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L-PI-06-045
10 CFR 50.90
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U S Nuclear Regulatory Commission
ATTN: Document Control Desk
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Prairie Island Nuclear Generating Plant Units 1 and 2
Dockets 50-282 and 50-306
License Nos. DPR-42 and DPR-60

License Amendment Request (LAR) to Incorporate Large Break Loss Of Coolant Accident (LOCA) Analyses Using ASTRUM

Pursuant to 10 CFR 50.90, the Nuclear Management Company, LLC (NMC) hereby requests an amendment to the Operating License for the Prairie Island Nuclear Generating Plant (PINGP) Units 1 and 2 to incorporate new Large Break LOCA (LBLOCA) analyses using the realistic LBLOCA methodology in the NRC approved WCAP-16009-P-A, "Realistic Large Break LOCA Evaluation Methodology Using Automated Statistical Treatment of Uncertainty Method (ASTRUM)" and revise Technical Specification (TS) 5.6.5.b to include reference to WCAP-16009-P-A. This LAR fulfills the commitment NMC made in letter L-PI-06-010, dated February 2, 2006, to provide a new Large Break Loss of Coolant Accident analysis for each Prairie Island Nuclear Generating Plant unit by July 31, 2006. NMC has evaluated the proposed changes in accordance with 10 CFR 50.92 and concluded that they involve no significant hazards consideration.

Exhibit A contains the licensee's evaluation of this LAR. Exhibit B provides a markup of TS and Bases pages. Exhibit C provides the retyped TS page.

NMC requests approval of this LAR within one calendar year of the submittal date. Upon NRC approval, NMC requests implementation of the license amendment for Unit 1 for the fuel cycle which commences following the refueling scheduled for Winter 2008 (Unit 1 Cycle 25) and within 90 days from issuance for Unit 2. The Unit 1 core which will be operating one year from the date of this submittal (Unit 1 Cycle 24) was designed before results from the proposed LBLOCA analyses using WCAP-16009-P-A were available. Thus implementation of this license amendment during Unit 1 Cycle 24 will require re-analysis. NMC proposes to implement this license amendment with the next fuel cycle (Unit 1 Cycle 25) commencing following the Winter 2008 refueling to avoid the additional expense of re-analysis. The design for the Unit 2 core which will be loaded during the Fall 2006 refueling outage (Unit 2 Cycle 24) included consideration of the LBLOCA analyses using WCAP-16009-P-A and therefore, this license amendment can

be implemented within 90 days from the date of issuance for Unit 2 without additional analysis or expense.

In accordance with 10 CFR 50.91, NMC is notifying the State of Minnesota of this LAR by transmitting a copy of this letter and attachments to the designated State Official.

Summary of Commitments

This letter contains no new commitments and no revisions to existing commitments.

I declare under penalty of perjury that the foregoing is true and correct.
Executed on JUL 06 2006

Thomas J. Palmisano
Site Vice President, Prairie Island Nuclear Generating Plant Units 1 and 2
Nuclear Management Company, LLC

cc: Administrator, Region III, USNRC
Project Manager, Prairie Island, USNRC
Resident Inspector, Prairie Island, USNRC
State of Minnesota

Exhibits:

A. Licensee's Evaluation
B. Proposed Technical Specification and Bases Changes (markup)
C. Proposed Technical Specification Changes (retyped)

Exhibit A

LICENSEE'S EVALUATION

License Amendment Request (LAR) to Incorporate Large Break Loss Of Coolant Accident (LOCA) Analyses Using ASTRUM

1.0 DESCRIPTION

This is an LAR to amend Operating Licenses DPR-42 and DPR-60 for Prairie Island Nuclear Generating Plant (PINGP) Units 1 and 2 to incorporate new Large Break LOCA (LBLOCA) analyses using the realistic LBLOCA methodology in the NRC approved WCAP-16009-P-A, "Realistic Large Break LOCA Evaluation Methodology Using Automated Statistical Treatment of Uncertainty Method (ASTRUM)" and revise Technical Specification (TS) 5.6.5.b to include reference to WCAP-16009-P-A. The Nuclear Management Company, LLC (NMC) requests Nuclear Regulatory Commission (NRC) review and approval of proposed revisions

2.0 PROPOSED CHANGE

A brief description of the associated proposed TS and TS Bases changes is provided below along with a discussion of the justification for each change. The specific wording changes to the TS and Bases are provided in Exhibits B and C.

> **TS 5.6.5 Paragraph b:** This license amendment request proposes to add reference 27 to the list of documents describing NRC approved analytical methods which may be used to determine core operating limits for the PINGP units. WCAP-16009-P-A, "Realistic Large Break LOCA Evaluation Methodology Using Automated Statistical Treatment of Uncertainty Method (ASTRUM)", will be included as Reference 27. This change is acceptable based on the discussions in Section 4.0 of this LAR.

The Bases will also be revised where necessary to support the licensing basis changes. Although the Bases changes are not a part of this LAR, marked up Bases pages are included for information.

3.0 BACKGROUND

PINGP is a two unit plant located on the right bank of the Mississippi River approximately 6 miles northwest of the city of Red Wing, Minnesota. The facility is owned by the Northern States Power Company (NSP) and operated by NMC. Each unit at PINGP employs a two-loop pressurized water reactor designed and supplied by Westinghouse Electric Corporation. The initial PINGP application for a Construction Permit and Operating License was submitted to the Atomic Energy Commission (AEC) in April 1967. The Final Safety Analysis Report (FSAR) was submitted for application of an Operating License in January 1971. Unit 1 began commercial operation in December 1973 and Unit 2 began commercial operation in December 1974.

The PINGP was designed and constructed to comply with NSP's understanding of the intent of the AEC General Design Criteria (GDC) for Nuclear Power Plant Construction Permits, as proposed on July 10, 1967. PINGP was not licensed to NUREG-0800, "Standard Review Plan (SRP)."

The current PINGP LBLOCA analyses of record were performed in 1995 using the Westinghouse SECY 83-472 Methodology approved by the NRC for Large Break LOCA Analyses for Upper Plenum Injection (UPI) Plants (Addendum 4 to WCAP-10924-P-A). By letter L-PI-04-111, "Clarification of Actions for Corrections to Emergency Core Cooling System (ECCS) Evaluation Models", dated September 24, 2004 and letter L-PI-06-010, "Revised Commitment to Submit Best-Estimate Loss of Coolant Accident (LOCA) Analysis", dated February 2, 2006, NMC committed to provide the NRC new LBLOCA analyses for PINGP. The best estimate LBLOCA analyses using ASTRUM and associated TS changes proposed in this LAR are presented in fulfillment of this commitment to the NRC.

Westinghouse obtained generic NRC approval of its original topical report describing best estimate large break LOCA methodology in 1996. NRC approval of the methodology is documented in the NRC safety evaluation report appended to the topical report (Reference 1). This methodology was later extended to two-loop Westinghouse plants with upper plenum injection (UPI) in 1999 as documented in the NRC safety evaluation report appended to the UPI topical report (Reference 2).

Westinghouse recently underwent a program to revise the statistical approach used to develop the peak cladding temperature (PCT) and oxidation results at the 95th percentile. This method is still based on the Code Qualification Document (CQD) methodology (Refs. 1 and 2) and follows the steps in the Code Scaling, Applicability, and Uncertainty (CSAU) methodology. However, the uncertainty analysis (Element 3 in CSAU) is replaced by a technique based on order statistics. The ASTRUM

methodology replaces the response surface technique with a statistical sampling method where the uncertainty parameters are simultaneously sampled for each case. The approved ASTRUM evaluation model is documented in WCAP-16009-P-A (Reference 3) which includes References 1 and 2.

Separate best estimate LBLOCA analyses have been completed for each PINGP unit. This LAR summarizes the application of the Westinghouse ASTRUM BELOCA evaluation model to the PINGP Units 1 and 2 for the large break LOCA accident analysis. Tables 1-1 and 2-1 list the major plant parameter assumptions used in the best estimate LOCA analysis for Units 1 and 2, respectively. Both NMC and the analysis vendor (Westinghouse) have interface processes which identify plant configuration changes potentially impacting safety analyses. These interface processes, along with vendor internal processes for assessing evaluation model changes and errors, are used to identify the need for LOCA analyses impact assessments.

The proposed licensing basis, TS and Bases changes provide new LBLOCA analyses using the NRC approved WCAP-16009-P-A ASTRUM methodology. With these changes, PINGP will continue to operate safely and in accordance with applicable regulations

4.0 TECHNICAL ANALYSIS

4.1 Proposed TS 5.6.5 changes

This LAR proposes to incorporate new LBLOCA analyses using the methodology in the NRC reviewed and approved WCAP-16009-P-A. The results of the LOCA analyses are used to determine core operating limits. In accordance with TS 5.6.5 paragraph b, analytical methods used to determine core operating limits must be listed in TS 5.6.5. This LAR proposes to add new item 27 which references WCAP-16009-P-A.

The user note in NUREG-1431, “Standard Technical Specifications, Westinghouse Plants”, Revision 3, TS 5.6.5 states,

> Identify the Topical Report(s) by number and title or identify the staff Safety Evaluation Report for a plant specific methodology by NRC letter and date. The COLR will contain the complete identification for each of the TS referenced topical reports used to prepare the COLR (I.e., report number, title, revision, date, and any supplements).

The proposed format and content of the proposed item 27 meets the NUREG-1431 guidance.

4.2 Proposed Application of Westinghouse Best Estimate LBLOCA Methodology

4.2.1 Methodology Background

When the final acceptance criteria (FAC) governing the LOCA for Light Water Reactors was issued in 10 CFR 50.46 (Reference 4), both the NRC and the nuclear industry recognized that stipulations of 10 CFR 50 Appendix K (Appendix K) were highly conservative. That is, using the then accepted analysis methods, the performance of the ECCS would be conservatively underestimated and result in predicted peak clad temperatures (PCTs) much higher than expected. At that time, however, the degree of conservatism in the analysis could not be quantified. As a result, the NRC began a large-scale confirmatory research program with the following objectives:

1. Identify, through separate effects and integral effects experiments, the degree of conservatism in those models required in the Appendix K rule. In this fashion, those areas in which a purposely prescriptive approach was used in the Appendix K rule could be quantified with additional data so that a less prescriptive future approach might be allowed.

2. Develop improved thermal-hydraulic computer codes and models so that more accurate and realistic accident analysis calculations could be performed. The purpose of this research was to develop an accurate predictive capability so that the uncertainties in the ECCS performance and the degree of conservatism with respect to the Appendix K limits could be quantified.

Since that time, the NRC and the nuclear industry have sponsored reactor safety research programs directed at meeting the above two objectives. The overall results have quantified the conservatism in the Appendix K rule for LOCA analyses and confirmed that some relaxation of the rule can be made without a loss in safety to the public. It was also found that some plants were being restricted in operating flexibility by overly conservative Appendix K requirements. In recognition of the Appendix K conservatism that was being quantified by the research programs, the NRC adopted an interim approach for evaluation methods. This interim approach is described in SECY-83-472 (Reference 5). The SECY-83-472 approach retained those features of Appendix K that were legal requirements, but permitted applicants to use best estimate thermal-hydraulic models in their ECCS evaluation model. Thus, SECY-83-472 represented an important step in basing licensing decisions on realistic calculations, as opposed to those calculations prescribed by Appendix K.

In 1988, the NRC Staff amended the requirements of 10 CFR 50.46 and Appendix K, "ECCS Evaluation Models", to permit the use of a realistic evaluation model to analyze the performance of the ECCS during a hypothetical LOCA. This decision was based on an improved understanding of LOCA thermal-hydraulic phenomena gained by extensive research programs. Under the amended rules, best estimate thermal-hydraulic models may be used in place of models with Appendix K features. The rule

change also requires, as part of the LOCA analysis, an assessment of the uncertainty of the best estimate calculations. It further requires that this analysis uncertainty be included when comparing the results of the calculations to the prescribed acceptance criteria of 10 CFR 50.46. Further guidance for the use of best estimate codes is provided in Regulatory Guide 1.157 (Reference 6).

To demonstrate use of the revised ECCS rule, the NRC and its consultants developed a method called the Code Scaling, Applicability, and Uncertainty (CSAU) evaluation methodology (Reference 7). This method outlined an approach for defining and qualifying a best estimate thermal-hydraulic code and quantifying the uncertainties in a LOCA analysis.

A LOCA evaluation methodology for three and four loop PWR plants based on the revised 10 CFR 50.46 rules was developed by Westinghouse with the support of EPRI and Consolidated Edison and was approved by the NRC (Ref. 1). This methodology was later extended to two-loop Westinghouse plants with upper plenum injection (UPI) in 1999 as documented in the NRC safety evaluation report appended to the UPI topical report (Ref.2).

More recently, Westinghouse developed an alternative uncertainty methodology called ASTRUM, which stands for Automated Statistical TReatment of Uncertainty Method (Ref. 3). This method is still based on the CQD methodology and follows the steps in the CSAU methodology. However, the uncertainty analysis (Element 3 in the CSAU) is replaced by a technique based on order statistics. The ASTRUM methodology replaces the response surface technique with a statistical sampling method where the uncertainty parameters are simultaneously sampled for each case. The ASTRUM methodology has received NRC approval for referencing in licensing calculations in WCAP-16009-P-A (Ref. 3). The ASTRUM methodology remains applicable to three and four loop pressurized water reactors (PWRs), as well as two-loop Westinghouse plants with UPI.

The ASTRUM methodology requires the execution of 124 transients to determine a bounding estimate of the 95th percentile of the PCT, local maximum oxidation (LMO), and core wide oxidation (CWO) with 95% confidence level. These parameters are needed to satisfy the 10 CFR 50.46 criteria with regard to PCT, LMO, and CWO.

Downcomer boiling is considered as appropriate in the ASTRUM methodology. The WCOBRA/TRAC computer code determines if downcomer boiling will occur for a particular transient. If downcomer boiling is determined to occur in a transient, WCOBRA/TRAC includes the effects of downcomer boiling in the transient calculation.

This analysis is in accordance with the applicability limits and usage conditions defined in Section 13-3 of WCAP-16009-P-A (Reference 3) as applicable to the ASTRUM methodology. Section 13-3 of WCAP-16009-P-A (Reference 3) was found to acceptably disposition each of the identified conditions and limitations related to WCOBRA/TRAC and the CQD uncertainty approach per Section 4.0 of the ASTRUM

Final Safety Evaluation Report appended to this WCAP. The Best Estimate LBLOCA analyses and associated models for PINGP Units 1 and 2 are each unit-specific.

4.2.2 Description of a Large Break LOCA Transient

Before the break occurs, the Reactor Coolant System (RCS) is assumed to be operating normally at full power in an equilibrium condition, that is, the heat generated in the core is being removed via the secondary system. A large break is assumed to open instantaneously in one of the main RCS cold leg pipes.

Immediately following the cold leg break, a rapid system depressurization occurs along with a core flow reversal due to a high discharge of sub-cooled fluid into the broken cold leg and out of the break. The fuel rods go through departure from nucleate boiling (DNB) and the cladding rapidly heats up, while the core power decreases due to voiding in the core. The hot water in the core, upper plenum, and upper head flashes to steam, and subsequently the cooler water in the lower plenum and downcomer begins to flash. Once the system has depressurized to the accumulator pressure, the accumulator begins to inject cold borated water into the intact cold leg. During the blowdown period, a portion of the injected ECCS water is calculated to be bypassed around the downcomer and out of the break. The bypass period ends as the system pressure continues to decrease and approaches the containment pressure, resulting in reduced break flow and consequently, reduced core flow.

As the refill period begins, the core continues to heat up as the vessel begins to fill with ECCS water. This phase continues until the lower plenum is filled, the bottom of the core begins to reflood, and entrainment begins.

During the reflood period, the core flow is oscillatory as ECCS water periodically rewets and quenches the hot fuel cladding, which generates steam and causes system re-pressurization. The steam and entrained water must pass through the vessel upper plenum, the hot legs, the steam generators, and the reactor coolant pumps before it is vented out of the break. This flow path resistance is overcome by the downcomer water elevation head, which provides the gravity driven reflood force. The pumped upper plenum and cold leg injection ECCS water aids in the filling of the vessel and downcomer, which subsequently supplies water to maintain the core and downcomer water levels and complete the reflood period.

4.3 PINGP Realistic Large Break LOCA Analysis Results

4.3.1 PINGP Unit 1

4.3.1.1 Analysis Results

Table 1-1 lists the major plant parameter assumptions used in the best estimate LOCA analysis for Unit 1. The results of the PINGP Unit 1 ASTRUM analysis are summarized in Table 1-2. Table 1-3 contains a sequence of events for the limiting PCT transient.

The scatter plot presented on Figure 1-1 shows the effect of the effective break area on the analysis PCT. The effective break area is calculated by multiplying the discharge coefficient C_D with the sampled value of the break area, normalized to the cold-leg cross sectional area. Figure 1-1 is provided to show the break area is a significant contributor to the variation in PCT.

From the 124 calculations performed as part of the ASTRUM analysis, different cases proved to be the limiting PCT and limiting LMO transients for PINGP Unit 1. Figure 1-2 shows the predicted clad temperature transient at the PCT limiting elevation for the limiting PCT case. Figure 1-3 presents the clad temperature transient predicted at the LMO elevation for the limiting LMO case. Due to the relatively low PCT results, CWO remained on the order of 0 (zero) for all cases.

Figures 1-4 through 1-18 illustrate the key major response parameters for the limiting PCT transient. The reference point for the lower plenum liquid level presented in Figure 1-12 is the bottom of the vessel (8.4 feet below the bottom of active fuel). The reference point for the downcomer liquid level presented in Figure 1-13 is the point at which the outside of the core barrel, if extended downward, intersects with the vessel wall (5.8 feet below the bottom of active fuel). The reference point for the core collapsed liquid levels presented in Figures 1-14 and 1-17 is the bottom of the active fuel.

The containment backpressure utilized for the LBLOCA analysis compared to the calculated containment backpressure is provided as Figure 1-19. The worst single failure for the LBLOCA analysis is the loss of one train of ECCS injection (consistent with the ASTRUM Topical), however, all containment systems which would reduce containment pressure are modeled for the LBLOCA containment backpressure calculation.

4.3.1.2 10 CFR 50.46 Requirements

Compliance with 10 CFR 50.46 requires demonstration that there is a high level of probability that the limits set forth in the regulation are met:

(b)(1) The limiting P CT corresponds to a bounding estimate of the 95th percentile PCT at the 95-percent confidence level. Since the resulting PCT for the limiting case is 1594°F for PINGP Unit 1, the analysis confirms that 10 CFR 50.46 acceptance criterion (b)(1), that is, "Peak Clad Temperature less than 2200 °F", is demonstrated. The result is shown in Table 1-2.

(b)(2) The maximum cladding oxidation corresponds to a bounding estimate of the 95th percentile LMO at the 95-percent confidence level. Since the resulting LMO for the limiting case is 0.2 percent for PINGP Unit 1, the analysis confirms that 10 CFR 50.46 acceptance criterion (b)(2), that is, "Local Maximum Oxidation of the cladding less than 17 percent", is demonstrated. The result is shown in Table 1-2.

(b)(3) The limiting core-wide oxidation corresponds to a bounding estimate of the 95th percentile CWO at the 95-percent confidence level. While the limiting LMO is determined based on the single Hot Rod, the CWO value can be conservatively chosen as that calculated for the limiting Hot Assembly Rod (HAR) when there is significant margin to the regulatory limit. The limiting HAR total maximum oxidation is 0 (zero) percent for PINGP Unit 1. Thus, a detailed CWO calculation is not needed because the calculations would include many lower power assemblies and the outcome would always be less than the limiting HAR total maximum oxidation. Therefore, the analysis confirms that 10 CFR 50.46 acceptance criterion (b)(3), that is, "Core-Wide Oxidation less than 1 percent", is demonstrated. The result is shown in Table 1-2.

(b)(4) 10 CFR 50.46 acceptance criterion (b)(4) requires that the calculated changes in core geometry are such that the core remains amenable to cooling. This criterion has historically been satisfied by adherence to criteria (b)(1) and (b)(2), and by assuring that fuel deformation due to combined LOCA and seismic loads is specifically addressed. The analysis demonstrated that the PCT and maximum cladding oxidation limits remain in effect for Best-Estimate LOCA applications. The grid crush calculations currently in place for PINGP Unit 1 remain unchanged with the application of the ASTRUM methodology (Reference 3), therefore, acceptance criterion (b)(4) is satisfied.

(b)(5) 10 CFR 50.46 acceptance criterion (b)(5) requires that long-term core cooling be provided following the successful initial operation of the ECCS. Long-term cooling is dependent on the demonstration of continued delivery of cooling water to the core. The actions, automatic or manual, that are currently in place at PINGP Unit 1 to maintain long-term cooling remain unchanged with the application of the best estimate LOCA ASTRUM methodology (Reference 3).

Based on the ASTRUM analysis results (see Table 1-2), it is concluded that PINGP Unit 1 continues to maintain a margin of safety to the limits prescribed by 10 CFR 50.46.

4.3.2 PINGP Unit 2

4.3.2.1 Analysis Results

Table 2-1 lists the major plant parameter assumptions used in the best estimate LOCA analysis for Unit 2. The results of the PINGP Unit 2 ASTRUM analysis are summarized in Table 2-2. Table 2-3 contains a sequence of events for the limiting PCT transient.

The scatter plot presented on Figure 2-1 shows the effect of the effective break area on the analysis PCT. The effective break area is calculated by multiplying the discharge coefficient, C_D, with the sample value of the break area, normalized to the cold-leg cross sectional area. Figure 2-1 is provided to show the break area is a significant contributor to the variation in PCT.

From the 124 calculation performed as part of the ASTRUM analysis, different cases proved to be the limiting PCT and limiting LMO transients for PINGP Unit 2. Figure 2-2

shows the predicted clad temperature transient at the PCT limiting elevation for the limiting PCT case. Figure 2-3 presents the clad temperature transient predicted at the LMO elevation for the limiting LMO case. Due to the relatively low PCT results, CWO remained on the order of 0 (zero) for all cases.

Figures 2-4 through 2-18 illustrate the key major response parameters for the limiting PCT transient. The reference point for the lower plenum liquid level presented in Figure 2-12 is the bottom of the vessel (8.4 feet below the bottom of active fuel). The reference point for the downcomer liquid level presented in Figure 2-13 is the point at which the outside of the core barrel, if extended downward, intersects with the vessel wall (5.8 feet below the bottom of active fuel). The reference point for the core collapsed liquid levels presented in Figures 2-14 and 2-17 is the bottom of the active fuel.

The containment backpressure utilized for the LBLOCA analysis compared to the calculated containment backpressure is provided as Figure 2-19. The worst single failure for the LBLOCA analysis is the loss of one train of ECCS injection (consistent with the ASTRUM Topical), however, all containment systems which would reduce containment pressure are modeled for the LBLOCA containment backpressure calculation.

4.3.2.2 10 CFR 50.46 Requirements

Compliance with 10 CFR 50.46 requires demonstration that there is a high level of probability that the limits set forth in the regulation are met:

(b)(1) The limiting PCT corresponds to a bounding estimate of the 95^{th} percentile PCT at the 95-percent confidence level. Since the resulting PCT for the limiting case is 1546°F for PINGP Unit 2, the analysis confirms that 10 CFR 50.46 acceptance criterion (b)(1), that is, "Peak Clad Temperature less than 2200 °F", is demonstrated. The result is shown in Table 2-2.

(b)(2) The maximum cladding oxidation corresponds to a bounding estimate of the 95^{th} percentile LMO at the 95-percent confidence level. Since the resulting LMO for the limiting case is 0.5 percent for PINGP Unit 2, the analysis confirms that 10 CFR 50.46 acceptance criterion (b)(2), that is, "Local Maximum Oxidation of the cladding less than 17 percent", is demonstrated. The result is shown in Table 2-2.

(b)(3) The limiting core -wide oxidation corresponds to a bounding estimate of the 95^{th} percentile CWO at the 95-percent confidence level. While the limiting LMO is determined based on the single Hot Rod, the CWO value can be conservatively chosen as that calculated for the limiting Hot Assembly Rod (HAR) when there is significant margin to the regulatory limit. The limiting HAR total maximum oxidation is 0 (zero) percent for PINGP Unit 2. Thus, a detailed CWO calculation is not needed because the calculations would include many lower power assemblies and the outcome would always be less than the limiting HAR total

maximum oxidation. Therefore, the analysis confirms that 10 CFR 50.46 acceptance criterion (b)(3), that is, "Core-Wide Oxidation less than 1 percent", is demonstrated. The result is shown in Table 2-2.

