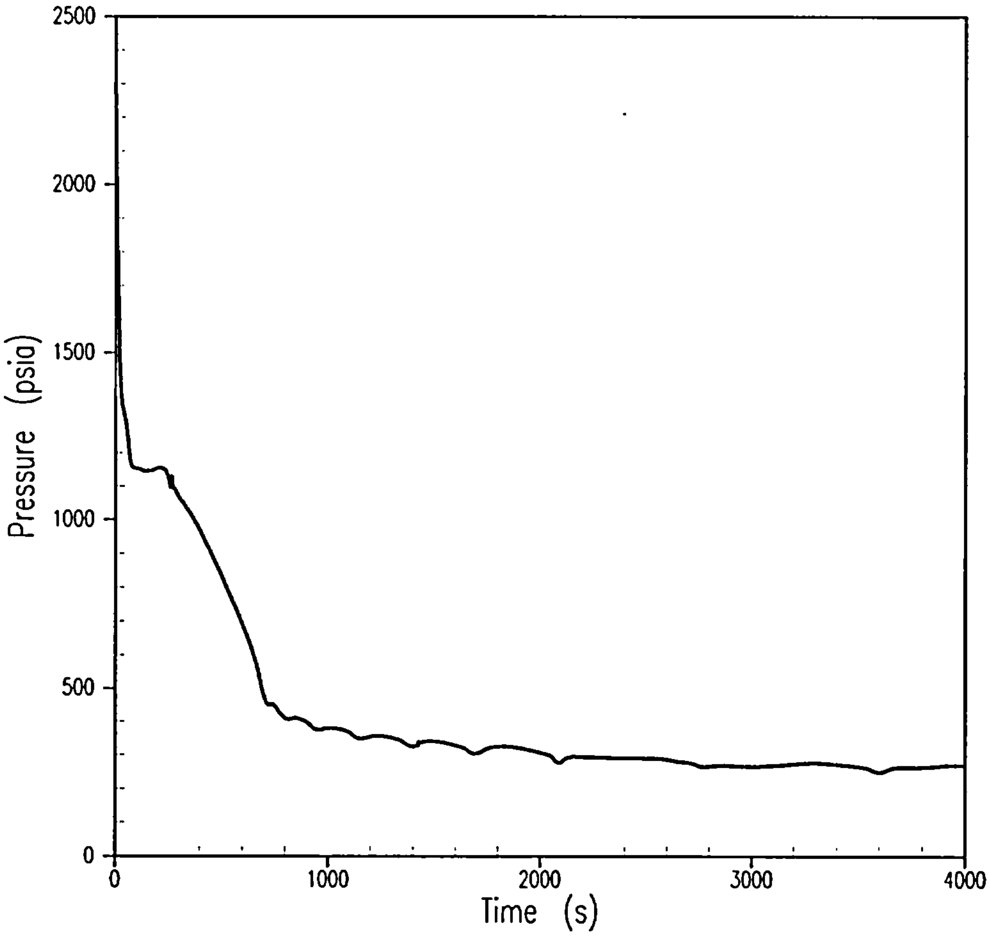

Figure 5.2.2-31A
BVPS-1 4-inch Break
RCS Pressure

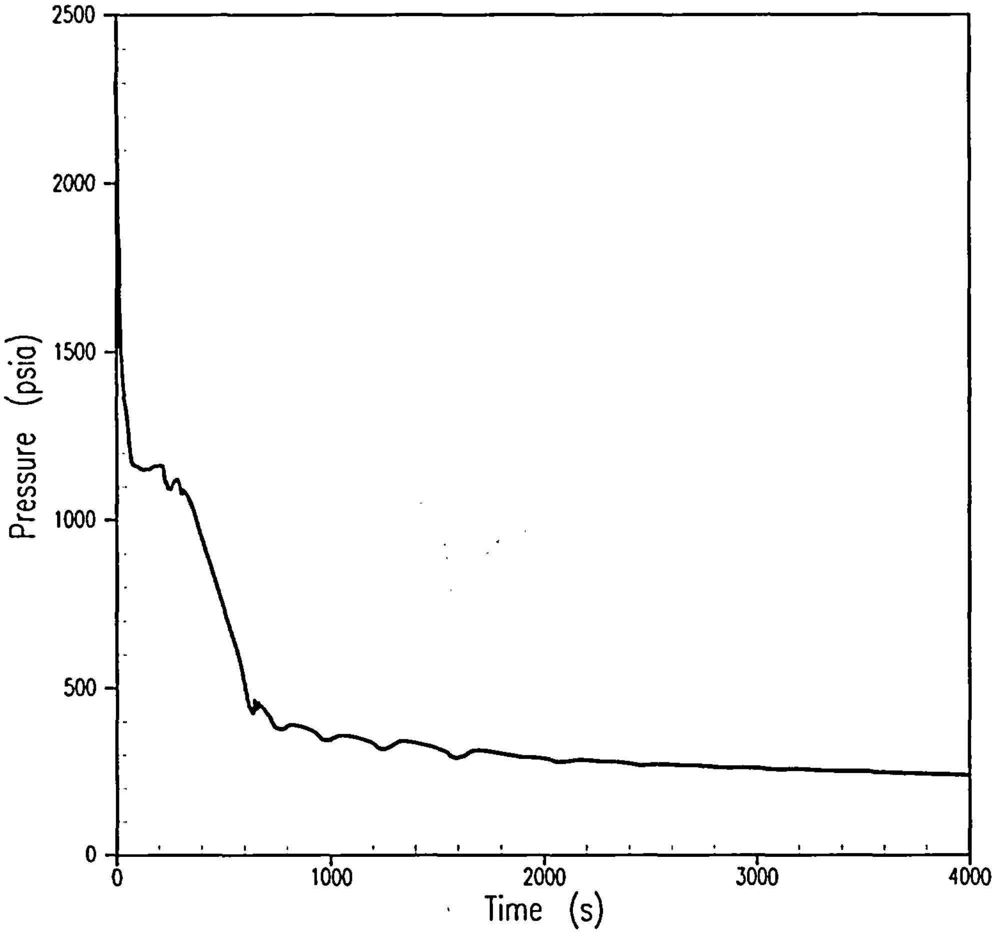

Figure 5.2.2-31B
