FENOC

FirstEnergy Nuclear Operating Company

Peter P. Sena III
Site Vice President

724-682-5234
Fax: 724-643-8069

July 6, 2009

L-09-141 10 CFR 50.90

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

SUBJECT:
Beaver Valley Power Station, Unit No. 1
Docket No. 50-334, License No. DPR-66
Application to Permit Operation with ASTRUM Best-Estimate Large Break Loss of Coolant Accident (LOCA) Methodology

In accordance with the provisions of 10 CFR 50.90, the FirstEnergy Nuclear Operating Company (FENOC) is submitting a request for an amendment to the operating license for Beaver Valley Power Station Unit No. 1 (BVPS-1).

The proposed amendment would revise Technical Specification 5.6.3, "Core Operating Limits Report," to allow use of the generically approved topical report, WCAP-16009-P-A, "Realistic Large Break LOCA Evaluation Methodology Using Automated Statistical Treatment of Uncertainty Method (ASTRUM)," for BVPS-1. The existing reference to WCAP-12945-P-A, "Code Qualification Document for Best Estimate LOCA Analysis," will be retained as it is still applicable to Beaver Valley Power Station, Unit Nos. 1 and 2.

By letter dated April 13, 2005, (ADAMS Accession No. ML051080236), FENOC committed to complete and submit a reanalysis of large break LOCA for BVPS-1 to the Nuclear Regulatory Commission. The regulatory commitment is addressed with the submittal of this amendment request.

FENOC requests approval of the proposed amendment within a nominal one year period of the date of this letter. The amendments shall be implemented within 30 days of approval.

A001
NRR

There are no regulatory commitments contained in this letter. If there are any questions or if additional information is required, please contact Mr. Thomas A. Lentz, Manager – Fleet Licensing, at 330-761-6071.

I declare under penalty of perjury that the foregoing is true and correct. Executed on July 6, 2009.

Sincerely,

Peter P. Sena III

Enclosure: FENOC Evaluation of the Proposed Changes

cc: NRC Region I Administrator
NRC Senior Resident Inspector
NRR Project Manager
Director BRP/DEP
Site Representative (BRP/DEP)

FENOC Evaluation of the Proposed Changes
Beaver Valley Power Station Unit No. 1
License Amendment Request No. 09-004

Subject: Application to Permit Operation with ASTRUM Best-Estimate Large Break LOCA Methodology.

Table of Contents

Attachments

1.0 SUMMARY DESCRIPTION

This evaluation supports a request to amend Operating License DPR-66 for Beaver Valley Power Station Unit No. 1 (BVPS-1).

The proposed change would revise the Technical Specifications to apply the Westinghouse Automated Statistical Treatment of Uncertainty Method (ASTRUM) best-estimate large break loss of coolant accident (LOCA) analysis methodology.

2.0 DETAILED DESCRIPTION

The proposed Technical Specification change is provided in Attachment 1. Retyped Technical Specification replacement pages are provided in Attachment 2 for information only. The proposed Technical Specification Bases and Licensing Requirements Manual (LRM) changes are provided in Attachments 3 and 4 respectively. The proposed Technical Specification Bases and LRM changes do not require Nuclear Regulatory Commission (NRC) approval. The Beaver Valley Power Station (BVPS) Technical Specification Bases Control Program controls the review, approval and implementation of Technical Specification Bases changes. The BVPS Licensing Document Control Program controls the review, approval and implementation of LRM changes. The Technical Specification Bases and LRM changes are provided for information only.

The proposed changes to the Technical Specifications, Technical Specification Bases and LRM have been prepared electronically. Deletions are shown with a strike-through and insertions are shown double-underlined.

To meet format requirements the Index, Technical Specifications, Technical Specification Bases and LRM pages will be revised and repaginated as necessary to reflect the proposed changes.

FirstEnergy Nuclear Operating Company (FENOC) requests that Specification 5.6.3, "Core Operating Limits Report (COLR)" be amended to allow use of the ASTRUM methodology. Specification 5.6.3.b lists applicable references for the analytical methods used to determine core operating limits identified in Specification 5.6.3.a. This list of references includes the Westinghouse topical report, "Code Qualification Document for Best Estimate LOCA Analysis," WCAP-12945-P-A (Reference 1) that documents the currently approved large break loss of coolant accident (LOCA) analysis methodology. It is proposed that an additional reference be added for BVPS-1. This reference is the generically approved topical report, "Realistic Large Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)," WCAP-16009-P-A (Reference 2). The existing reference to WCAP-12945-P-A (Reference 1) will be retained as it is still applicable to both Beaver Valley Power Station Unit Nos. 1 and 2.

