FPL Energy Point Beach, LLC, 6610 Nuclear Road, Two Rivers, WI 54241

FPL Energy.
Point Beach Nuclear Plant

November 25, 2008

NRC 2008-0086
10 CFR 50.90
10 CFR 50.46

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

Point Beach Nuclear Plant, Units 1 and 2
Dockets 50-266 and 50-301
Renewed License Nos. DPR-24 and DPR-27

License Amendment Request 258
Incorporate Best Estimate Large Break Loss Of Coolant Accident (LOCA)
Analyses Using ASTRUM

Pursuant to 10 CFR 50.90, FPL Energy Point Beach, LLC hereby requests an amendment to the Renewed Operating Licenses for Point Beach Nuclear Plant (PBNP), Units 1 and 2 to incorporate new best estimate (BE) large break loss-of-coolant accident (LBLOCA) analyses. This amendment requests use of the realistic LBLOCA methodology presented in the NRC approved WCAP-16009-P-A, "Realistic Large Break LOCA Evaluation Methodology Using Automated Statistical Treatment of Uncertainty Method (ASTRUM)," (ML043100073) and to revise Technical Specification (TS) 5.6.4.b to include a reference to WCAP-16009-P-A.

This request also proposes to implement Technical Specification Task Force (TSTF) Traveler-363A. TSTF-363A is incorporated into the current version of NUREG-1431, Standardized Technical Specifications for Westinghouse Pressurized Water Reactors. This application was approved by the NRC staff on July 9, 2002, for Oconee Nuclear Station, Units 1, 2 and 3 (ML021900580) and has been incorporated into NUREG-1431, Standard Technical Specifications for Westinghouse Plants, Revision 3.1. TSTF-363A eliminates the revision numbers and dates from the list of topical reports in TS 5.6.4.b. TS 5.6.4.b provides the analytical methods used to determine the core operating limits. Relocation of the complete citations from TS 5.6.4.b, in accordance with the approved TSTF which is incorporated into Version 3.1 of NUREG-1431, to the core operating limits report (COLR) will enable the current revisions of these topical reports to be used.

Submittal of this license amendment request fulfills the commitment made by Nuclear Management Company, LLC (NMC), former license holder for PBNP, in letter NRC 2007-0081, dated September 14, 2007 (ML072570301), and modified by FPL Energy Point Beach letter NRC 2008-0080 dated October 30, 2008, to provide a new LBLOCA analysis for PBNP by November 28, 2008. The commitments were made as follow-up to the NMC 30-day notification, NRC 2007-0055, (ML071860067) to the NRC dated July 3, 2007, regarding errors in the emergency core cooling system (ECCS) evaluation model as required by 10 CFR 50.46.

The new analyses have been performed in support of the planned PBNP extended power uprate (EPU) to 1800 MWt. The new LBLOCA analyses bound the current licensed power level of 1540 MWt and are applicable to the current licensed thermal power level. NRC approval is being requested for the current licensed power level of 1540 MWt. The EPU license amendment will incorporate this application by reference for approval of the LBLOCA at the uprated power level.

FPL Energy Point Beach requests approval of this license amendment request in approximately one calendar year. The license amendment will be implemented 60 days following approval.

Enclosure 1 contains the licensee's evaluation of this license amendment request, including an evaluation determining that the proposed change involves no significant hazards as defined in 10 CFR 50.92, and an evaluation that concludes this change satisfies the criteria of 10 CFR 51.22 for categorical exclusion from the requirements for an environmental assessment. Enclosure 2 provides a markup of TS 5.6.4 changes.

In accordance with 10 CFR 50.91, a copy of this license amendment request has been provided to the designated State of Wisconsin Official. This license amendment request has been reviewed by the Plant Operations Review Committee.

Summary of Commitments

This letter contains no new commitments and no revisions to existing commitments. This letter fulfills a Regulatory Commitment stated in NRC 2008-0080 dated October 30, 2008, to resubmit the LBLOCA analyses by November 28, 2008, to demonstrate continued compliance with the provisions of 10 CFR 50.46.

I declare under penalty of perjury that the foregoing is true and correct.
Executed on November 25, 2008

Very truly yours,

FPL Energy Point Beach, LLC

11/25/08

Larry Meyer
Site Vice President

Enclosures

ENCLOSURE 1

FPL ENERGY POINT BEACH, LLC
POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

LICENSE AMENDMENT REQUEST 258
INCORPORATE BEST ESTIMATE LARGE BREAK LOSS OF COOLANT ACCIDENT (LOCA) ANALYSES USING ASTRUM

52 pages follow

1.0 SUMMARY DESCRIPTION

This license amendment request proposes to amend Renewed Operating Licenses DPR-24 and DPR-27 for Point Beach Nuclear Plant (PBNP) Units 1 and 2 to incorporate new Large Break LOCA (LBLOCA) analyses using the realistic LBLOCA methodology contained in NRC-approved WCAP-16009-P-A, "Realistic Large Break LOCA Evaluation Methodology Using Automated Statistical Treatment of Uncertainty Method (ASTRUM)," and to revise Technical Specification (TS) 5.6.4.b to include reference to WCAP-16009-P-A. FPL Energy Point Beach requests Nuclear Regulatory Commission (NRC) review and approval of this proposed licensing basis change.

Based on the ASTRUM analyses results (Table 2), FPL Energy Point Beach concludes that PBNP Units 1 and 2 continue to maintain a margin of safety to the limits prescribed by 10 CFR 50.46.

This request also proposes to implement Technical Specification Task Force (TSTF) Traveler-363A. TSTF-363A is incorporated into the current version of NUREG-1431, Standardized Technical Specifications for Westinghouse Pressurized Water Reactors. The application was approved by the NRC staff on July 9, 2002, for Oconee Nuclear Station, Units 1, 2 and 3 (ML021900580) and has been incorporated into NUREG-1431, Standard Technical Specifications for Westinghouse Plants, Revision 3.1. TSTF-363A eliminates the revision numbers and dates from the list of topical reports in TS 5.6.4.b. TS 5.6.4.b provides the analytical methods used to determine the core operating limits. Relocation of the complete citations to the core operating limits report (COLR) will enable the current revisions of these topical reports to be used.

2.0 DETAILED DESCRIPTION

A description of the associated proposed TS changes is provided below along with a discussion of the justification for each change. The specific wording changes to TS 5.6.4.b are provided in Enclosure 2. This request also proposes to implement TSTF Traveler-363A, which relocates the complete citations for analytical methods listed in TS 5.6.4.b(1) through (12) to the COLR.

This license amendment request proposes to add Reference (13) to the list of documents contained in TS 5.6.4.b describing NRC approved analytical methods which may be used to determine core operating limits for the PBNP units. WCAP-16009-P-A, "Realistic Large-Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)", will be included as Reference (13). The PBNP Units 1 and 2 plant-specific best estimate LBLOCA analyses are not based on the model or analysis of any other plant. The analyses are PBNP unit-specific. The LBLOCA licensing basis in place previous to this submittal was applicable to both Units 1 and 2 and resulted in a PCT of 2131°F, including all rackup items.

TSTF-363A is incorporated into the current version of NUREG-1431, Standardized Technical Specifications for Westinghouse Pressurized Water Reactors. This application was approved by the NRC staff on July 9, 2002, for Oconee Nuclear Station, Units 1, 2 and 3 (ML021900580) and has been incorporated into NUREG-1431, Standard Technical Specifications for Westinghouse Plants, Revision 3.1. TSTF-363A eliminates the revision numbers and dates from the list of topical reports in TS 5.6.4.b. TS 5.6.4.b provides the analytical methods used to determine the core operating limits. Relocation of the complete citations to the core operating

COLR will enable the current revisions of these topical reports to be used without obtaining NRC approval for revisions to the topical reports.

The proposed changes are acceptable based upon the discussions in Section 4.0 of this license amendment request.

PBNP is a two-unit plant located on the west side of Lake Michigan approximately 35 miles southeast of Green Bay, Wisconsin. The facility is owned by FPL Energy Point Beach, LLC. Each unit at PBNP employs a two-loop pressurized water reactor designed and supplied by Westinghouse Electric Corporation. Units 1 and 2 were granted Operating Licenses by the AEC via a safety evaluation dated July 5, 1970. Unit 1 began commercial operation in December 1970 and Unit 2 began commercial operation in September 1972.

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

The current PBNP LBLOCA analyses of record were performed using:

- WCAP-14449-P-A, "Application of Best Estimate Large Break LOCA Methodology to Westinghouse PWRs with Upper Plenum Injection;"
- WCAP-10924-P-A, "Large Break LOCA Best Estimate Methodology, Volume 2: Application to Two-Loop PWRs Equipped with Upper Plenum Injection;" and
- WCAP-10924-P-A, "LBLOCA Best Estimate Methodology: Volume I, Model Description and Validation: Model Revisions."

This license amendment request fulfills the commitment made by Nuclear Management Company, LLC (NMC), former license holder for PBNP, in letter NRC 2007-0081, dated September 14, 2007 (ML072570301), and modified by FPL Energy Point Beach via letter NRC 2008-0080 dated October 30, 2008, to provide a new LBLOCA analysis for each PBNP unit by November 28, 2008. As follow-up to the NMC 30-day notification (ML071860067) to the NRC dated July 3, 2007, the commitments were made as required by 10 CFR 50.46 regarding errors in the Emergency Core Cooling System (ECCS) evaluation model . The best estimate LBLOCA analyses using ASTRUM and associated TS changes proposed in this request are presented in fulfillment of this commitment to the NRC.

Westinghouse obtained generic NRC approval of its original topical report describing best estimate LBLOCA methodology in 1996. NRC approval of the methodology is documented in the NRC safety evaluation report appended to topical report WCAP-12945-P-A (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 (References 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).

This request summarizes the application of the Westinghouse ASTRUM best estimate LBLOCA evaluation model to PBNP Units 1 and 2 for the LBLOCA accident analysis. Table 1 lists the major plant parameter assumptions used in the BELOCA analysis for Units 1 and 2, respectively.

FPL Energy Point Beach and its vendor, Westinghouse Electric Company LLC, continue to have ongoing processes which ensure that LOCA analysis input values conservatively bound current operating values and expected plant operating values at extended power uprate (EPU) conditions. The analyses presented in this application were performed modeling a homogeneous core of Westinghouse 14x14 Vantage + fuel and bounds operation at the current licensed power level.

The new analyses have been performed in support of the planned PBNP EPU to 1800 MWt; however, these analyses bound the current licensed power level of 1540 MWt. As such, the applicability limit is met. This application requests NRC approval of the revised methodology for the current licensed power level of 1540 MWt to provide continued compliance with the requirements of 10 CFR 50.46. The EPU license amendment will incorporate this application by reference for approval of the LBLOCA at the uprated power level.

The proposed licensing basis and associated Technical Specification change provide new LBLOCA analyses using the NRC-approved WCAP-16009-P-A ASTRUM methodology. With these changes, PBNP will continue to provide assurance of operation in accordance with applicable regulations.

3.0 TECHNICAL EVALUATION

3.1 Proposed TS 5.6.4 Changes

This application 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 5.6.4.b, analytical methods used to determine core operating limits must be listed in TS 5.6.4. This license amendment request proposes to add new analytical method (13) to reference WCAP-16009-P-A.

The user note in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants", Revision 3.1, 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 analytical method in the new Reference (13) meet the NUREG-1431 guidance, with the exception that in the PBNP Technical Specifications, TS 5.6.4 is the correct citation for this requirement.

The proposed change to relocate the complete citation for currently approved analytical methods listed in TS 5.6.4.b Items (1) through (12) to the COLR also meets the NUREG-1431 Revision 3.1 guidance.

3.2 Proposed Application of Westinghouse Best Estimate LBLOCA Methodology

3.2.1 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 (Reference 1). This methodology is known as the Code Quantification Document (CQD). This methodology was later extended to two-loop Westinghouse plants with UPI in 1999 as documented in the NRC safety evaluation report appended to the UPI topical report (Reference 2). PBNP is a two-loop Westinghouse plant with UPI.

More recently, Westinghouse developed an alternative uncertainty methodology called ASTRUM , which stands for Automated Statistical TReatment of Uncertainty Method (Reference 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 (Reference 3). The ASTRUM methodology remains applicable to three and four-loop pressurized water reactors (PWRs), as well as two-loop Westinghouse plants with UPI. This method was also extended to Combustion Engineering designed PWRs.

The ASTRUM methodology requires the execution of 124 transients to determine a bounding estimate of the 95th percentile of the peak clad temperature (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 specifically modeled 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 WCAP-16009-P-A (Reference 3).

The Best Estimate LBLOCA analysis and associated model for PBNP Units 1 and 2 are unit-specific, (i.e., a separate analysis was performed for each unit due to the differences in the steam generators). Since there were two separate analyses performed, a different set of uncertainty attributes was generated for each unit.

3.2.2 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.

3.3 ASTRUM Analysis Results for PBNP

3.3.1 Analysis Results

The results of the PBNP ASTRUM analyses are summarized in Table 2 and a sequence of events for Units 1 and 2 limiting cases are given in Tables 3 and 4, respectively.

The scatter plots presented in Figure 1 (Unit 1) and Figure 19 (Unit 2) show the effect of the effective break area on the analyses PCT. The effective break area is calculated by multiplying the discharge coefficient CD with the sample value of the break area, normalized to the cold leg cross sectional area. Figures 1 and 19 are provided to show the break area is a significant contributor to the variation in PCT.

From the 124 calculations performed as part of the Unit 1 ASTRUM analysis, the same case proved to be both the limiting PCT and limiting LMO transient. Figure 2 shows the predicted clad temperature transient at the PCT limiting elevation for the limiting Unit 1 case. For the 124 Unit 1 cases, the maximum CWO calculated remained well below the 10 CFR 50.46 requirement as described below.

From the 124 calculations performed as part of the Unit 2 ASTRUM analysis, the same case proved to be both the limiting PCT and limiting LMO transient. Figure 20 shows the predicted clad temperature transient at the PCT limiting elevation for the limiting Unit 2 case. For the 124 Unit 2 cases, the maximum CWO calculated remained well below the 10 CFR 50.46 requirement as described in Section 3.3.2 below.

Figures 3 through 16 illustrate the key major response parameters for the Unit 1 limiting PCT transient. Figures 21 through 34 illustrate the key major response parameters for the Unit 2 limiting PCT transient. Note that the limiting cases depicted by Figures 3 through 16 and 21 through 34 (and Table 2) reflect loss-of-offsite-power (LOOP) transient cases since LOOP cases were determined to be limiting (versus no-LOOP cases) by each unit's Confirmatory Study as documented by the vendor. The containment backpressure utilized for the LBLOCA analysis compared to the calculated containment backpressure is provided in Figure 17 for Unit 1 and Figure 35 for Unit 2. The worst single failure for the LBLOCA analysis is the loss of one train of ECCS injection (consistent with the ASTRUM Topical Report); however, all containment systems which would reduce containment pressure are modeled for the LBLOCA containment backpressure calculation.

The sampling range of power shape used is provided in Figure 18. This range was used for the analysis for each of the PBNP units.

3.3.2 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. These limits are complied with as demonstrated below:

(b)(1) The limiting PCT 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 1975°F for Unit 1 and 1810°F for Unit 2, the analyses confirm that 10 CFR 50.46 acceptance criterion (b)(1), i.e., "Peak Clad Temperature less than 2200°F," is demonstrated. The results for both units are shown in Table 2.

(b)(2) The maximum cladding oxidation corresponds to a bounding estimate of the 95th percentile LMO at the 95-percent confidence level. The limiting transient LMO is 2.61% for Unit 1 and 2.57% for Unit 2. Therefore, the analyses confirm that 10 CFR 50.46 acceptance criterion (b)(2), i.e., "Local Maximum Oxidation of the cladding less than 17 percent," is demonstrated. The results of the transient oxidation for both units are shown in Table 2.

(b)(3) The limiting CWO corresponds to a bounding estimate of the 95th percentile CWO at the 95-percent confidence level. The limiting Hot Assembly Rod (HAR) total maximum oxidation is 0.386% for Unit 1 and 0.154% for Unit 2. A detailed CWO calculation takes advantage of the core power census that includes many lower power assemblies. Because there is significant margin to the regulatory limit for each unit, the CWO value can be conservatively chosen as that calculated for the limiting HAR. A unit-specific and detailed CWO calculation is not needed because the outcome is always less than the limiting HAR. Therefore, the analyses confirm that 10 CFR 50.46 acceptance criterion (b)(3), i.e., "Core-Wide Oxidation less than 1 percent," is demonstrated. The results for both units are shown in Table 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. It has been demonstrated that for each unit, the PCT and maximum cladding oxidation limits remain in effect for Best-Estimate LOCA applications. The actions, automatic or manual, that are currently in place at these plants to maintain that fuel deformation due to combined LOCA and seismic loads remain unchanged with the application of the ASTRUM methodology (Reference 3).

(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 these plants to maintain long-term cooling remain unchanged with the application of the ASTRUM methodology (Reference 3).

Based on the ASTRUM Analyses results (Table 2), it is concluded that PBNP Units 1 and 2 continue to maintain a margin of safety to the limits prescribed by 10 CFR 50.46.

3.4 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. PBNP Units 1 and 2 are Westinghouse two-loop plants with UPI.

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

This license amendment request proposes to incorporate the best estimate LBLOCA analyses using ASTRUM in the PBNP licensing basis by including WCAP-16009-P-A in the list of NRC approved methods for establishing core operating limits as TS 5.6.4.b Item (13).

This request also proposes to implement TSTF Traveler-363A, which relocates the complete citations for analytical methods listed in TS 5.6.4.b to the Core Operating Limits Report (COLR). This application was approved by the NRC staff on July 9, 2002, for Oconee Nuclear Station, Units 1, 2 and 3 (ML021900580) and has been incorporated into NUREG-1431, Standard Technical Specifications for Westinghouse Plants, Revision 3.1.

The proposed changes will continue to protect the health and safety of the public. Operation and maintenance of the PBNP with the proposed licensing basis changes will continue to comply with all applicable rules and regulations. Implementation of TSTF-363A is an administrative change and has no impact upon the health and safety of the public.

4.0 REGULATORY EVALUATION

4.1 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 LBLOCA 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 PWRs with UPI. WCAP-16009-P-A is applicable to PBNP Units 1 and 2 and the plant-specific application of the ASTRUM methodology to the PBNP, Units 1 and 2, LBLOCA analyses have been performed in accordance with the conditions and limitations of the topical report and the associated NRC Safety Evaluation for WCAP-16009-P-A.

The licensing basis changes proposed in this license amendment request meet the requirements of 10 CFR 50.46 and provide the basis for continued safe plant operation. Relocation of the complete citations for core operating limits analytical methods from the Technical Specifications to the Core Operating Limits Report is an administrative change that has no effect upon the continued safe operation of PBNP.

NUREG-1431, Standard Technical Specifications, Westinghouse Plants,

NUREG-1431, "Standard Technical Specifications, Westinghouse Plants," (NUREG-1431) provides guidance for Technical Specifications for plants with Westinghouse Nuclear Steam Supply Systems and has been approved for use by the NRC. This license amendment request proposes to incorporate new LBLOCA analyses using the ASTRUM methodology in the NRC reviewed and approved WCAP-16009-P-A. In accordance with TS 5.6.4.b, analytical methods used to determine core operating limits must be listed in TS 5.6.4. This license amendment request proposes to add new Item (13) 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 of NUREG-1431 is met as discussed above and the plant Technical Specifications will continue to provide the basis for safe plant operation even though the applicable PBNP TS section in which the COLR NRC-approved analytical methods are placed is TS 5.6.4, rather than TS 5.6.5, as defined in NUREG-1431, Revision 3.