BVPS-2 4-inch Break
RCS Pressure

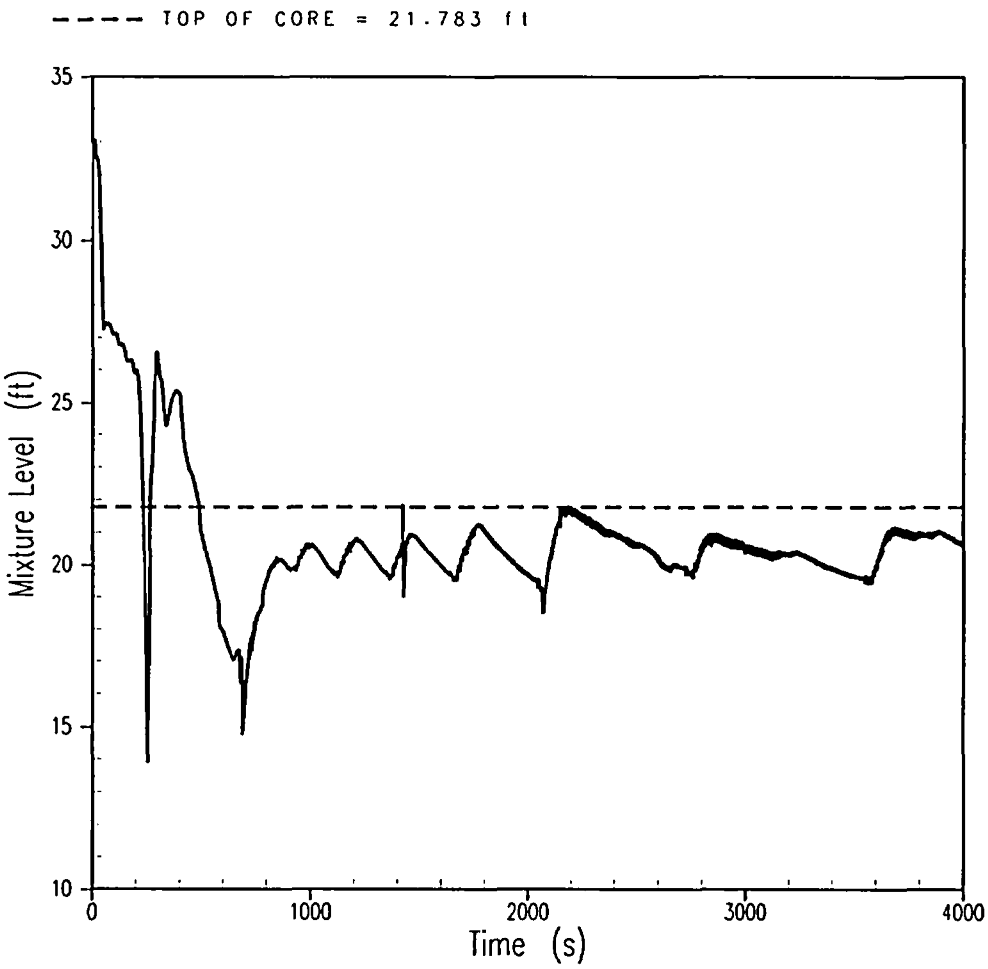

Figure 5.2.2-32A
BVPS-1 4-inch Break
Core Mixture Level