(b)(4) 10 CFR 50.46 acceptance criterion (b)(4) requires that the calculated changes in core geometry are such that the core remains amenable to cooling. This criterion has historically been satisfied by adherence to criteria (b)(1) and (b)(2), and by assuring that fuel deformation due to combined LOCA and seismic loads is specifically addressed. The analysis demonstrated that the PCT and maximum cladding oxidation limits remain in effect for Best-Estimate LOCA applications. The grid crush calculations currently in place for PINGP Unit 2 remain unchanged with the application of the ASTRUM methodology (Reference 3), therefore, acceptance criterion (b)(4) is satisfied.

(b)(5) 10 CFR 50.46 acceptance criterion (b)(5) requires that long -term core cooling be provided following the successful initial operation of the ECCS. Long-term cooling is dependent on the demonstration of continued delivery of cooling water to the core. The actions, automatic or manual, that are currently in place at PINGP Unit 2 to maintain long-term cooling remain unchanged with the application of the use of best estimate LOCA ASTRUM methodology (Reference 3).

Based on the ASTRUM analysis results (see Table 2-2), it is concluded that PINGP Unit 2 continues to maintain a margin of safety to the limits prescribed by 10 CFR 50.46.

4.4 Conclusions

Since the issuance of 10 CFR 50 Appendix K, the NRC and nuclear industry have developed improved thermal-hydraulic computer codes and models that more accurately and realistically perform accident analysis calculations. Westinghouse has developed the ASTRUM methodology for performing best estimate LBLOCA analyses as documented in WCAP-16009-P-A. The NRC has approved WCAP-16009-P-A for application to Westinghouse two-loop plants with UPI. PINGP is a Westinghouse two-loop plant with UPI.

LBLOCA analyses have been performed for each PINGP unit using the ASTRUM methodology. The results demonstrate that the acceptance criteria of 10 CFR 50.46 are met for both units.

This LAR proposes to incorporate the best estimate LBLOCA analyses using ASTRUM in the PINGP licensing basis and make concomitant TS changes to include WCAP-16009-P-A in the list of NRC approved methods for establishing core operating limits. Operation and maintenance of the Prairie Island Nuclear Generating Plant with the proposed licensing basis changes and Technical Specification revisions will continue to protect the health and safety of the public.

5.0 REGULATORY SAFETY ANALYSIS

5.1 No Significant Hazards Consideration

The Nuclear Management Company has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below for each of these characterizations:

1. **Do the proposed changes involve a significant increase in the probability or consequences of an accident previously evaluated?**

   Response: No

   This license amendment request proposes to incorporate large break loss of coolant accident analyses using the ASTRUM methodology, documented in WCAP-16009-P-A, "Realistic Large Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)", in the Prairie Island Nuclear Generating Plant licensing basis and add reference to WCAP-16009-P-A in the Technical Specification's list of approved methodologies for establishing core operating limits.

   Accident analyses are not accident initiators, therefore, this proposed licensing basis change does not involve a significant increase in the probability of an accident. The analyses using ASTRUM demonstrated that the acceptance criteria in 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors", were met. The NRC has approved WCAP-16009-P-A for application to two-loop Westinghouse plants with upper plenum injection. Since the Prairie Island Nuclear Generating Plant is a two-loop Westinghouse plants with upper head injection and the analysis results meet the 10 CFR 50.46 acceptance criteria, this change does not involve a significant increase in the consequences of an accident.

   Addition of the reference to WCAP-16009-P-A in the Technical Specifications is an administrative change that does not affect the probability or consequences of an accident previously evaluated.

   The changes proposed in this license amendment do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Do the proposed changes create the possibility of a new or different kind of accident from any accident previously evaluated?**

   Response: No

This license amendment request proposes to incorporate large break loss of coolant accident analyses using the ASTRUM methodology, documented in WCAP-16009-P-A, "Realistic Large Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)", in the Prairie Island Nuclear Generating Plant licensing basis and add reference to WCAP-16009-P-A in the Technical Specification's list of approved methodologies for establishing core operating limits.

There are no physical changes being made to the plant as a result of using the Westinghouse ASTRUM analysis methodology in WCAP-16009-P-A for performance of the large break loss of coolant accident analyses. No new modes of plant operation are being introduced. The configuration, operation and accident response of the structures or components are unchanged by utilization of the new analysis methodology. Analyses of transient events have confirmed that no transient event results in a new sequence of events that could lead to a new accident scenario. The parameters assumed in the analysis are within the design limits of existing plant equipment.

In addition, employing the Westinghouse ASTRUM large break loss of coolant accident analysis methodology does not create any new failure modes that could lead to a different kind of accident. The design of all systems remains unchanged and no new equipment or systems have been installed which could potentially introduce new failure modes or accident sequences. No changes have been made to any reactor protection system or emergency safeguards features instrumentation actuation setpoints.

Based on this review, it is concluded that no new accident scenarios, failure mechanisms or limiting single failures are introduced as a result of the proposed methodology changes.

Addition of the reference to WCAP-16009-P-A in the Technical Specifications is an administrative change that does not create the possibility of a new or different kind of accident.

The licensing basis and Technical Specification changes proposed in this license amendment do not create the possibility of a new or different kind of accident from any previously evaluated.

3. Do the proposed changes involve a significant reduction in a margin of safety?

Response: No

This license amendment request proposes to incorporate large break loss of coolant accident analyses using the ASTRUM methodology, documented in WCAP-16009-P-A, "Realistic Large Break LOCA Evaluation Methodology Using

the Automated Statistical Treatment of Uncertainty Method (ASTRUM)", in the Prairie Island Nuclear Generating Plant licensing basis and add reference to WCAP-16009-P-A in the Technical Specification's list of approved methodologies for establishing core operating limits.

The analyses using ASTRUM demonstrated that the applicable acceptance criteria in 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors" are met. Margins of safety for large break loss of coolant accidents include quantitative limits for fuel performance established in 10 CFR 50.46. These acceptance criteria and the associated margins of safety are not being changed by this proposed new methodology. The NRC has approved WCAP-16009-P-A for application to two-loop Westinghouse plants with upper head injection. Since the Prairie Island Nuclear Generating Plant is a two-loop Westinghouse plants with upper plenum injection and the analysis results meet the 10 CFR 50.46 acceptance criteria, this change does not involve a significant reduction in a margin of safety.

Addition of the reference to WCAP-16009-P-A in the Technical Specifications is an administrative change that does not involve a significant reduction in a margin of safety.

The licensing basis and Technical Specification changes proposed in this license amendment do not involve a significant reduction in a margin of safety.

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

5.2 Applicable Regulatory Requirements/Criteria

Title 10 Code of Federal Regulations 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors"

The applicable regulatory requirement for this license amendment request is 10 CFR 50.46, which includes requirements and acceptance criteria pertaining to the evaluation of post accident emergency core cooling system performance.

This regulation includes the requirement that "... uncertainties in the analysis method and inputs must be identified and assessed so that the uncertainty in the calculated results can be estimated. This uncertainty must be accounted for, so that, when the calculated ECCS [emergency core cooling system] cooling performance is compared to the criteria ... there is a high level of probability that the criteria would not be exceeded."

This license amendment request proposes to use the ASTRUM methodology (WCAP-16009-P-A, "Realistic Large Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)") for the performance of large break loss of coolant accident analyses, including treatment of uncertainties in the inputs used for the analysis. No change is proposed to the analysis acceptance criteria specified in the regulations. The NRC has reviewed WCAP-16009-P-A and found it acceptable for referencing in licensing applications for Westinghouse designed two-loop pressurized water reactors with upper plenum injection. WCAP-16009-P-A is applicable to the Prairie Island Nuclear Generating Plant, Units 1 and 2 and the plant-specific application of the ASTRUM methodology to the Prairie Island Nuclear Generating Plant, Units 1 and 2, large break loss of coolant accident analyses have been performed in accordance with the conditions and limitations of the topical report and the associated NRC Safety Evaluation.

The licensing basis changes proposed in this license amendment request meet the requirements of 10 CFR 50.46 and provide the basis for safe plant operation.

<u>NUREG-1431 Standard Technical Specifications, Westinghouse Plants, Revision 3.0</u>

NUREG-1431, "Standard Technical Specifications, Westinghouse Plants," Revision 3.0 (NUREG-1431) provides guidance for Technical Specifications for plants with Westinghouse Nuclear Steam Supply Systems and has been approved for use by the Nuclear Regulatory Commission. This LAR proposes to incorporate new large break loss of coolant accident analyses using the ASTRUM methodology in the NRC reviewed and approved WCAP-16009-P-A. In accordance with TS 5.6.5 paragraph b, analytical methods used to determine core operating limits must be listed in TS 5.6.5. This LAR proposes to add new item 27 which references WCAP-16009-P-A.

The user note in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants", Revision 3, TS 5.6.5 states,

> Identify the Topical Report(s) by number and title or identify the staff Safety Evaluation Report for a plant specific methodology by NRC letter and date. The COLR will contain the complete identification for each of the TS referenced topical reports used to prepare the COLR (I.e., report number, title, revision, date, and any supplements).

Thus, with the changes proposed in this license amendment request, the format and content guidance NUREG-1431 is met as discussed above and the plant Technical Specifications will continue to provide the basis for safe plant operation.

<u>Regulatory Requirements/Criteria Conclusions</u>

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the

Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

6.0 ENVIRONMENTAL CONSIDERATION

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

7.0 REFERENCES

1. Bajorek, S. M., et. al., 1998, "Code Qualification Document for Best Estimate LOCA Analysis," WCAP-12945-P-A, Volume 1, Revision 2 and Volumes 2 through 5, Revision 1, and WCAP-14747 (Non-Proprietary).

2. Dederer, S. I., et. al., 1999, "Application of Best Estimate Large Break LOCA Methodology to Westinghouse PWRs with Upper Plenum Injection," WCAP-14449-P-A, Revision 1 and WCAP-14450, Revision 1 (Non-Proprietary).

3. Nissley, M. E., et.al., 2005, "Realistic Large Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)," WCAP-16009-P-A and WCAP-16009-NP-A (Non-Proprietary).

4. "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Cooled Nuclear Power Reactors", 10 CFR 50.46 and Appendix K of 10 CFR 50, Federal Register, Volume 39, Number 3, January 4, 1974.

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

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

7. Boyack, B., et. al., 1989, "Quantifying Reactor Safety Margins: Application of Code Scaling Applicability and Uncertainty (CSAU) Evaluation Methodology to a Large Break Loss-of-Coolant-Accident," NUREG/CR-5249.

8.0 TABLE AND FIGURES

8.1 PINGP Unit 1 Tables and Figures

Table 1-1 - Major Plant Parameter Assumptions Used in the BE LOCA Analysis for PINGP Unit 1

Parameter	Value
Plant Physical Description	
• SG[1] Tube Plugging	≤10% Unit 1 (RSG[2])
Plant Initial Operating Conditions	
• Reactor Power	≤100% of 1683 MWt
• Peaking Factors	$F_Q \leq 2.5$ $F_{\Delta H} \leq 1.77$
• Axial Power Distribution	See Figure 1-20
Fluid Conditions	
• T_{AVG}	T_{AVG} = 560.0 +/- 4 °F
• Pressurizer Pressure	2190 psia ≤ P_{RCS} ≤ 2310 psia
• Reactor Coolant Flow	≥ 178,000 gpm
• Accumulator Temperature	70 °F ≤ T_{ACC} ≤ 120 °F
• Accumulator Pressure	699.7 psia ≤ P_{ACC} ≤ 809.7 psia
• Accumulator Water Volume	1245 ft^3 ≤ V_{ACC} ≤ 1295 ft^3
• Accumulator Boron Concentration	≥ 1900 ppm
Accident Boundary Conditions	
• Single Failure Assumptions	Loss of one ECCS train
• Safety Injection (SI) Flow	Minimum
• Safety Injection Temperature	60 °F ≤ T_{SI} ≤ 120 °F
• Low Head Safety Injection Initiation Delay Time	≤ 15 sec (with offsite power) ≤ 28 sec (without offsite power)
• High Head Safety Injection Initiation Delay Time	≤ 10 sec (with offsite power) ≤ 28 sec (without offsite power)
• Containment Pressure	Bounded (minimum)

[1] Steam Generator
[2] Replacement Steam Generator

Table 1-2 - PINGP Unit 1 Best Estimate Large Break LOCA Results

10 CFR 50.46 Requirement	Value	Criteria
95/95 PCT[3] (°F)	1594	< 2200
95/95 LMO[4] (%)	0.2	< 17
95/95 CWO[5] (%)	0	< 1

Table 1-3 - PINGP Unit 1 Best Estimate Large Break Sequence of Events for the Limiting PCT Case

Event	Time (sec)
Start of Transient	0.0
Safety Injection Signal	3.8
PCT Occurs	6.0
Accumulator Injection Begins	9.0
High Head Safety Injection Begins	13.8
Low Head Safety Injection Begins	18.8
End of Blowdown	25.0
Bottom of Core Recovery	34.0
Accumulator Empty	~35.0
End of Transient	450.0

[3] Peak Clad Temperature
[4] Local Maximum Oxidation
[5] Core Wide Oxidation

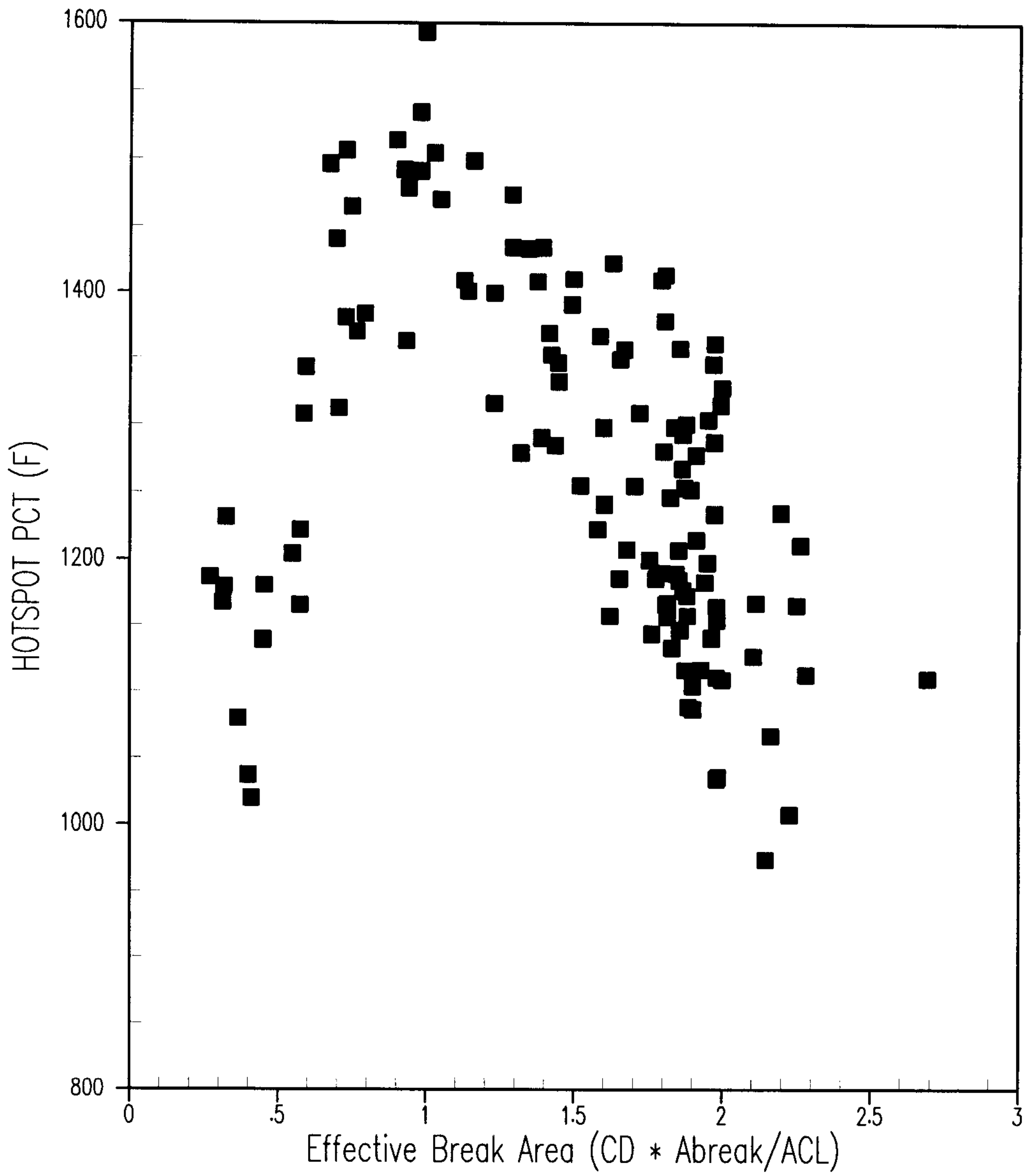

Figure 1-1 – HOTSPOT PCT versus Effective Break Area Scatter Plot (CD = Discharge Coefficient, Abreak = Break Area, ACL = Cold Leg Area)

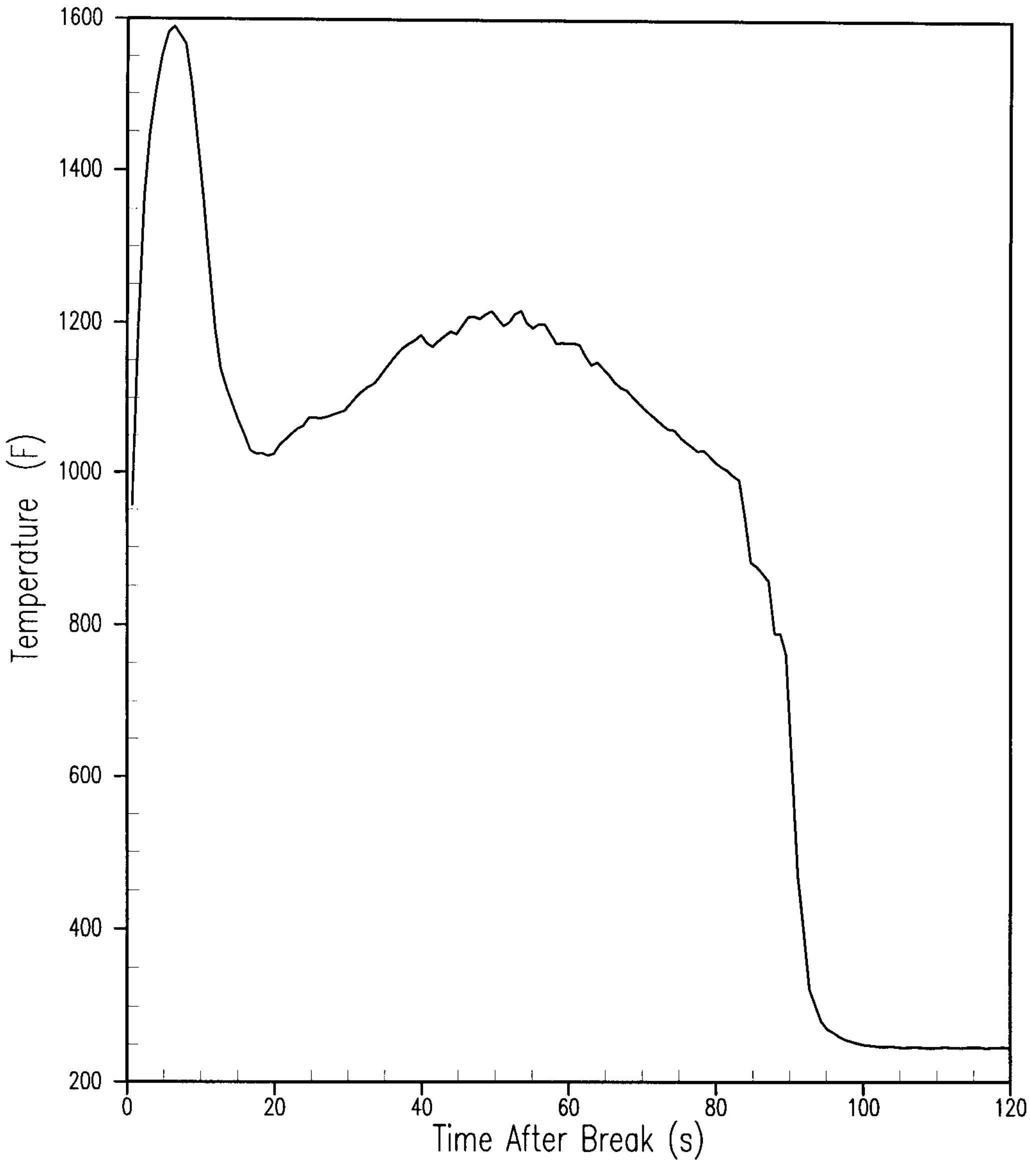

Figure 1-2 –HOTSPOT Clad Temperature Transient at the Limiting Elevation for the Limiting PCT Case

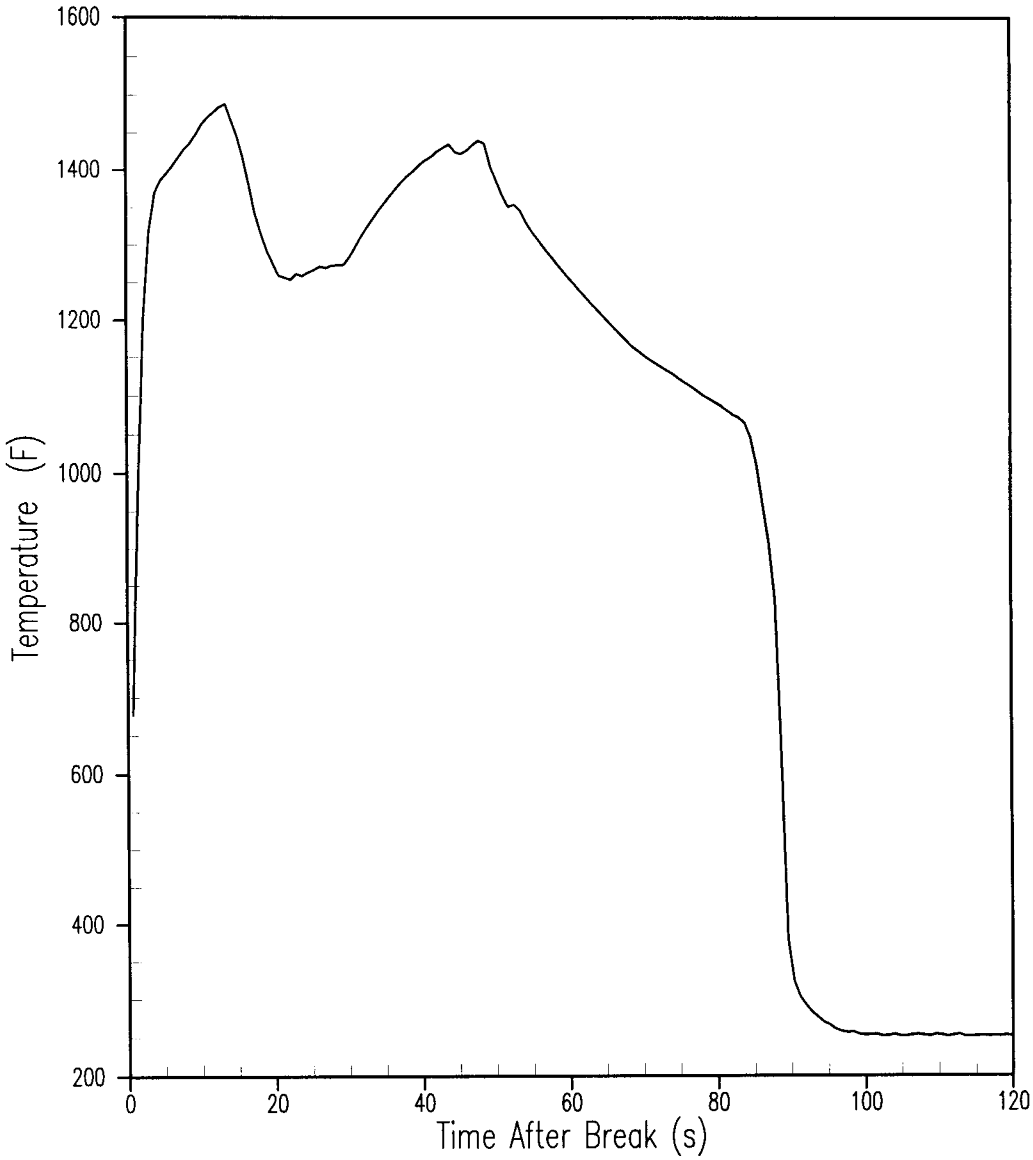

Figure 1-3 –HOTSPOT Clad Temperature Transient at the Limiting Elevation for the Limiting LMO Case