4.2 Precedents

The NRC has approved the use of the ASTRUM methodology for a number of plants including the following plants:

- Joseph M. Farley, Units 1 and 2 (ML0611810338)
- Indian Point Nuclear Generating Station, Unit 2 (ML06170291)
- Comanche Peak Steam Electric Generating Station (ML080500627)
- Prairie Island Nuclear Generating Plant, Units 1 and 2 (ML071230789)
- R.E. Ginna Nuclear Power Plant (ML061180353)

The license amendment request for PBNP differs somewhat from the above listed NRC-approved amendments in that while the new best estimate LBLOCA analyses were performed for a higher thermal power level of 1810 MWt, the analyses bound the current licensed power level of 1540 MWt. Approval to use the ASTRUM methodology is being requested for continued operation at 1540 MWt via incorporation of WCAP-16009-P-A as Item (13) into TS 5.6.4.b. Implementation of the proposed amendments will take place within 60 days of NRC approval of this license amendment request.

The submitted analyses are linked to the planned EPU application. However, this application is independent of that effort although the supporting analysis have been performed for a power level of 1810 MWt intended for the future EPU submittal. The EPU application will request NRC staff approval at the uprated power level via reference to this license amendment request.

Finally, this request proposes to implement TSTF Traveler 363A. The application of TSTF-363A was approved by the NRC staff on July 9, 2002, for Oconee Nuclear Station, Units 1, 2 and 3 (ML021900580). TSTF-363A has been incorporated into NUREG-1431, Standard Technical Specifications for Westinghouse Plants, Revision 3.1.

4.3 Significant Hazards Consideration

FPL Energy Point Beach, LLC has evaluated whether significant hazards are 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 application proposes to incorporate LBLOCA 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 PBNP licensing basis, add reference to WCAP-16009-P-A in the Technical Specification 5.6.4.b list of approved methodologies for establishing core operating limits, and relocate topical report detailed reference citations from TS 5.6.4.b to the COLR.

   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 (UPI). Since the PBNP Units 1 and 2 are two-loop Westinghouse plants with UPI 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 TS 5.6.4.b and relocation of topical report detailed citations to the COLR are administrative changes that do 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 LBLOCA 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 PBNP licensing basis, add a reference to WCAP-16009-P-A in the Technical Specification list of approved methodologies for establishing core operating limits, and relocate topical report detailed reference citations from TS 5.6.4.b to the COLR in accordance with approved TSTF-363A.

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 LBLOCA 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 LBLOCA 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. Relocation of topical report detailed citations from the Technical Specifications to the core operating limits report in accordance with approved TSTF-363A 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 application proposes to incorporate LBLOCA 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 PBNP licensing basis, add a reference to WCAP-16009-P-A in the Technical Specifications list of approved methodologies for establishing core operating limits, and relocate topical report detailed reference citations from Technical Specification 5.6.4.b to the COLR.

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 LBLOCAs 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 UPI. Since the PBNP is a two-loop Westinghouse plant with UPI 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. The analysis results using this methodology improve the margin of safety of PBNP.

Addition of the reference to WCAP-16009-P-A in the Technical Specifications and implementation of TSTF-363A are administrative changes that do not involve a significant reduction in a margin of safety.

4.4 Conclusions

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

5.0 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.

6.0 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.

7.0 TABLES AND FIGURES

Table 1

Major Plant Parameter Assumptions Used in the LOCA Analysis

Parameter	Value
Plant Physical Description	
• SG Tube Plugging	$\leq$10%
Plant Initial Operating Conditions	
• Reactor Power	$\leq$100% of 1811 MWt; 100.6% of 1800 MWt
• Peaking Factors	$F_Q \leq 2.6$ $F_{\Delta H} \leq 1.68$
• Axial Power Distribution	Bounded; See Figure 18
Fluid Conditions	
• T_{avg}	558.0 - 6.4°F $\leq T_{avg} \leq$ 577.0 + 6.4°F
• Pressurizer Pressure	2250 - 50 psia $\leq P_{RCS} \leq$ 2250 + 50 psia
• Reactor Coolant Flow	$\geq$89,000 gpm/loop
• Accumulator Temperature	60°F $\leq T_{ACC} \leq$ 120°F
• Accumulator Pressure	689.7 psia $\leq P_{ACC} \leq$ 839.7 psia
• Accumulator Water Volume	1068 $ft^3 \leq V_{ACC} \leq$ 1168 ft^3
• Accumulator Boron Concentration	$\geq$2600 ppm
Accident Boundary Conditions	
• Single Failure Assumptions	Loss of one ECCS train
• Safety Injection (SI) Flow	Minimum
• Safety Injection Temperature	32°F $\leq T_{SI} \leq$ 120°F
• High Head Safety Injection Initiation Delay Time	$\leq$13 sec (with offsite power) $\leq$28 sec (without offsite power)
• Low Head Safety Injection Initiation Delay Time	$\leq$23.7 sec (with offsite power) $\leq$37 sec (without offsite power)
• Containment Pressure	Bounded (minimum); see Figures 17 and 35
• Steam Generator Modeled	Unit 1: Model 44F / Unit 2: Model Delta 47

Table 2

PBNP Best Estimate LBLOCA ASTRUM Results*

10 CFR 50.46 Requirement	Unit 1 Value	Unit 2 Value	Criteria
95/95 PCT (°F)	1975	1810	<2200
95/95 LMO[1] (%)	2.61	2.57	<17
95/95 CWO[2] (%)	0.386	0.154	<1
Coolable Geometry	Criteria Acceptance Unaffected by Implementation of ASTRUM Methodology		
Long-Term Cooling	Criteria Acceptance Unaffected by Implementation of ASTRUM Methodology		

[1] Local Maximum Oxidation
[2] Core Wide Oxidation

* Note: PCTs, LMOs, and CWOs for the two units were generated with a different set of ASTRUM uncertainty attributes.

Table 3

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	4.25
Accumulator Injection Begins	9.0
High Head Safety Injection Begins	27.25
End of Blowdown	31.5
Low Head Safety Injection Begins	41.25
Bottom of Core Recovery	43.5
Accumulator Empty	~45
HSPCT Occurs	47.5
Time of Quench	~235
End of Analysis	500.0

Table 4

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	4.25
Accumulator Injection Begins	12.0
High Head Safety Injection Begins	27.25
End of Blowdown	30.0
Low Head Safety Injection Begins	40.0
Bottom of Core Recovery	41.25
Accumulator Empty	~44
HSPCT Occurs	80.0
Time of Quench	~150
End of Analysis	500.0

Figure 1

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

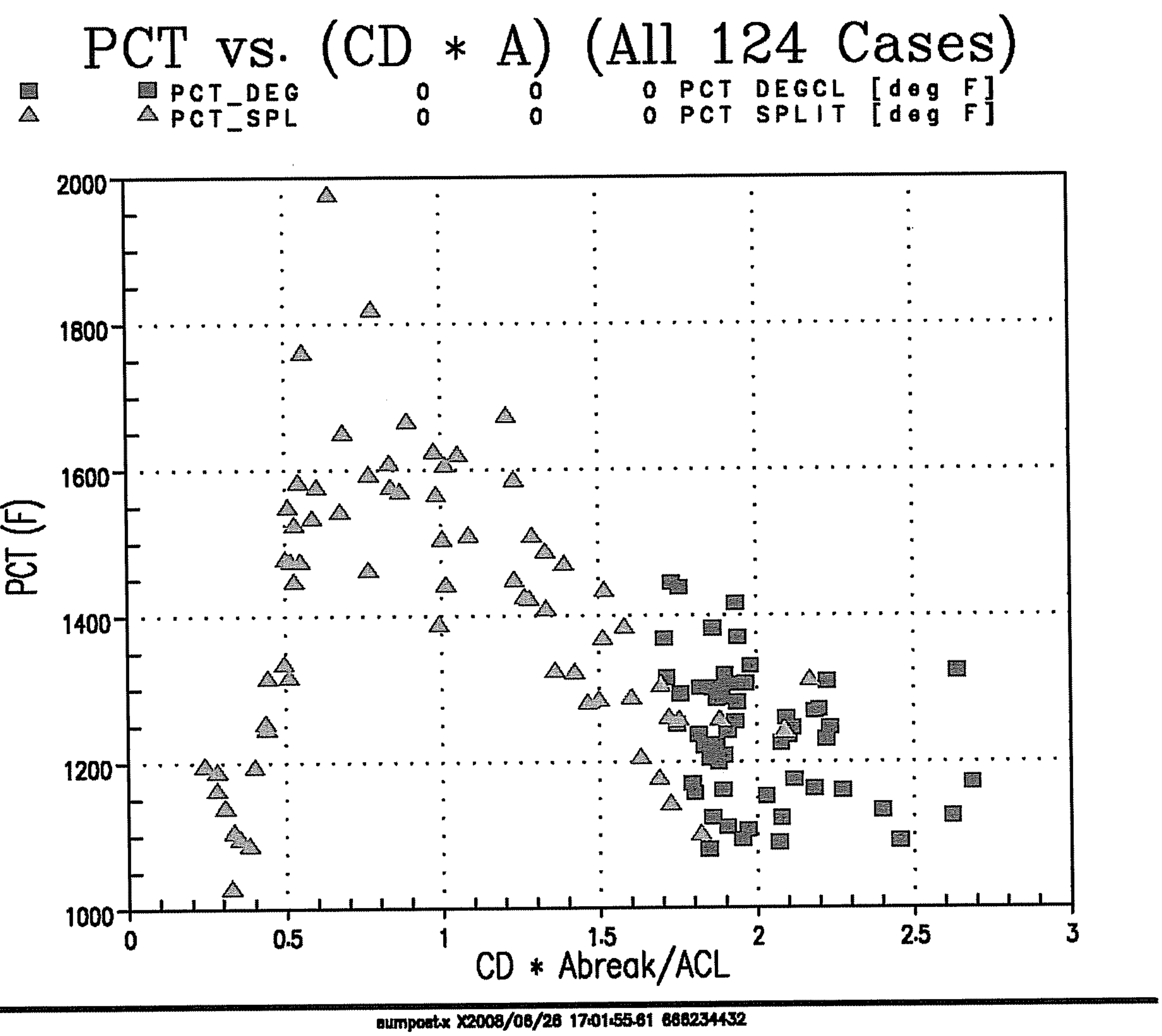

Figure 2

Unit 1 HOTSPOT Clad Temperature transient at the Limiting Elevation for the Limiting PCT Case