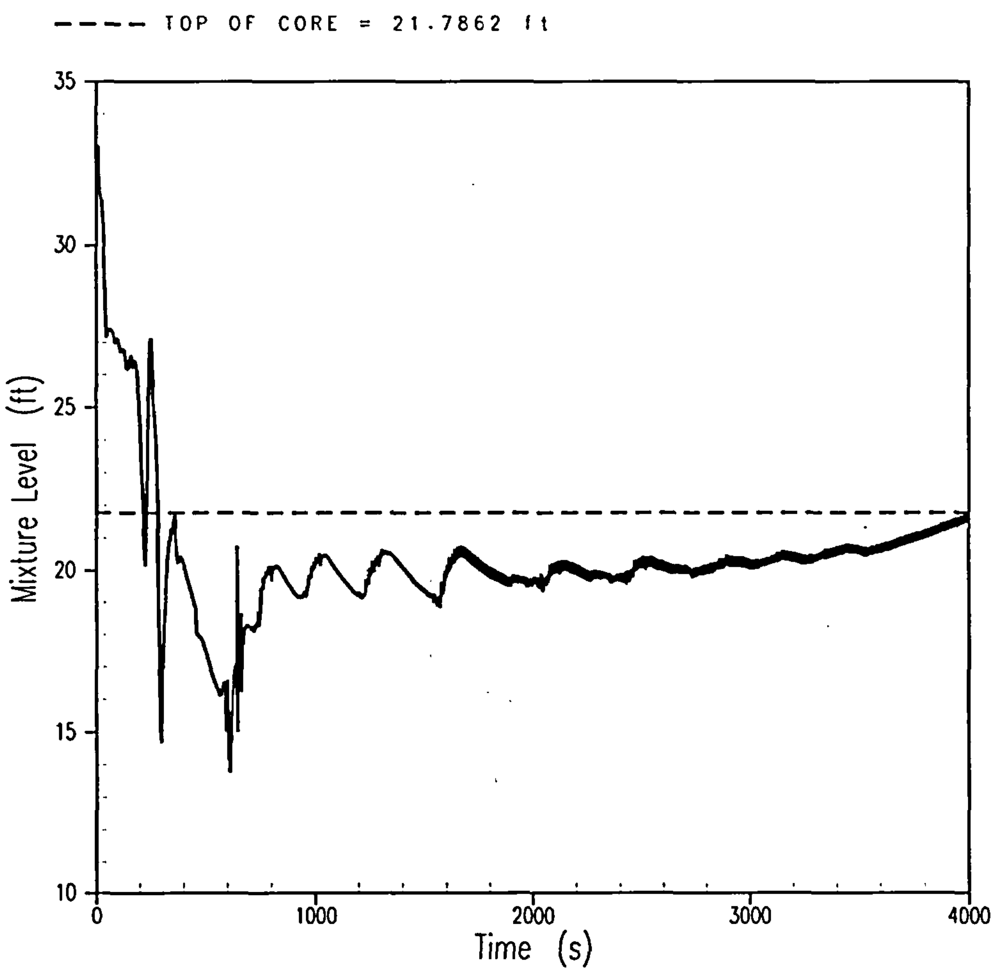

Figure 5.2.2-32B
BVPS-2 4-inch Break
Core Mixture Level

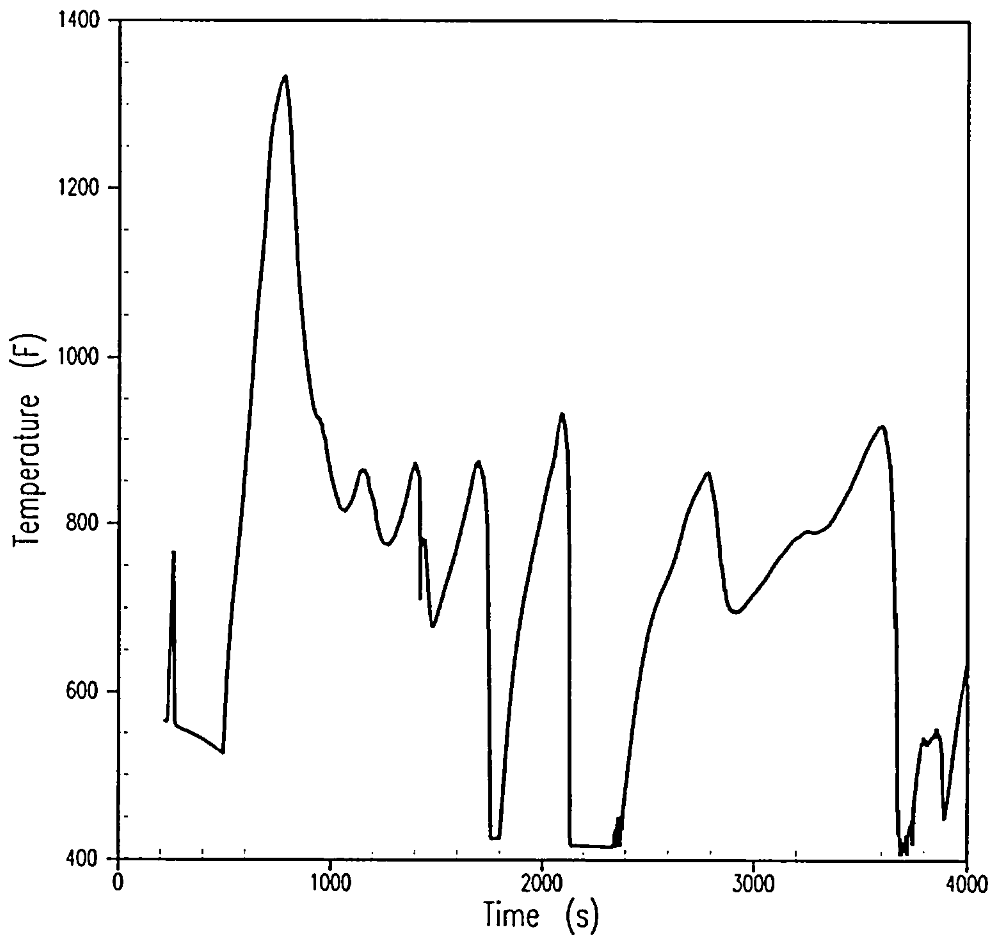

Figure 5.2.2-33A