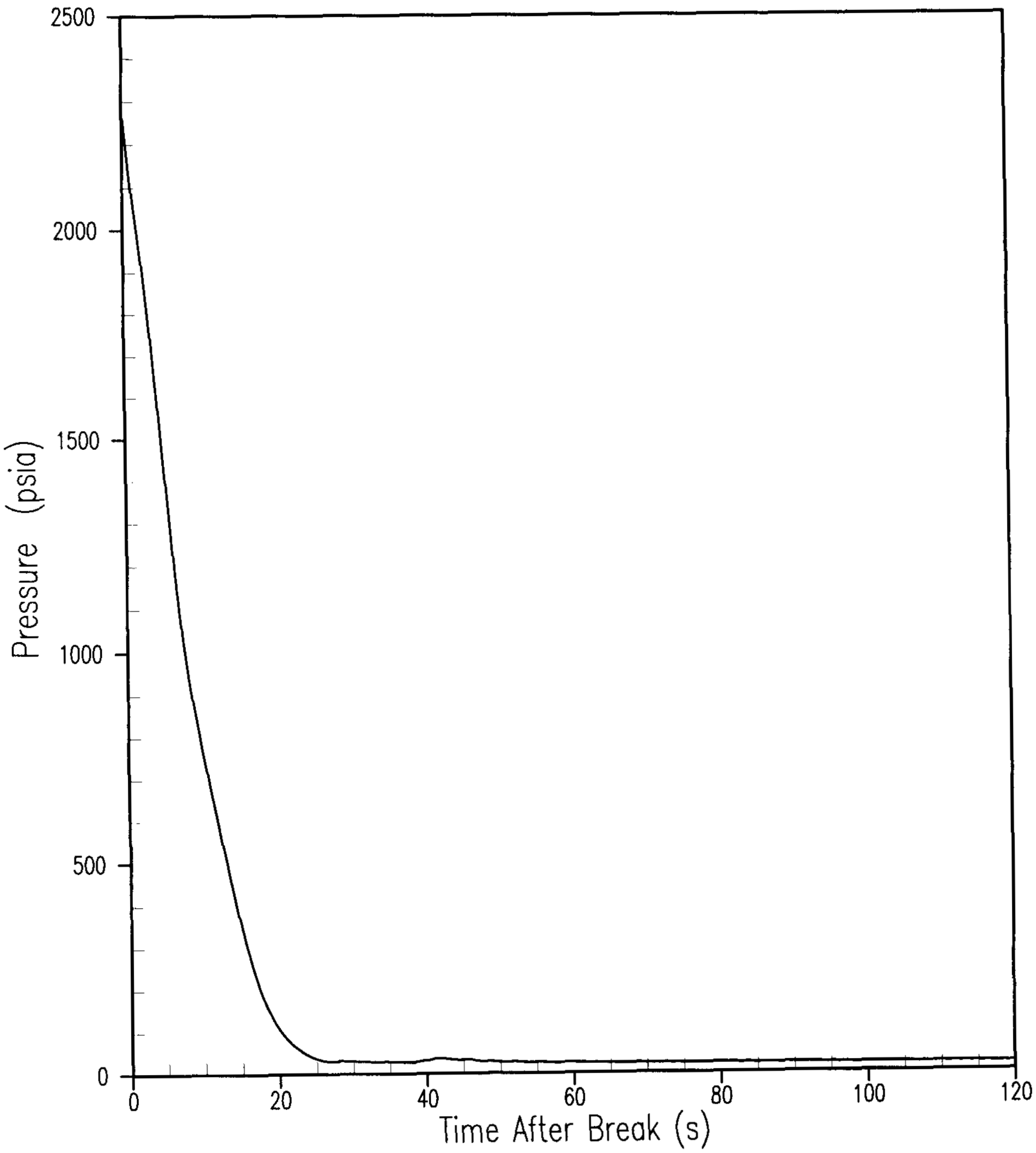

Figure 1-4 –Pressurizer Pressure for the Limiting PCT Case

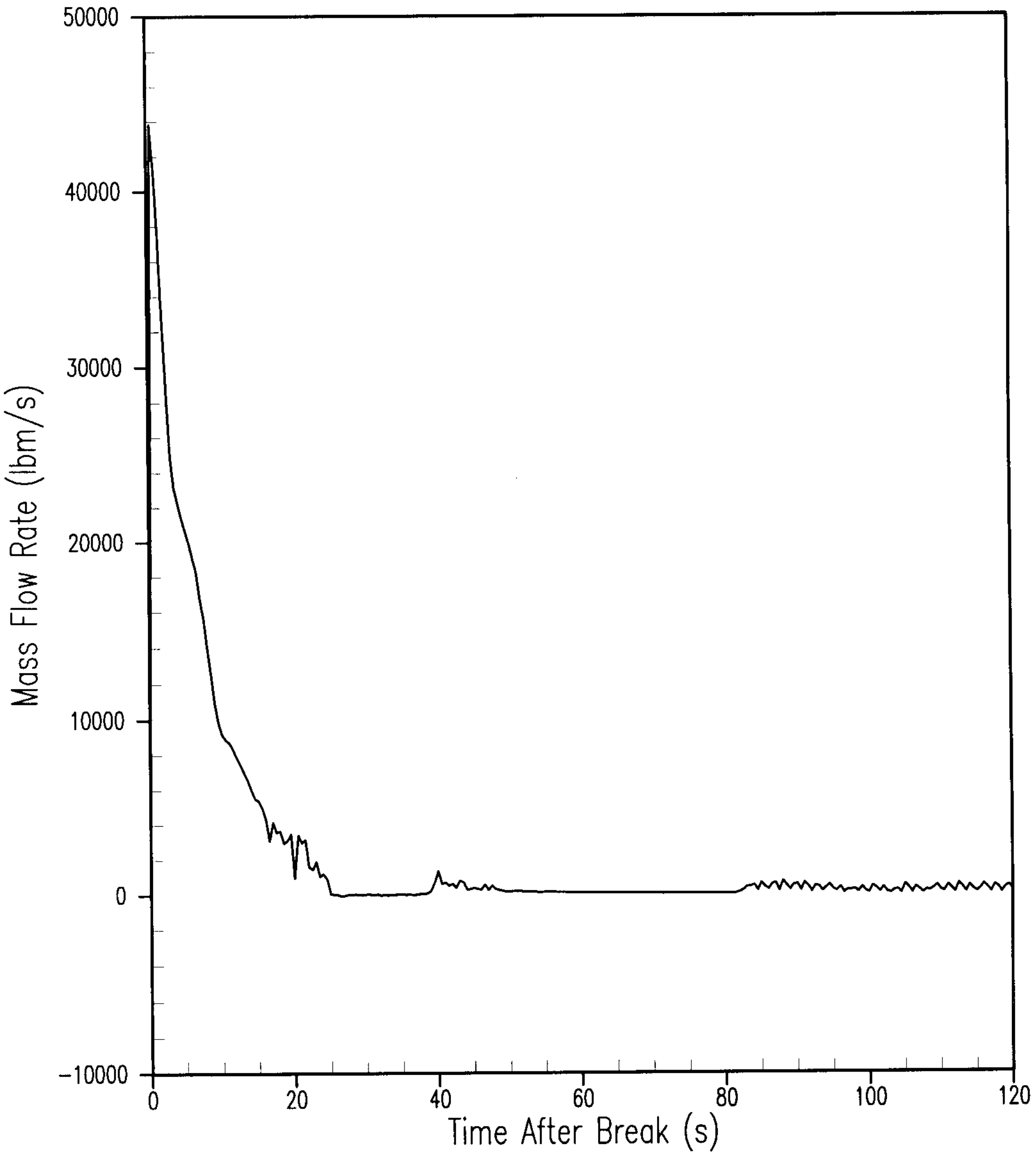

Figure 1-5 – Vessel Side Break Flow for the Limiting PCT Case

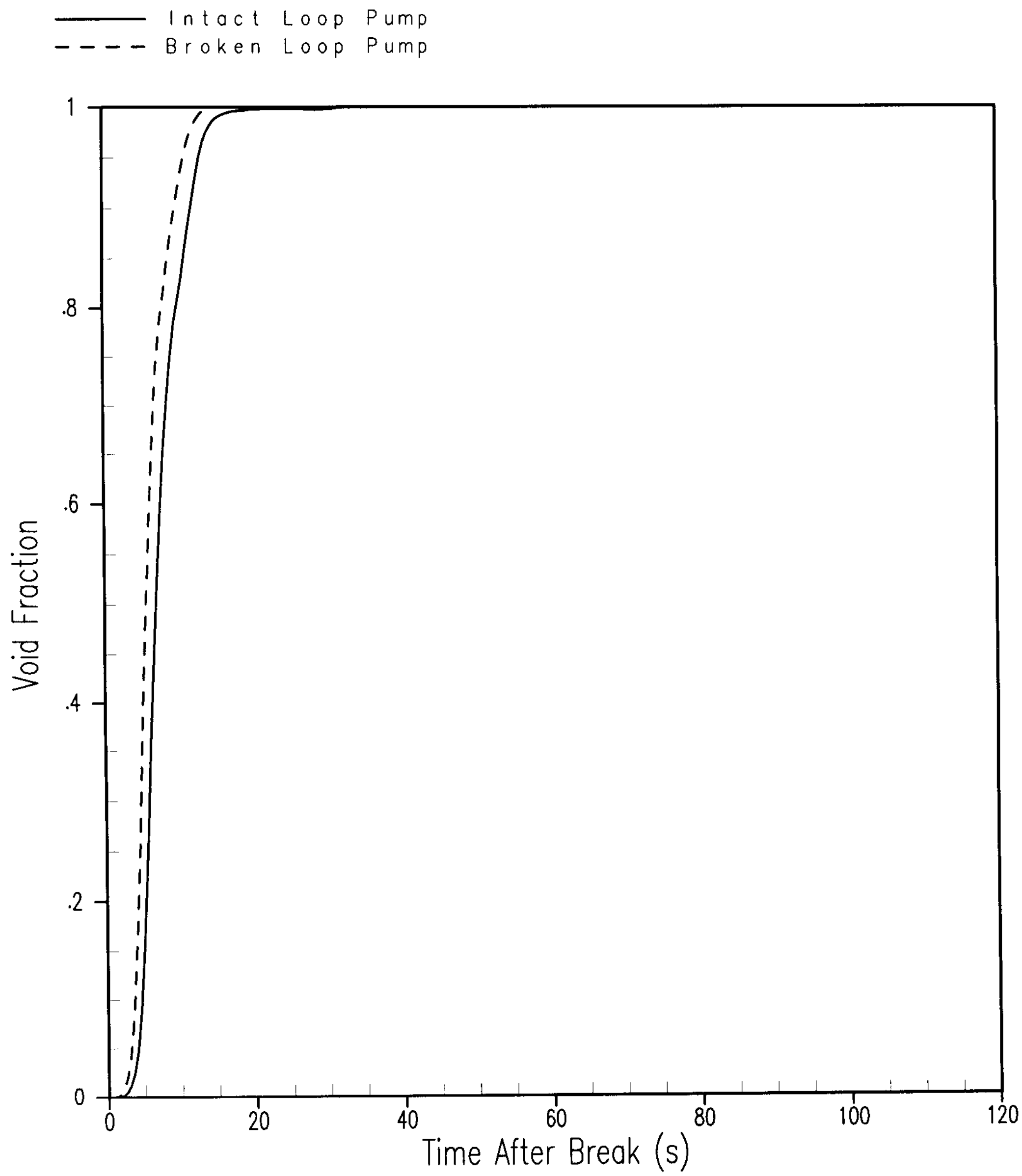

Figure 1-6 - Void Fraction in Pumps for the Limiting PCT Case

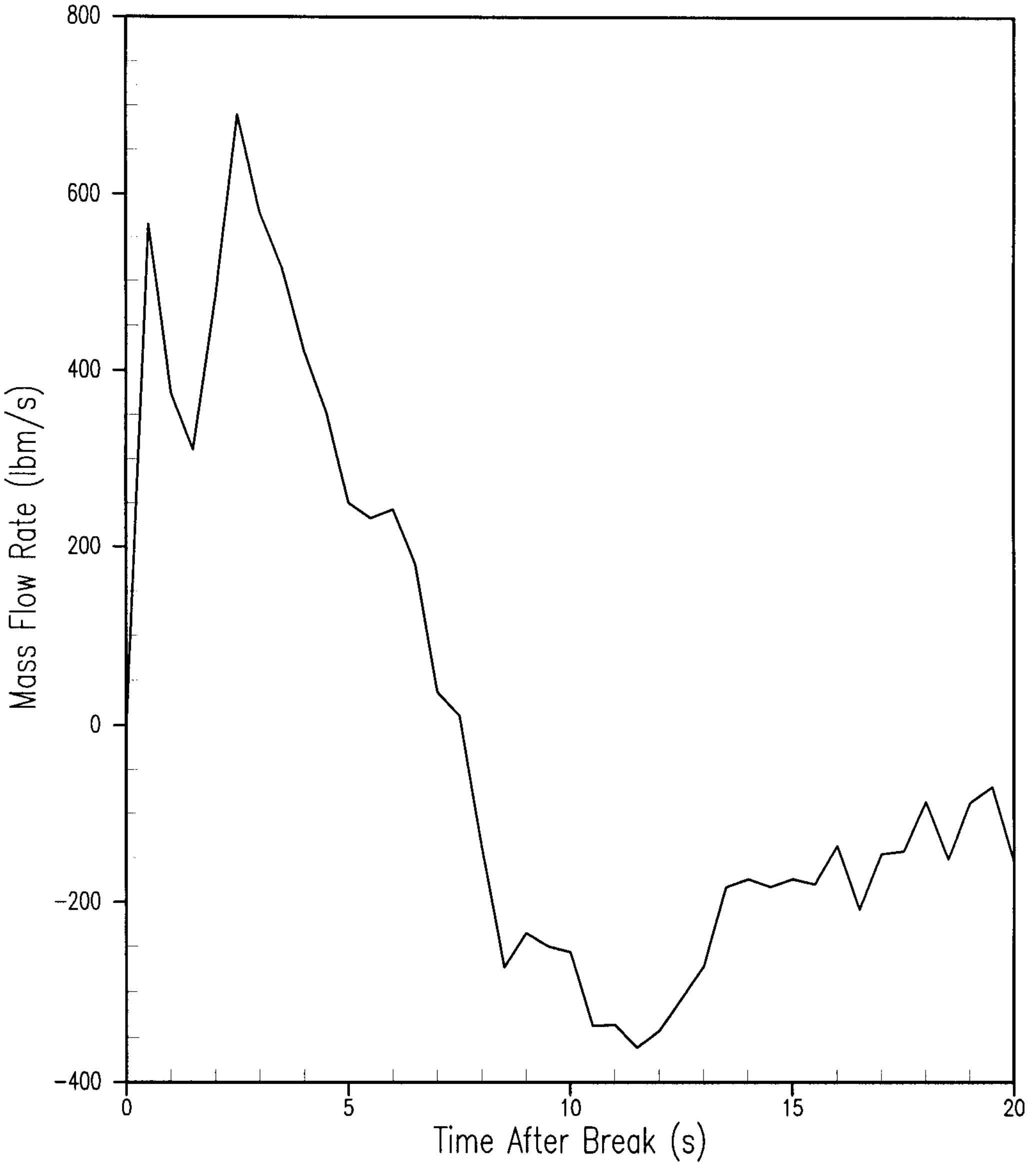

Figure 1-7 – Vapor Flow at Top of Core for the Limiting PCT Case

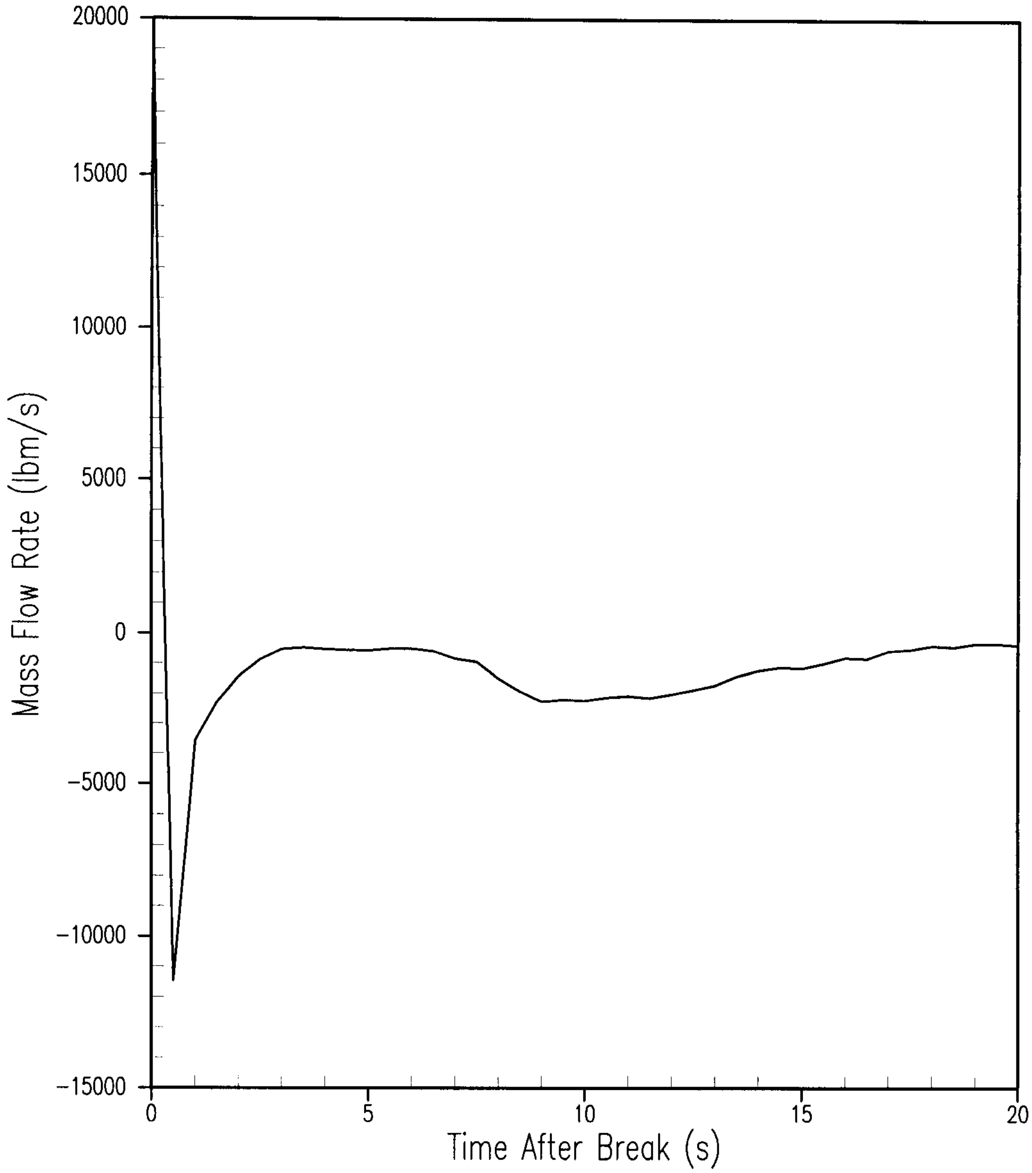

Figure 1-8 – Total Flow at Bottom of Core for the Limiting PCT Case

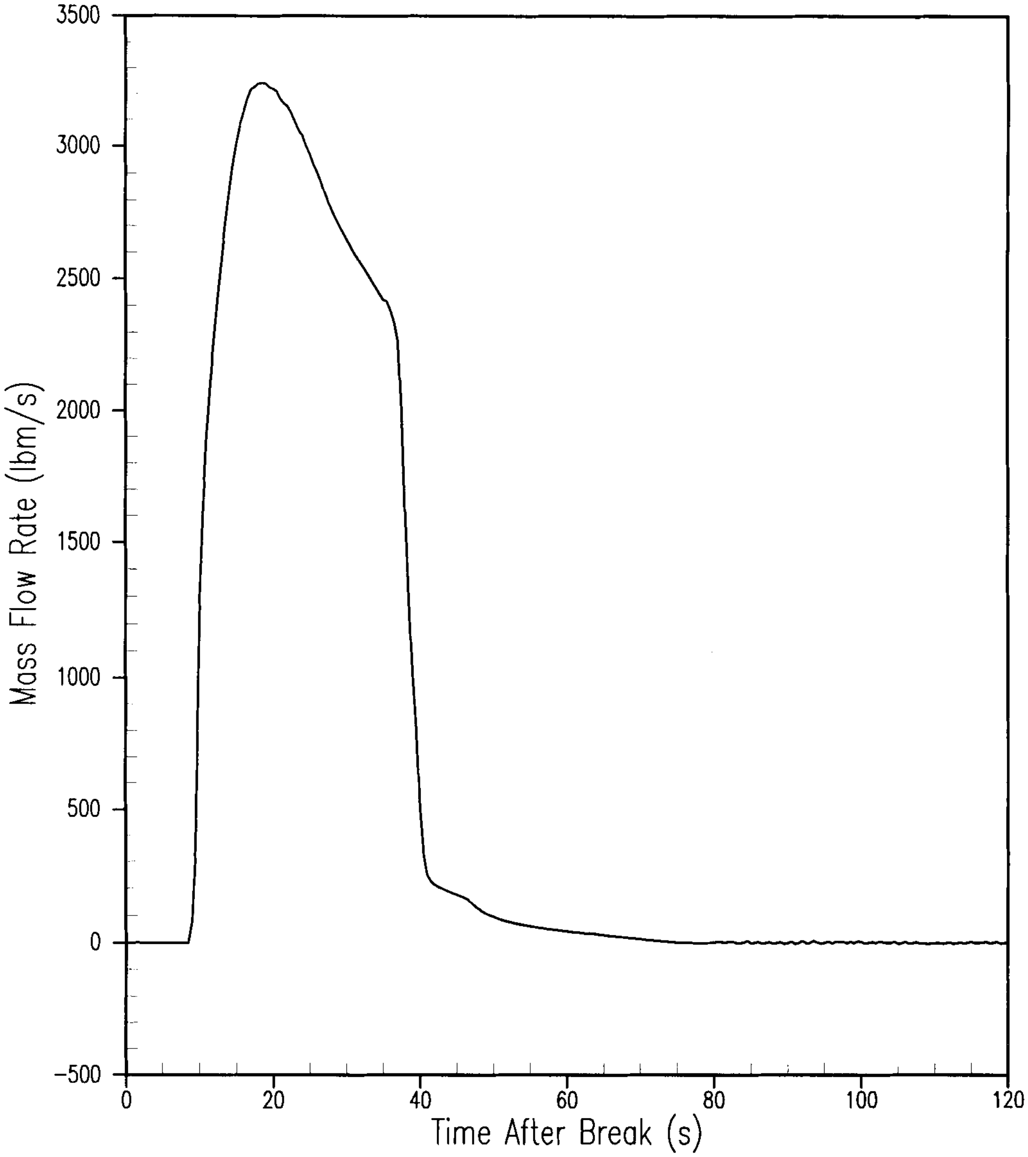

Figure 1-9 – Accumulator Injection Flow for the Limiting PCT Case

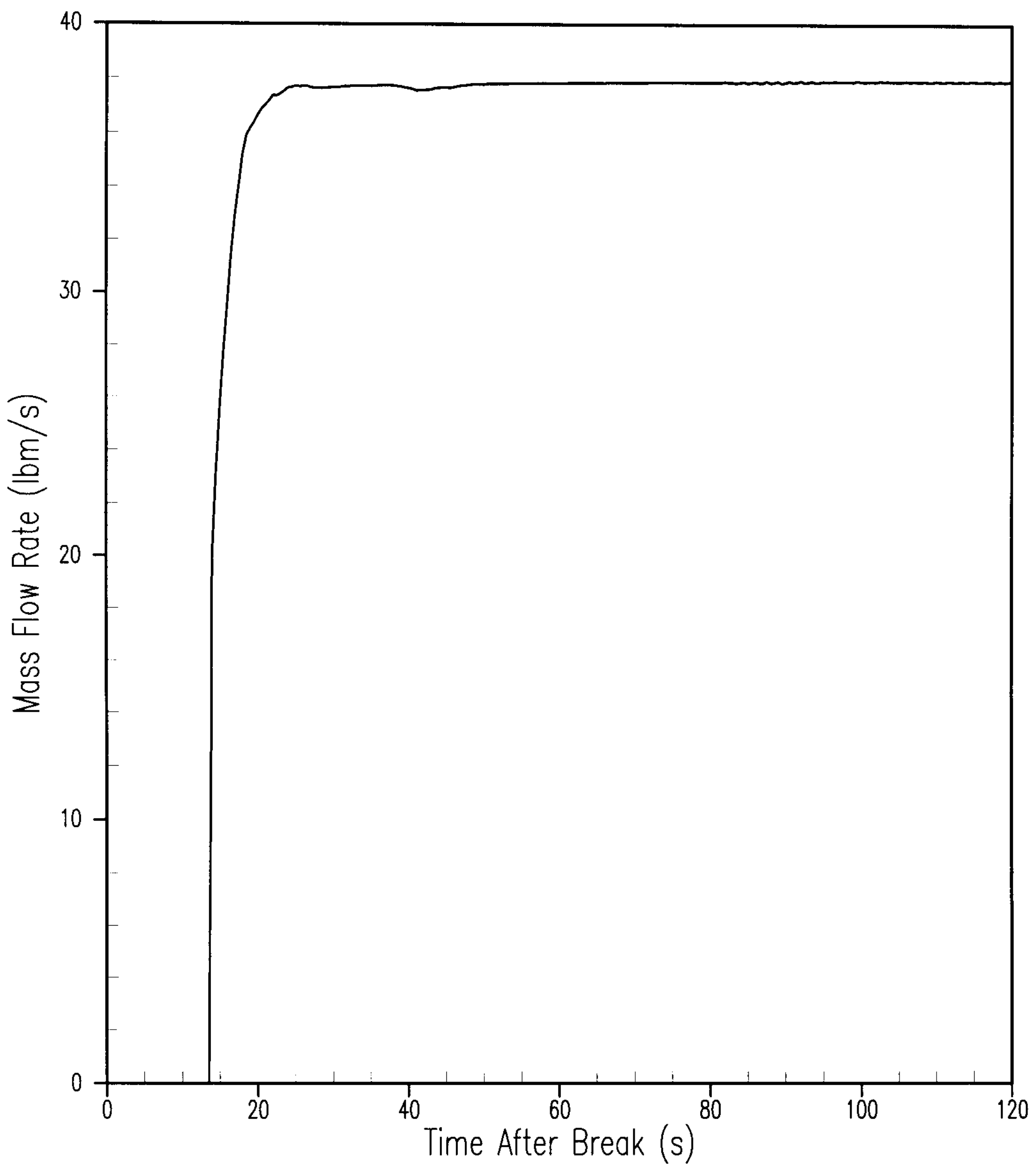

Figure 1-10 – High Head Safety Injection Flow for the Limiting PCT Case