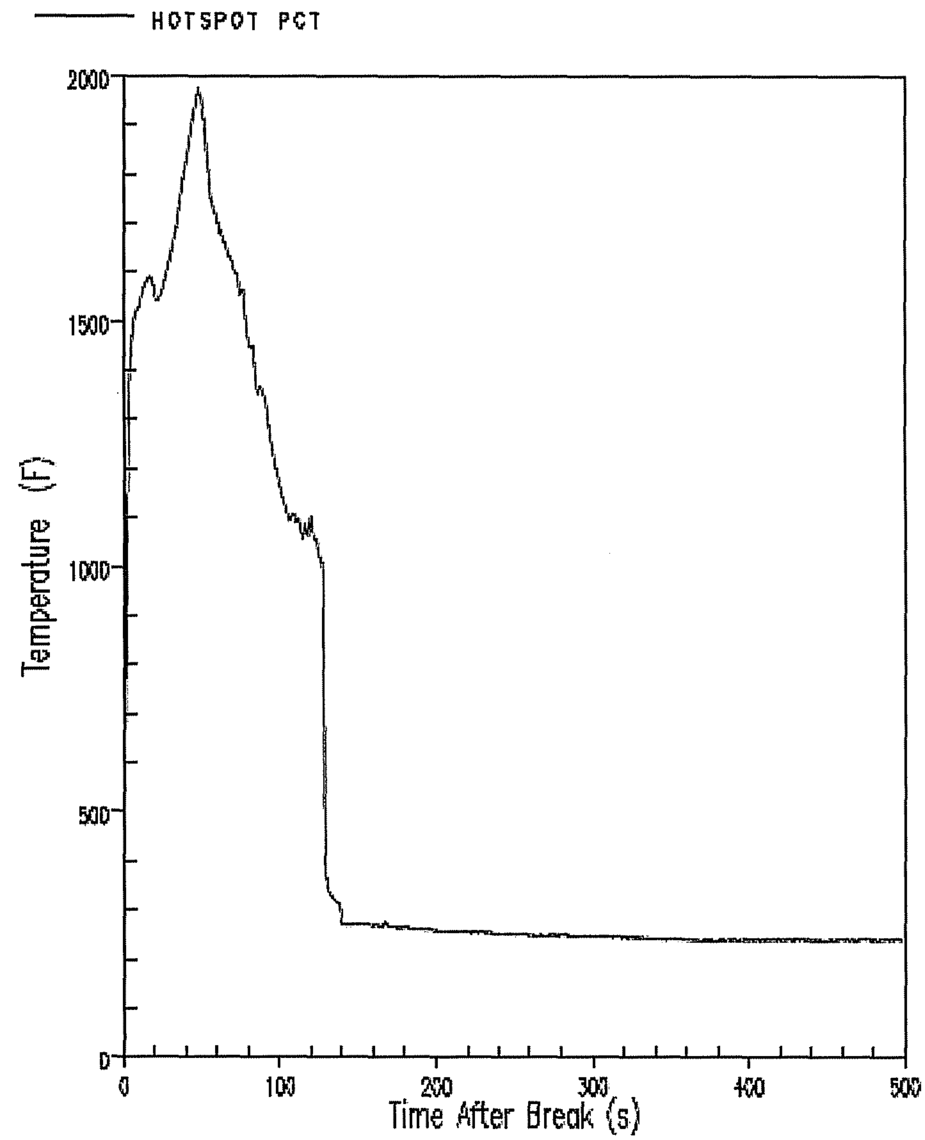

Figure 3

Unit 1 Pressurizer Pressure for the Limiting PCT Case

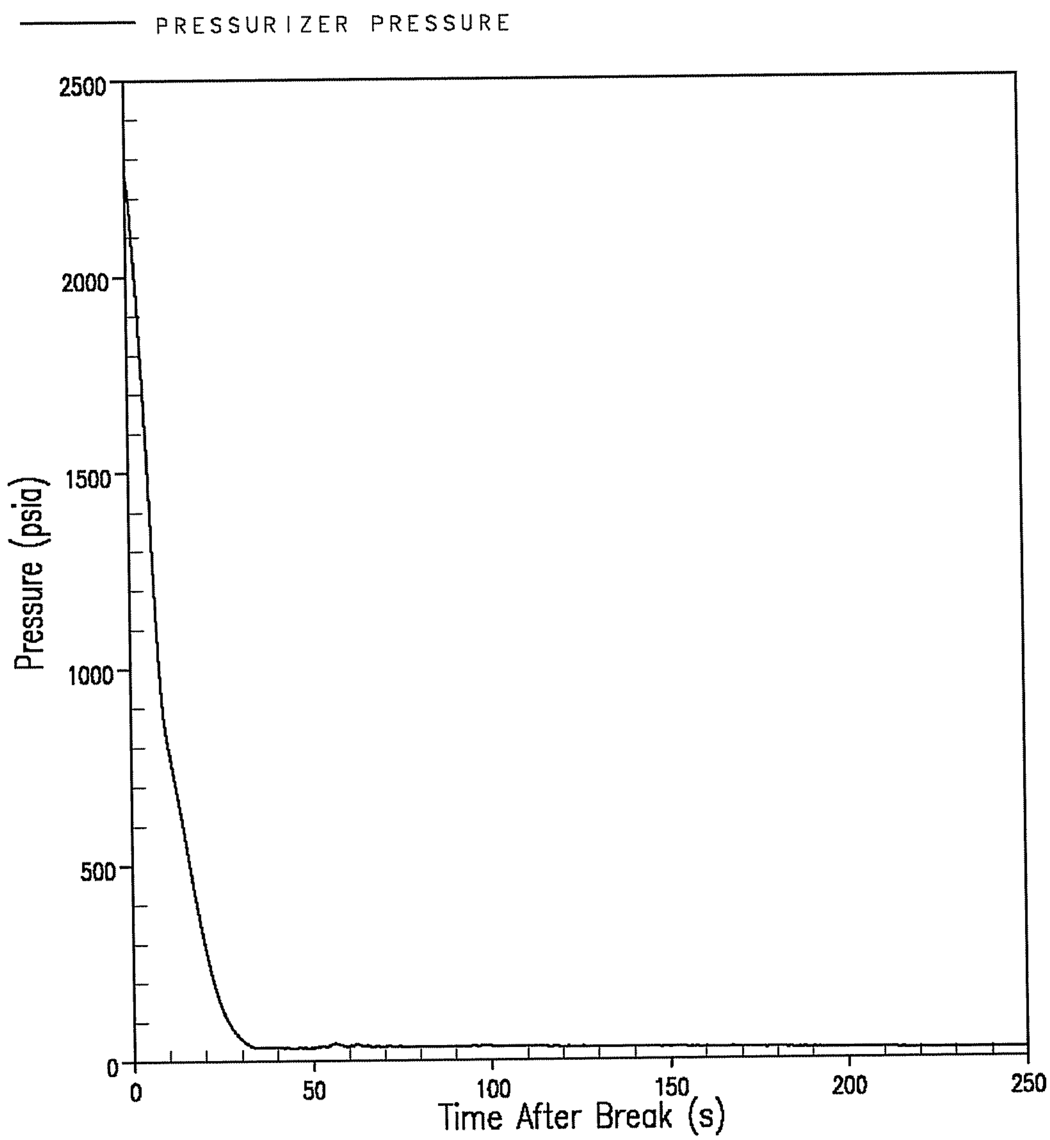

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Figure 4

Unit 1 Vessel Side Break Flow for the Limiting PCT Case

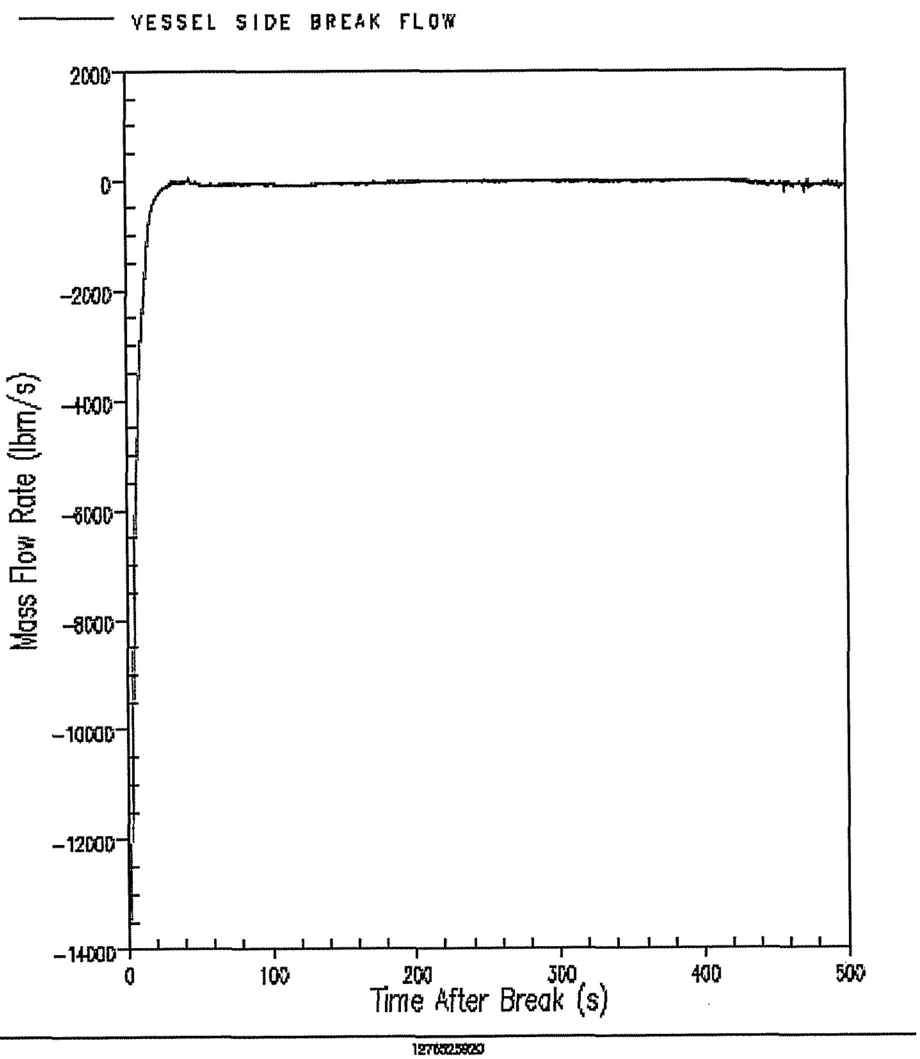

Figure 5

Unit 1 Pump Side Break Flow for the Limiting PCT Case

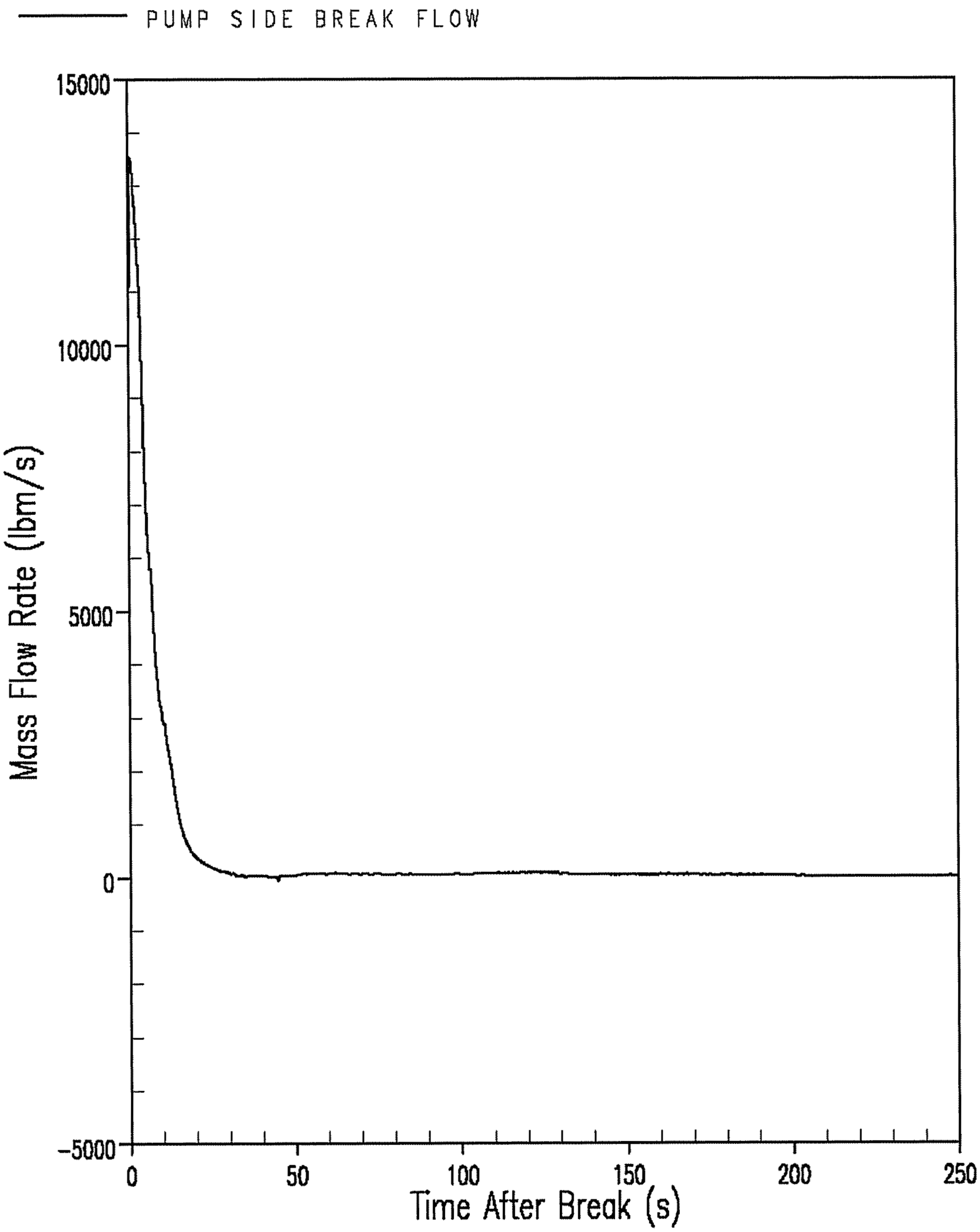

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Figure 6

Unit 1 Void Fraction in Pumps for the Limiting PCT Case

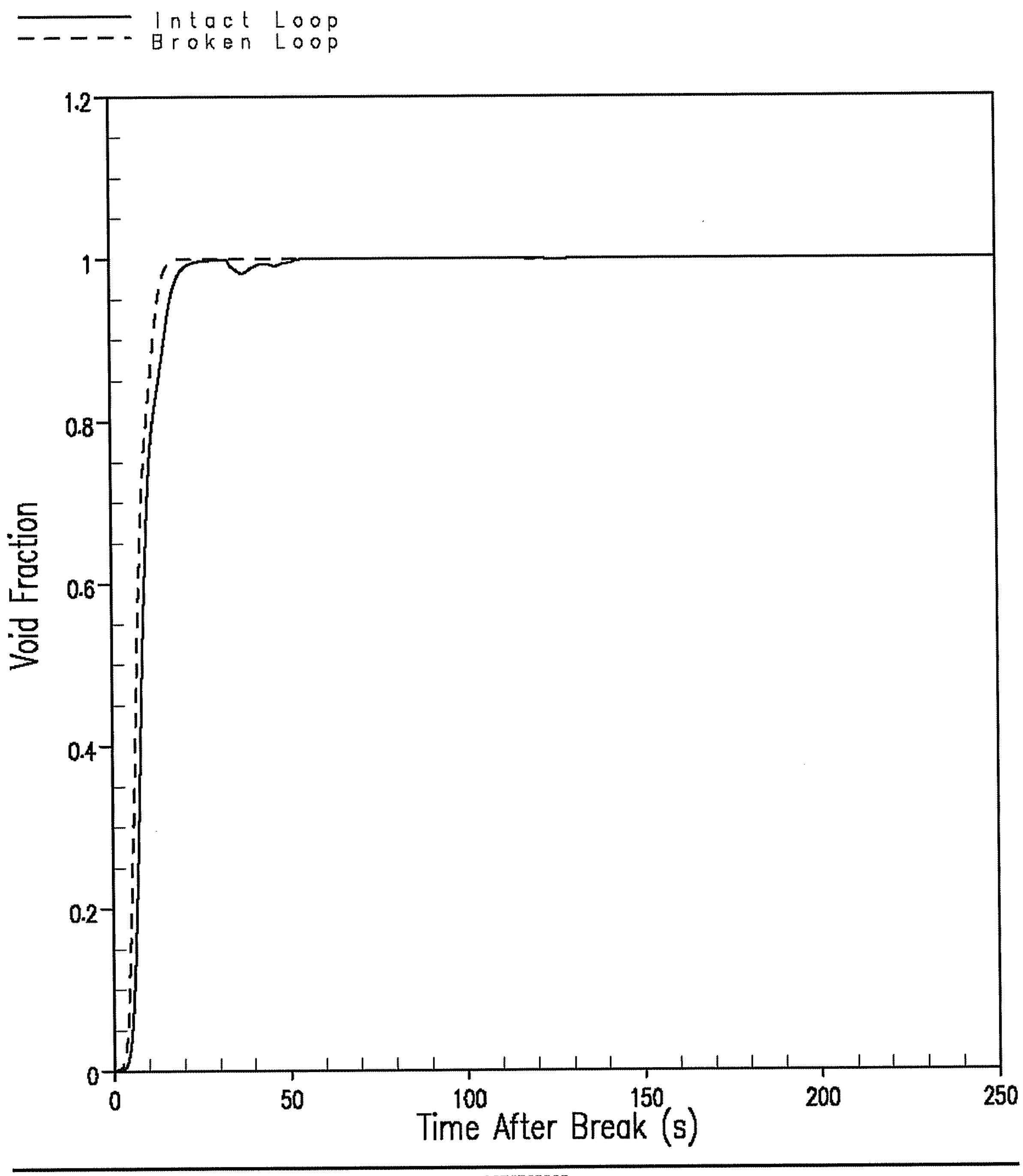

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Figure 7

Unit 1 Vapor Flow for Hot Assembly Near Top of Core for the Limiting PCT Case

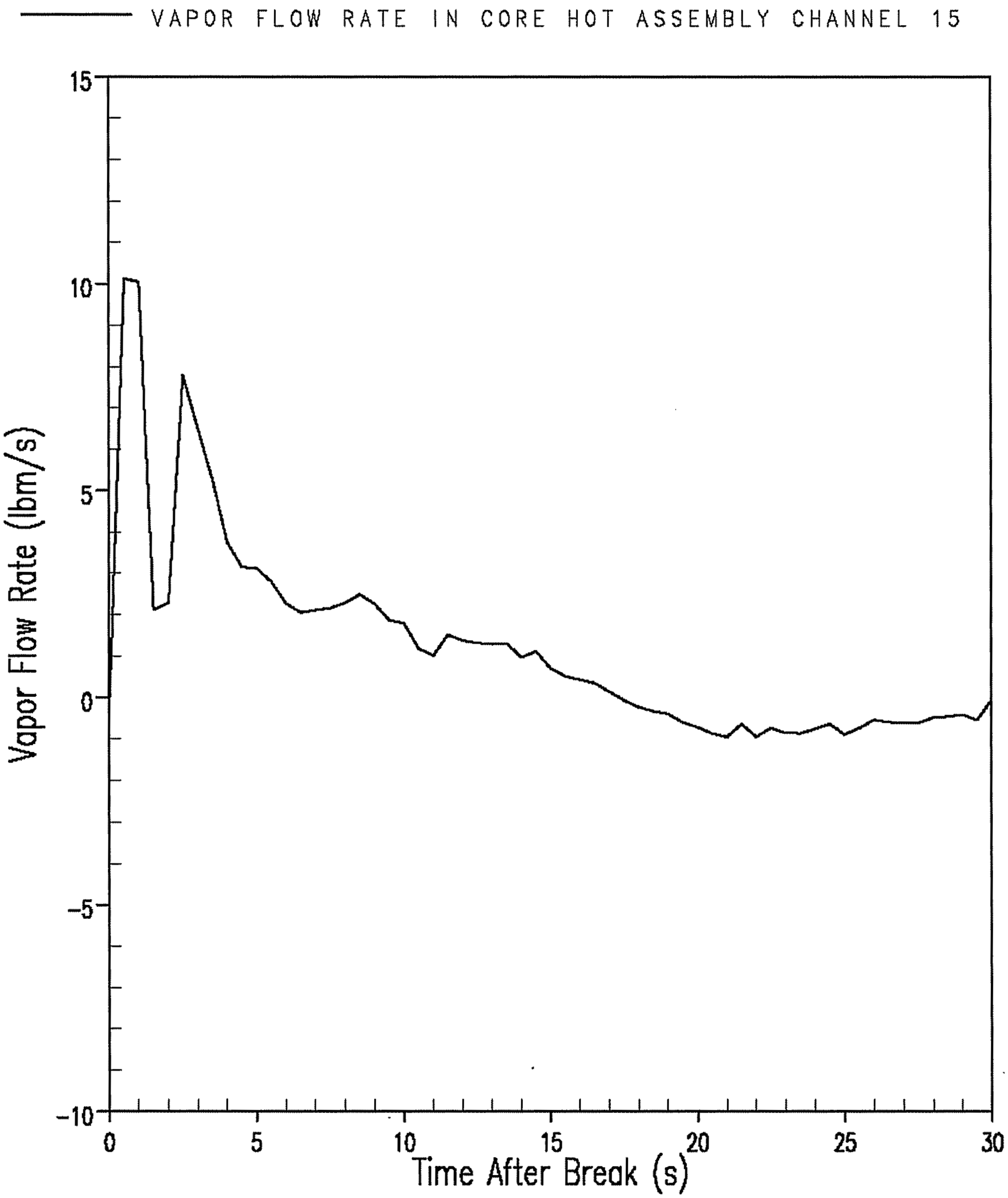

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Figure 8

Unit 1 Accumulator Injection flow for the Limiting PCT Case