BVPS-1 4-inch Break
Peak Clad Temperature

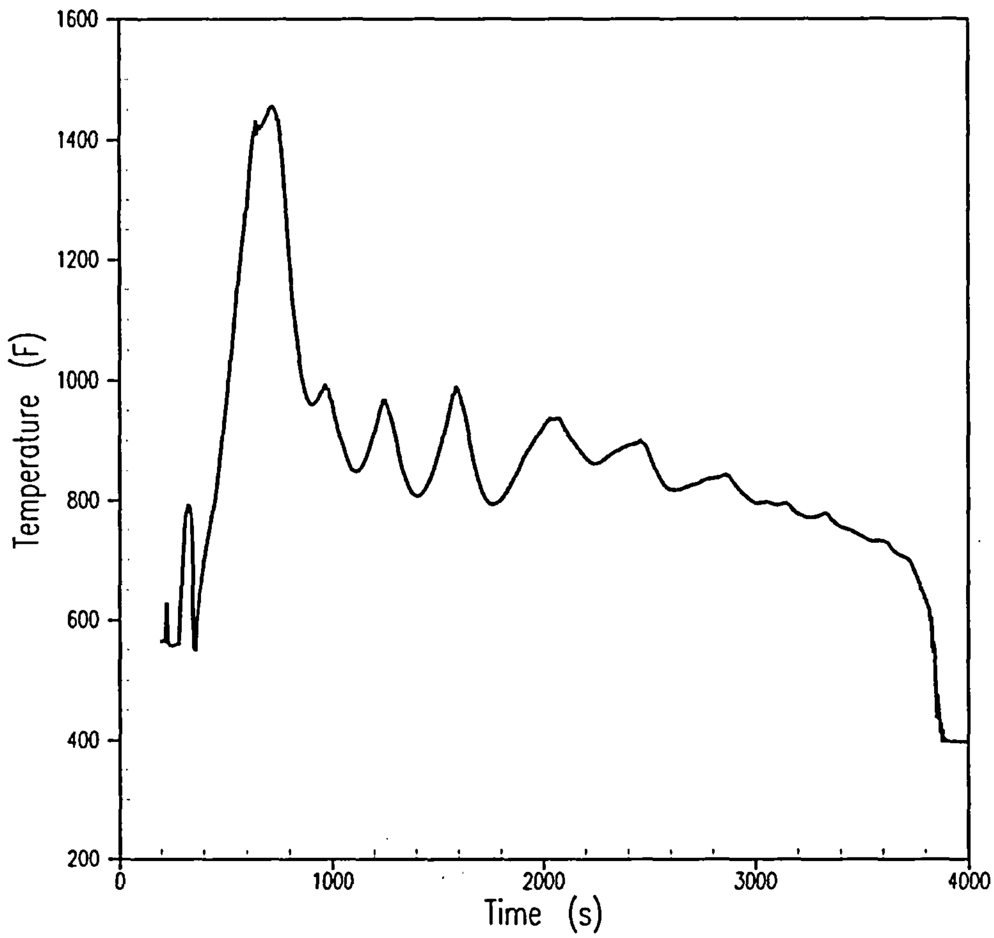

Figure 5.2.2-33B
BVPS-2 4-inch Break
Peak Clad Temperature