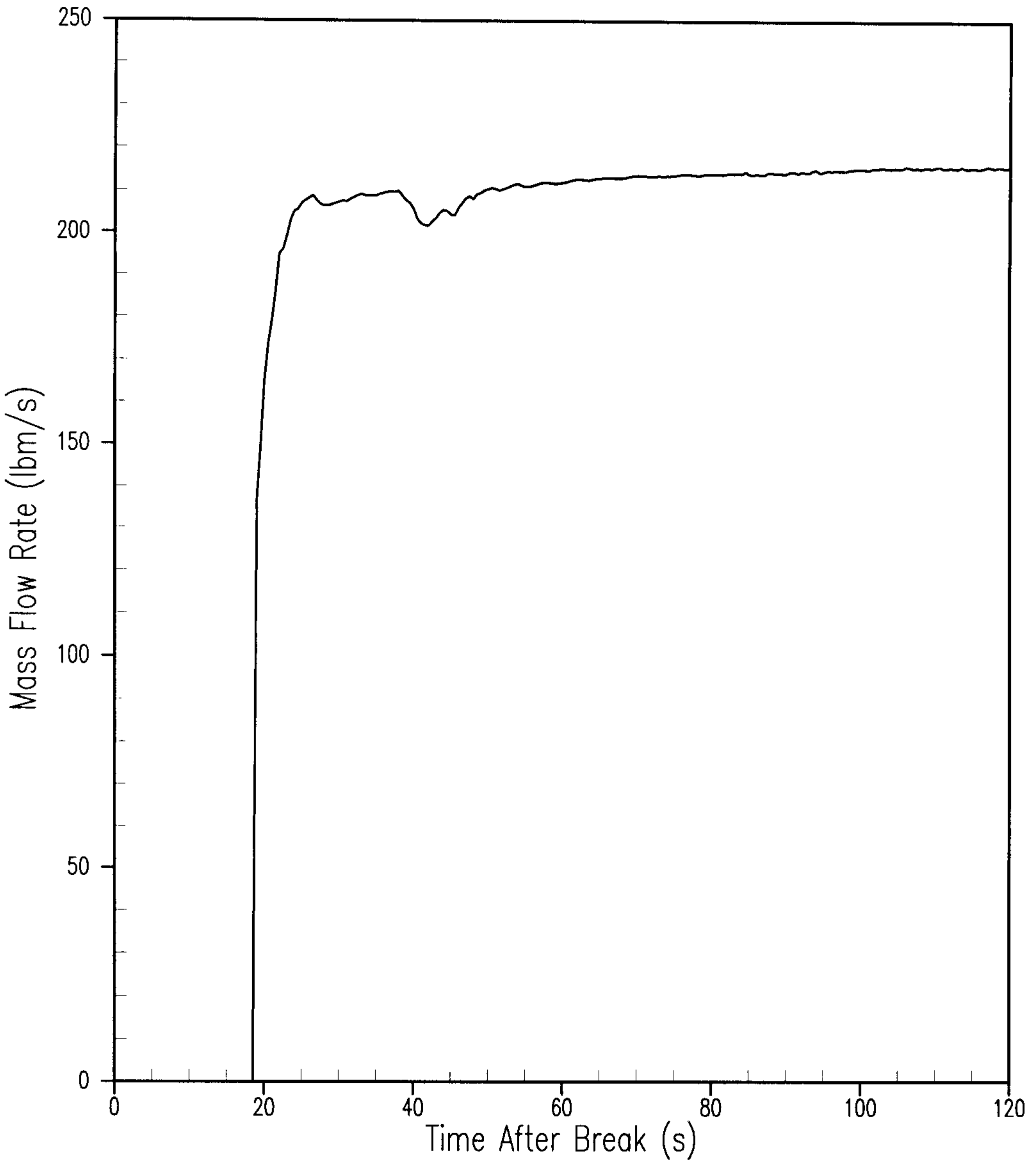

Figure 1-11 – Low Head Safety Injection Flow for the Limiting PCT Case

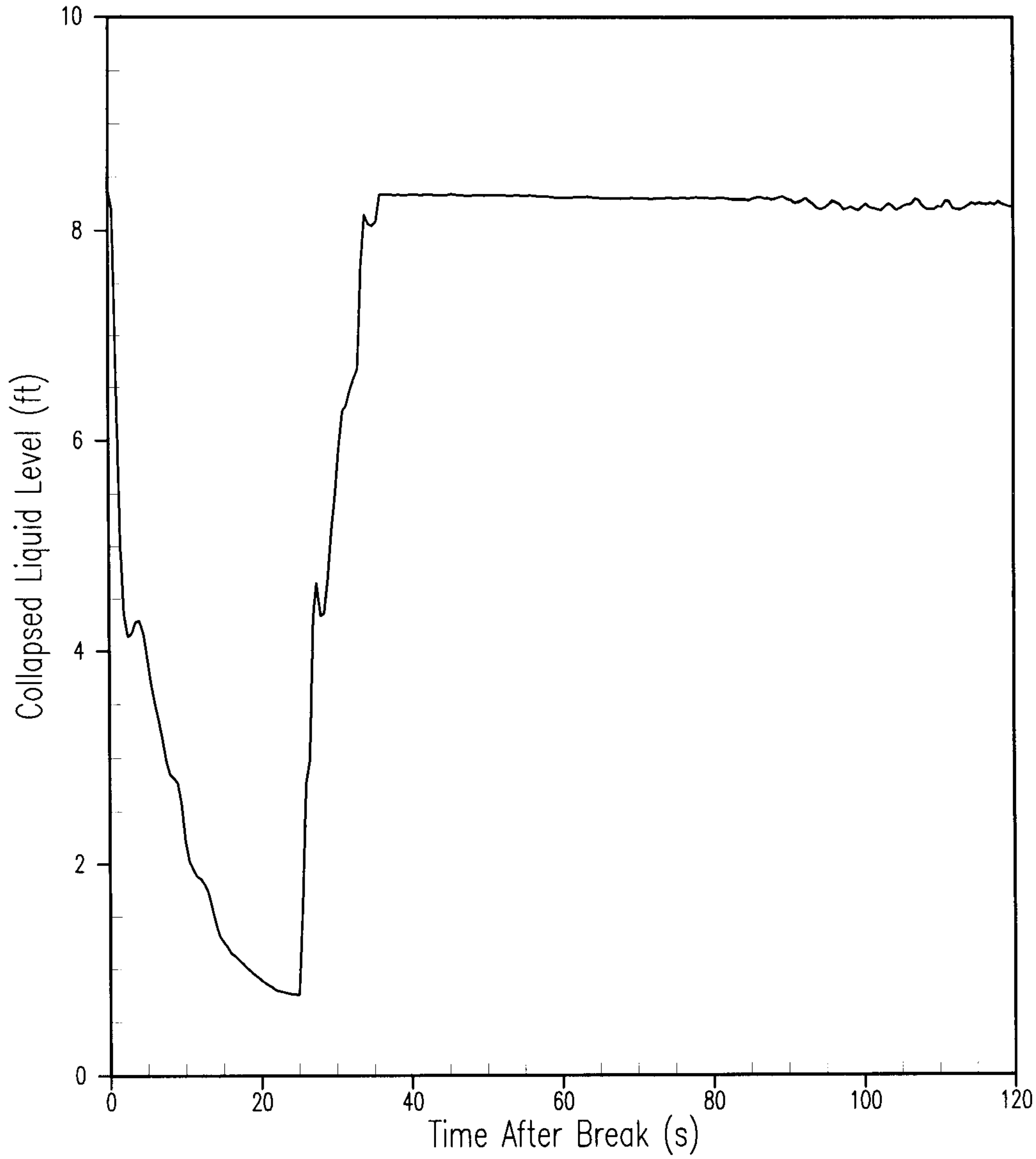

Figure 1-12 – Lower Plenum Collapsed Liquid Level for the Limiting PCT Case

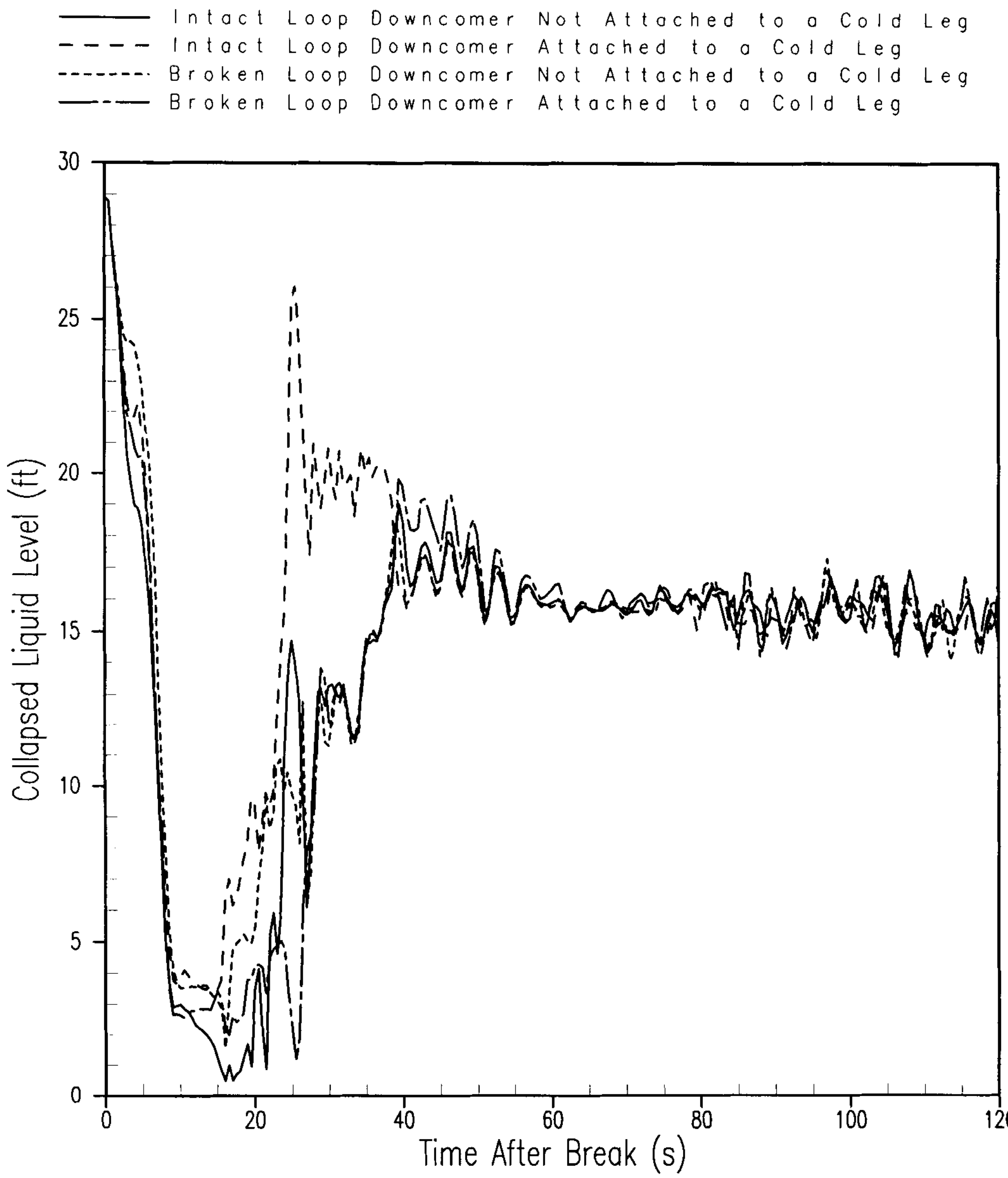

Figure 1-13 – Downcomer Collapsed Liquid Levels for the Limiting PCT Case

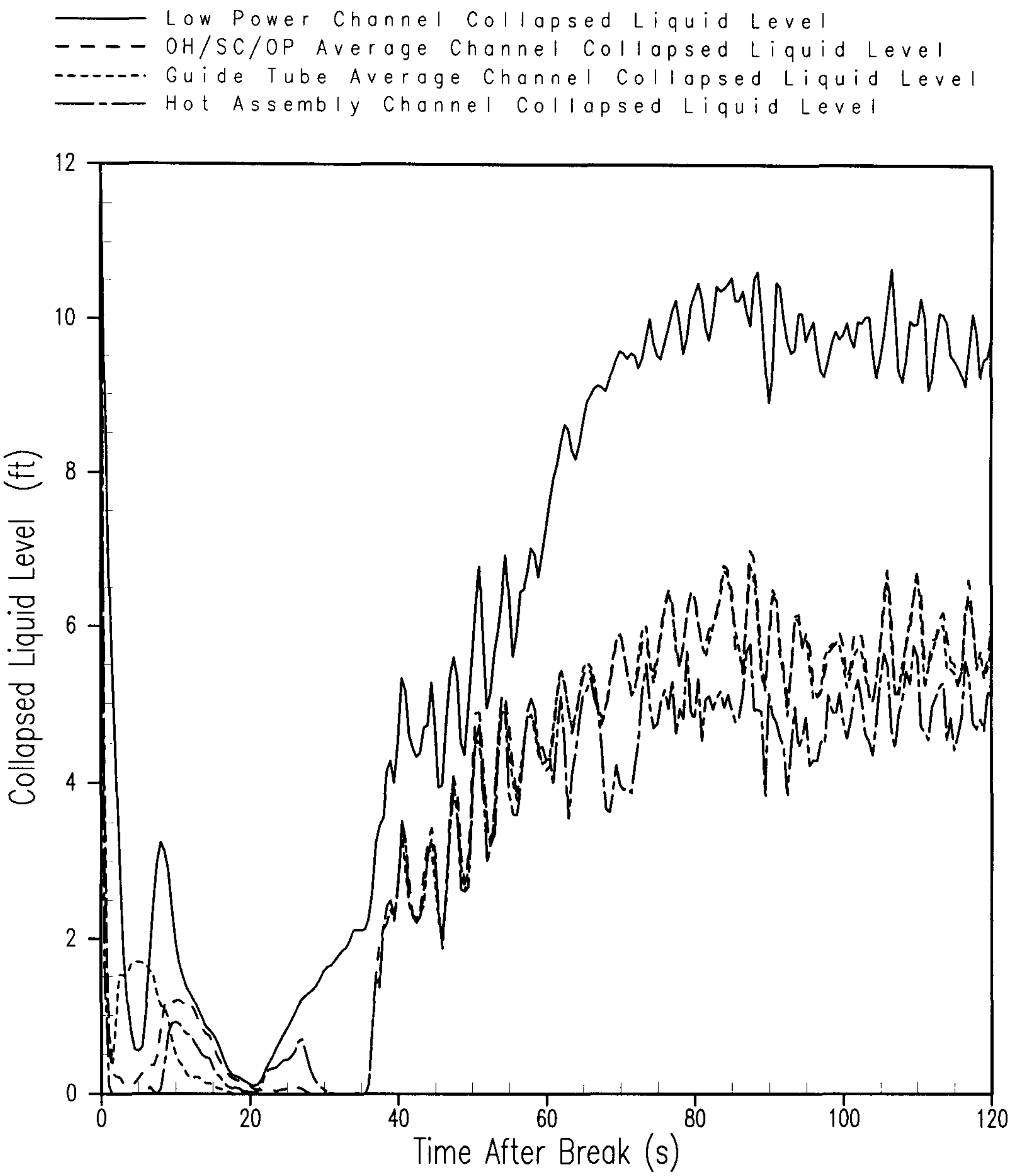

Figure 1-14 – Core Collapsed Liquid Levels for the Limiting PCT Case (OH = Open Holes, SC = Support Column, OP = Orifice Plate)

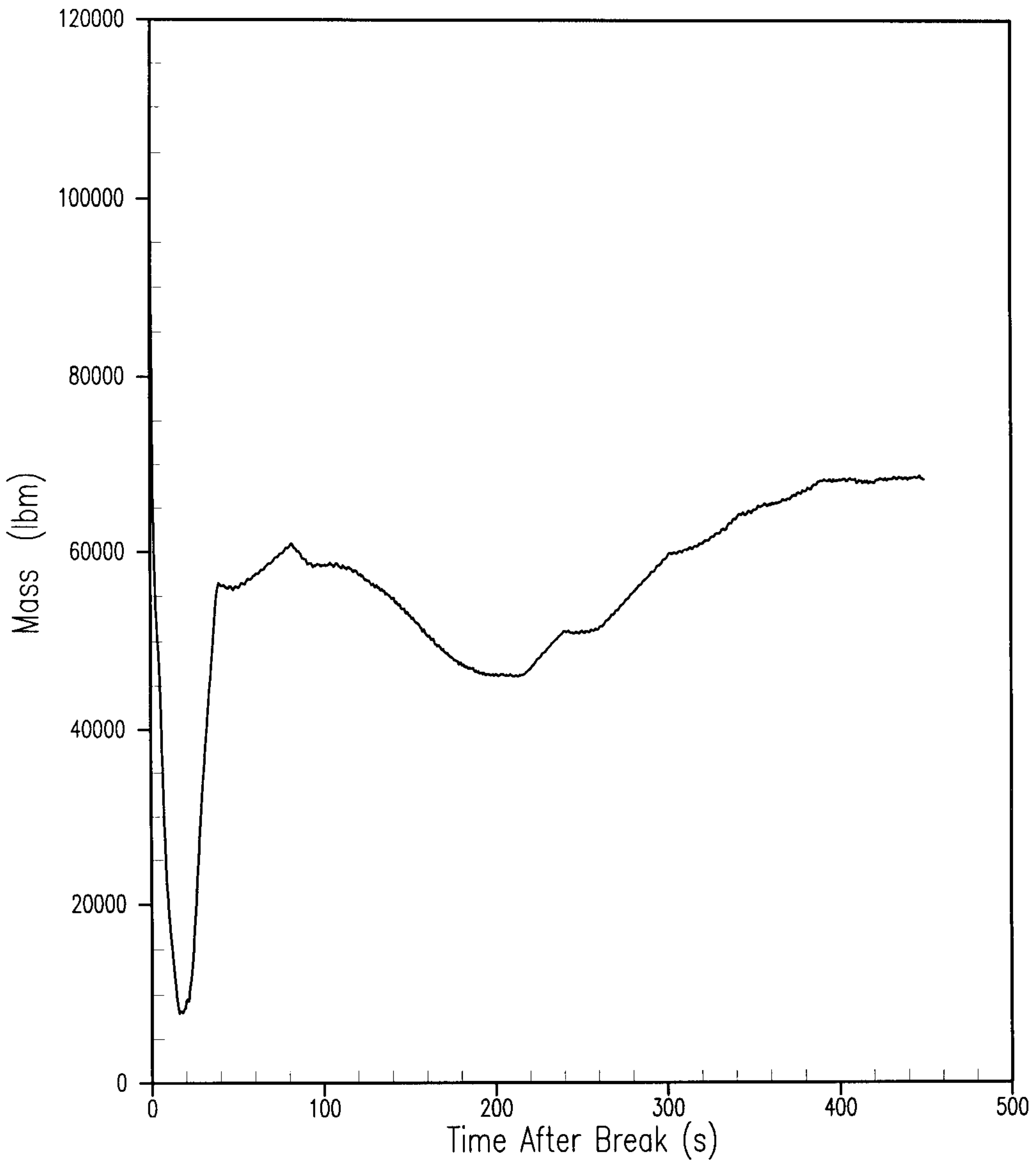

Figure 1-15 – Vessel Fluid Mass for the Limiting PCT Case

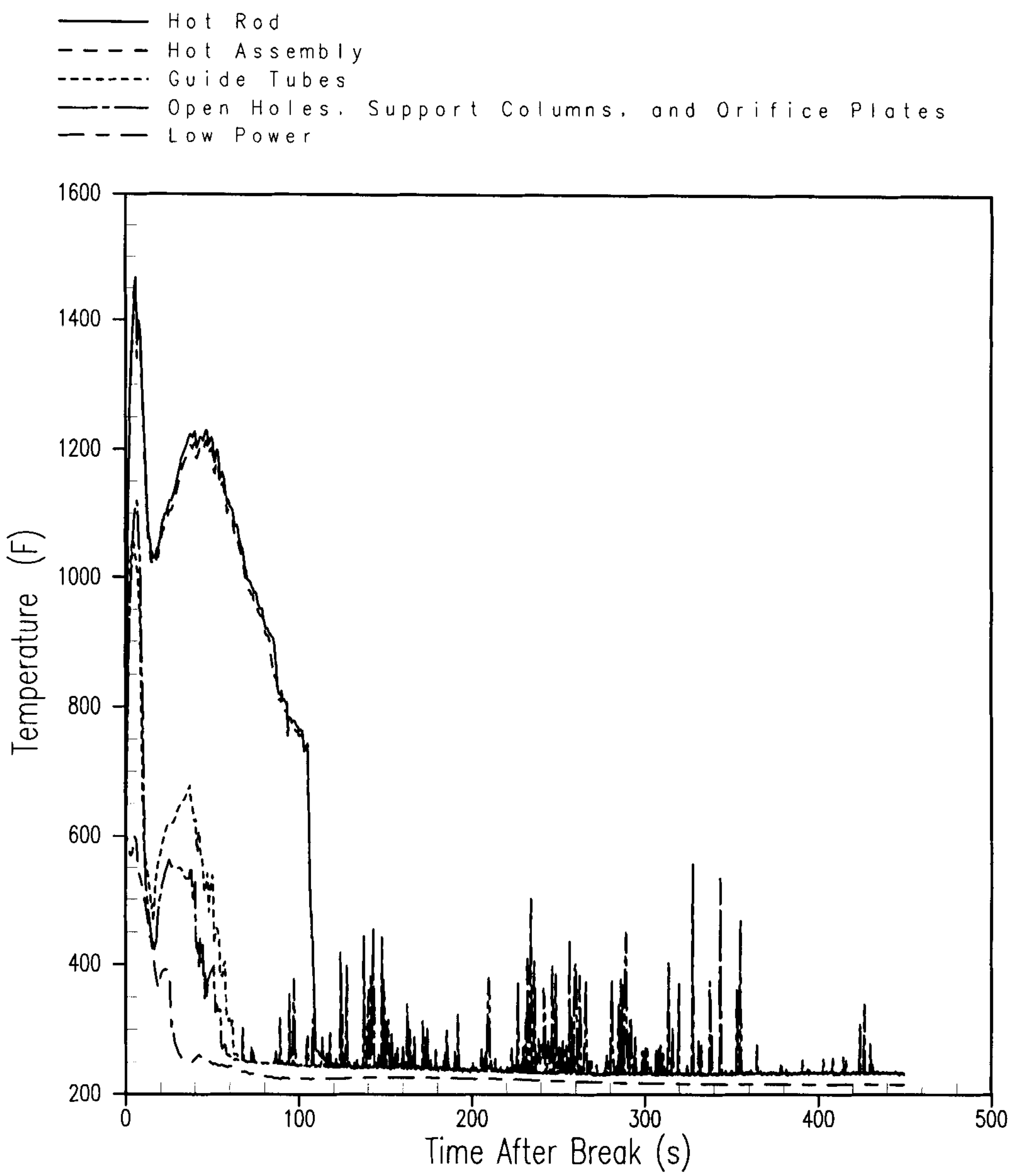

Figure 1-16 – <u>W</u>COBRA/TRAC Peak Clad Temperature for all 5 Rod Groups for the Limiting PCT Case