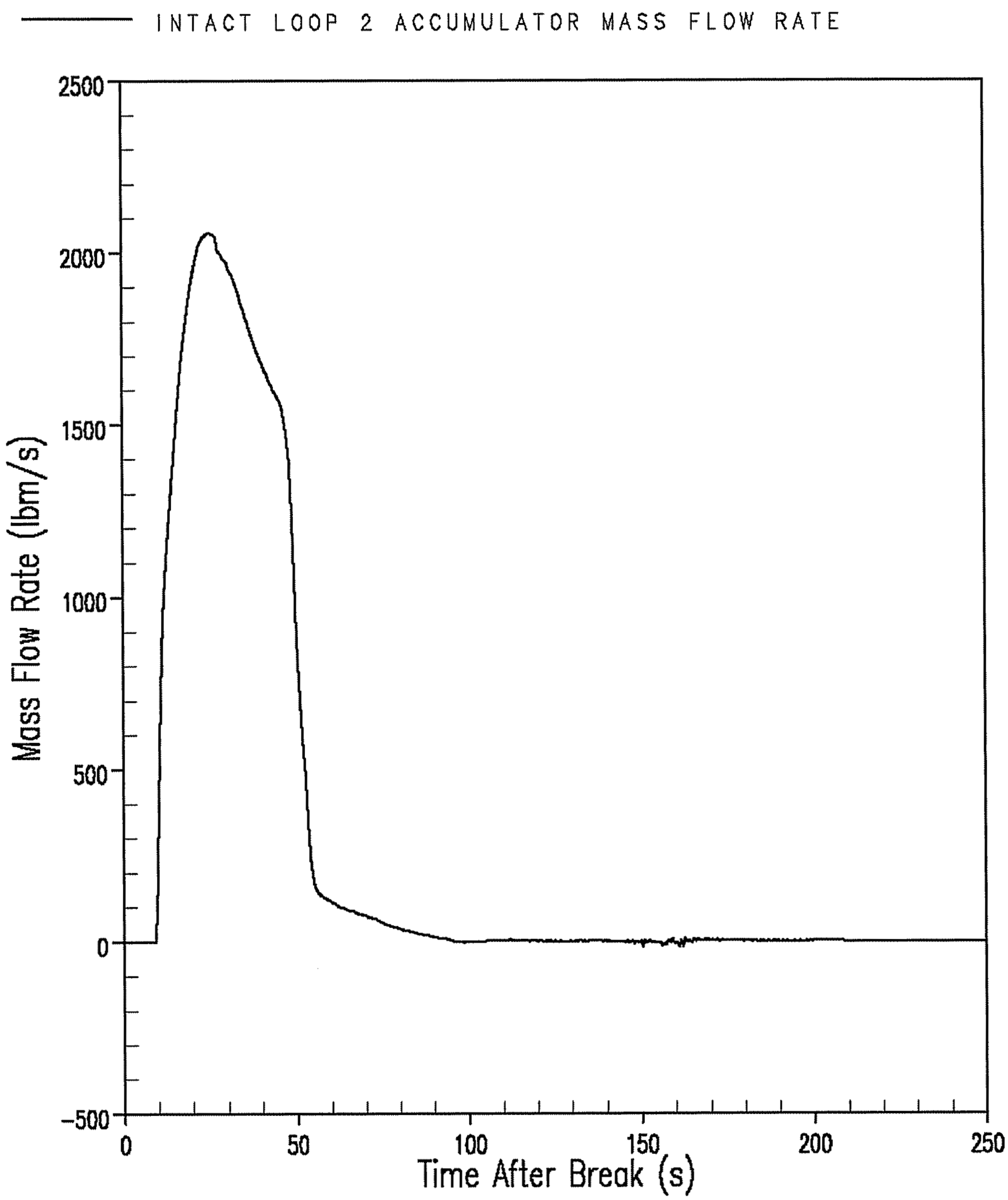

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Figure 9

Unit 1 High Head Safety Injection Flow for the Limiting PCT Case

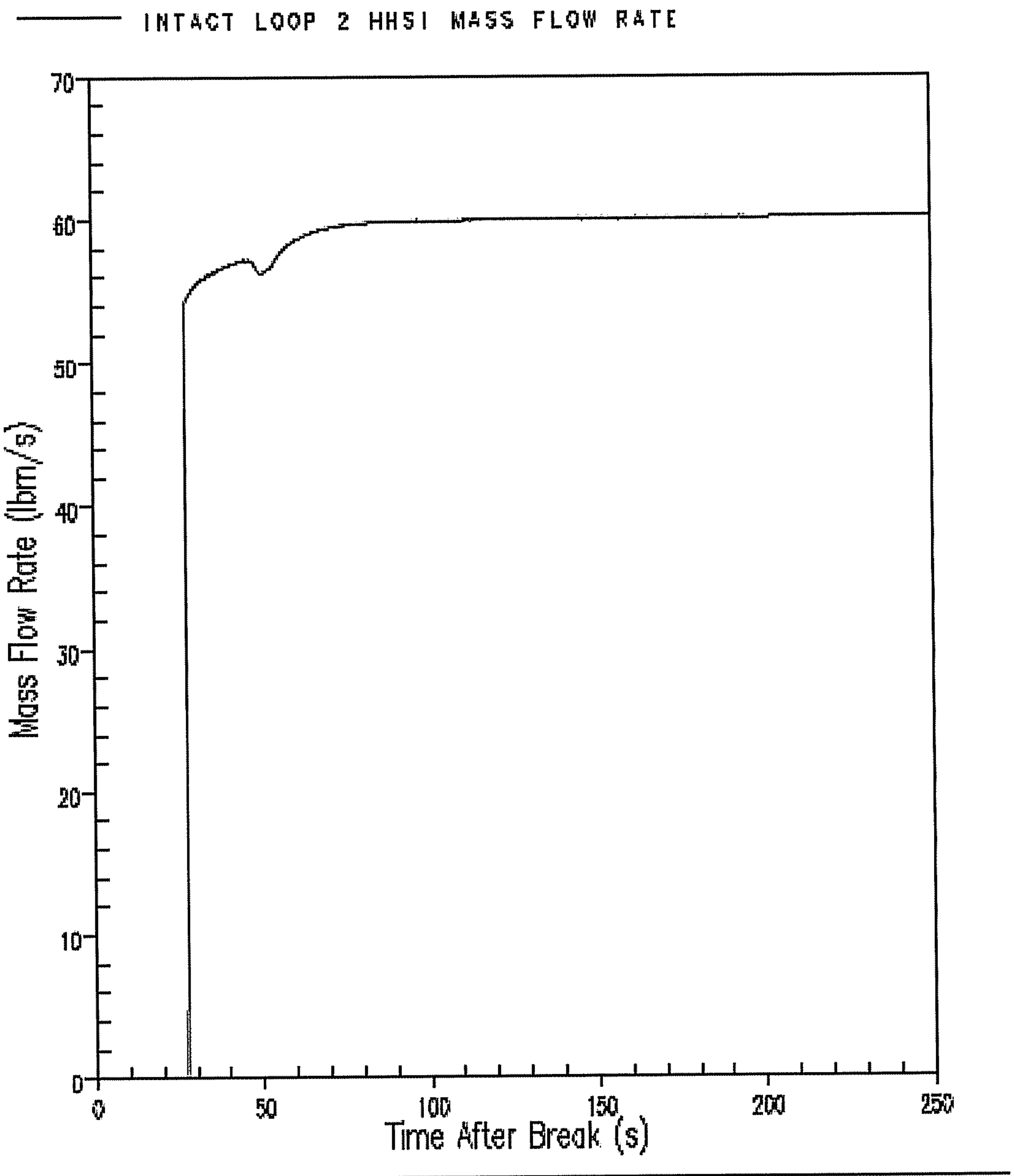


Figure 10

Unit 1 Low Head Safety Injection Flow for the Limiting PCT Case

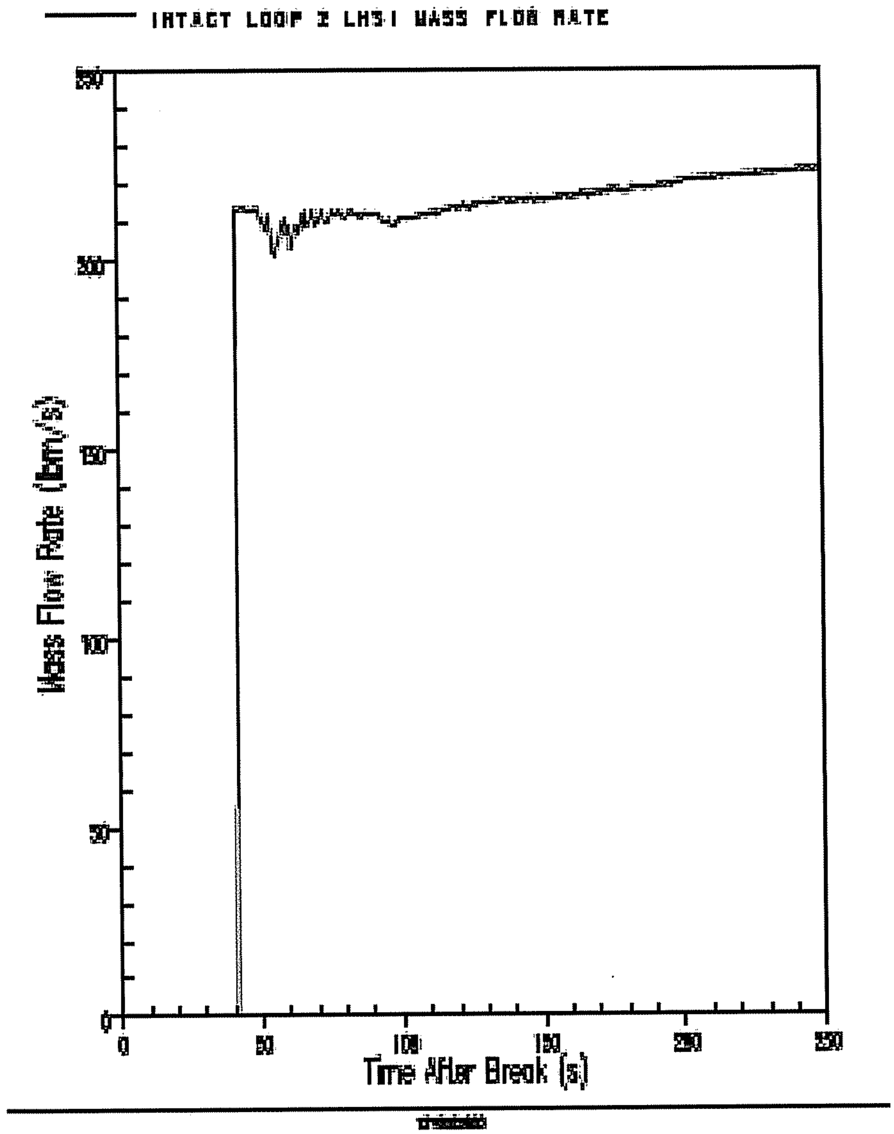


Figure 11

Unit 1 Lower Plenum Collapsed Liquid Level for the Limiting PCT Case

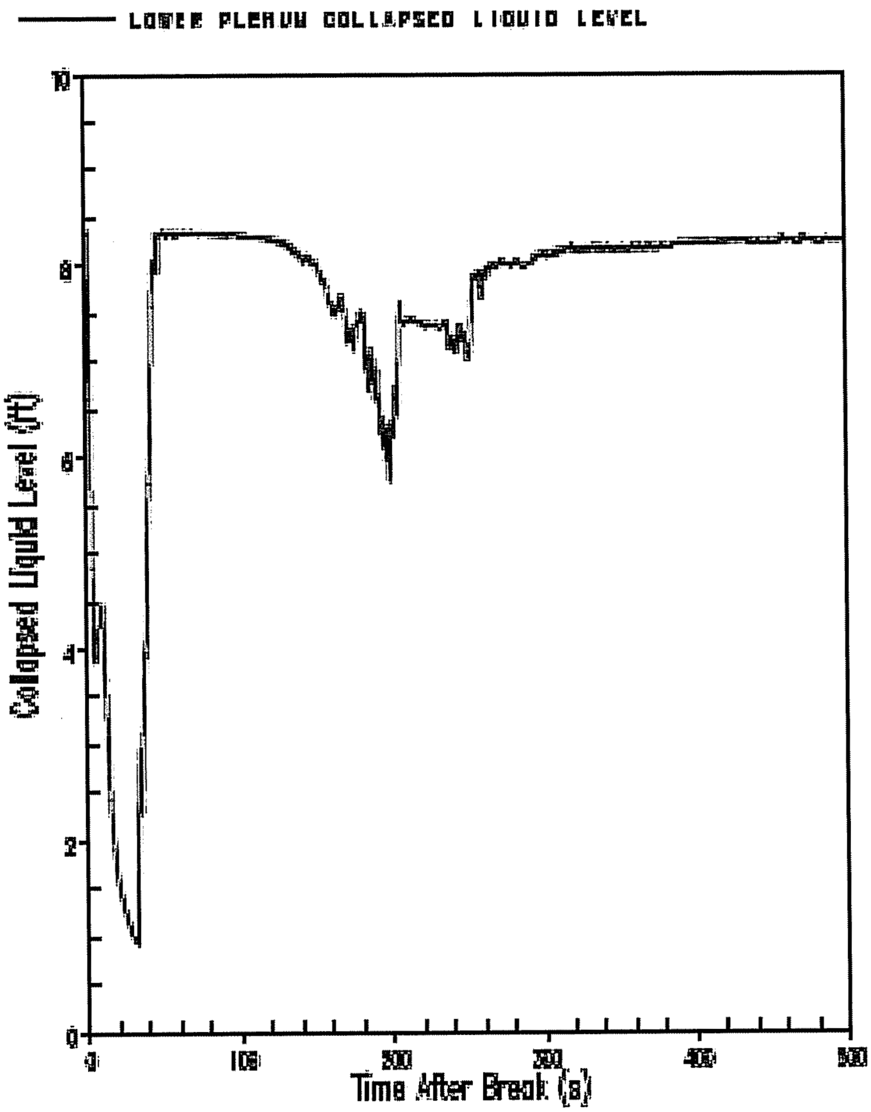


Figure 12

Unit1 Downcomer Collapsed Liquid Level for the Limiting PCT Case

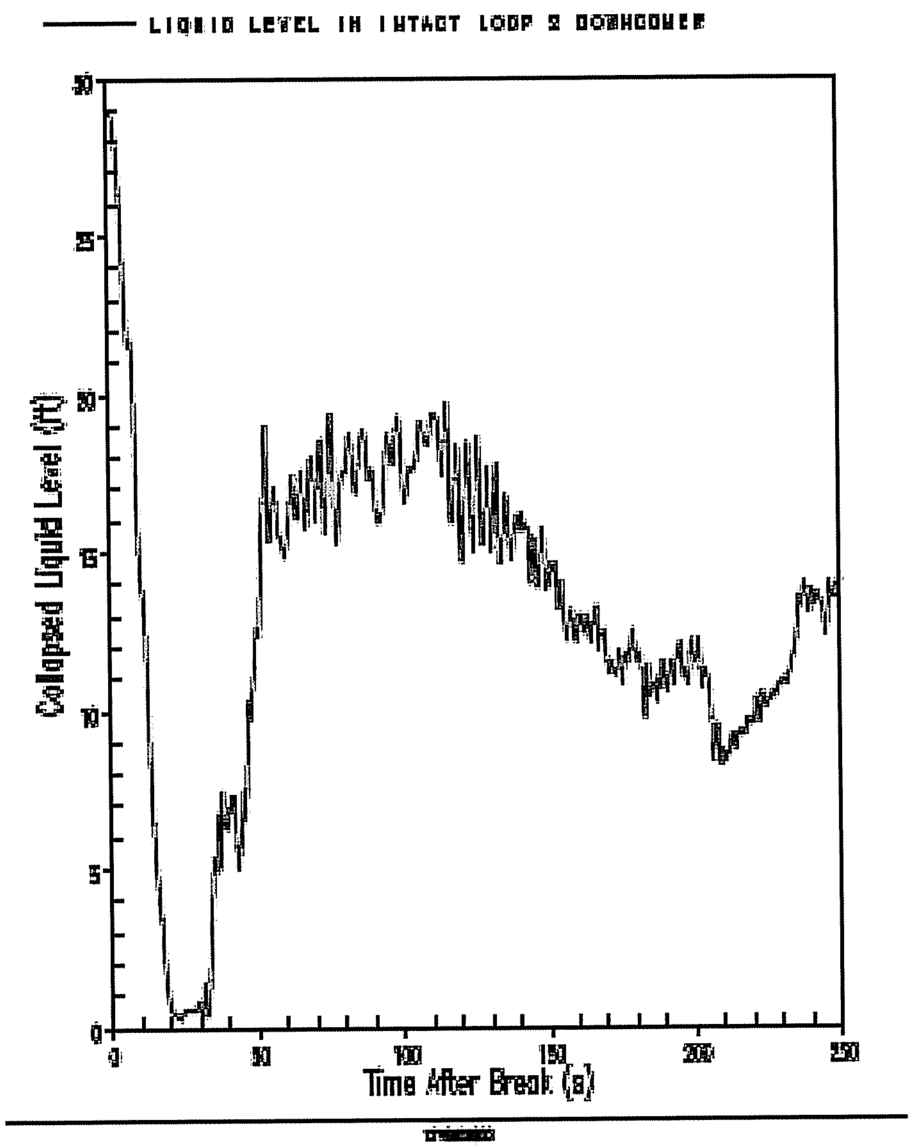


Figure 13

Unit 1 Core Average Channel Collapsed Liquid Level
For the Limiting PCT Case

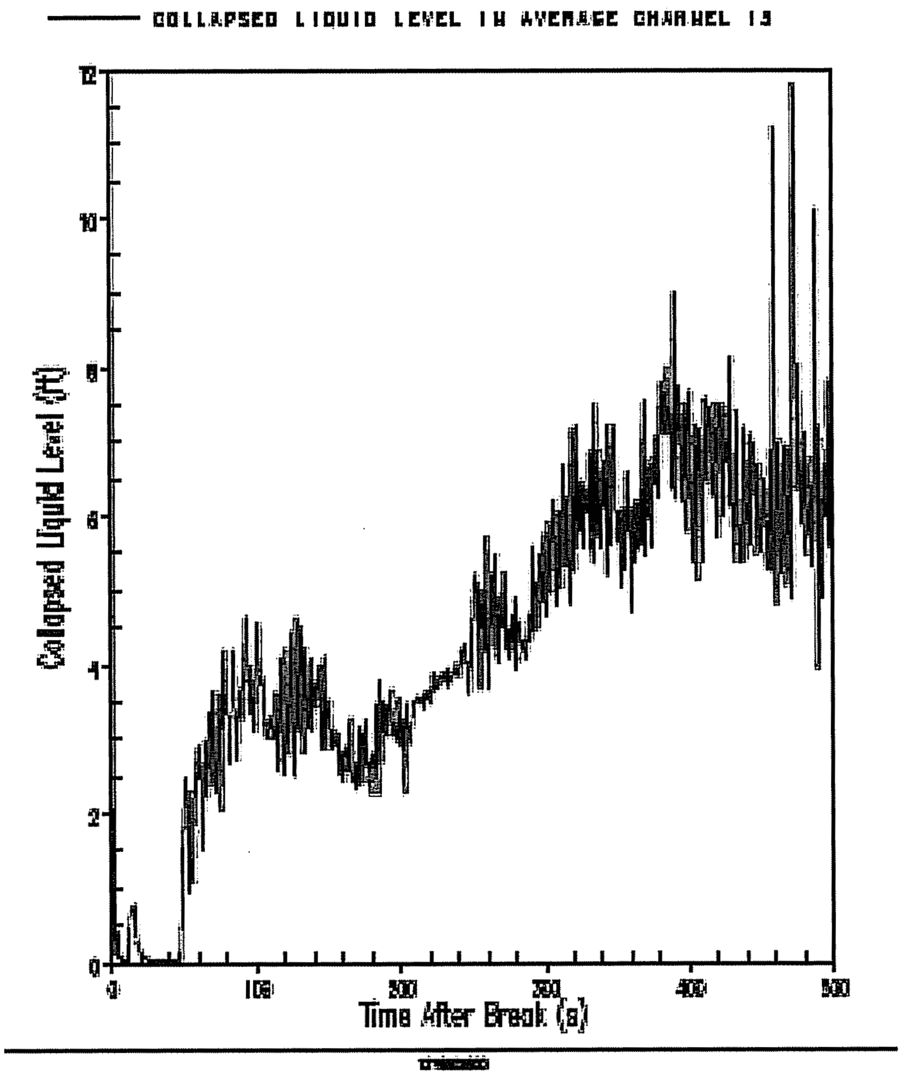


Figure 14

Unit 1 Vessel Liquid Mass for the Limiting PCT Case

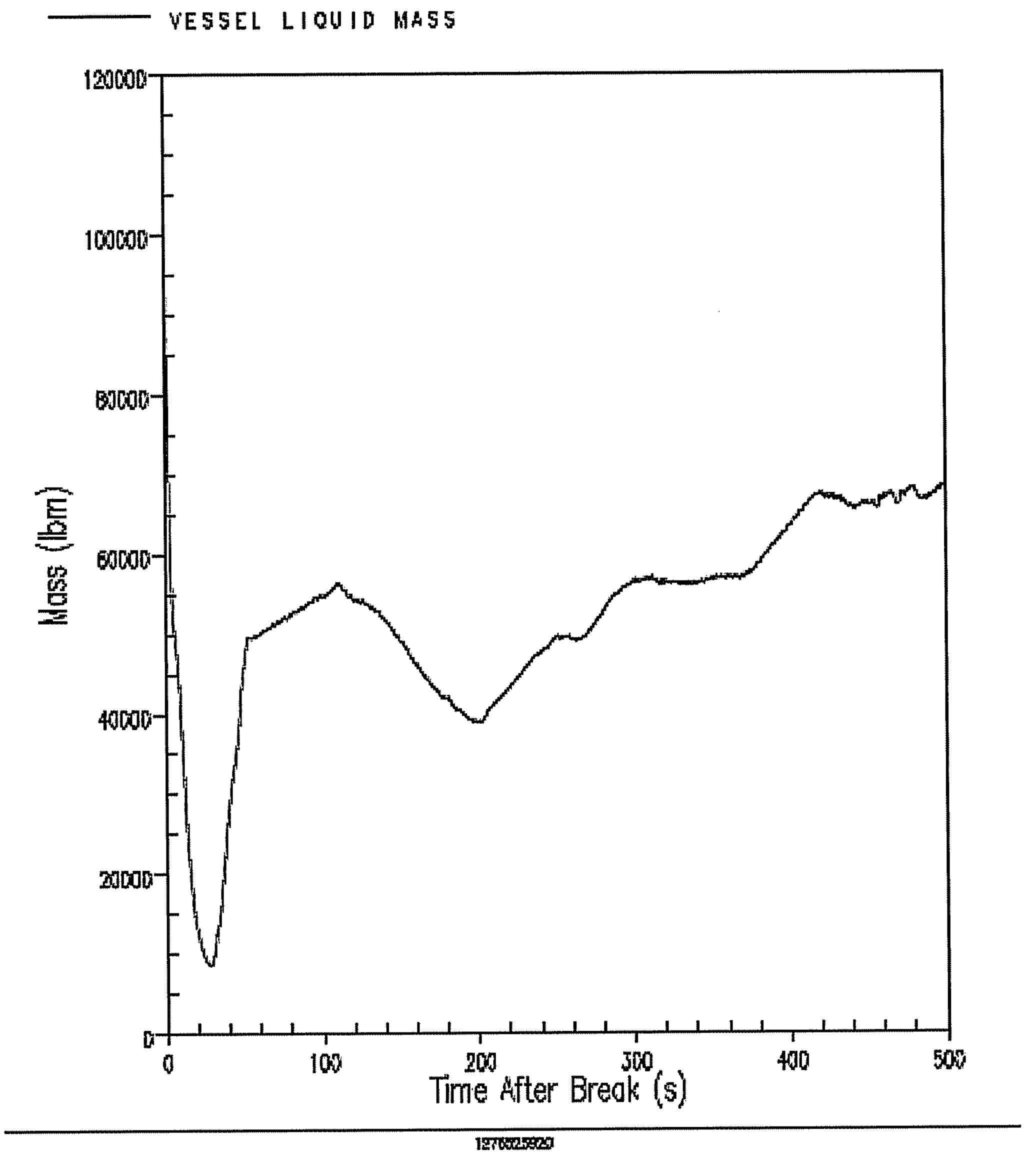


Figure 15

Unit 1 WCOBRA/TRAC Peak Clad Temperature for All Five Rod Groups for the Limiting PCT Case