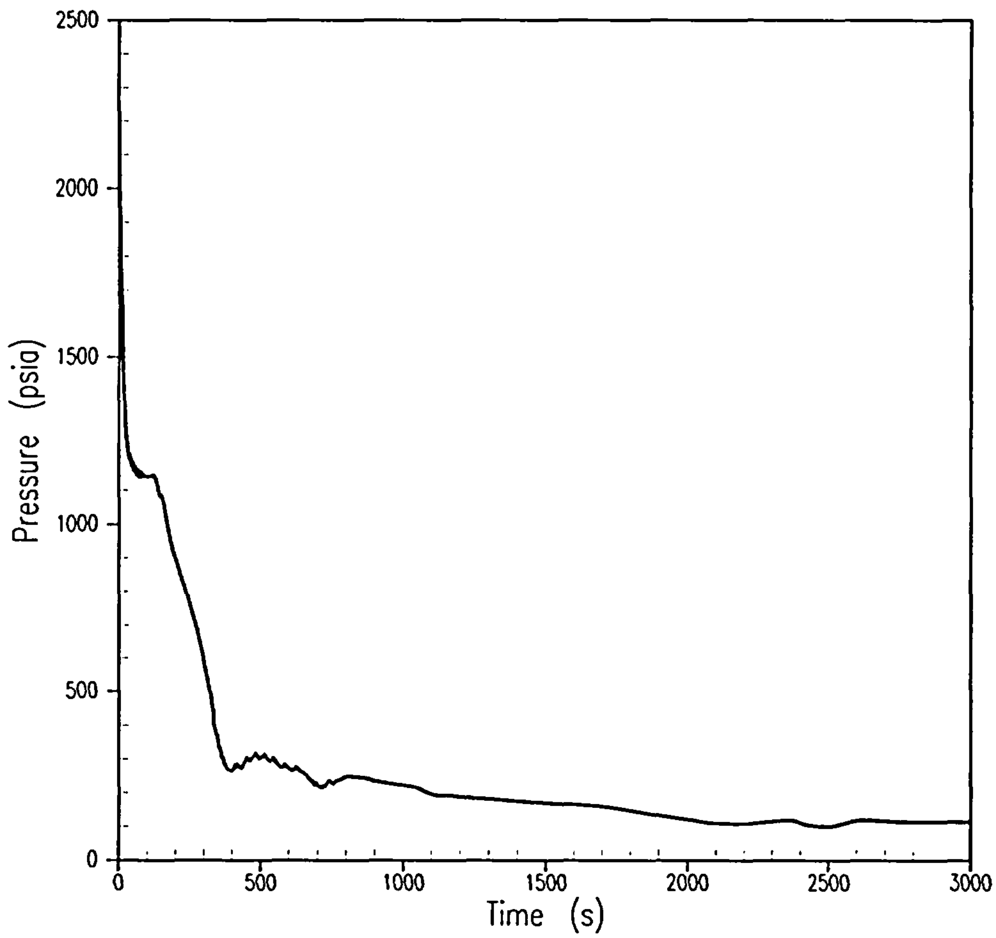

Figure 5.2.2-34A
BVPS-1 6-inch Break
RCS Pressure

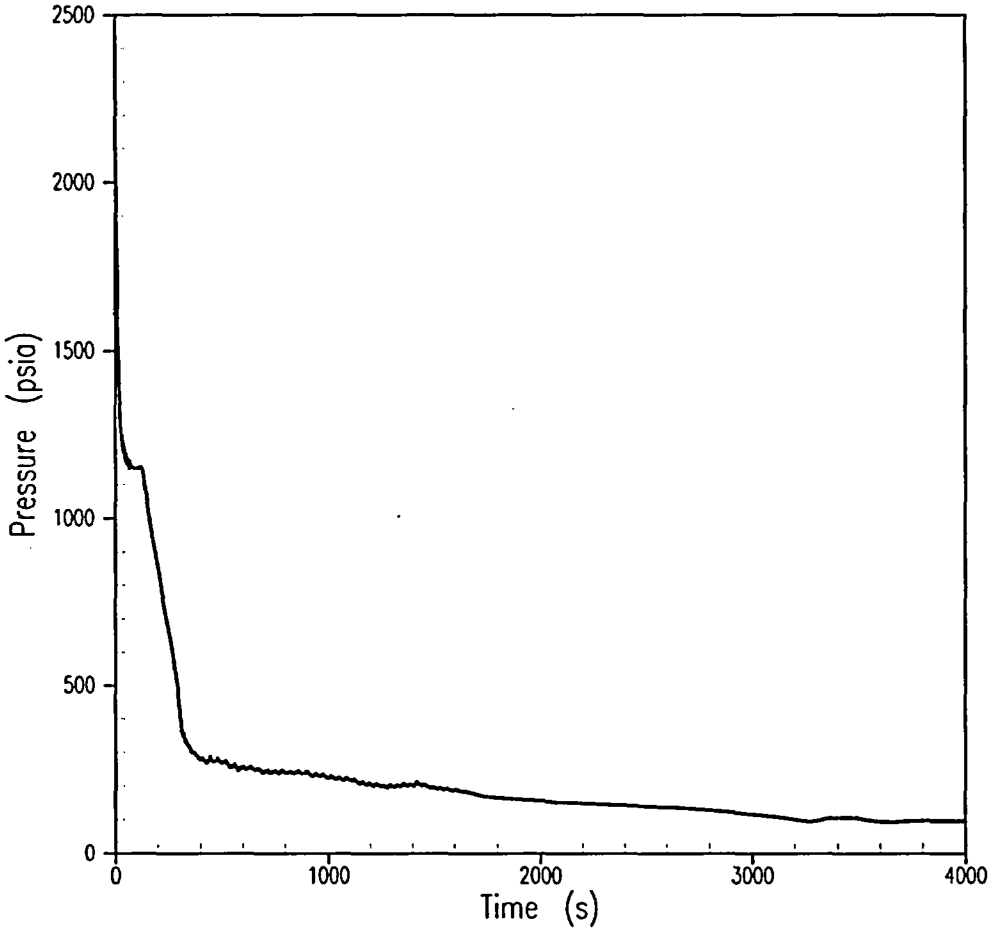

Figure 5.2.2-34B