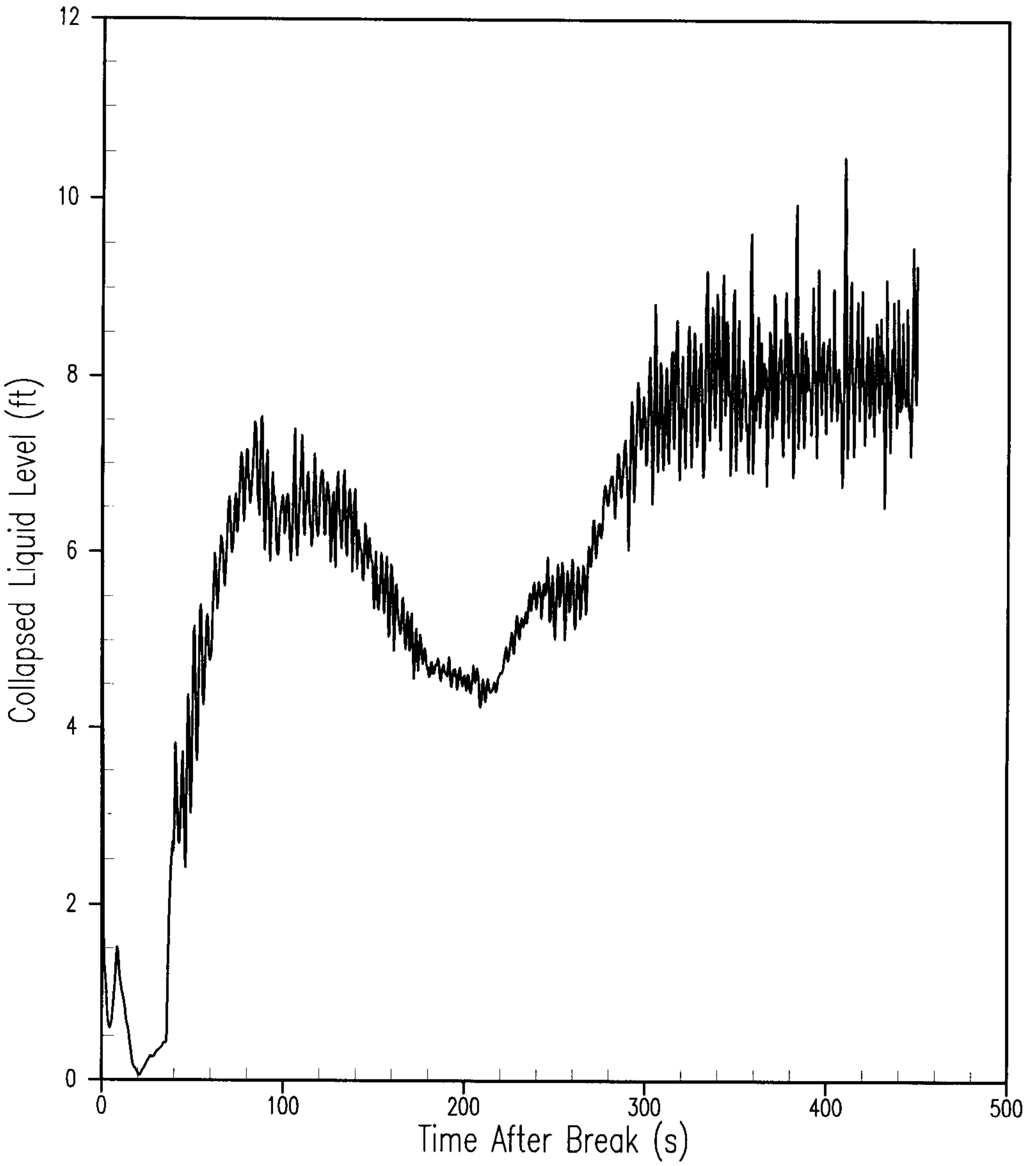

Figure 1-17 – Average Core Collapsed Liquid Level per Assembly for the Limiting PCT Case

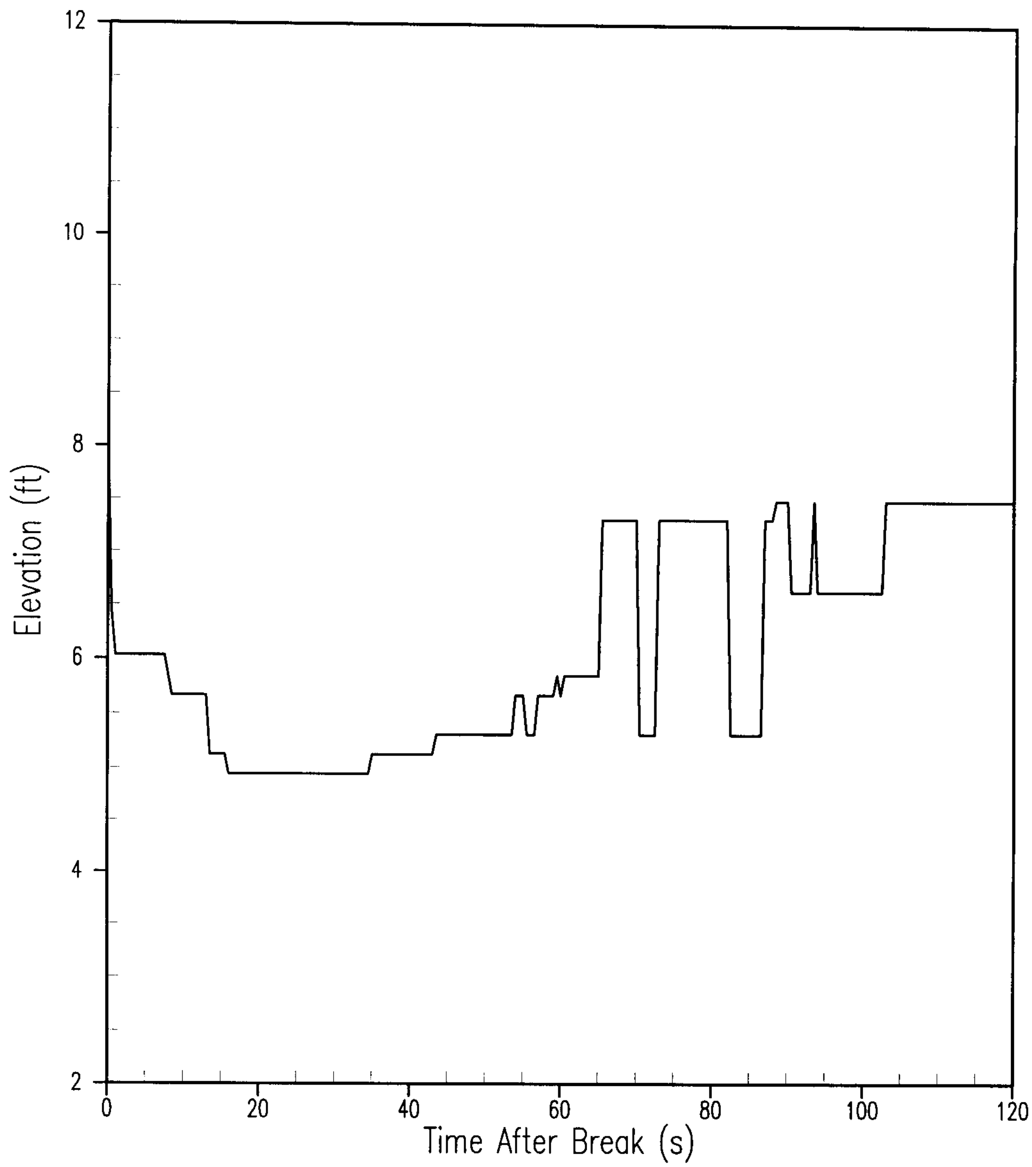

Figure 1-18 - Peak Clad Temperature Elevation for the Hot Rod for the Limiting PCT Case

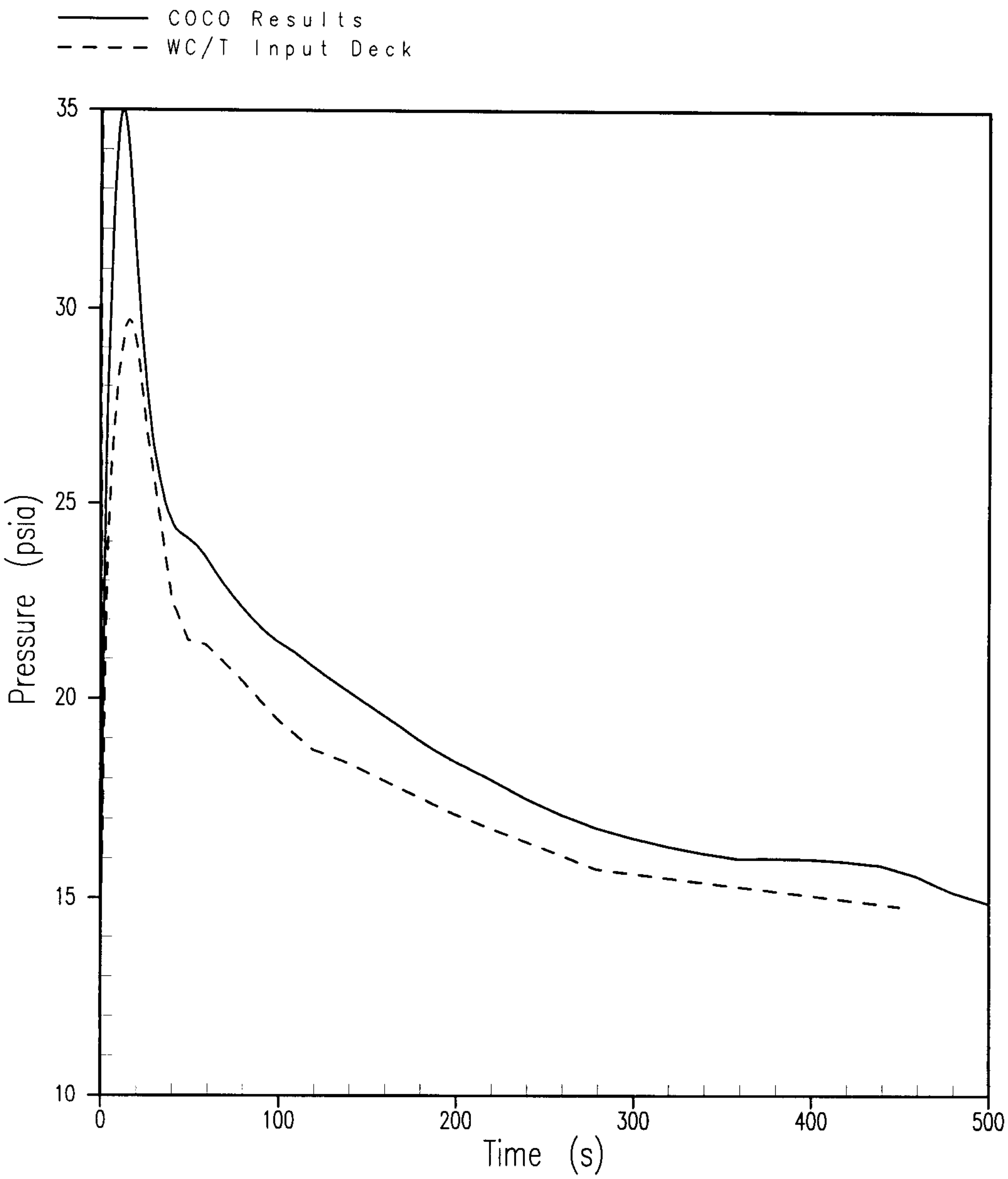

Figure 1-19 – Analysis Versus Calculated Containment Backpressure

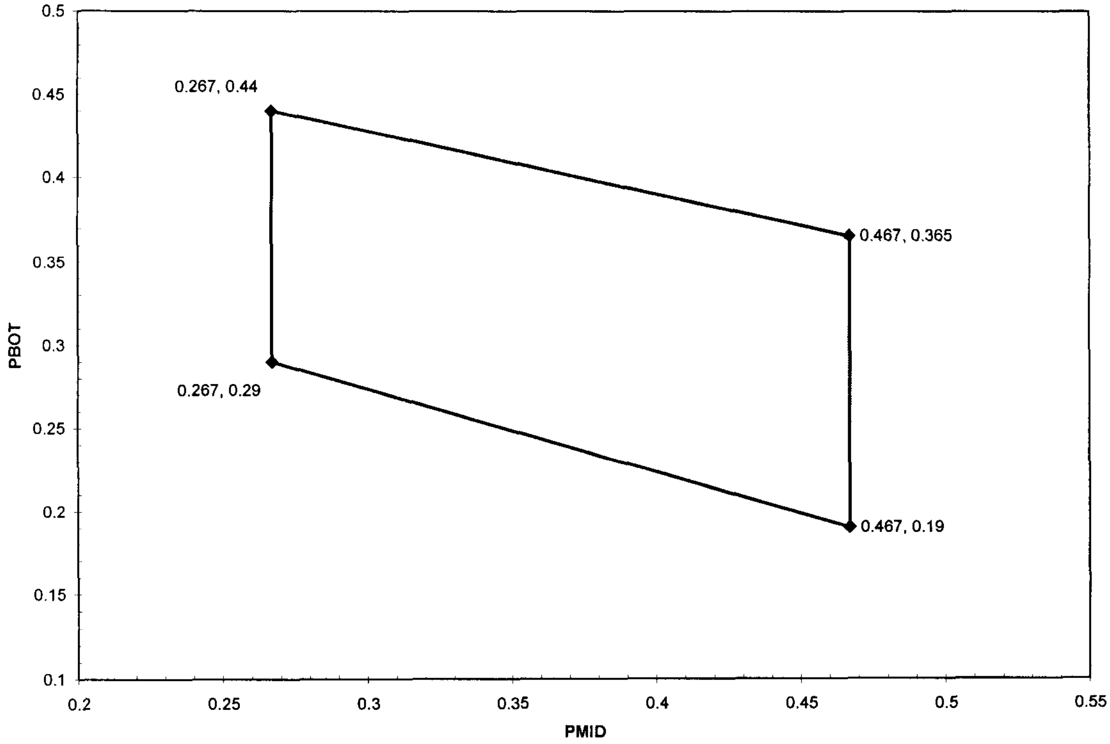

Figure 1-20 - PINGP Unit 1 BELOCA Analysis Axial Power Shape Operating Space Envelope

PBOT = integrated power fraction in the bottom third of the core

PMID = integrated power fraction in the middle third of the core

8.2 PINGP Unit 2 Table and Figures

Table 2-1 - Major Plant Parameter Assumptions Used in the BE LOCA Analysis for PINGP Unit 2

Parameter	Value
Plant Physical Description	
• SG Tube Plugging	≤25% Unit 2 (OSG[6])
Plant Initial Operating Conditions	
• Reactor Power	≤100% of 1683 MWt
• Peaking Factors	$F_Q \le 2.5$ $F_{\Delta H} \le 1.77$
• Axial Power Distribution	See Figure 2-20
Fluid Conditions	
• T_{AVG}	T_{AVG} = 560.0 +/- 4 °F
• Pressurizer Pressure	2190 psia ≤ P_{RCS} ≤ 2310 psia
• Reactor Coolant Flow	≥ 178,000 gpm
• Accumulator Temperature	70 °F ≤ T_{ACC} ≤ 120 °F
• Accumulator Pressure	699.7 psia ≤ P_{ACC} ≤ 809.7 psia
• Accumulator Water Volume	1245 ft^3 ≤ V_{ACC} ≤ 1295 ft^3
• Accumulator Boron Concentration	≥ 1900 ppm
Accident Boundary Conditions	
• Single Failure Assumptions	Loss of one ECCS train
• Safety Injection Flow	Minimum
• Safety Injection Temperature	60 °F ≤ T_{SI} ≤ 120 °F
• Low Head Safety Injection Initiation Delay Time	≤ 15 sec (with offsite power) ≤ 28 sec (without offsite power)
• High Head Safety Injection Initiation Delay Time	≤ 10 sec (with offsite power) ≤ 28 sec (without offsite power)
• Containment Pressure	Bounded (minimum)

[6] Original Steam Generator

Table 2-2 - PINGP Unit 2 Best Estimate Large Break LOCA Results

10 CFR 50.46 Requirement	Value	Criteria
95/95 PCT (°F)	1546	< 2200
95/95 LMO (%)	0.5	< 17
95/95 CWO (%)	0	< 1

Table 2-3 - PINGP Unit 2 Best Estimate Large Break Sequence of Events for the Limiting PCT Case

Event	Time (sec)
Start of Transient	0.0
Safety Injection Signal	4.3
PCT Occurs	6.0
Accumulator Injection Begins	7.0
End of Blowdown	22.0
Bottom of Core Recovery	29.5
Low Head Safety Injection Begins	32.3
High Head Safety Injection Begins	32.3
Accumulator Empty	~35.0
End of Transient	450.0

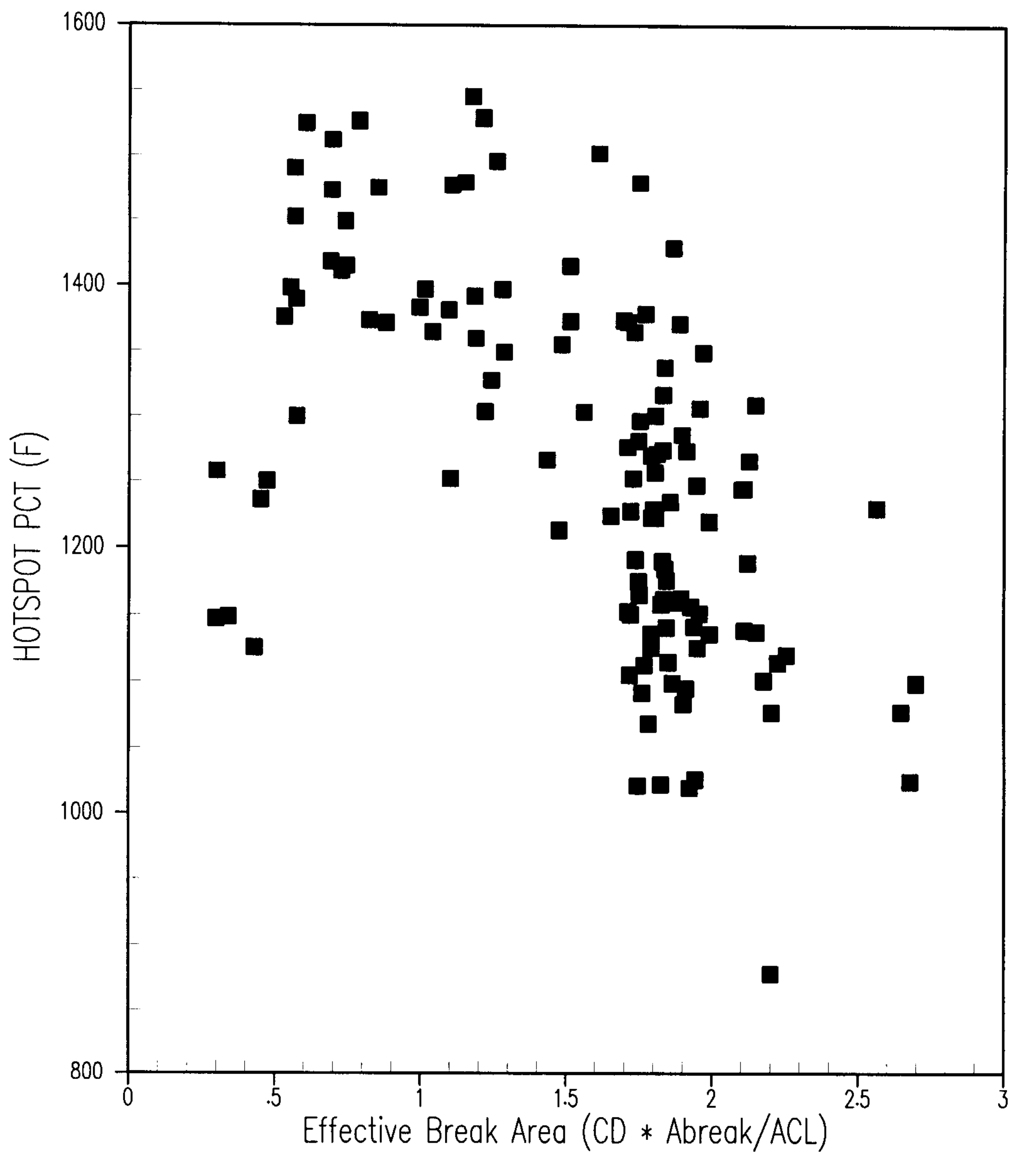

Figure 2-1 – HOTSPOT PCT versus Effective Break Area Scatter Plot (CD = Discharge Coefficient, Abreak = Break Area, ACL = Cold Leg Area)

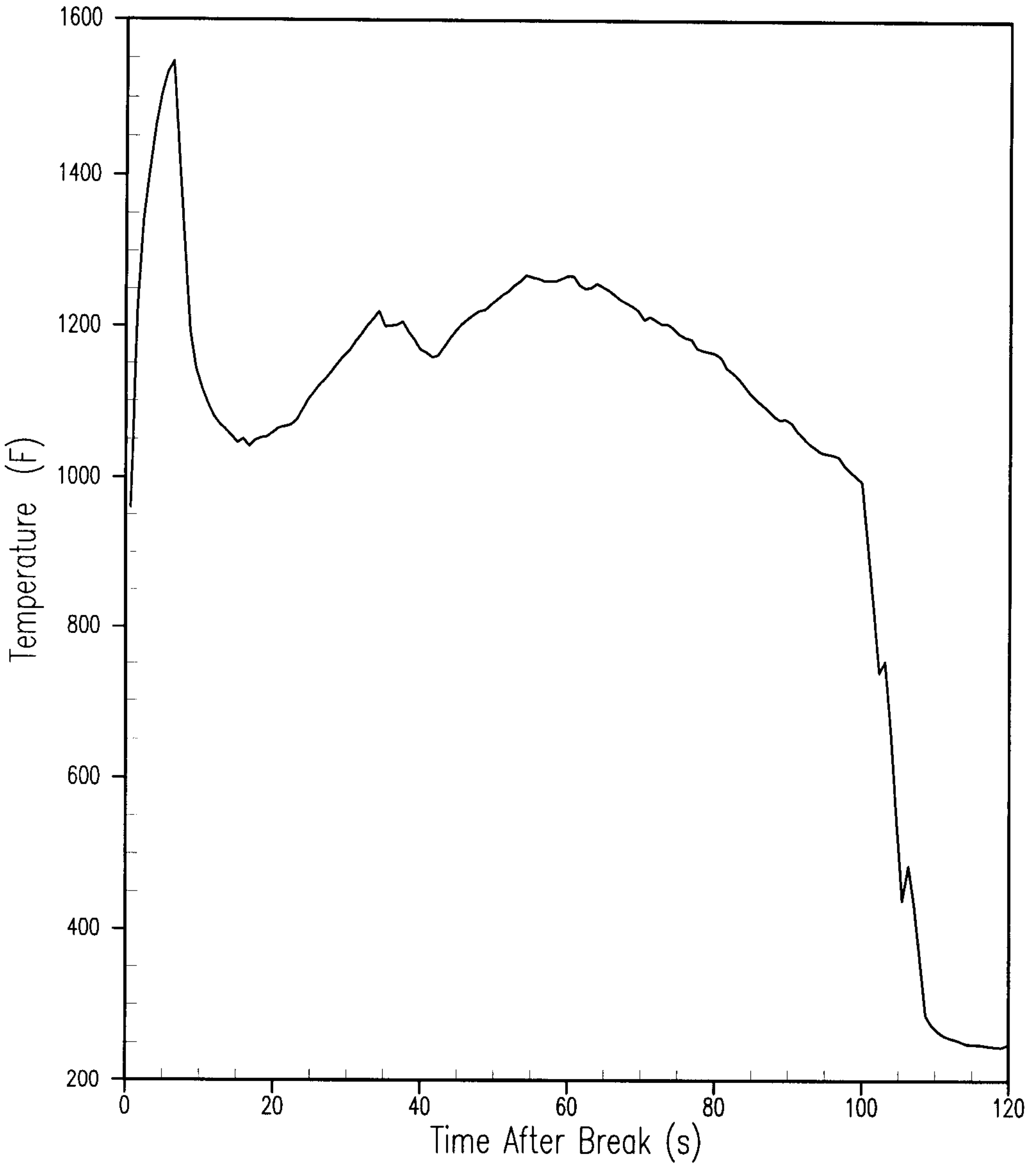

Figure 2-2 –HOTSPOT Clad Temperature Transient at the Limiting Elevation for the Limiting PCT Case