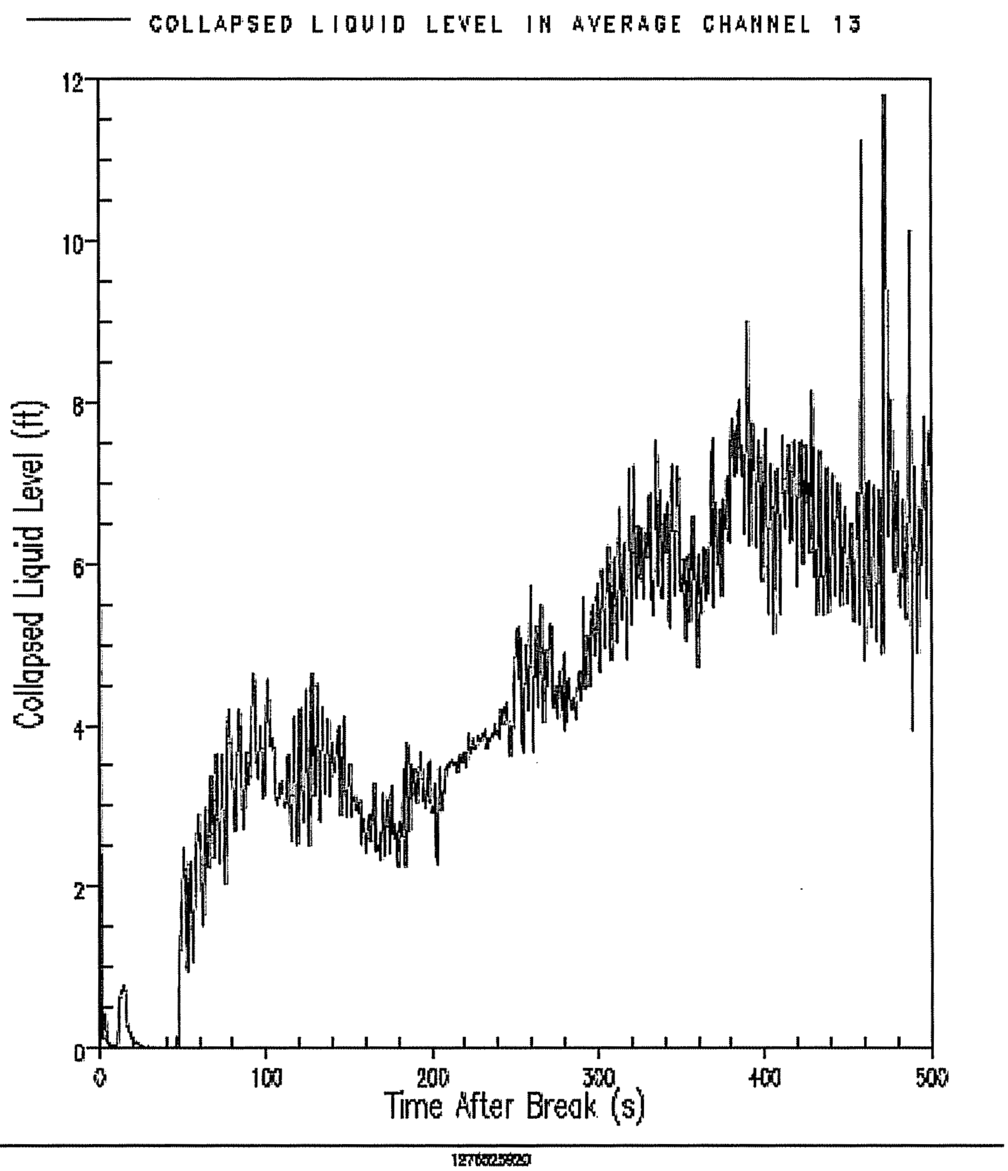


Figure 16

Unit 1 Peak Clad Temperature Elevation for the Hot Rod
For the Limiting PCT Case

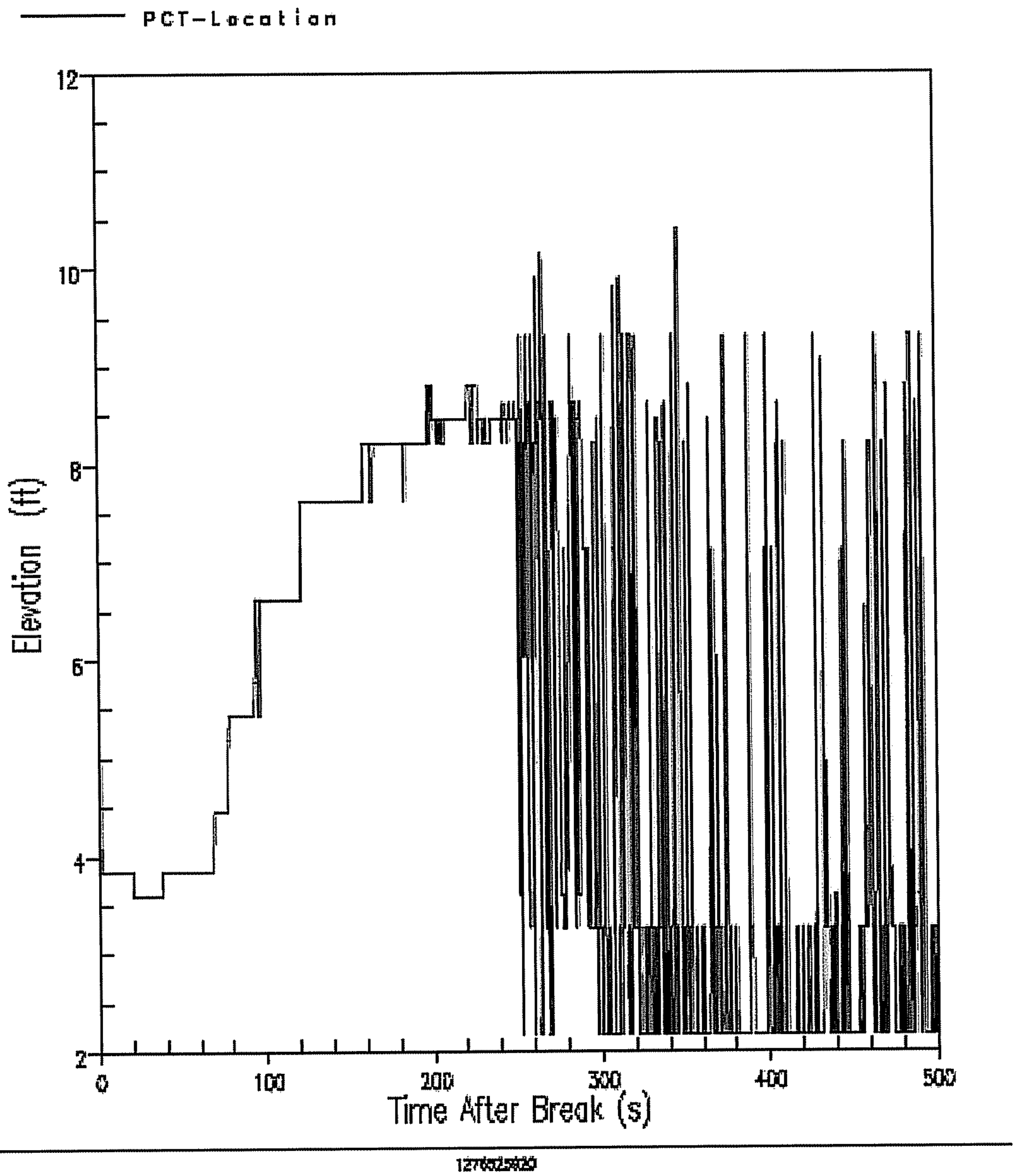


Figure 17

Unit 1 Analysis Versus Calculated Containment Backpressure

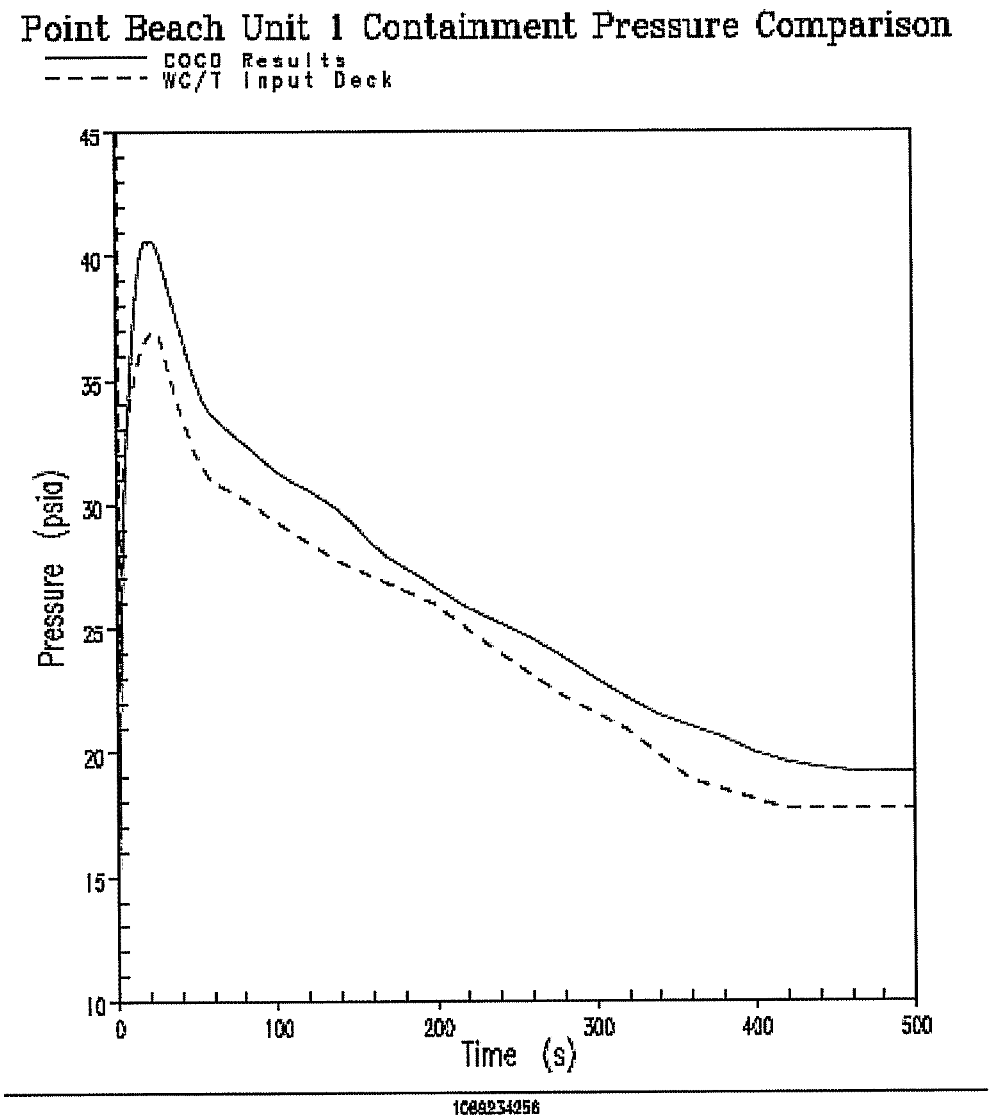
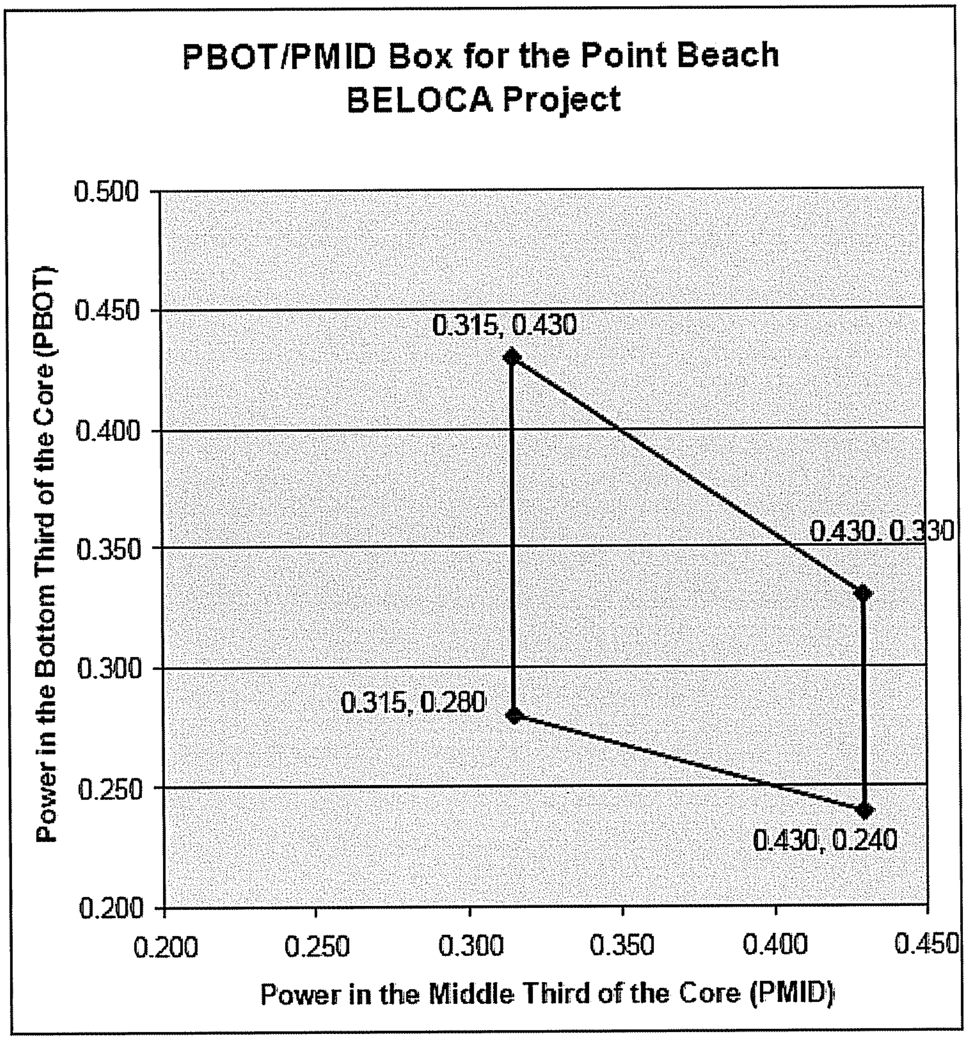


Figure 18

PBNP BELOCA Analysis
Axial Power Shape Operating Space Envelope for Units 1 and 2*



*Specific axial power distribution for each of 124 cases generated automatically (WCAP-16009-P-A, Table 1-10)

Figure 19

Unit 2 HOTSPOT PCT Versus Effective Break Area Scatter Plot
(CD = Discharge Coefficient, Abreak = Break Area, ACL = Cold Leg Area)

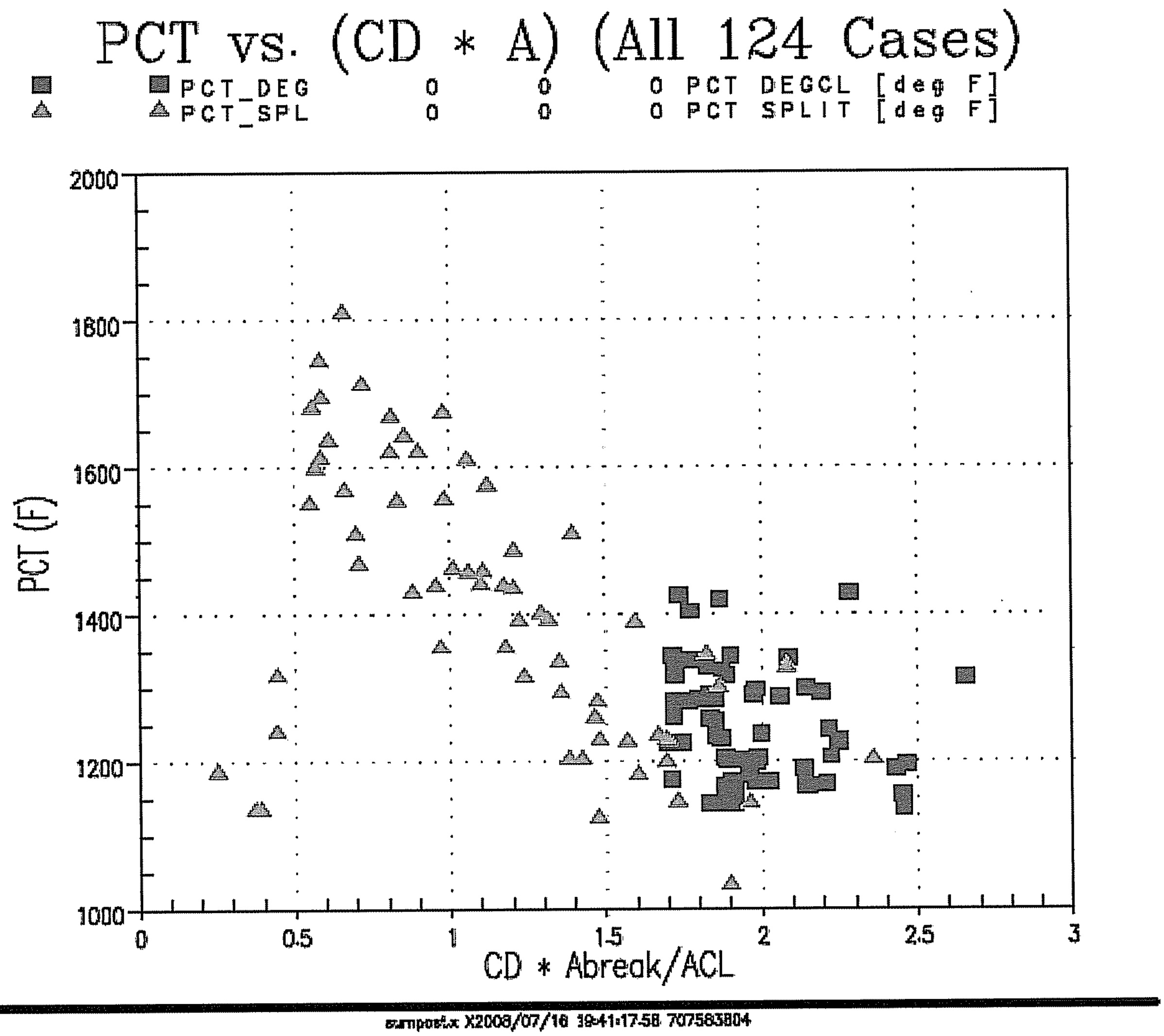


Figure 20

Unit 2 HOTSPOT Clad Temperature Transient at the Limiting Elevation for the Limiting PCT Case