BVPS-2 6-inch Break
RCS Pressure

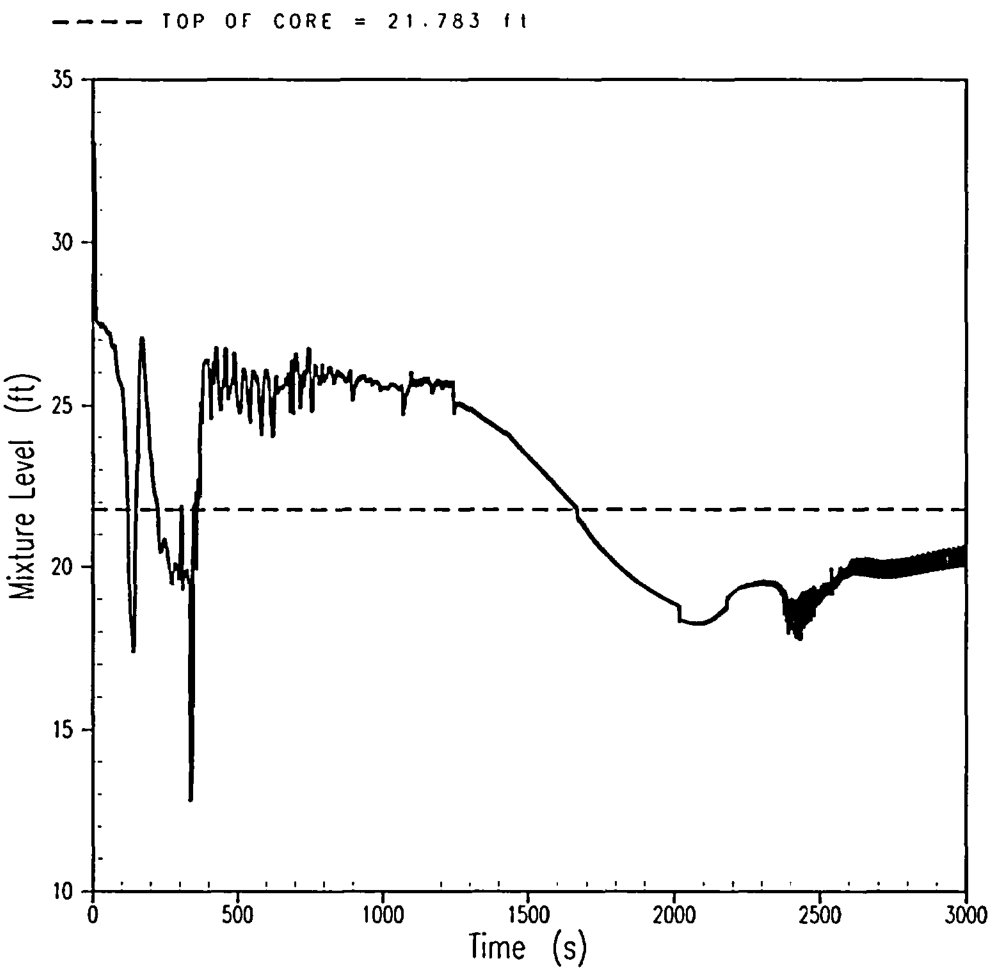

Figure 5.2.2-35A
BVPS-1 6-inch Break
Core Mixture Level