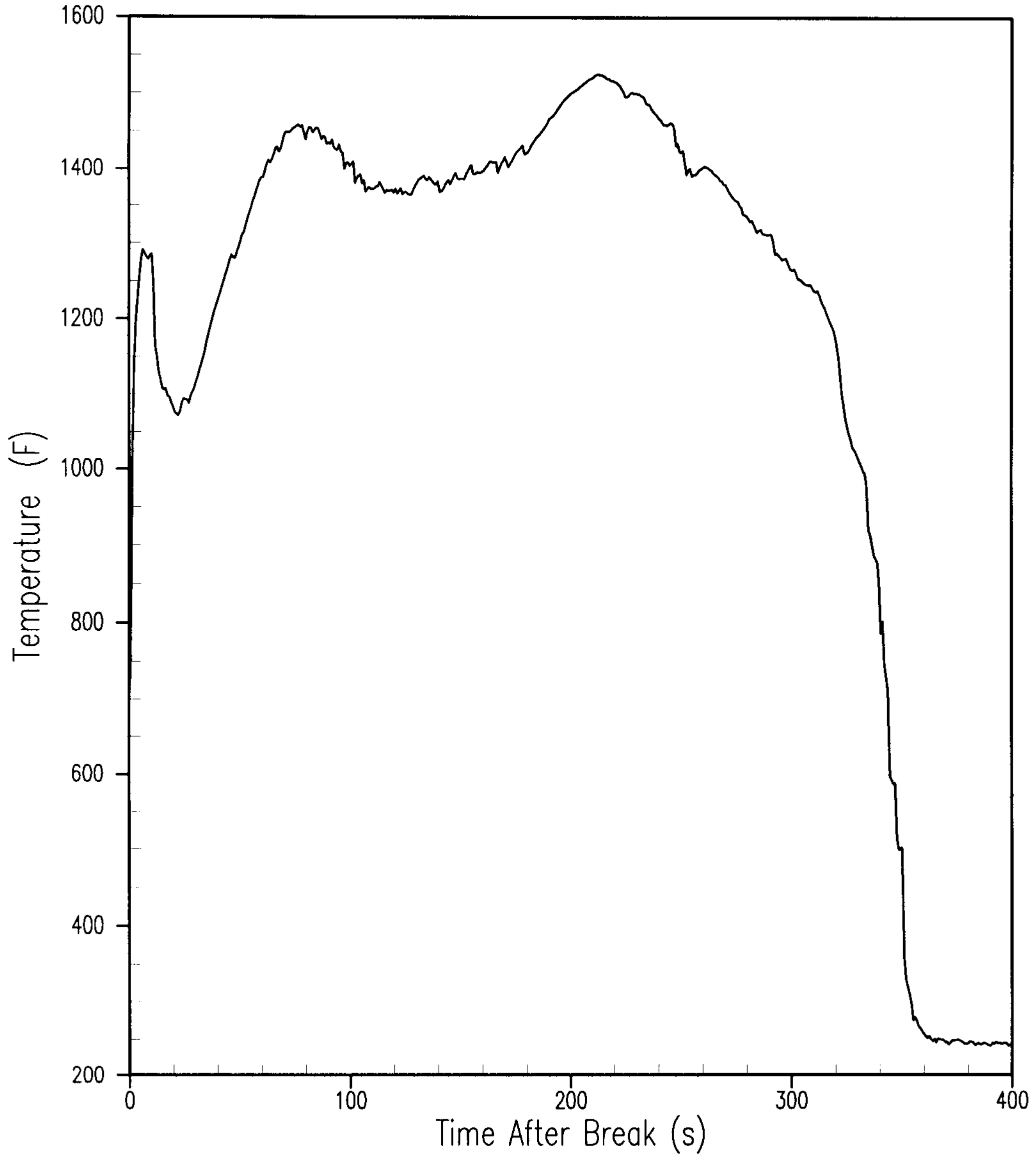

Figure 2-3 –HOTSPOT Clad Temperature Transient at the Limiting Elevation for the Limiting LMO Case