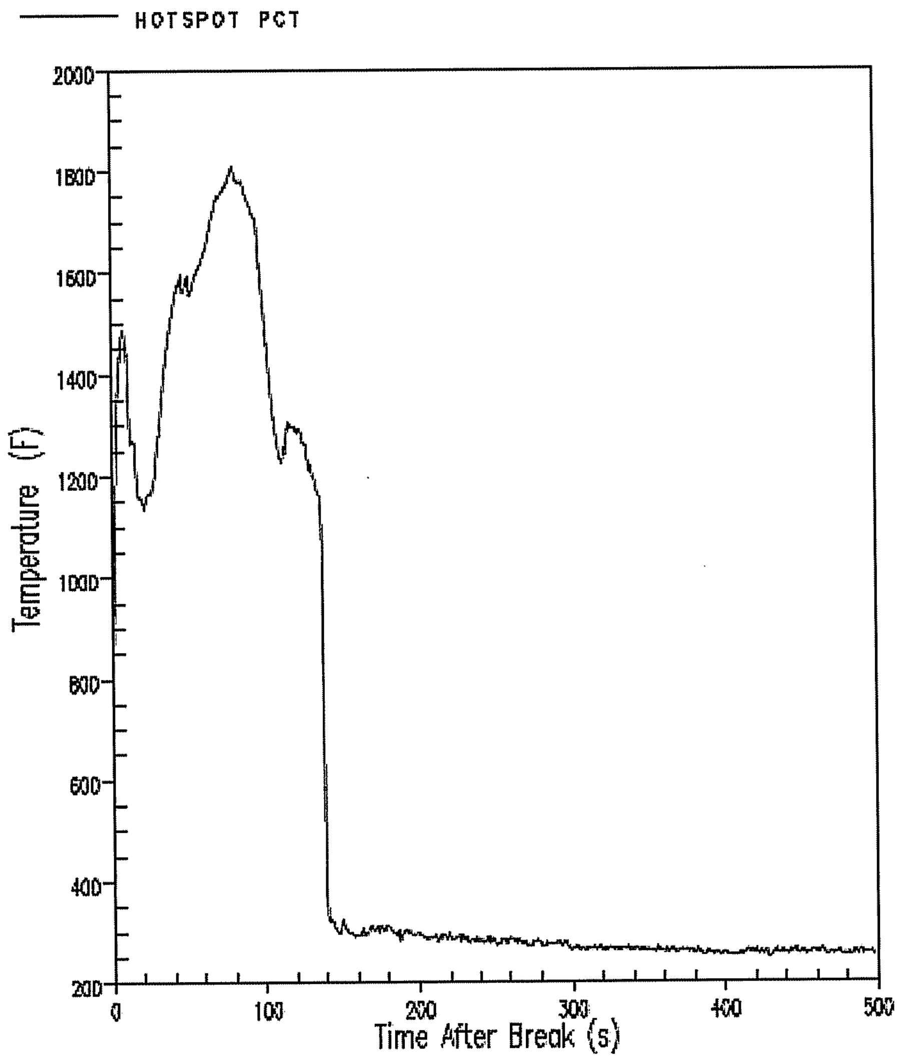


Figure 21

Unit 2 Pressurizer Pressure for the Limiting PCT Case

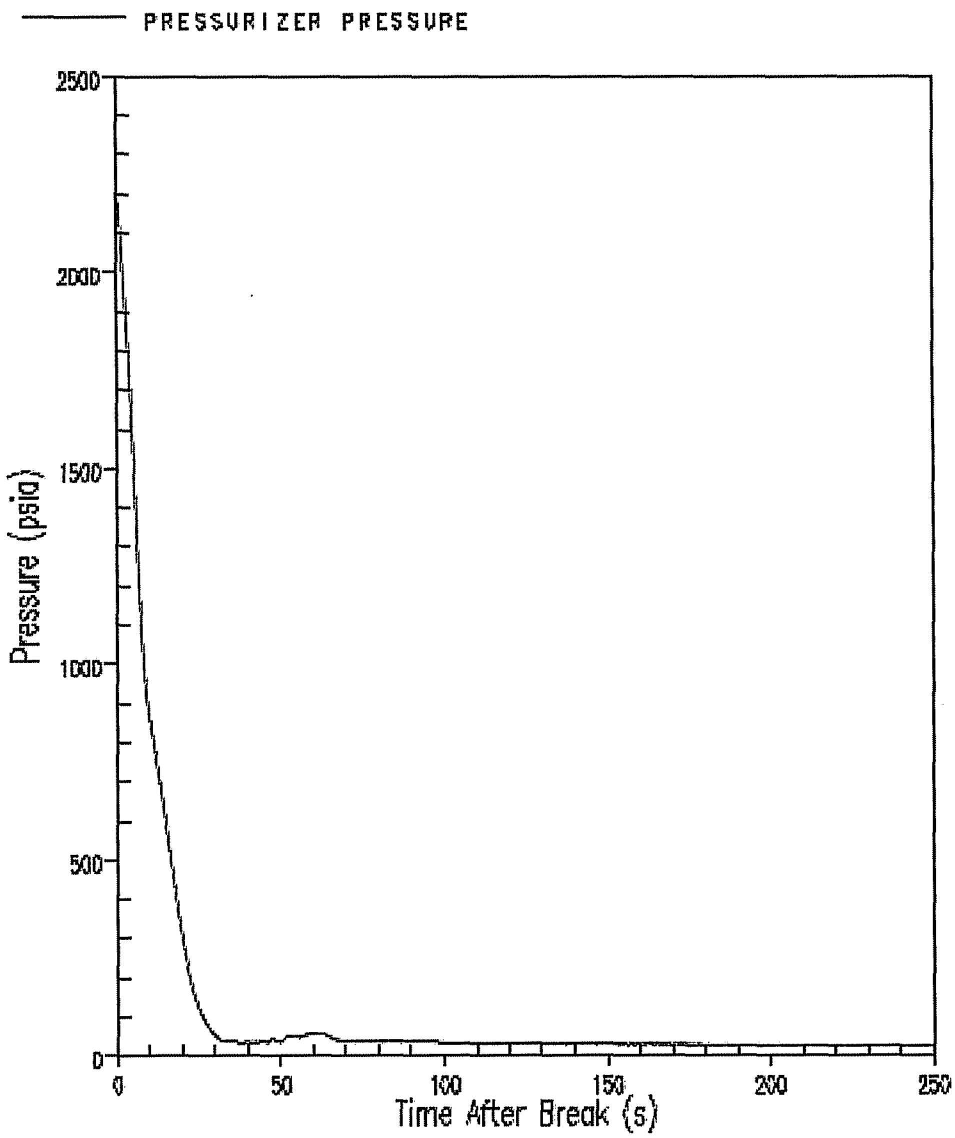


Figure 22

Unit 2 Vessel Side Break Flow for the Limiting PCT Case

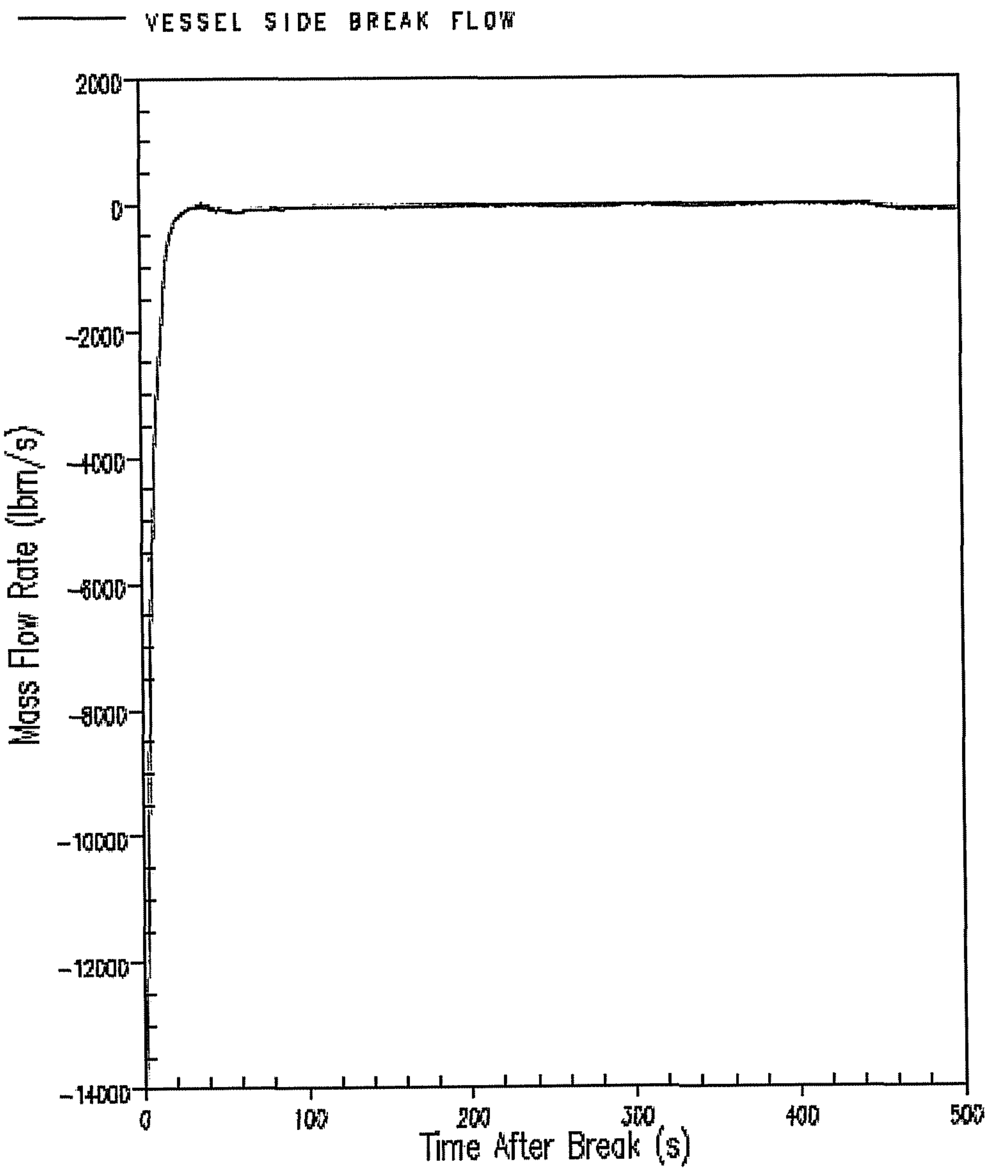


Figure 23

Unit 2 Pump Side Break Flow for the Limiting PCT Case

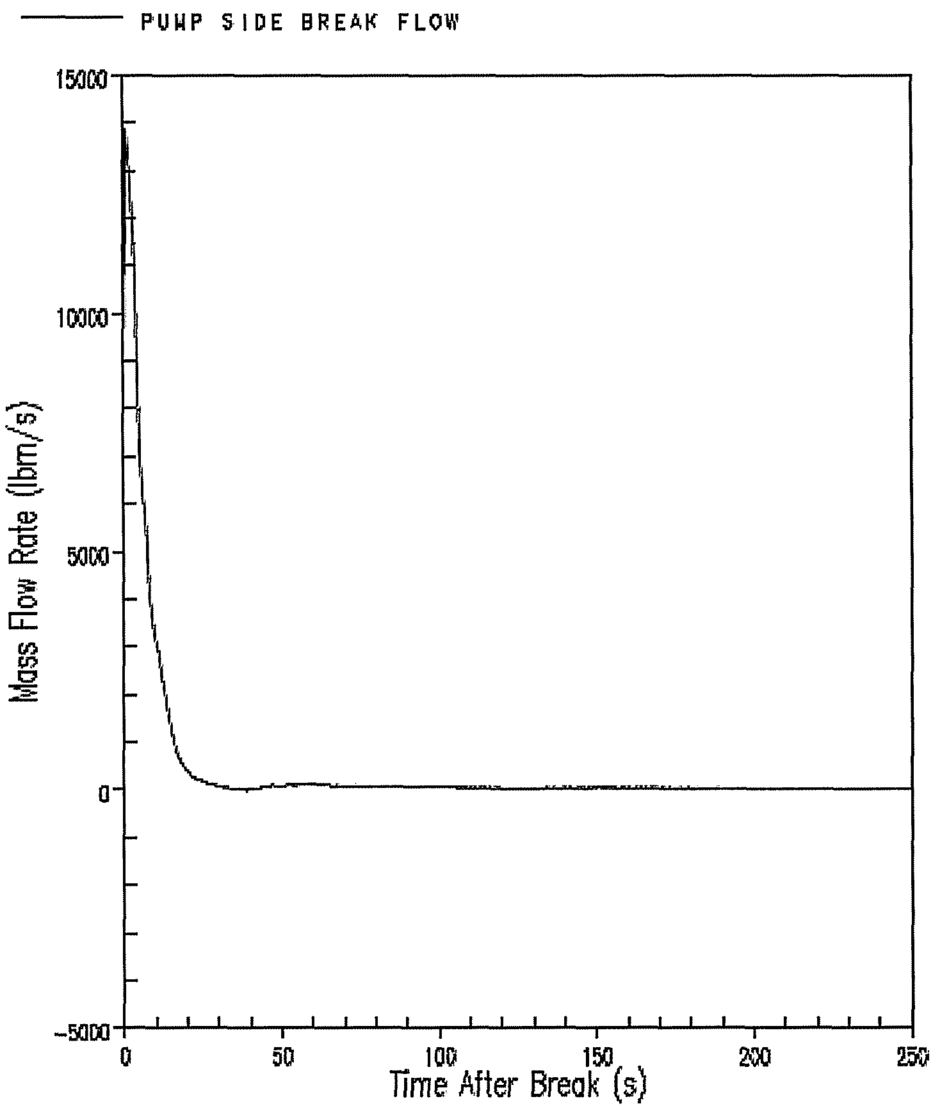


Figure 24

Unit 2 Void Fraction in Pumps for the Limiting PCT Case

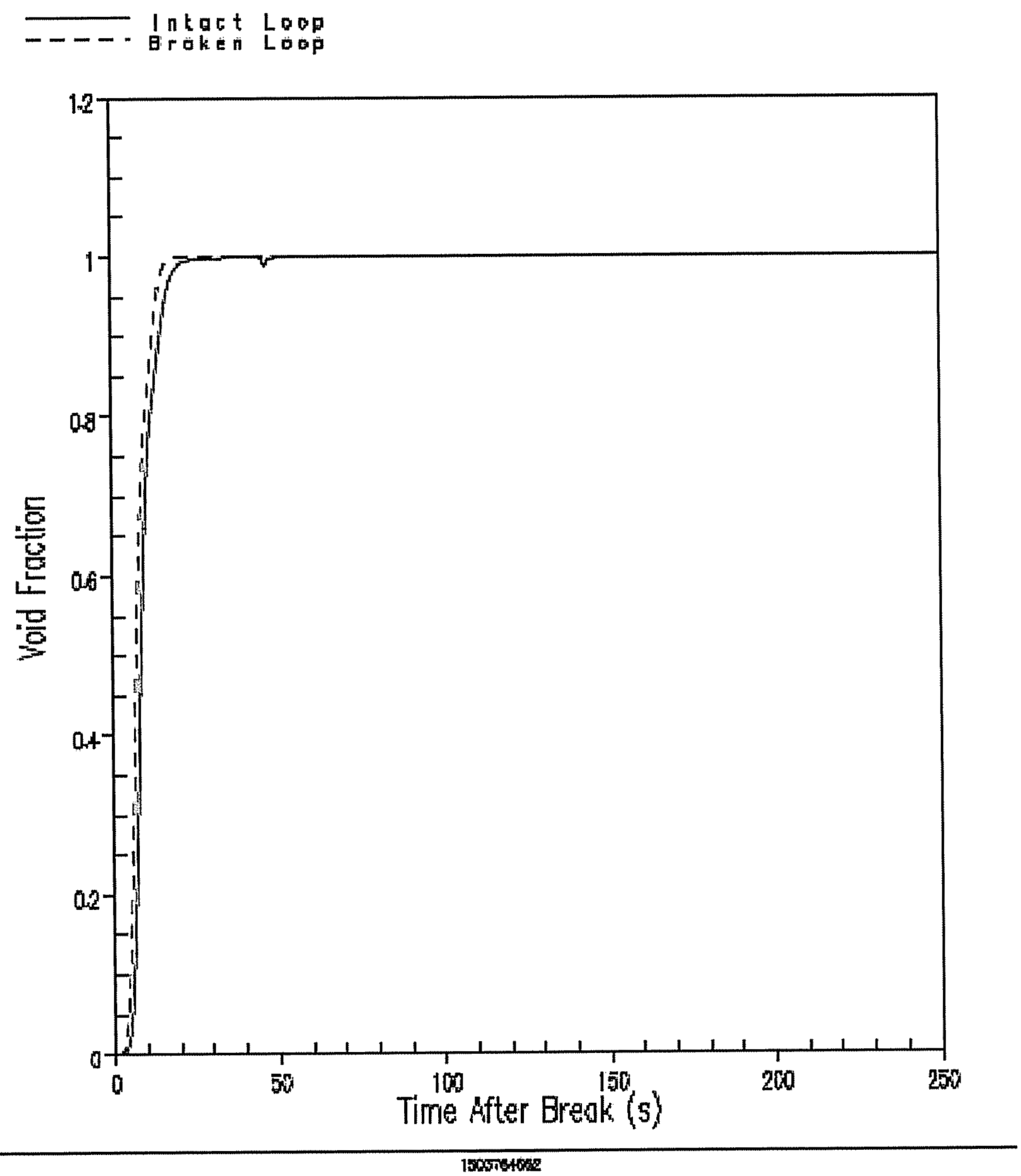


Figure 25

Unit 2 Vapor Flow for Hot Assembly Near Top of Core
for the Limiting PCT Case

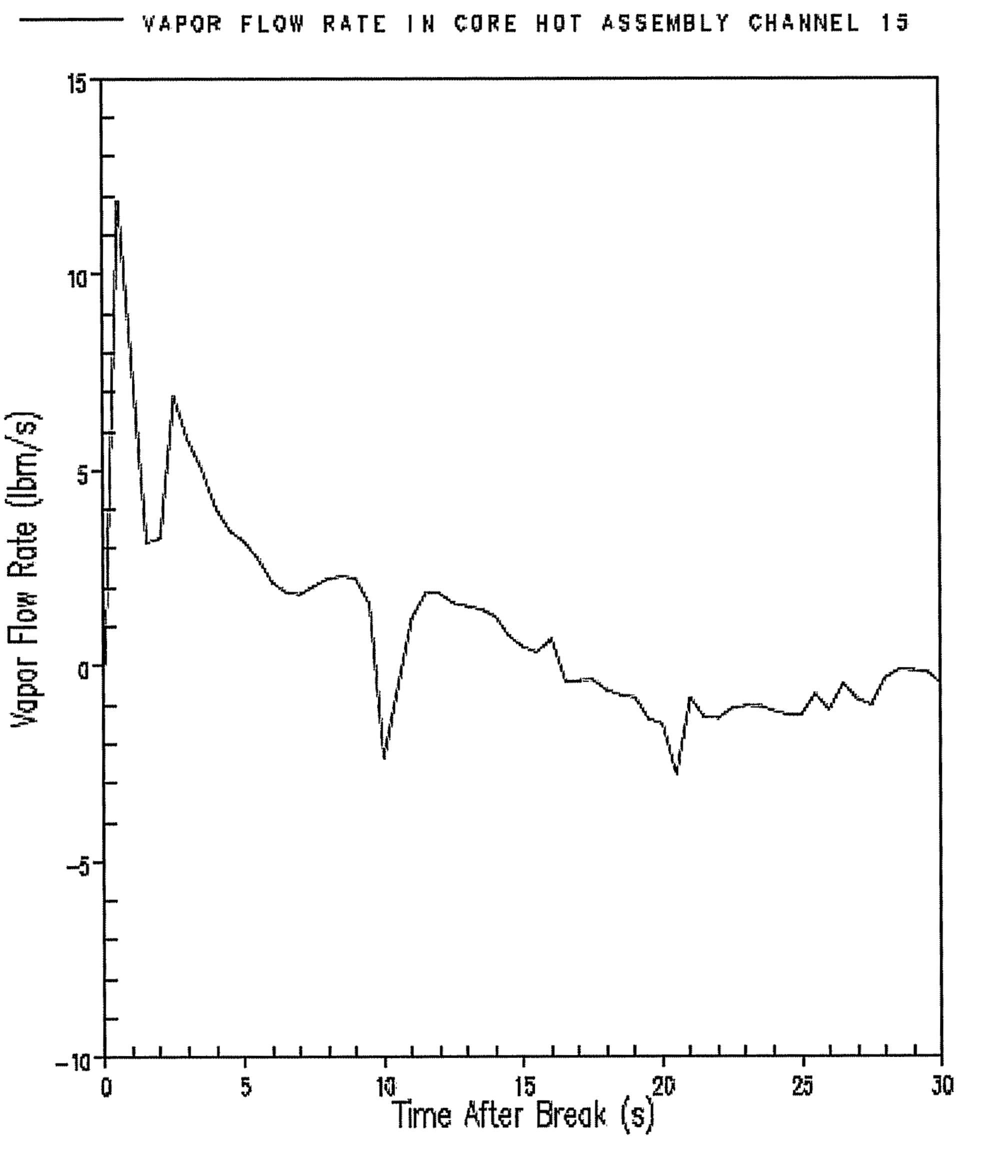


Figure 26

Unit 2 Accumulator Injection Flow for the Limiting PCT Case

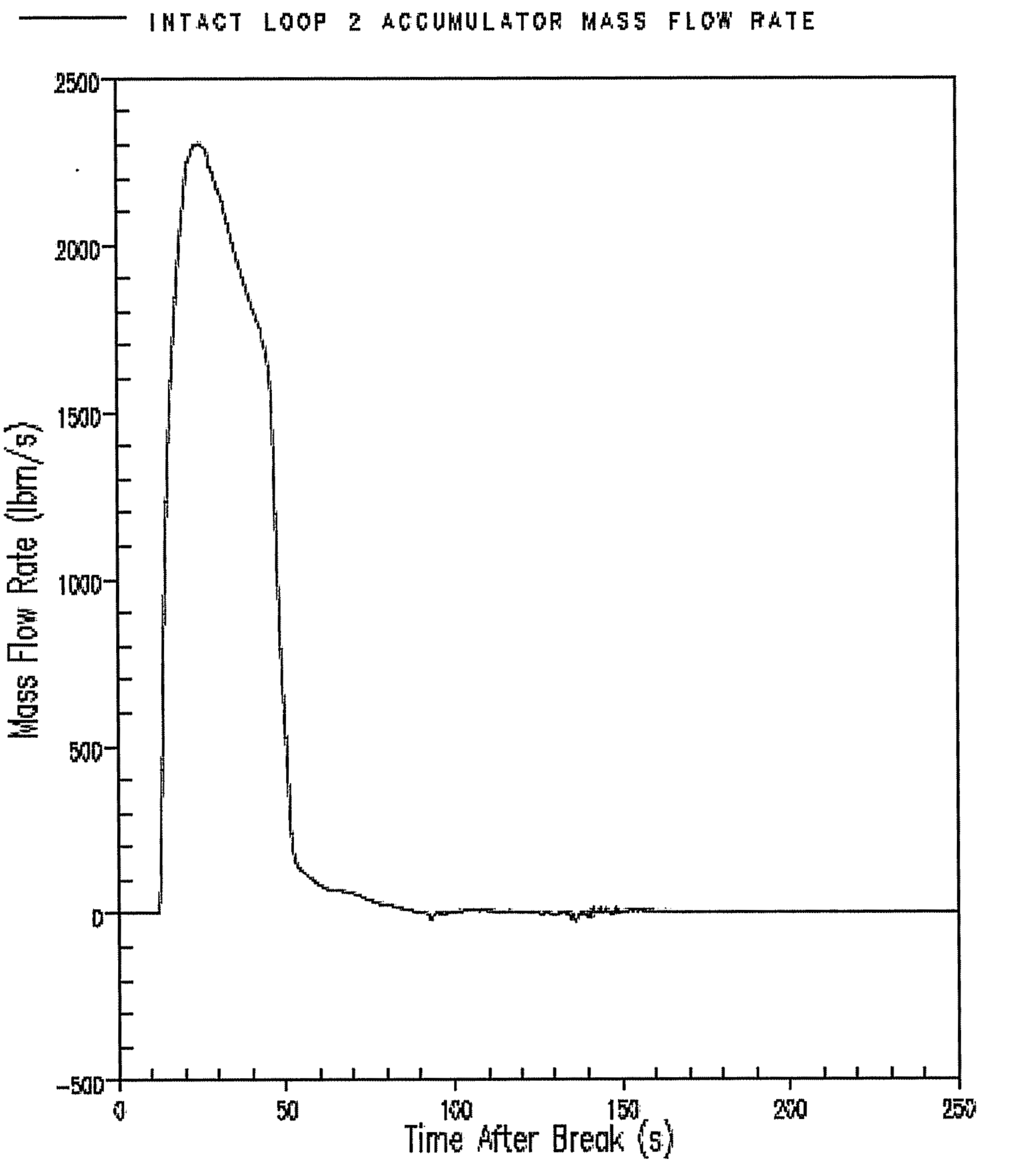


Figure 27

High Head Safety Injection Flow for the Limiting PCT Case

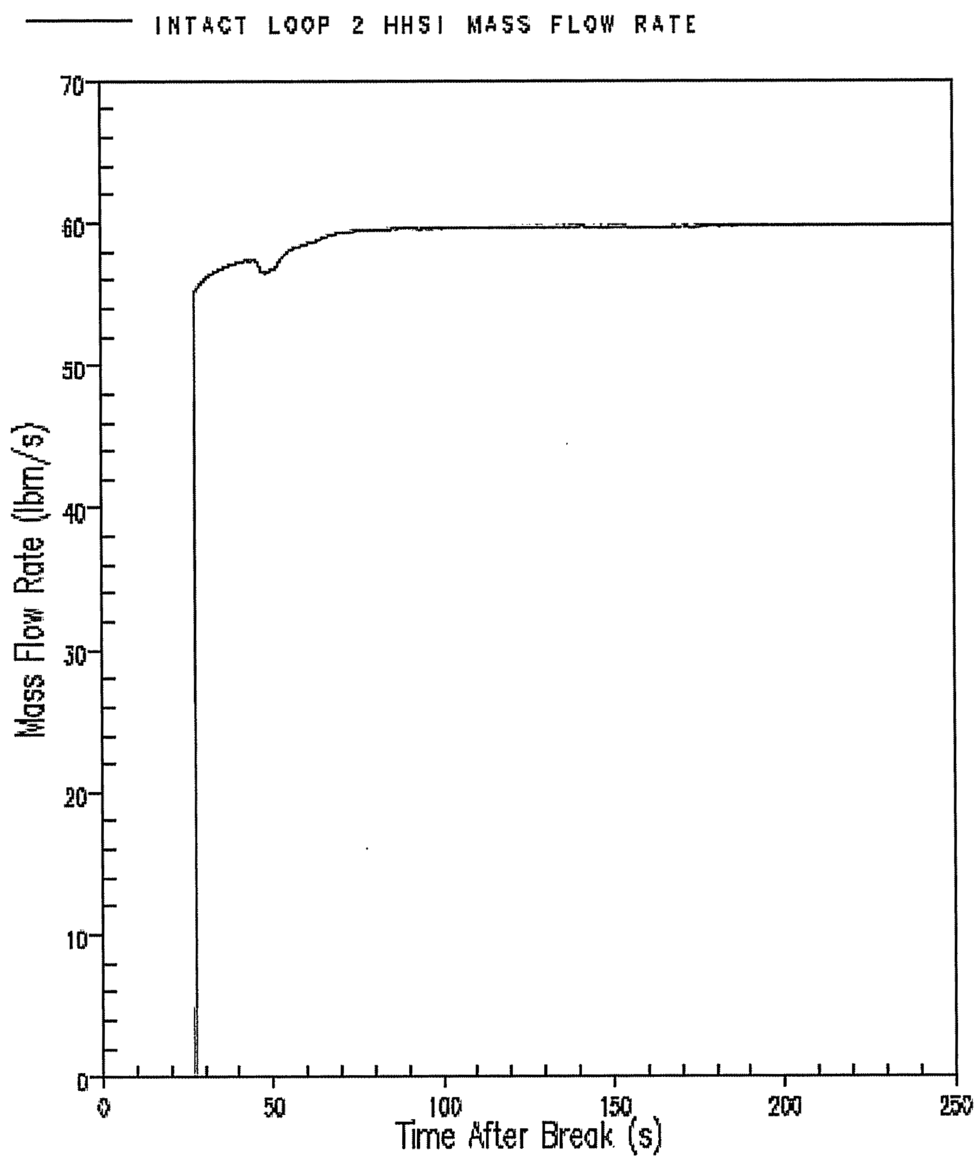


Figure 28

Unit 2 Low Head Safety Injection Flow for the Limiting PCT Case

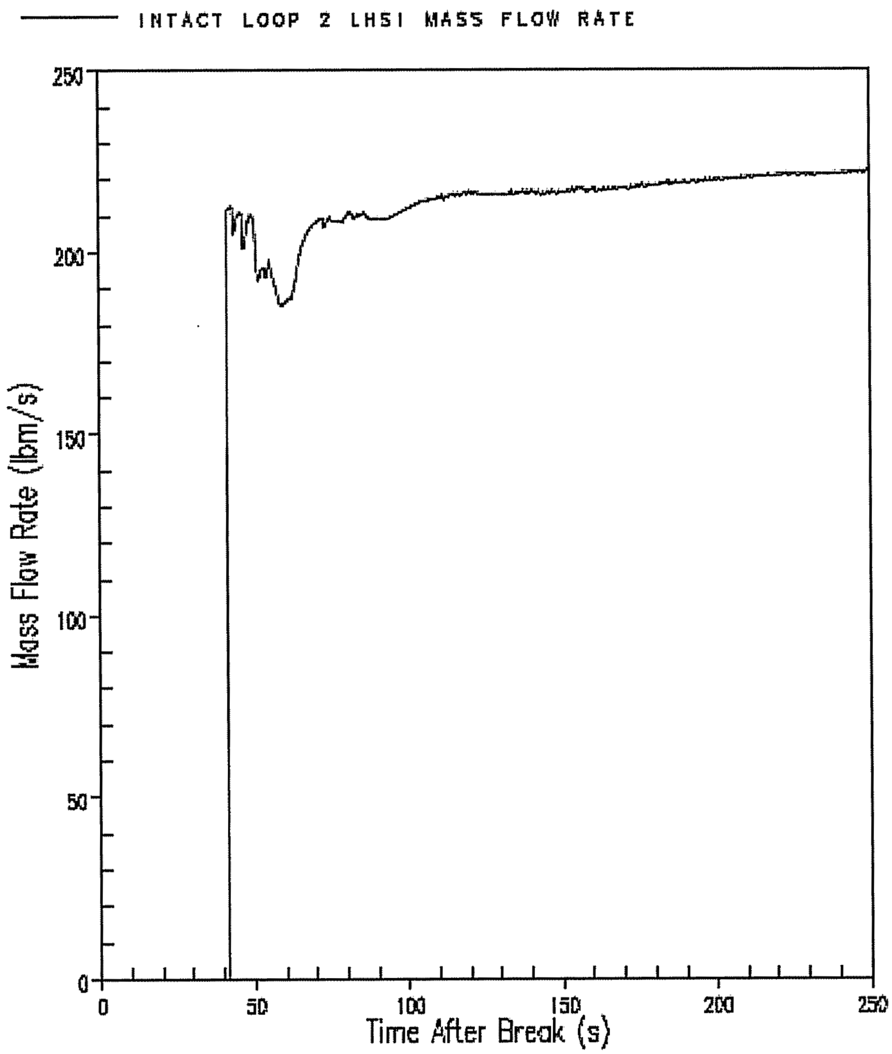


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Figure 29

Unit 2 Lower Plenum Collapsed Liquid Level for the Limiting PCT Case

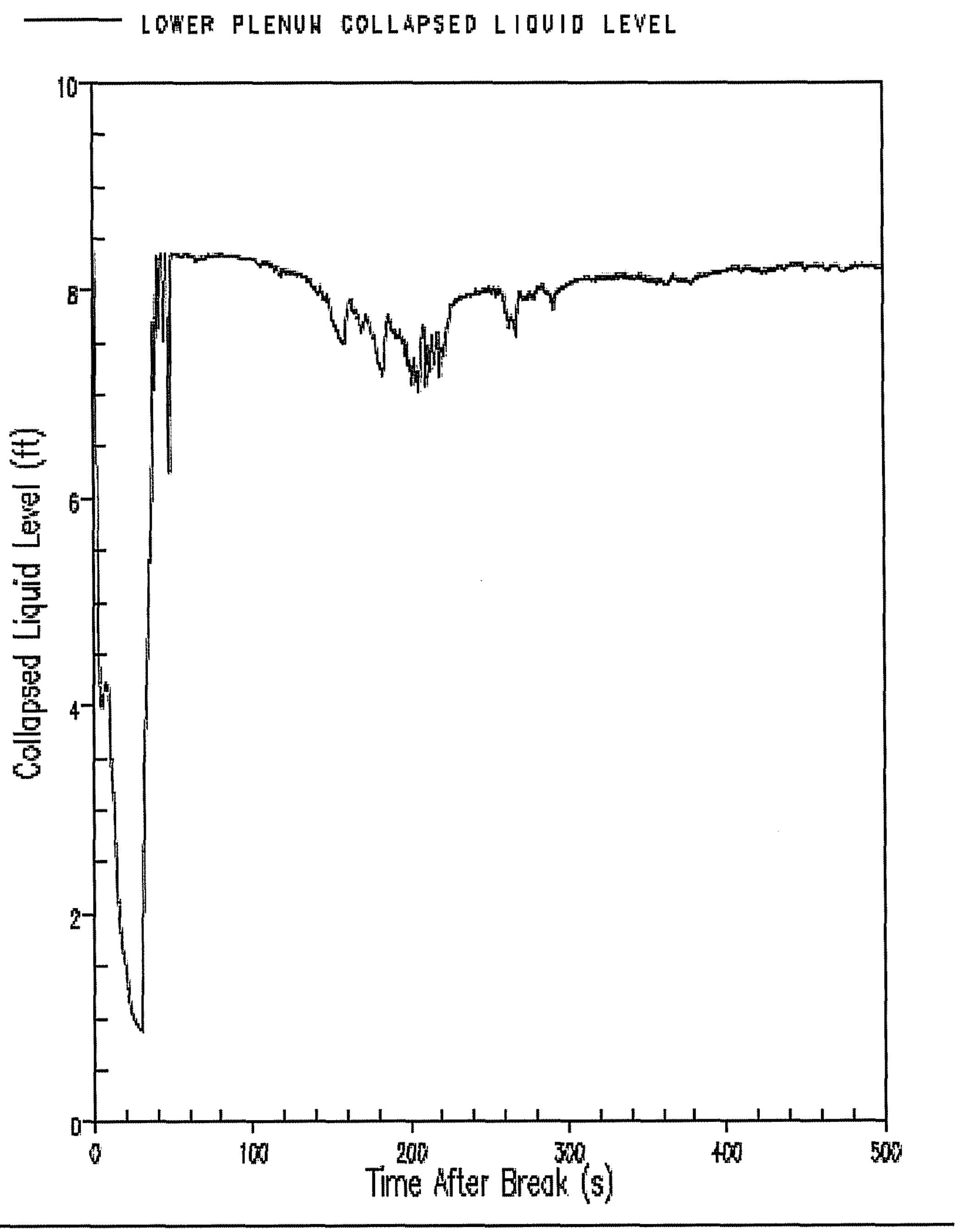


Figure 30

Unit 2 Downcomer Collapsed Liquid Level for the Limiting PCT Case

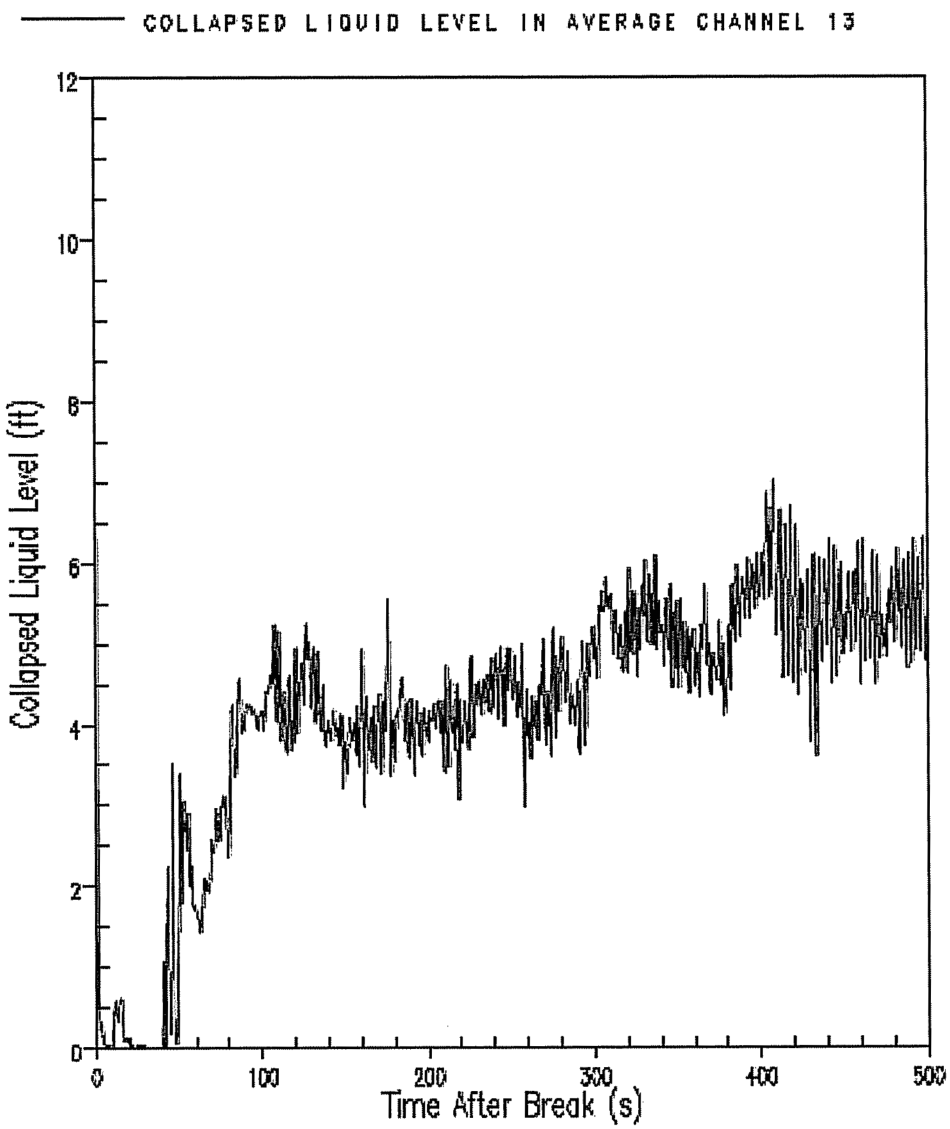


Figure 31

Unit 2 Core Average Channel Collapsed Liquid Level for the Limiting PCT Case

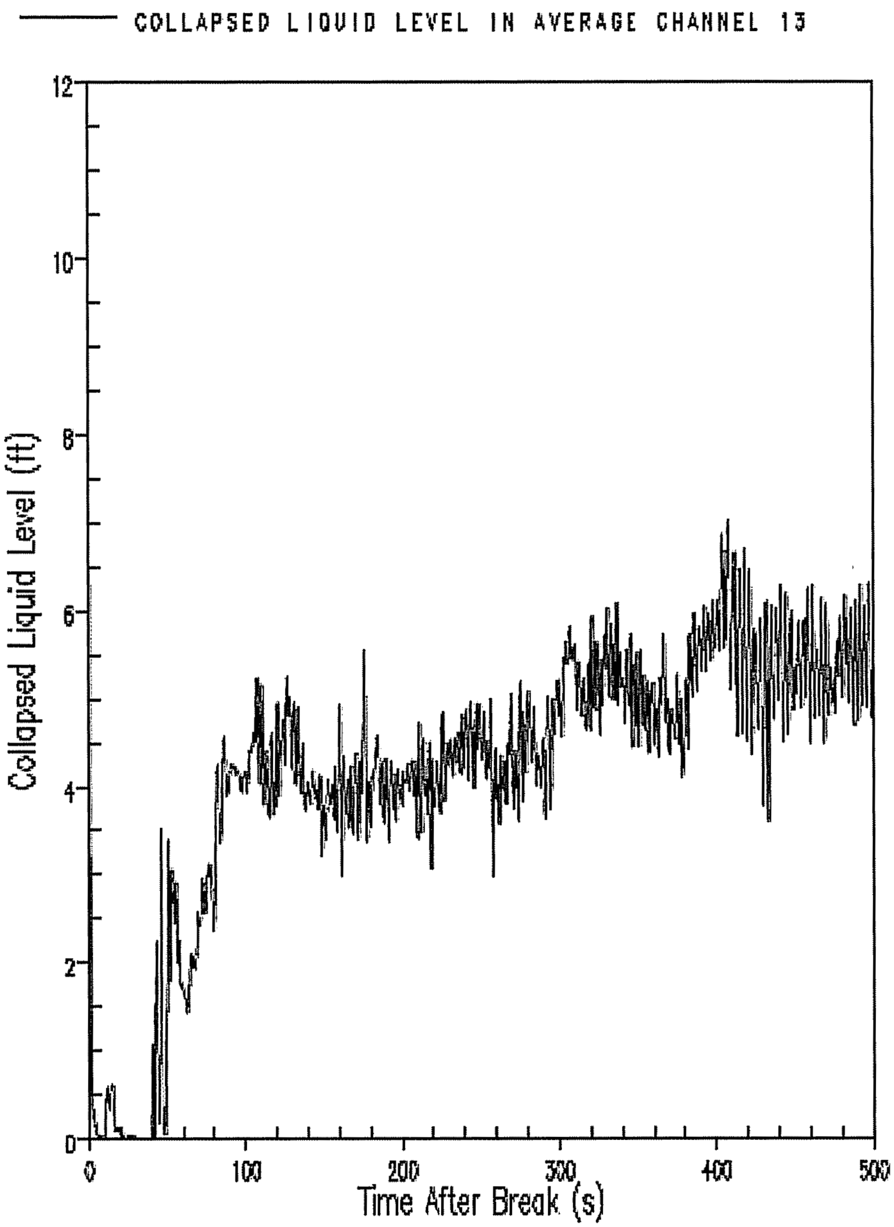


Figure 32

Unit 2 Vessel Liquid Mass for the Limiting PCT Case

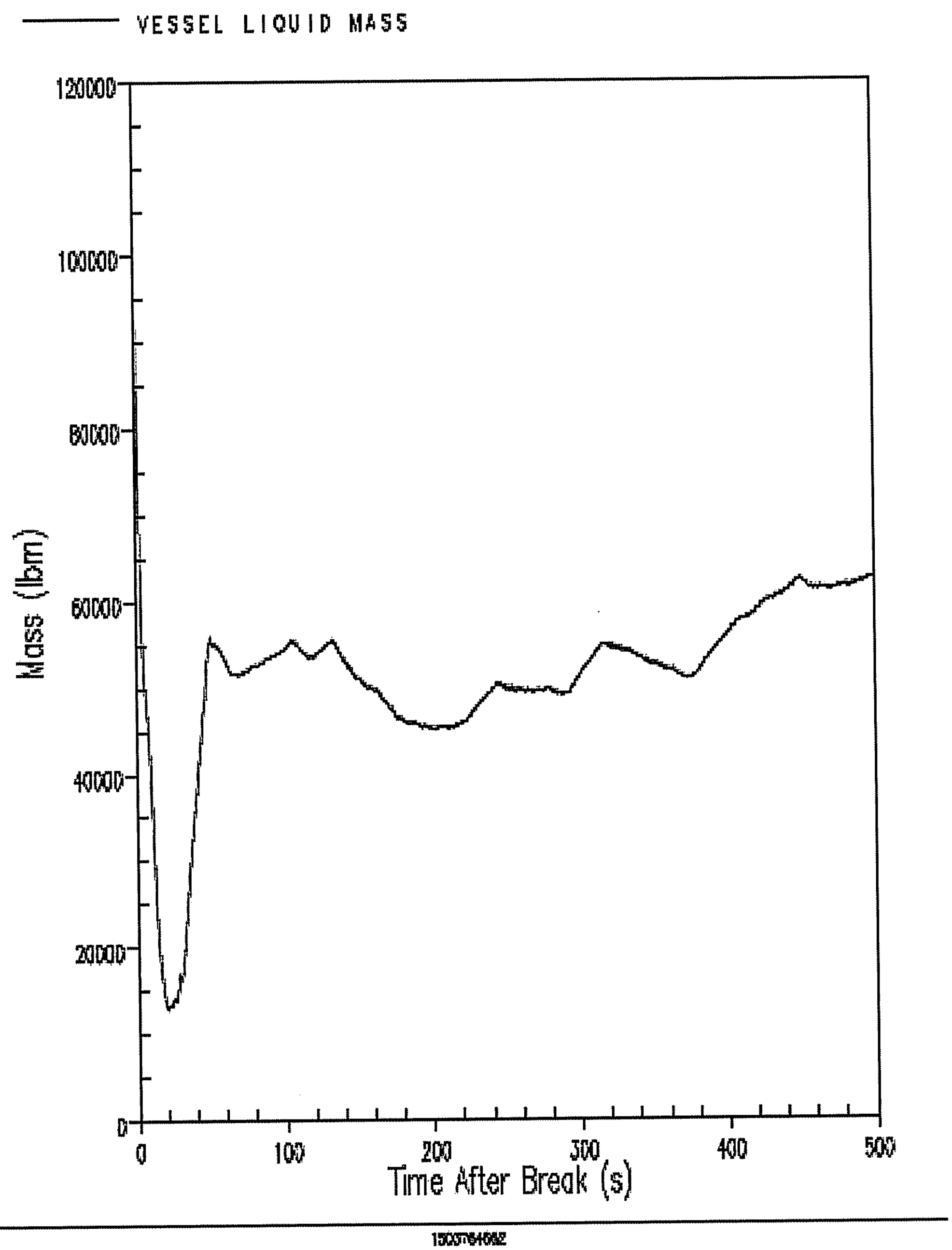


Figure 33
Unit 2 WCOBRA/TRAC Peak Clad Temperature for
All Five Rod Groups for the Limiting PCT Case

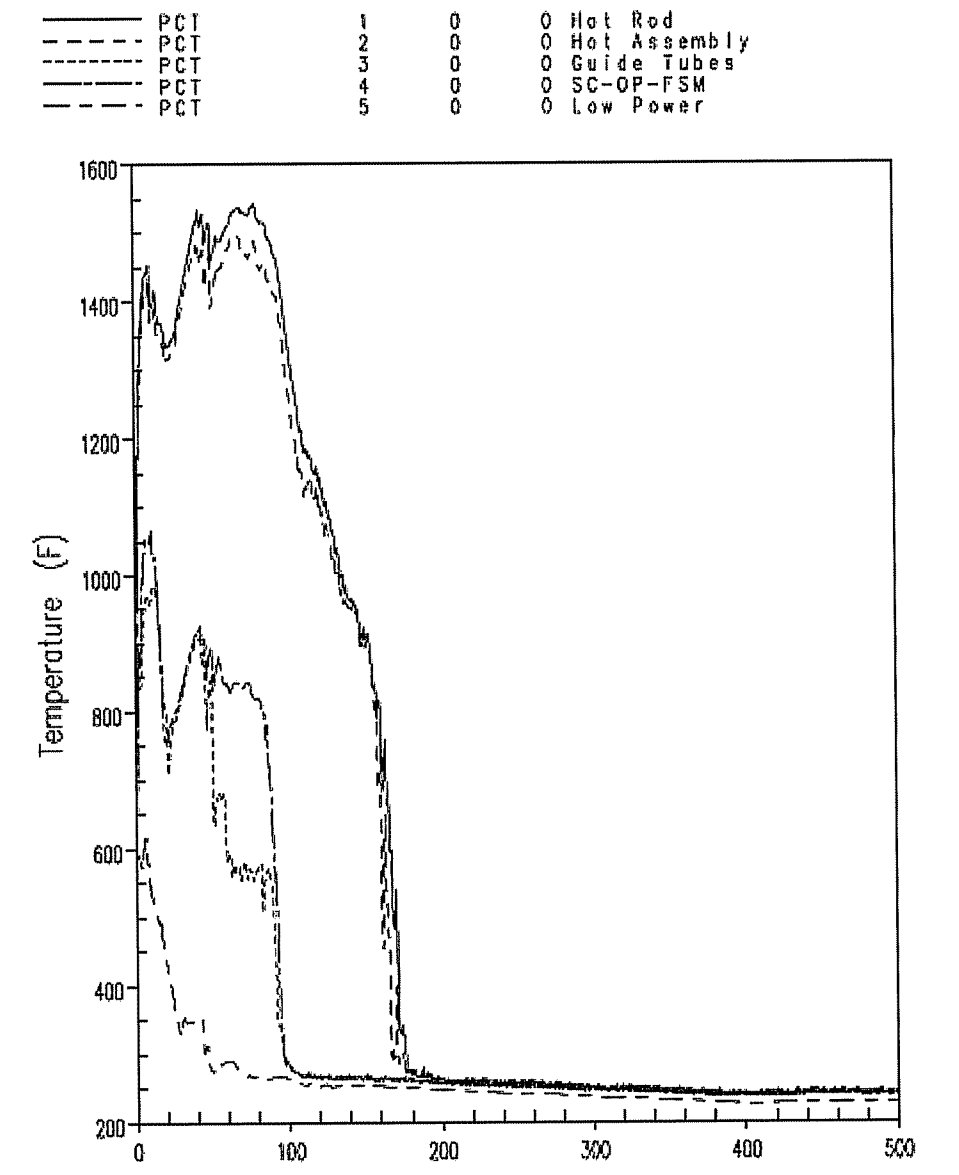


Figure 34
Unit 2 Peak Clad Temperature Elevation for the
Hot Rod for the Limiting PCT Case

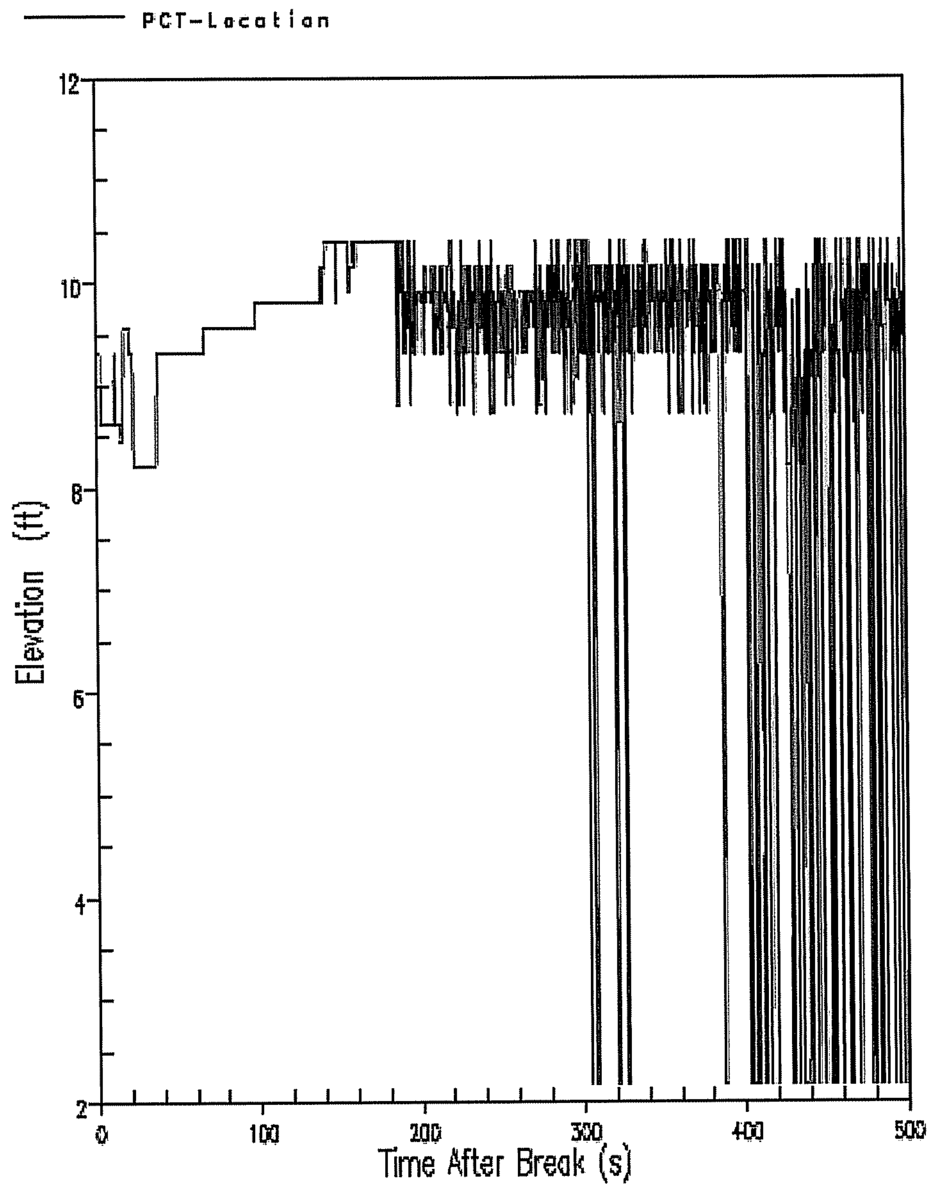
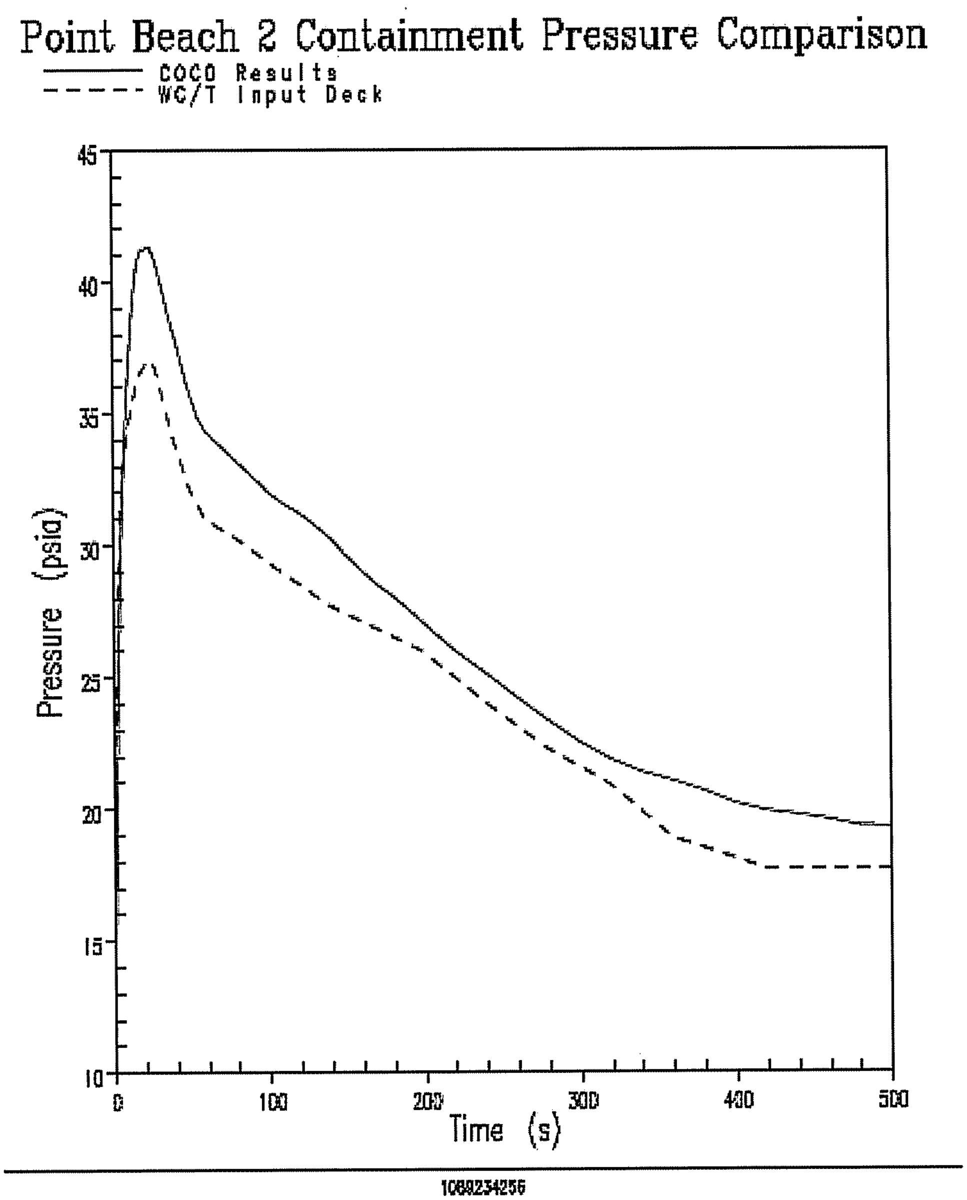


Figure 35

Unit 2 Analysis Versus Calculated Containment Backpressure



ENCLOSURE 2