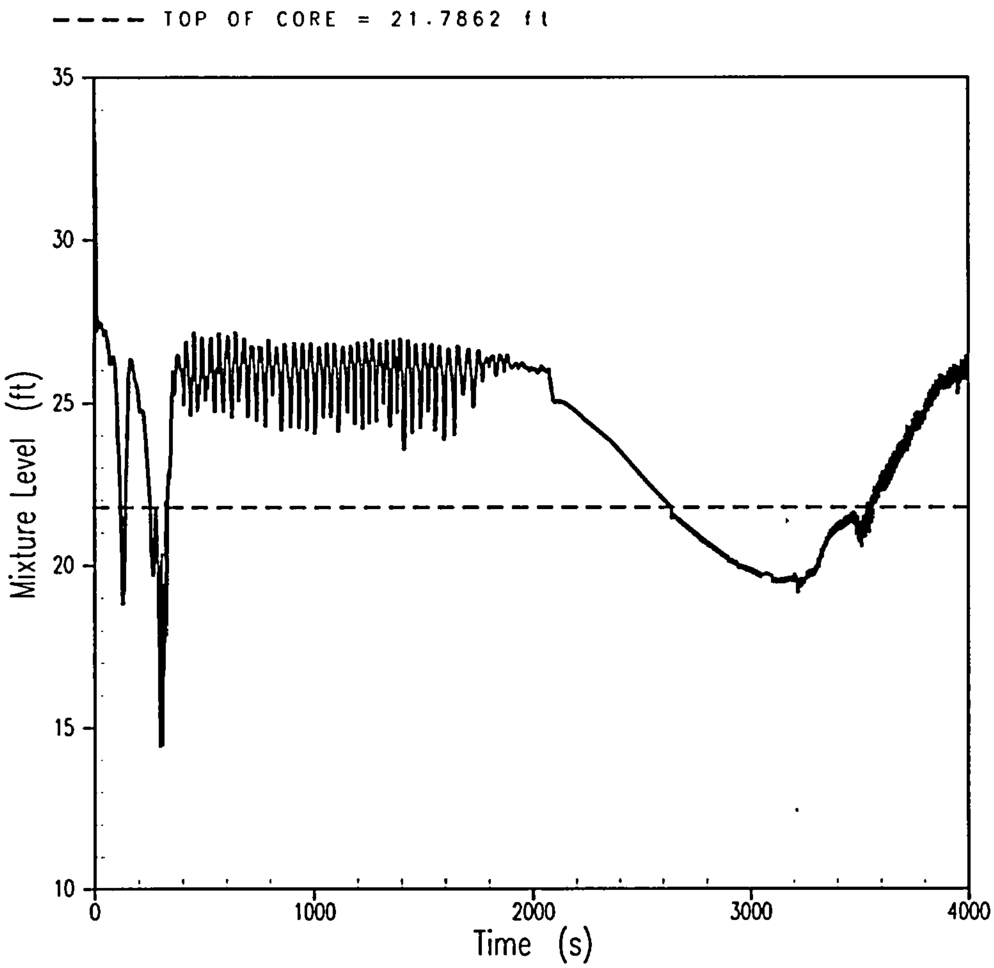

Figure 5.2.2-35B
BVPS-2 6-inch Break
Core Mixture Level

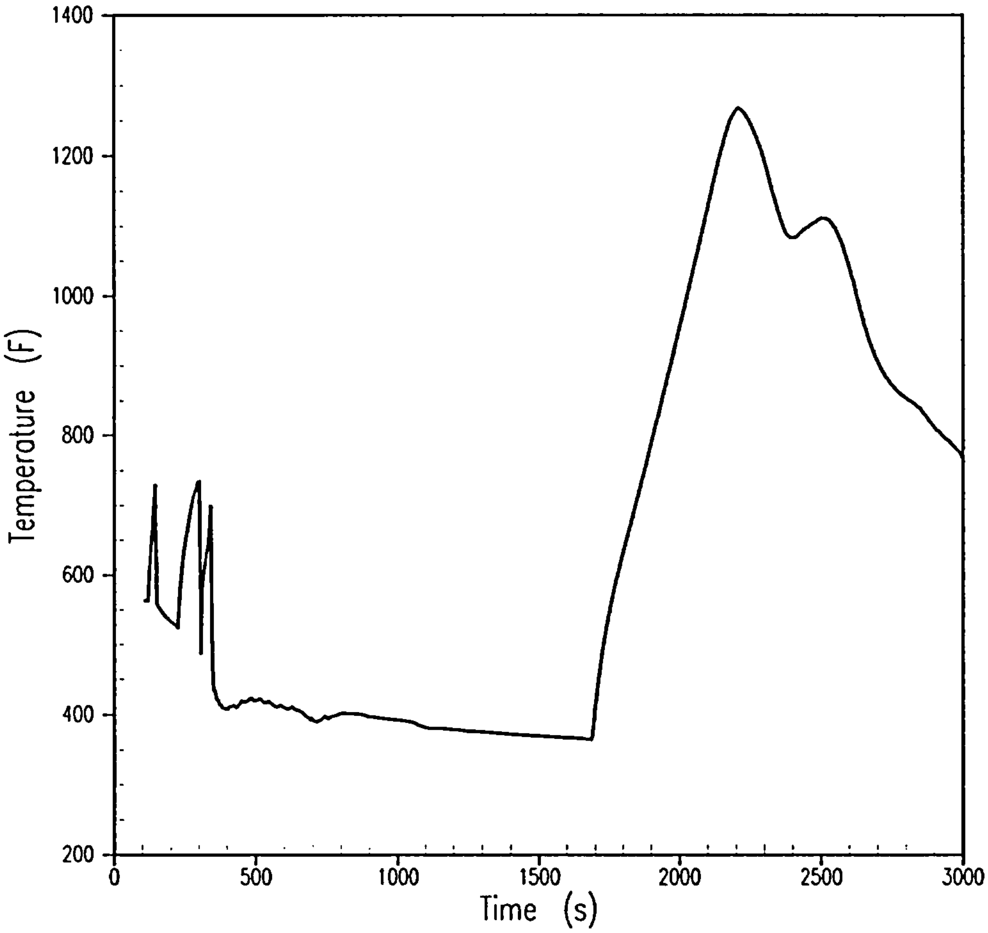

Figure 5.2.2-36A