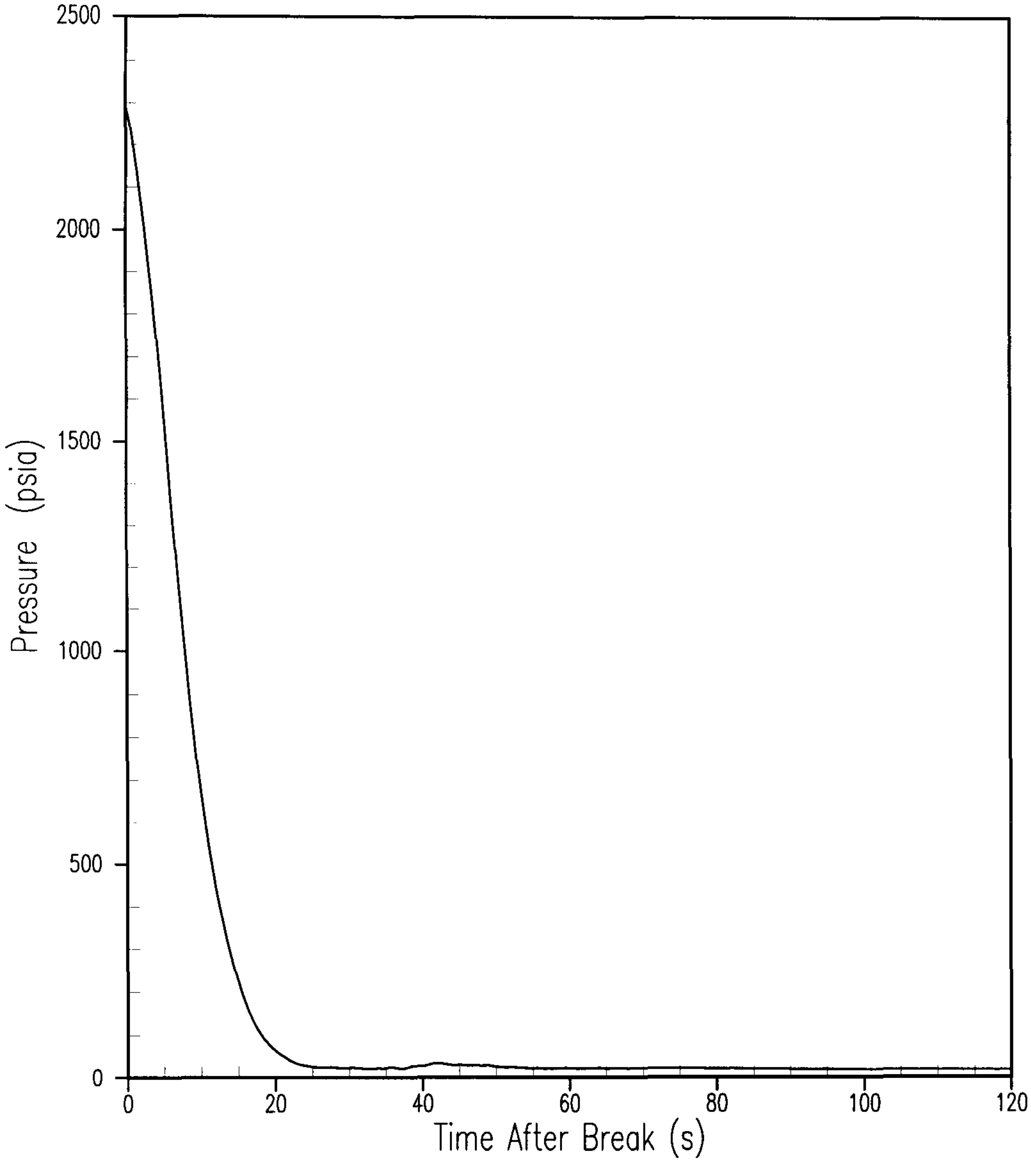

Figure 2-4 –Pressurizer Pressure for the Limiting PCT Case

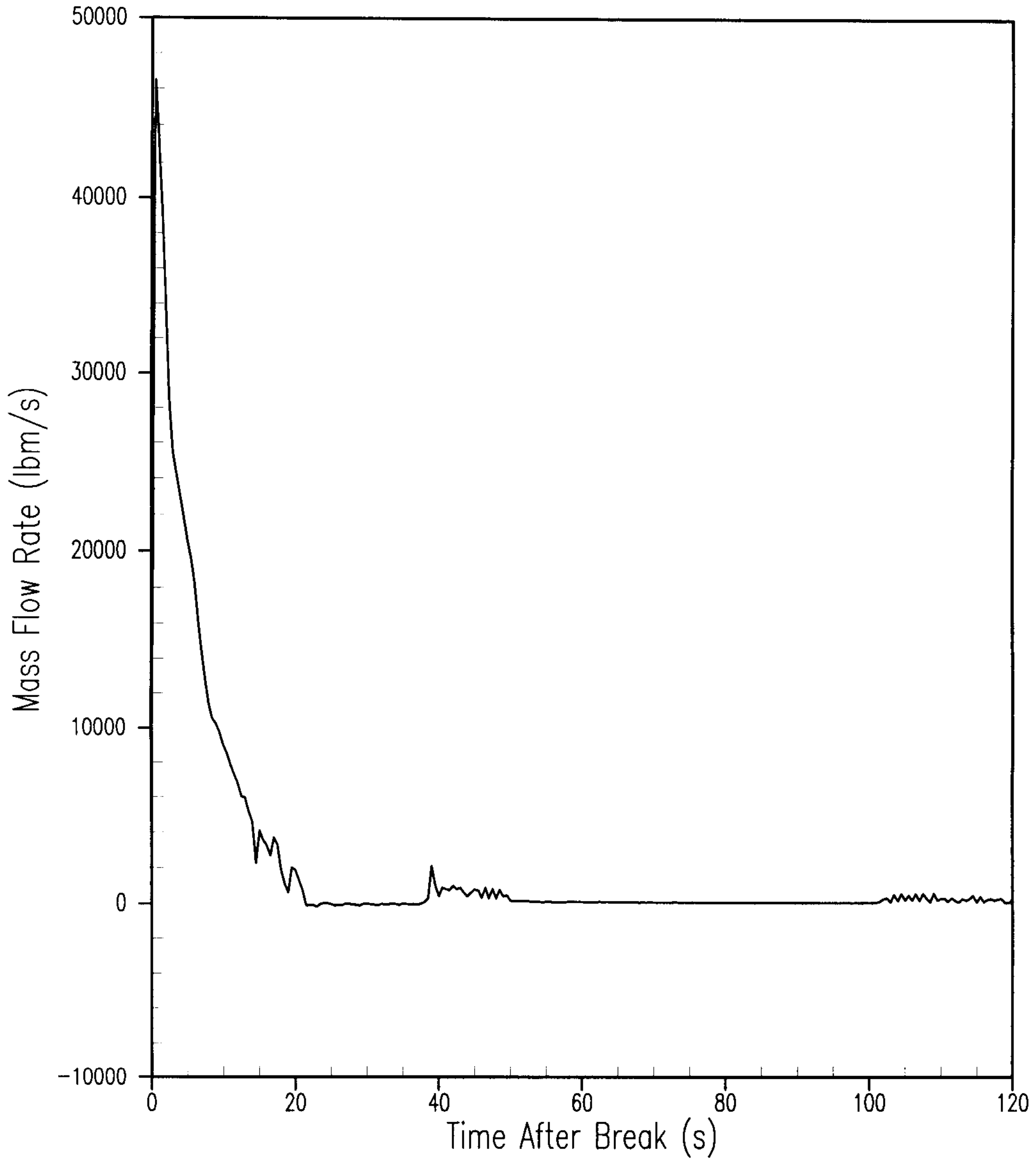

Figure 2-5 – Vessel Side Break Flow for the Limiting PCT Case

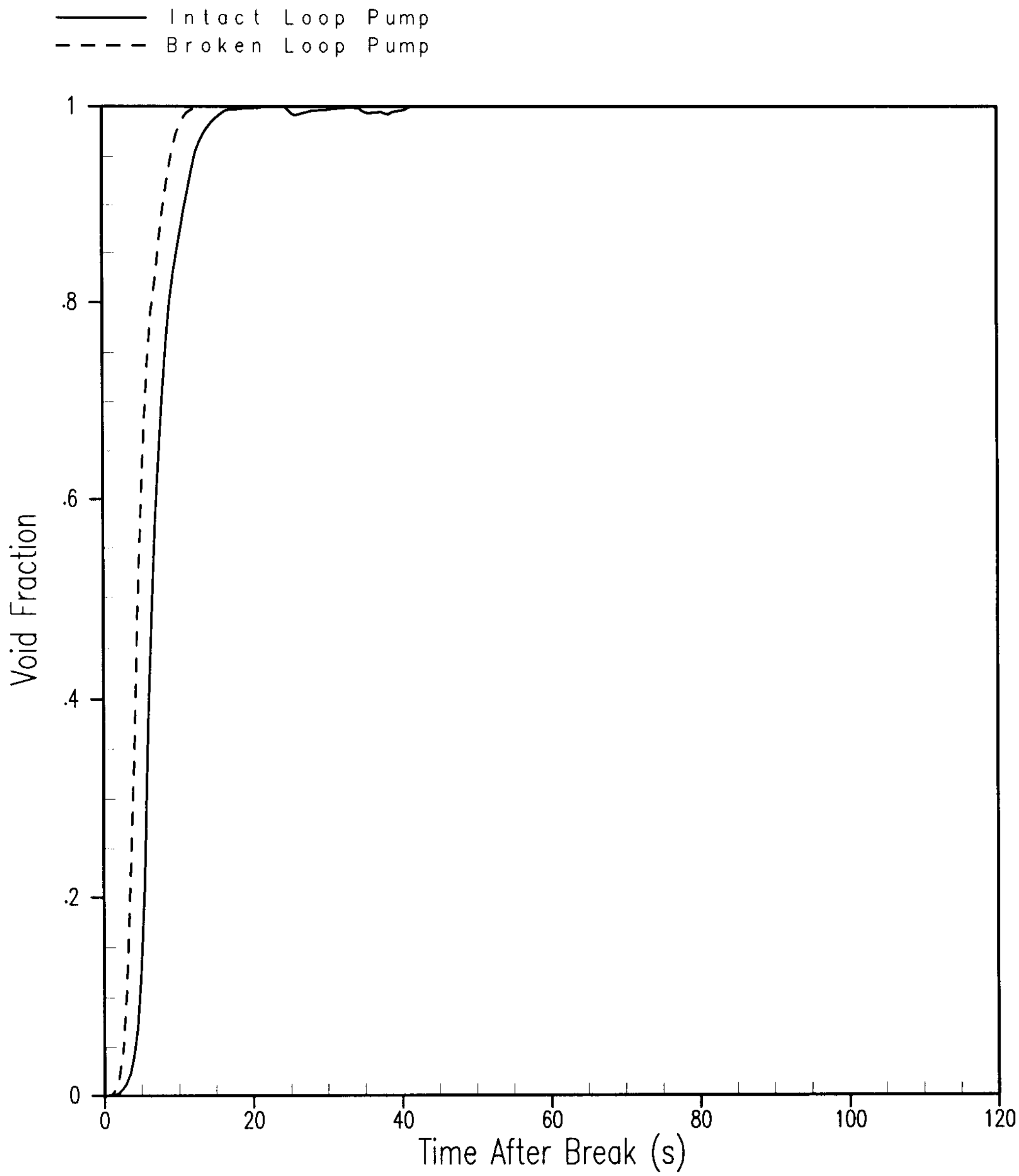

Figure 2-6 - Void Fraction in Pumps for the Limiting PCT Case

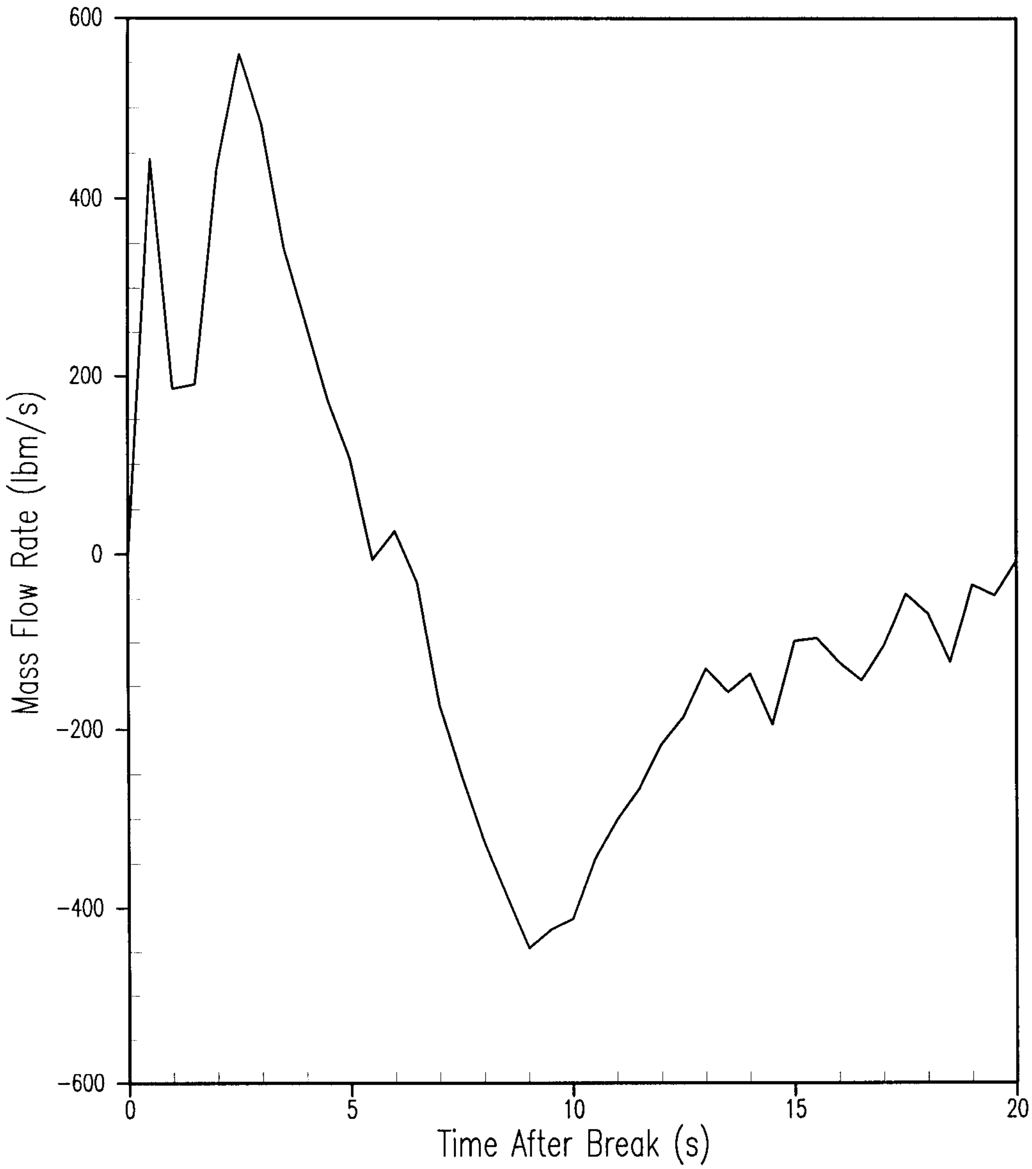

Figure 2-7 – Vapor Flow at Top of Core for the Limiting PCT Case

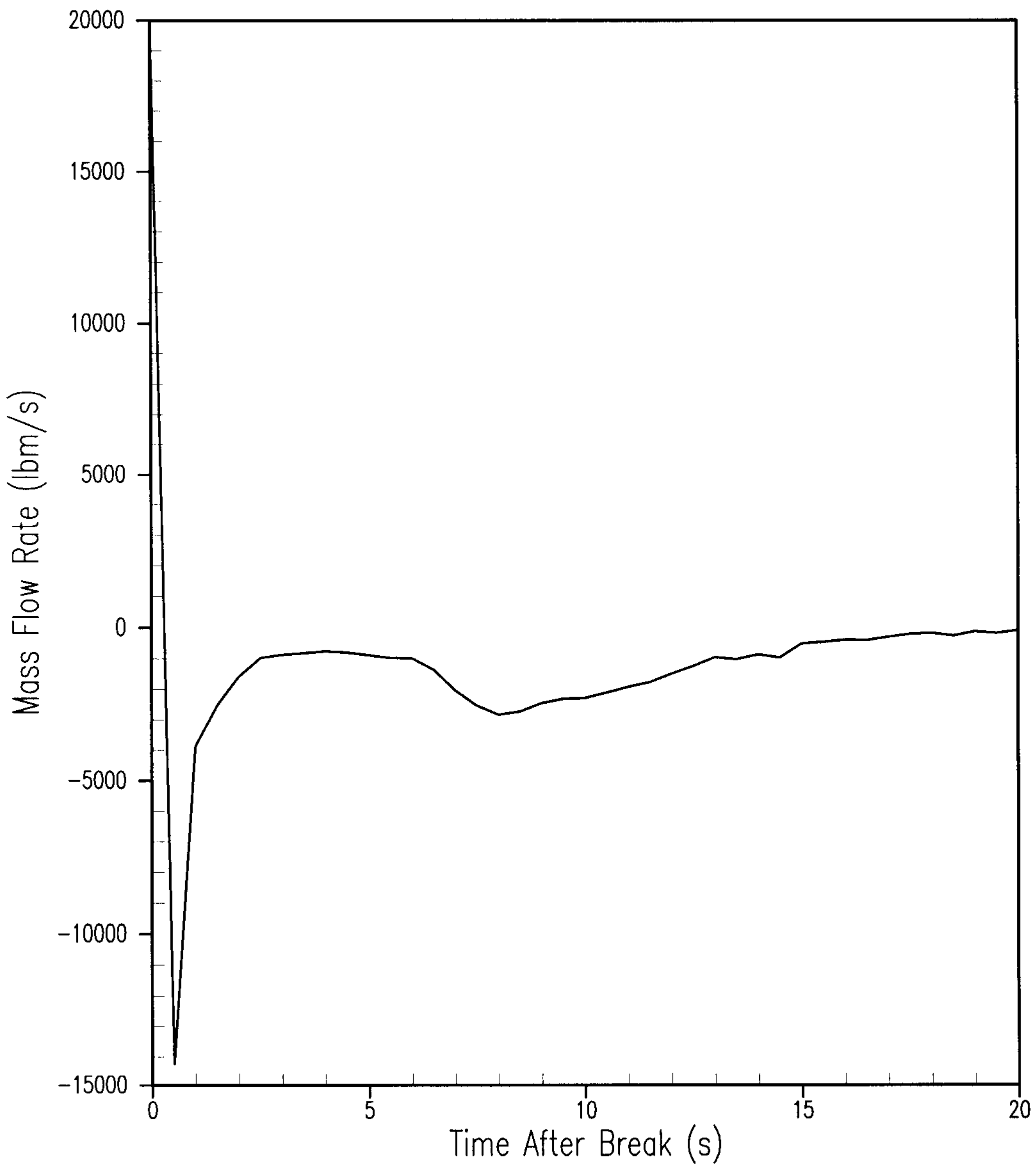

Figure 2-8 – Total Flow at Bottom of Core for the Limiting PCT Case

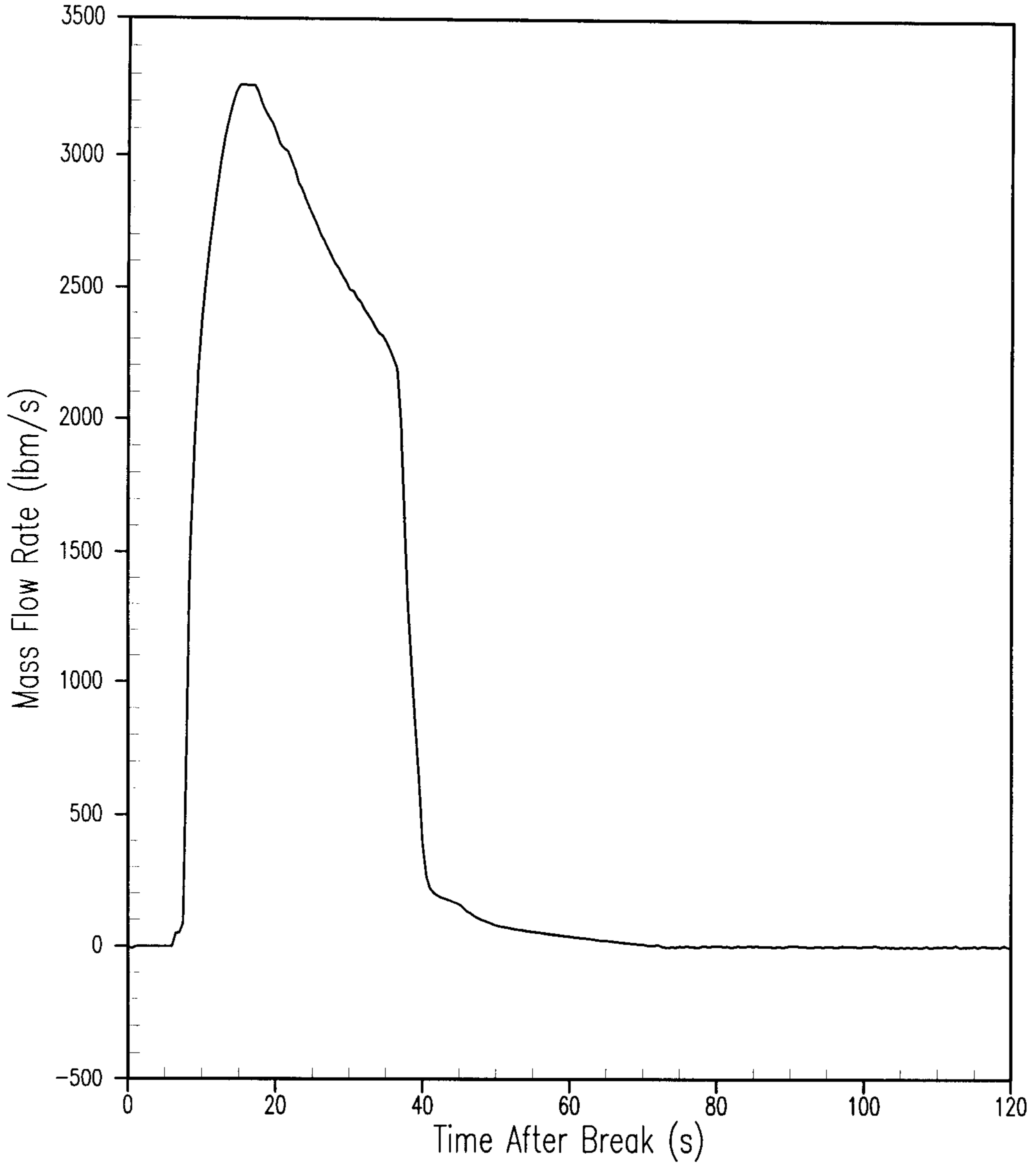

Figure 2-9 – Accumulator Injection Flow for the Limiting PCT Case

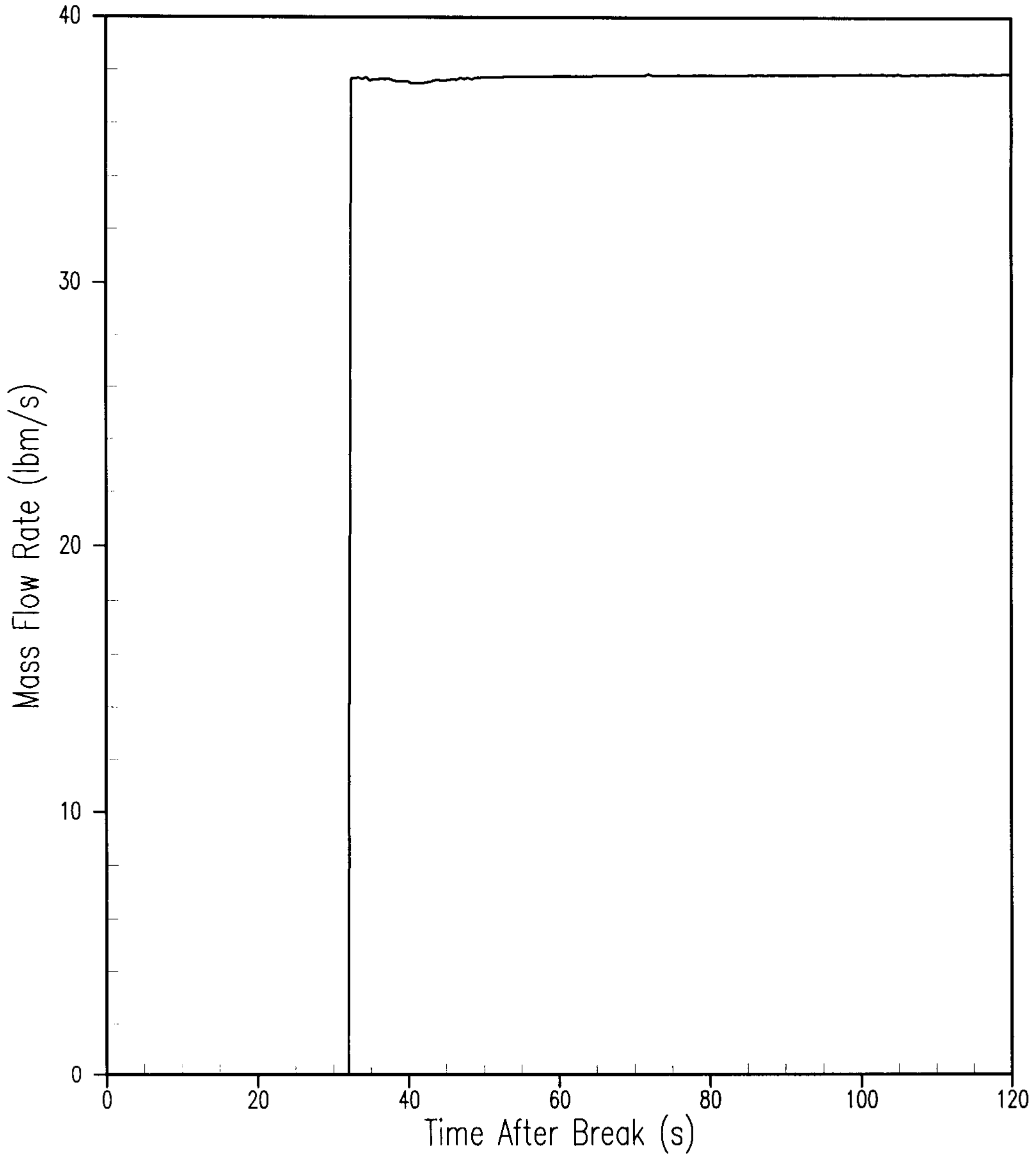

Figure 2-10 – High Head Safety Injection Flow for the Limiting PCT Case

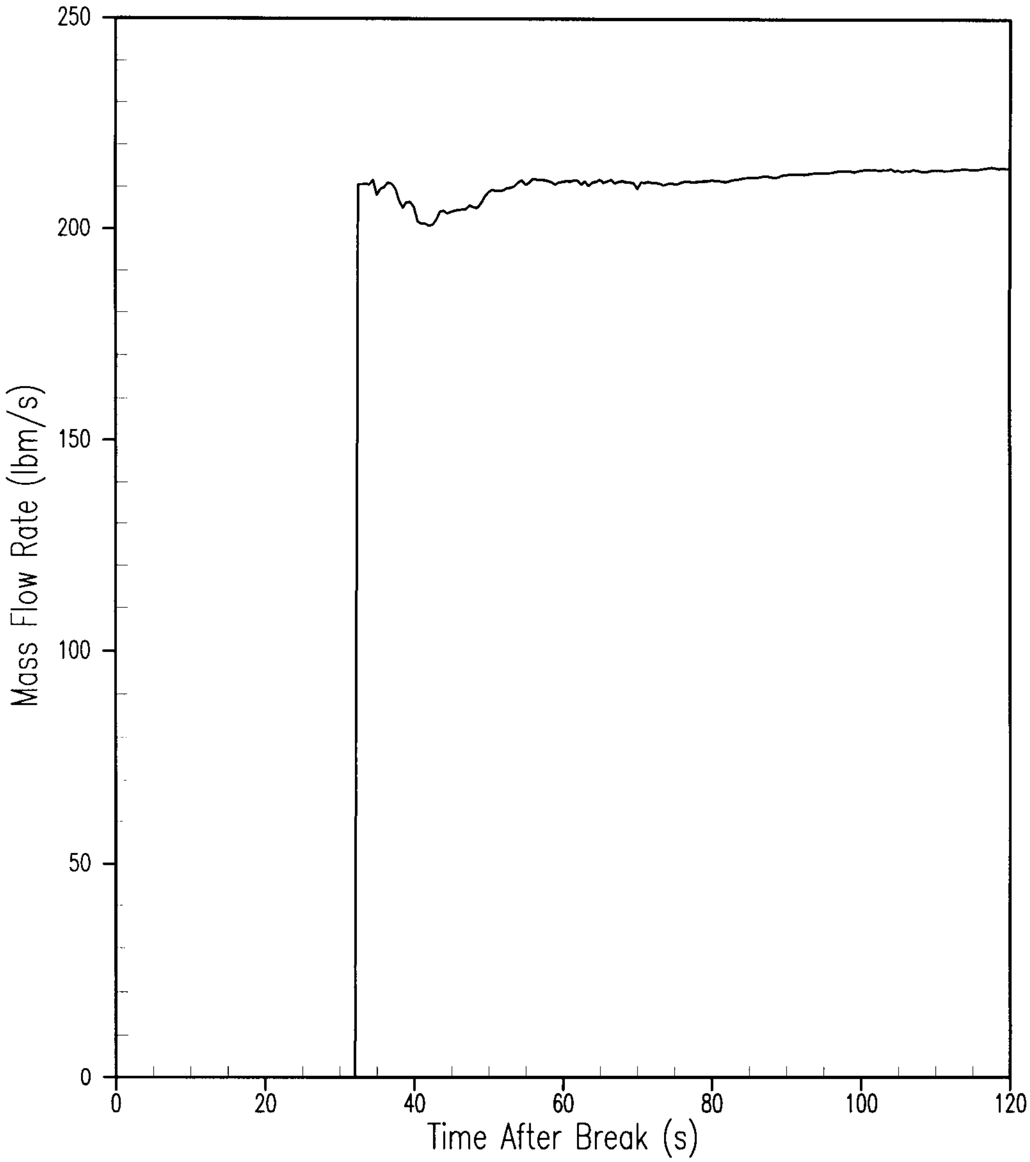

Figure 2-11 – Low Head Safety Injection Flow for the Limiting PCT Case

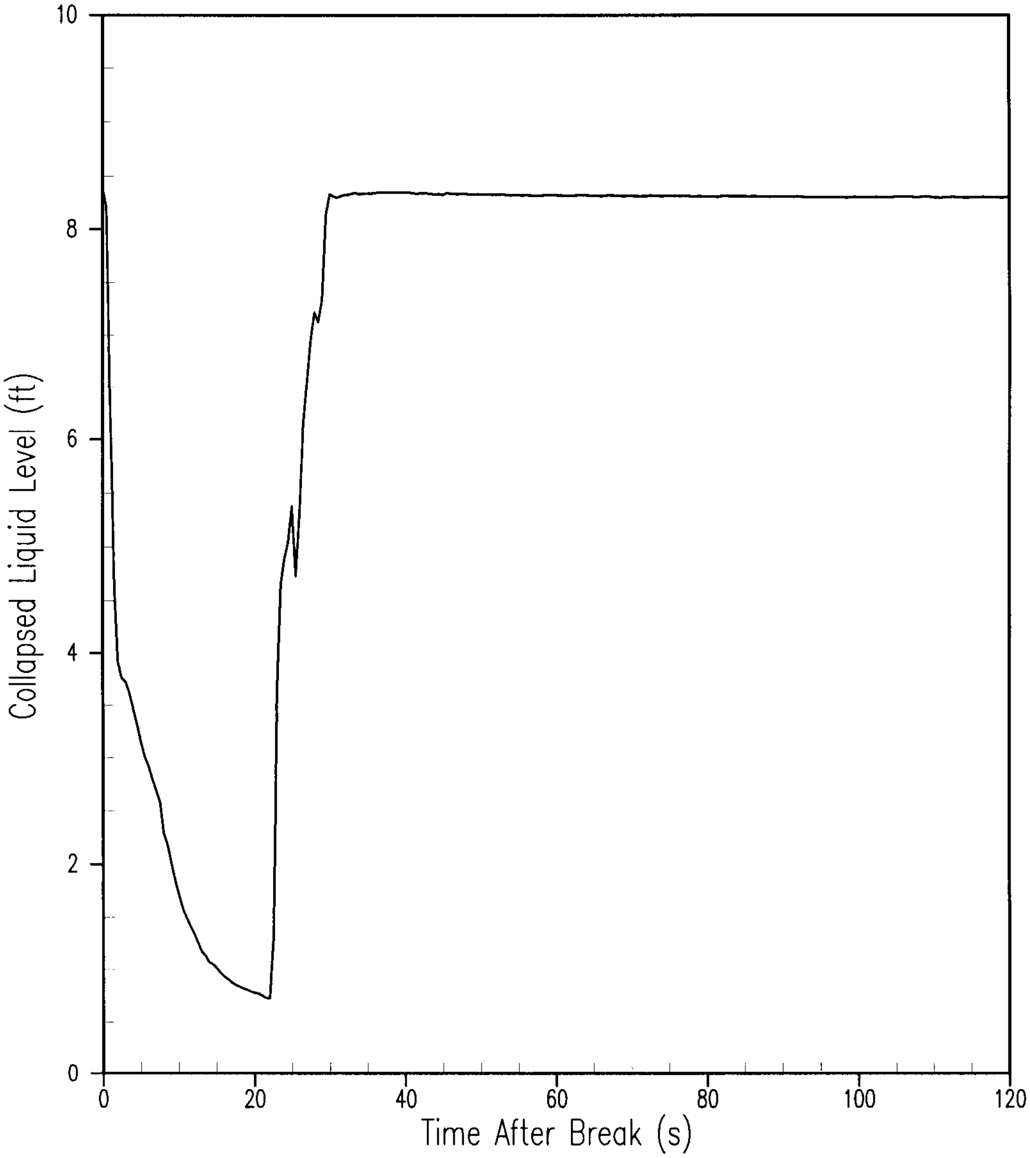

Figure 2-12 – Lower Plenum Collapsed Liquid Level for the Limiting PCT Case

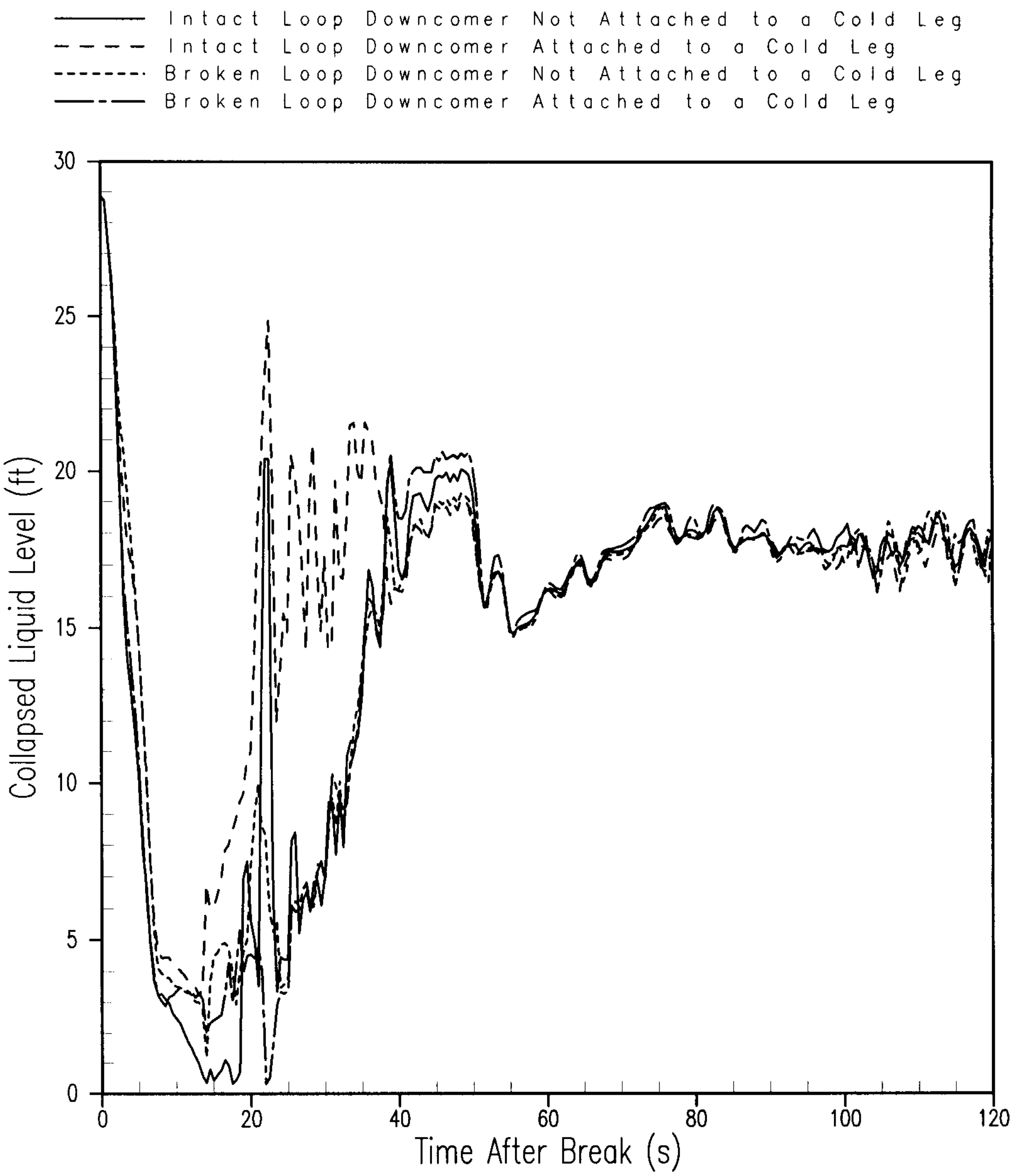

Figure 2-13 – Downcomer Collapsed Liquid Levels for the Limiting PCT Case

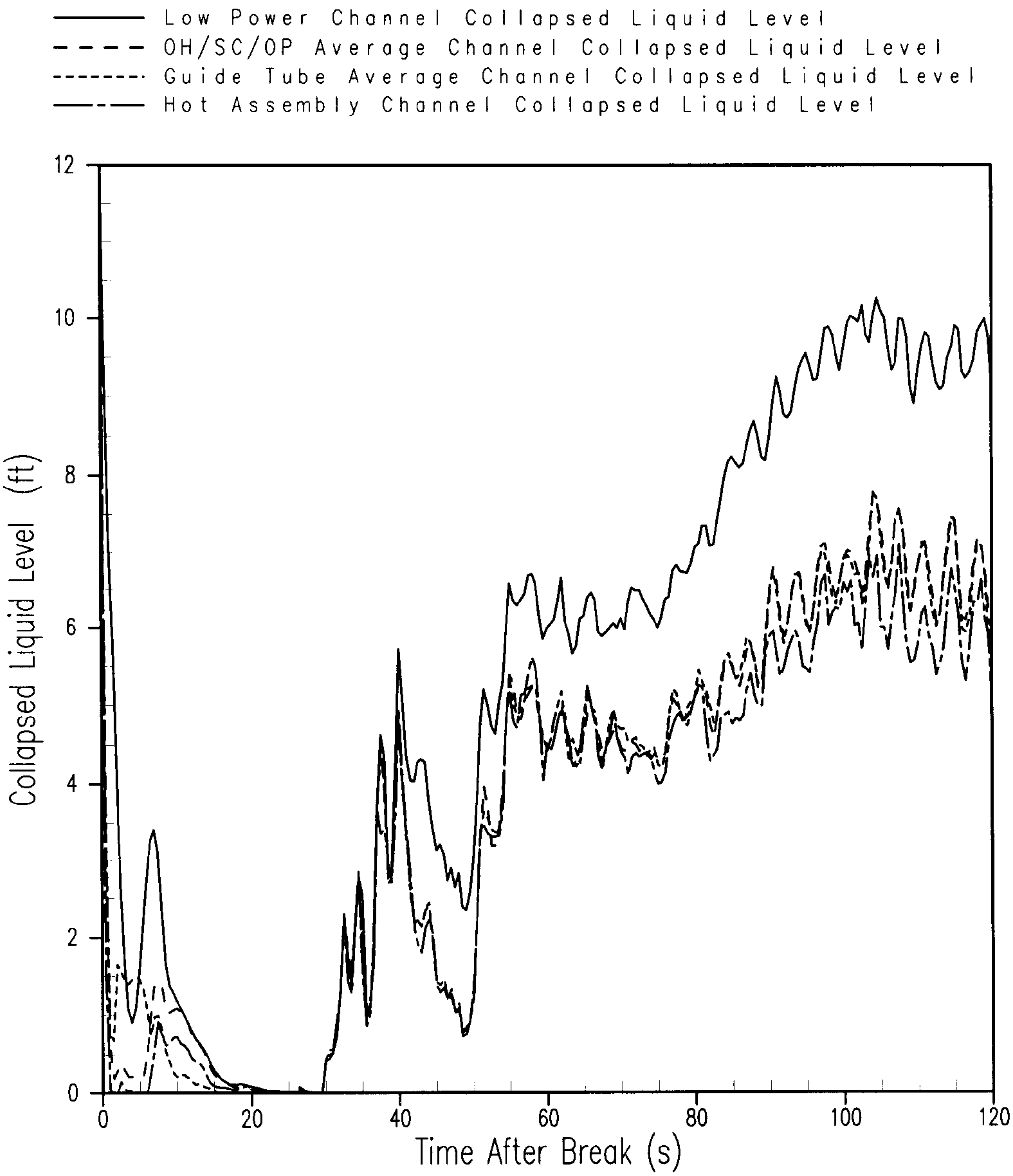

Figure 2-14 – Core Collapsed Liquid Levels for the Limiting PCT Case (OH = Open Holes, SC = Support Column, OP = Orifice Plate)