FPL ENERGY POINT BEACH, LLC
POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

LICENSE AMENDMENT REQUEST 258
INCORPORATE BEST ESTIMATE LARGE BREAK LOSS OF COOLANT ACCIDENT (LOCA) ANALYSES USING ASTRUM

MARKED-UP TECHNICAL SPECIFICATION PAGES

5.6 Reporting Requirements

5.6.4 CORE OPERATING LIMITS REPORT (COLR) (continued)

(8) LCO 3.2.3, "Axial Flux Difference (AFD)"
(9) LCO 3.3.1, "Reactor Protection System (RPS) Instrumentation - Overtemperature ΔT"
(10) LCO 3.3.1, "Reactor Protection System (RPS) Instrumentation - Overpower ΔT"
(11) LCO 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits"
(12) LCO 3.9.1, "Boron Concentration"

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC. When an initial assumed power level of 102 percent of the original rated thermal power is specified in a previously approved method, 100.6 percent of uprated rated thermal power may be used only when the main feedwater flow measurement (used as the input for reactor thermal output) is provided by the Caldon leading edge flowmeter (LEFM) as described in reports 11 and 12 listed below. When main feedwater flow measurements from the LEFM are unavailable, a power measurement uncertainty consistent with the instruments used shall be applied.

Future revisions of approved analytical methods listed in this Technical Specification that currently reference the original Appendix K uncertainty of 102 percent of the original rated thermal power should include the condition given above allowing use of 100.6 percent of uprated rated thermal power in the safety analysis methodology when the LEFM is used for main feedwater flow measurement.

The approved analytical methods are described in the following documents:

(1) WCAP-14449-P-A, "Application of Best Estimate Large Break LOCA Methodology to Westinghouse PWR's with Upper Plenum Injection~~,~~" ~~Revision 1, October 1999.~~ (cores containing 422V+ fuel)