BVPS-1 6-inch Break
Peak Clad Temperature

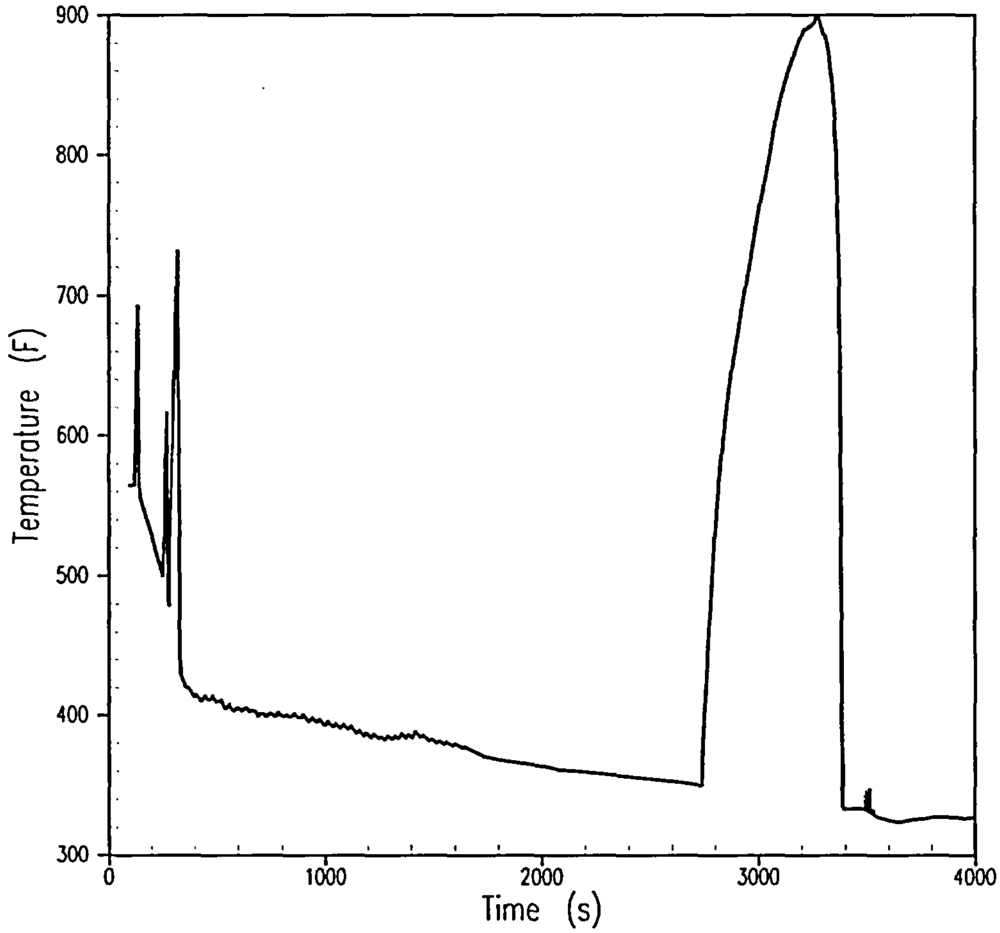

Figure 5.2.2-36B
BVPS-2 6-inch Break
Peak Clad Temperature