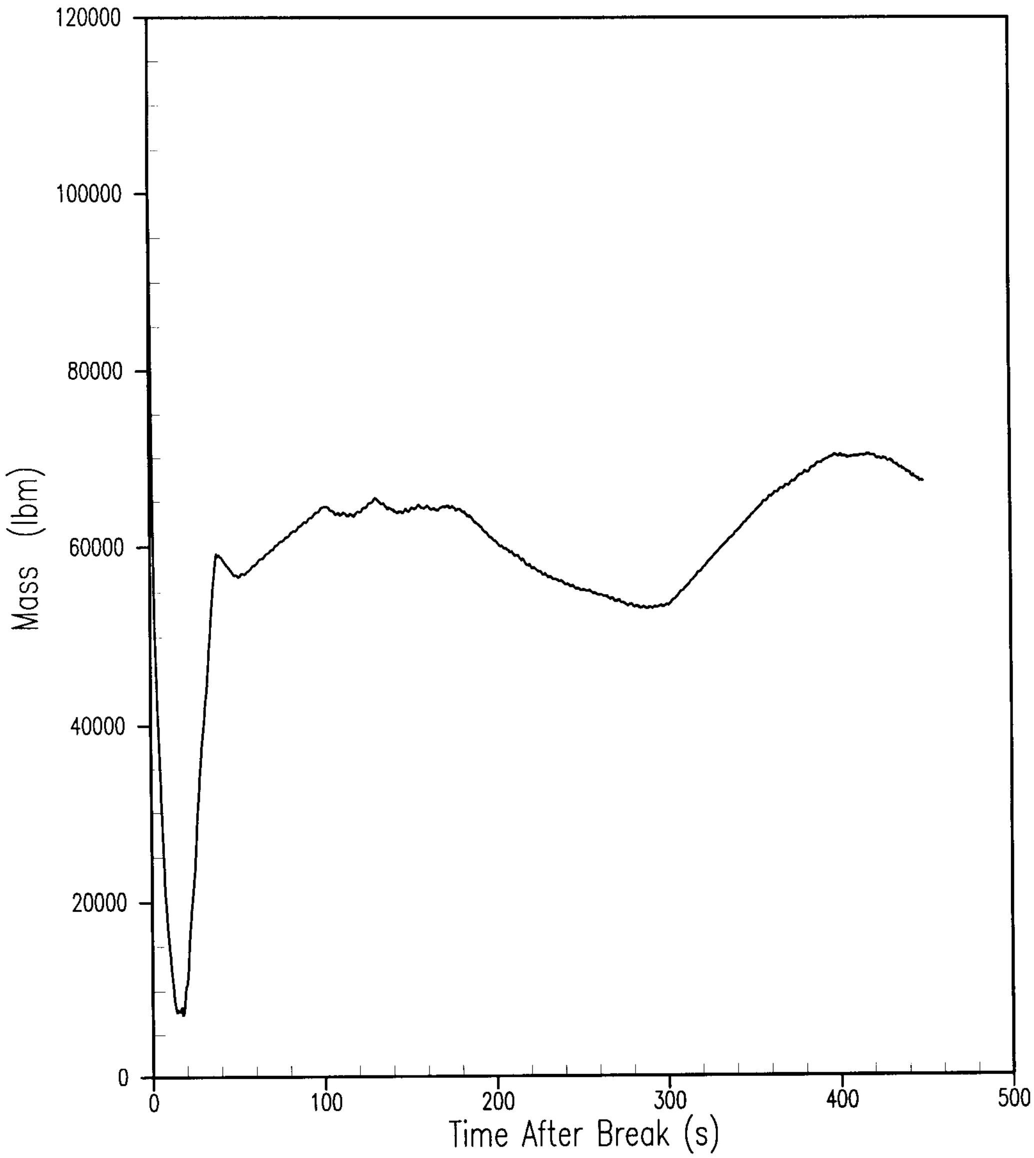

Figure 2-15 – Vessel Fluid Mass for the Limiting PCT Case

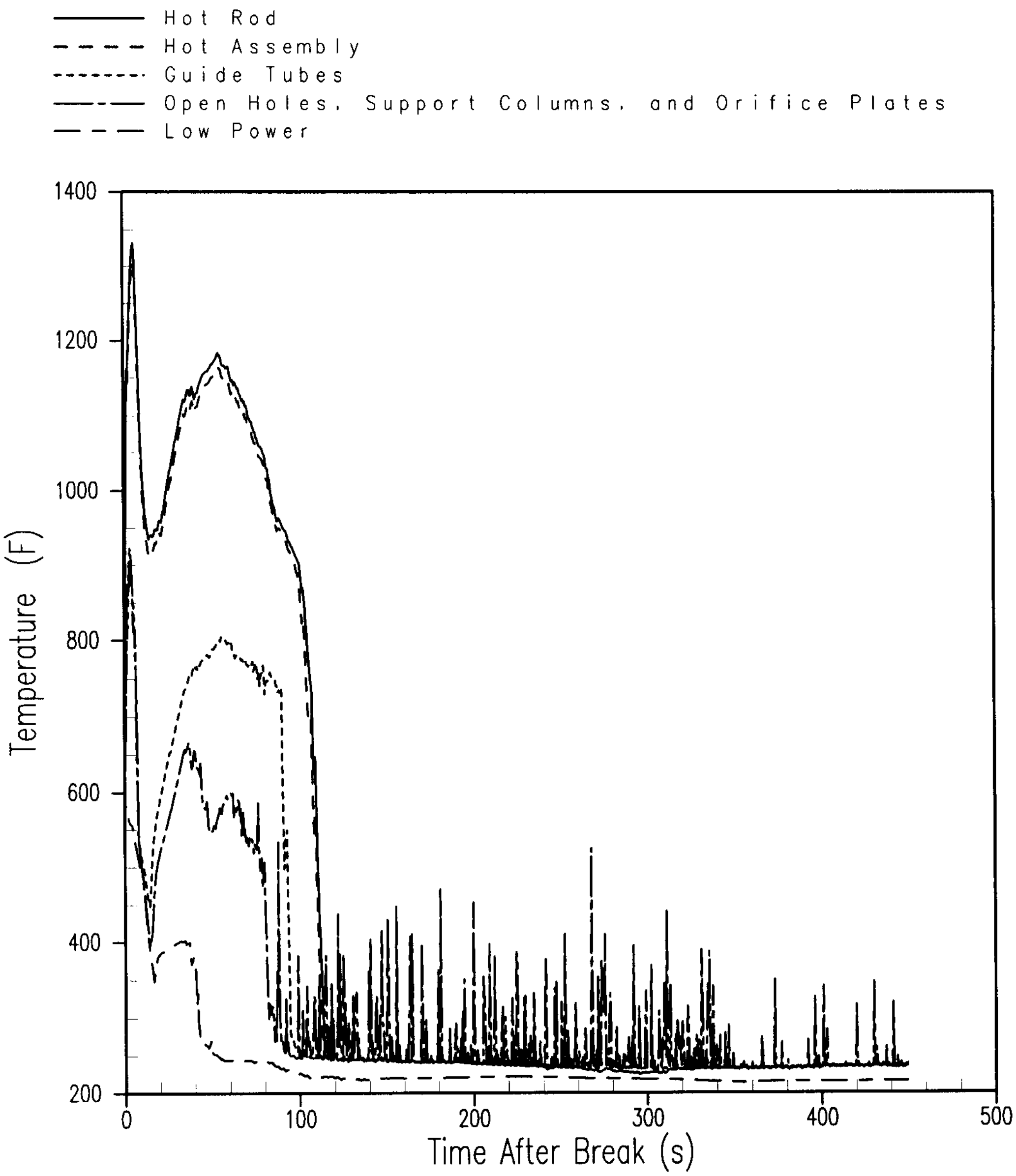

Figure 2-16 – WCOBRA/TRAC Peak Clad Temperature for all 5 Rod Groups for the Limiting PCT Case

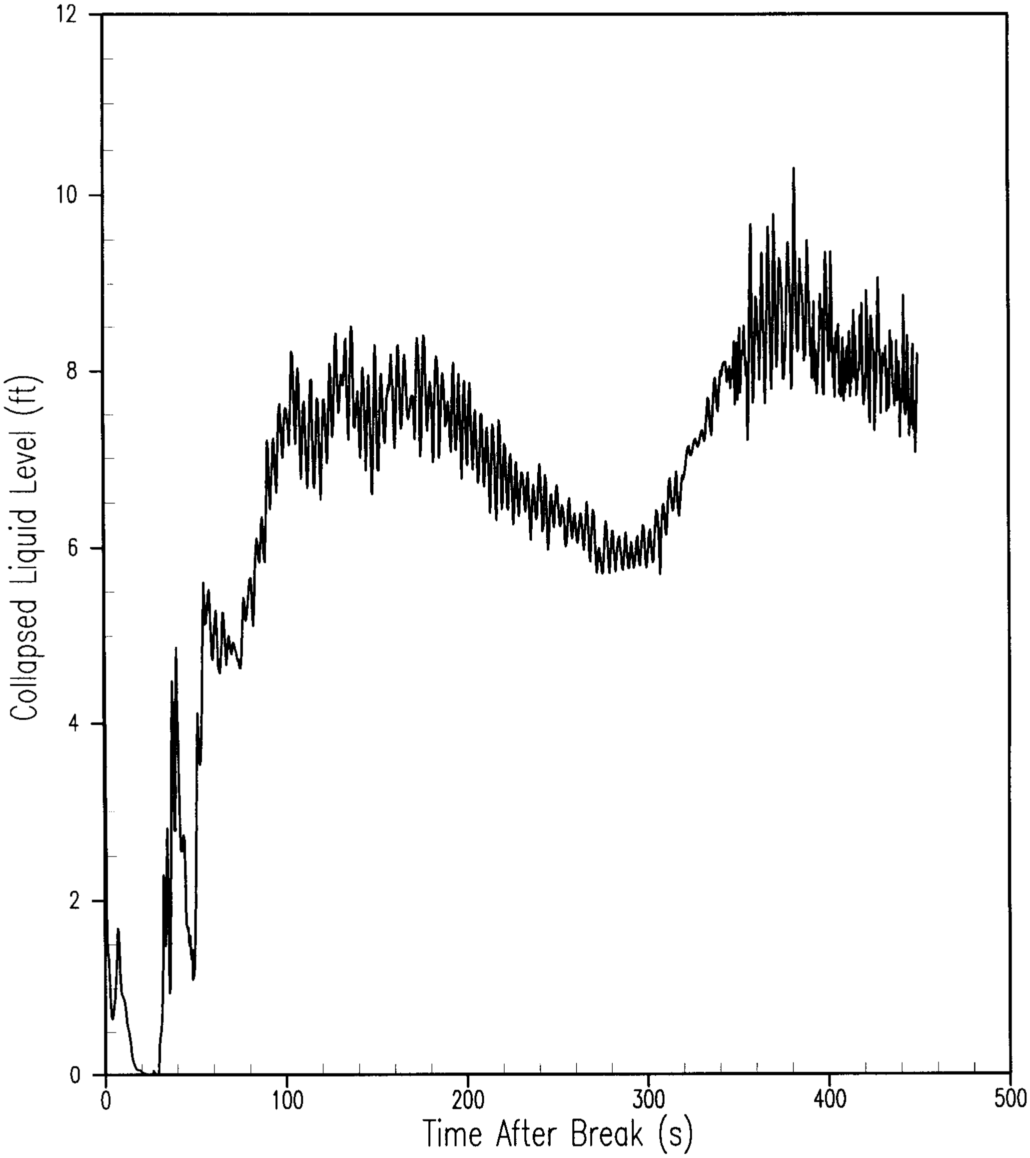

Figure 2-17 – Average Core Collapsed Liquid Level per Assembly for the Limiting PCT Case

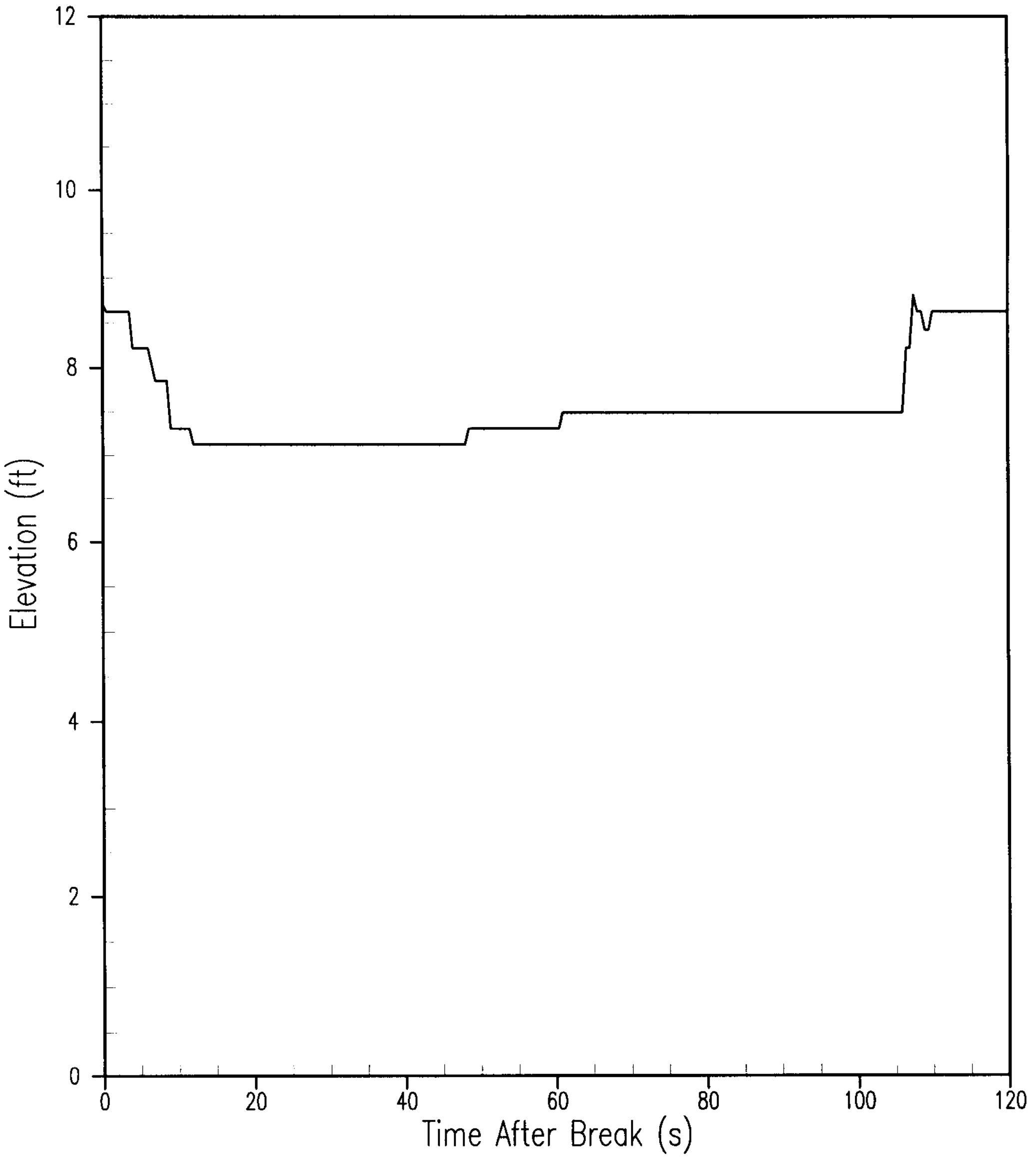

Figure 2-18 – Peak Clad Temperature Elevation for the Hot Rod for the Limiting PCT Case

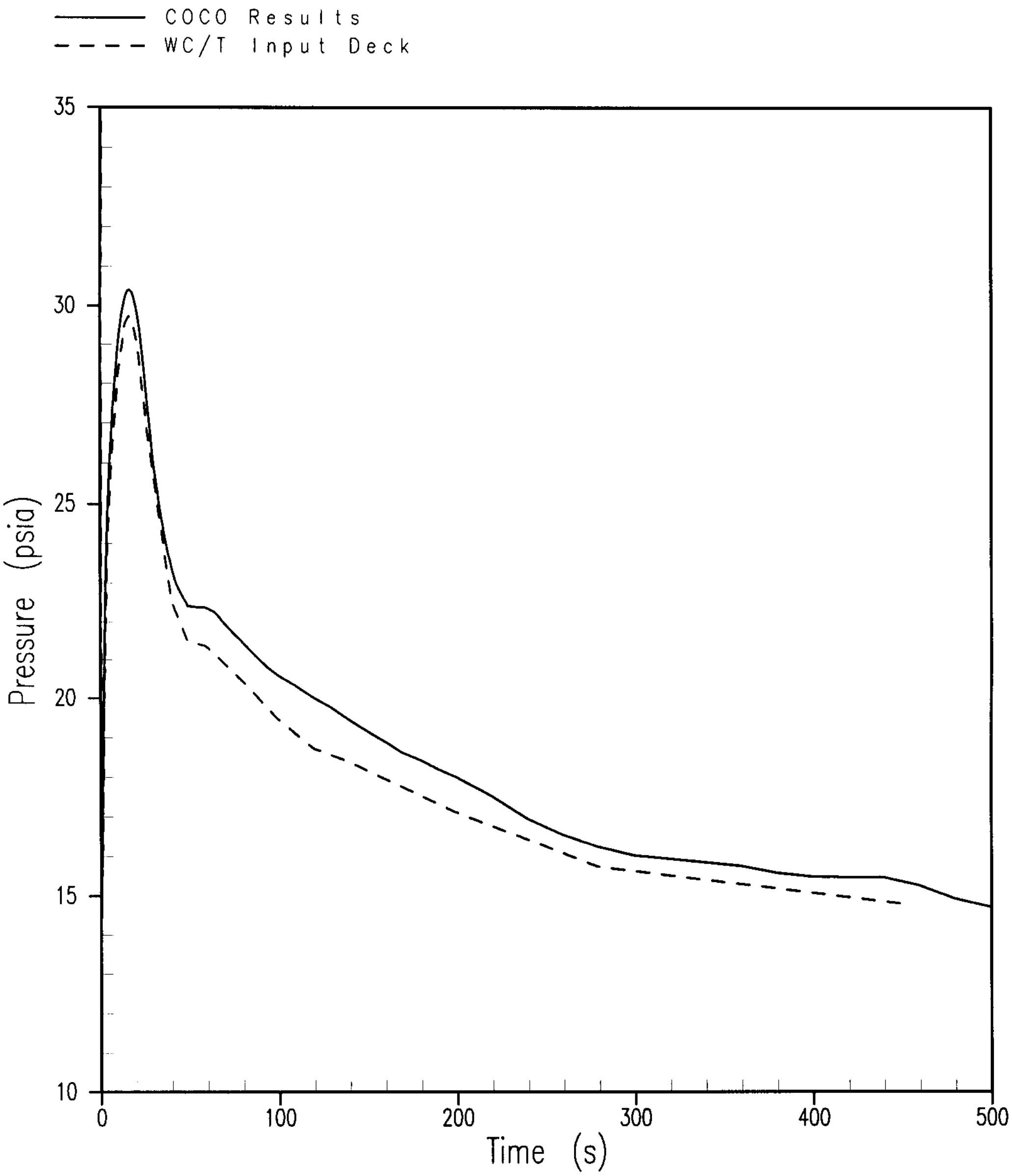

Figure 2-19 – Analysis Versus Calculated Containment Backpressure

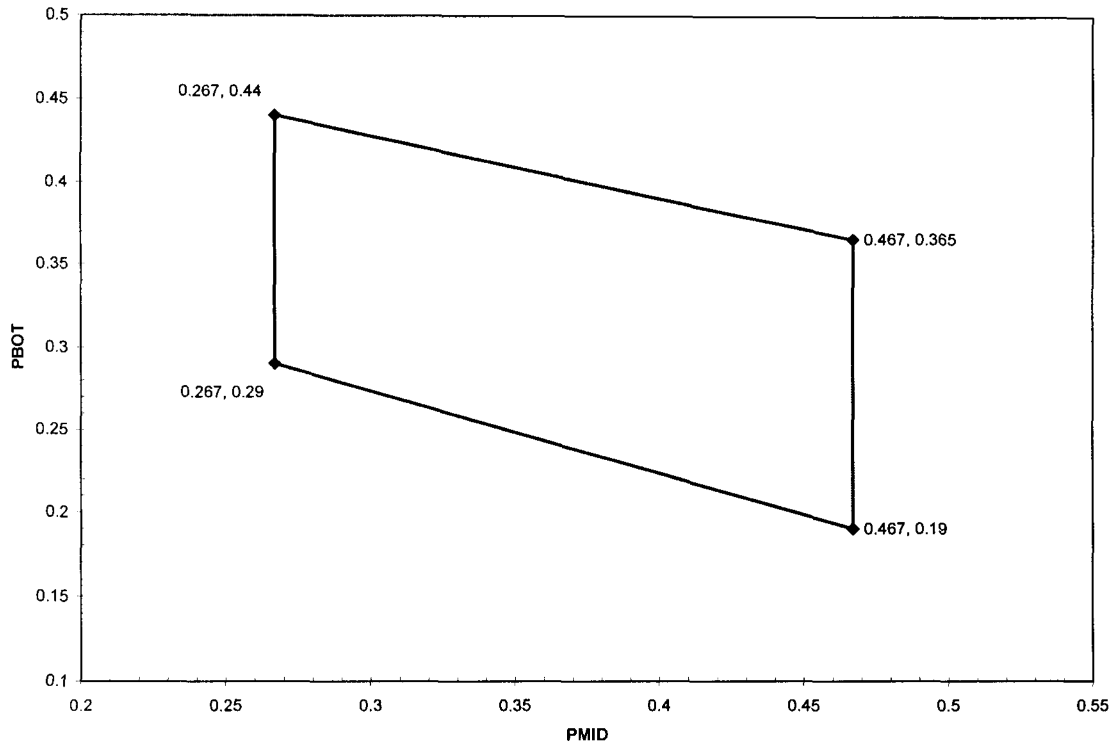

Figure 2-20- PINGP Unit 2 BELOCA Analysis Axial Power Shape Operating Space Envelope

PBOT = *integrated power fraction in the bottom third of the core*

PMID = integrated power fraction in the middle third of the core

Exhibit B

Proposed Technical Specification and Bases Changes (markup)

Technical Specification Page

5.0-36

Bases pages
(for information only)

B 3.5.1-1
B 3.5.1-3
B 3.5.1-4
B 3.6.5-5

5 pages follow

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

18. WCAP-7908-A, "FACTRAN – A FORTRAN IV Code for Thermal Transients in a UO_2 Fuel Rod";

19. WCAP-7907-P-A, "LOFTRAN Code Description";

20. WCAP-7979-P-A, "TWINKLE – A Multidimensional Neutron Kinetics Computer Code";

21. WCAP-10965-P-A, "ANC: A Westinghouse Advanced Nodal Computer Code";

22. WCAP-11394-P-A, "Methodology for the Analysis of the Dropped Rod Event";

23. WCAP-11596-P-A, "Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurized Water Reactor Cores";

24. WCAP-12910 Rev. 1-A, "Pressurizer Safety Valve Set Pressure Shift";

25. WCAP-14565-P-A, "VIPRE-01 Modeling and Qualification for pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis"; ~~and~~

26. WCAP-14882-P-A, "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses"~~.~~; and

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

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.1 Accumulators

BASES

BACKGROUND

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

The blowdown phase of a large break LOCA is the initial period of the transient during which the RCS departs from equilibrium conditions, and heat from fission product decay, hot internals, and the vessel continues to be transferred to the reactor coolant. The reactor coolant inventory is vacating the core during this phase through steam flashing and ejection out through the break. The blowdown phase of the transient ends when the collapsed liquid level in the lower plenum reaches a minimum and begins to increase~~RCS pressure falls to a value approaching that of the containment atmosphere~~.

In the refill phase of a LOCA, which immediately follows the blowdown phase, reactor coolant inventory has vacated the core through steam flashing and ejection out through the break. The core is essentially in adiabatic heatup. The balance of accumulator inventory is available to help fill voids in the lower plenum and reactor vessel downcomer, and to help the ongoing reflood of the core with the addition of water.

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

BASES

APPLICABLE SAFETY ANALYSES (continued)

The largest break area considered for a~~limiting~~ large break LOCA is a double ended guillotine break in the RCS cold leg~~at the discharge of the reactor coolant pump~~. During this event, the accumulators~~s~~ discharges to the RCS as soon as RCS pressure decreases to below accumulator pressure.

As a conservative estimate, no credit is taken for ECCS pump flow until an effective delay has elapsed. This delay accounts for safety injection (SI) signal generation, the diesels starting (for loss of offsite power assumption) and the pumps being loaded and delivering full flow. Prior to this delay elapsing~~The SI signal generation occurs approximately 2 seconds into the transient. During this time~~, the accumulators are analyzed as providing the sole source of emergency core cooling. No operator action is assumed during the blowdown stage of a large break LOCA.

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

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

a. The calculated peak fuel element cladding temperature is below the requirement of 2200°F;

b. The cladding temperature transient is terminated at a time when the core geometry is still amenable to cooling. The localized cladding oxidation limits of 17% are not exceeded during or after quenching;

BASES

APPLICABLE SAFETY ANALYSES (continued)

c. The amount of hydrogen generated by fuel element cladding that reacts chemically with water or steam does not exceed an amount corresponding to interaction of 1% of the total amount of Zircaloy in the reactor; and

d. The core remains amenable to cooling during and after the break.

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

~~For~~ T~~t~~he large break LOCA analysi~~es~~ considers a range of accumulator water volumes ~~, a nominal contained accumulator water volume of 1270 cubic feet is used~~ based on minimum and maximum volumes of 1250 cubic feet (25% indicated level) and 1290 cubic feet (91% indicated level).

The contained water volume is the same as the deliverable volume for the accumulators, since the accumulators are emptied, once discharged. For large breaks, an increase in water volume can be either a peak clad temperature penalty or benefit, depending on downcomer filling and subsequent spill through the break during the core reflooding portion of the transient. Prairie Island is a two loop plant with Upper Plenum Injection (UPI) LOCA analyses. For UPI plant small breaks, a decrease in water volume is a peak clad temperature penalty; thus, a minimum contained water volume is assumed. Both large and small break analyses use a nominal accumulator line water volume from the accumulator to the check valve.

The minimum boron concentration setpoint is used in the post LOCA boron concentration calculation. The calculation is performed to assure reactor subcriticality in a post LOCA environment. Of particular interest is the large break LOCA, since no credit is taken for control rod assembly insertion. A reduction in the accumulator minimum boron concentration would produce a subsequent

BASES

APPLICABLE SAFETY ANALYSES (continued)

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

The effect of an inadvertent containment spray actuation has been analyzed. An inadvertent spray actuation results in a containment pressure reduction associated with the sudden cooling effect in the interior of the leak tight containment. Additional discussion is provided in the Bases for LCO 3.6.8.

The modeled Containment Spray System actuation from the containment analysis is based on a response time associated with exceeding the containment High-High pressure setpoint to achieving full flow through the containment spray nozzles.

The analyses of the Main Steam Line Break (MSLB) and LOCA incorporated delays in Containment Spray actuation to account for load restoration, discharge valve opening, containment spray pump windup, and spray line filling (Ref. 3).

Containment cooling train performance for post accident conditions is given in Reference 4. The result of the analyses is that one train of containment cooling with one train of containment spray can provide 100% of the required peak cooling capacity during post accident conditions. The train post accident cooling capacity under varying containment ambient conditions, required to perform the accident analyses, is also shown in Reference 5.

The modeled Containment Cooling System actuation from the containment analysis is based upon a response time associated with receiving a safety injection (SI) signal to achieving full

Exhibit C

Proposed Technical Specification and Bases Changes (markup)

Technical Specification Page

5.0-36

1 page follows

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

18. WCAP-7908-A, "FACTRAN – A FORTRAN IV Code for Thermal Transients in a UO_2 Fuel Rod";

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