(2) WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology" ~~July 1985.~~
(3) WCAP-11397-P-A, "Revised Thermal Design Procedure~~,~~" ~~April 1989.~~

5.6.4 CORE OPERATING LIMITS REPORT (COLR) (continued)

(4) WCAP-14787-P, ~~Rev. 2,~~ "Revised Thermal Design Procedure Instrument Uncertainty Methodology for Wisconsin Electric Power Company Point Beach Units 1 & 2 (Fuel Upgrade & Uprate to 1656 MWt-NSSS Power with Feedwater Venturis, or 1679 MWt-NSSS Power with LEFM on Feedwater Header)"~~,~~ ~~October, 2002 (approved by NRC Safety Evaluation, November 29, 2002).~~

(5) WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model Using The NOTRUMP Code," ~~August 1985.~~

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

(7) WCAP-8745-P-A, "Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions~~,~~" ~~September 1986.~~

(8) WCAP-10216-P-A, "Relaxation of Constant Axial Offset Control," ~~Revision 1A, February 1994.~~

(9) WCAP-10924-P-A, "Large Break LOCA Best Estimate Methodology, Volume 2: Application to Two-Loop PWRs Equipped with Upper Plenum Injection~~,~~" ~~and Addenda~~, ~~December 1988.~~ (cores not containing 422 V+ fuel)

(10) WCAP-10924-P-A, "LBLOCA Best Estimate Methodology: Model Description and Validation: Model Revisions~~,~~" ~~Volume 1~~, ~~Addendum 4, August 1990.~~ (cores not containing 422 V+ fuel)

(11) Caldon, Inc., Engineering Report-80P, "TOPICAL REPORT: Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM✓™ System~~,~~" ~~Revision 0, March 1997~~.

(12) Caldon, Inc., Engineering Report-160P, "Supplement to Topical Report ER-80P: Basis for a Power Uprate With the LEFM✓™ System~~,~~" ~~Revision 0, May 2000.~~

(13) <u>WCAP-16009-P-A, "Realistic Large-Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)"</u>

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

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