3.0 TECHNICAL EVALUATION

3.1 Method of Thermal Analysis Used for BVPS-1

After obtaining generic NRC approval of WCAP-12945-P-A (Reference 1) describing best-estimate large break LOCA methodology, Westinghouse underwent a program to revise the statistical approach used to develop the peak cladding temperature (PCT) and oxidation results at the 95^{th} percentile. This method is still based on the Code Qualification Document (CQD) methodology of WCAP-12945-P-A (Reference 1) and follows the steps in the code scaling applicability and uncertainty (CSAU) methodology

(Reference 3). However, the uncertainty analysis (Element 3 in CSAU) is replaced by a technique based on order statistics. The Automated Statistical Treatment of Uncertainty Method (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 2).

This License Amendment Request summarizes the application of the Westinghouse ASTRUM best-estimate LOCA evaluation model to BVPS-1 for the large break LOCA accident analysis. Table 1 lists the major plant parameter assumptions used in the best-estimate LOCA evaluation analysis for BVPS-1.

The BVPS-1 analysis supports a Rated Thermal Power of 2900 megawatts thermal with ± 0.6 percent uncertainties. At 2900 megawatts thermal, the core average linear power is 5.83 kilowatt per foot (kW/ft). The hot rod average linear heat rate and hot assembly average linear heat rate supported by the ASTRUM analysis are 10.2 kW/ft and 9.8 kW/ft maximum, respectively. In the limiting PCT case, the core average channel fuel assemblies are modeled with an average power of 6.9 kW/ft, the assembly peak linear heat rate is 12.8 kW/ft and the hot rod peak linear rate is 13.3 kW/ft. For every run, the peak and average linear heat rates are sampled according to the ASTRUM methodology.

Both FENOC and the ASTRUM 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 ASTRUM methodology replaces the response surface technique with a statistical sampling method where the uncertainty parameters are simultaneously sampled for each case. It requires the execution of 124 calculations to determine a bounding estimate of the 95th percentile of the PCT, local maximum oxidation (LMO), and core wide oxidation (CWO) with 95 percent 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 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.

The BVPS-1 analysis is in accordance with the applicability limits and usage conditions defined in Section 13-3 of WCAP-16009-P-A (Reference 2) as applicable to the ASTRUM methodology. Section 13-3 of WCAP-16009-P-A (Reference 2) 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 large break LOCA (LBLOCA) analysis and associated model for BVPS-1 is unit-specific.

3.2 Description of the 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. 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. WCAP-16009-P-A (Reference 2) uses a double-ended guillotine break for plant-specific confirmatory analyses.

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 begin to flash. Once the system has depressurized to the accumulator pressure, the accumulator begins to inject cold borated water into the cold legs. During the blowdown phase, a portion of the injected emergency core cooling system (ECCS) water is calculated to be bypassed around the downcomer and out of the break. This 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.

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

During the reflood phase, 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 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 phase.

3.3 ASTRUM Analysis Results for BVPS-1

The major plant assumptions used in the ASTRUM analysis for BVPS-1 are provided in Table 1. The total minimum injected safety injection (SI) flows for given RCS pressures are provided in Table 1A. The results of the BVPS-1 ASTRUM analysis are summarized in Table 2. Table 3 contains a sequence of events for the limiting PCT transient.

The scatter plot presented in Figure 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 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, the same case proved to be the limiting PCT and limiting LMO transient for BVPS-1. Figure 2 shows the predicted clad temperature transient at the PCT limiting elevation for the limiting case.

Figures 3 through 15 illustrate the key major response parameters for the limiting PCT transient. The containment backpressure utilized for the LBLOCA analysis compared to the calculated containment backpressure is provided in Figure 16. The worst single failure for the LBLOCA analysis is the loss of one train of ECCS injection (consistent with WCAP-16009-P-A (Reference 2)); however, all containment systems that would reduce containment pressure are modeled for the LBLOCA containment backpressure calculation.

Figure 17 illustrates the operating limits for the Integral of the Power Generated in the Bottom Third of the Core (PBOT) and the Integral of the Power Generated in the Middle Third of the Core (PMID).

The ASTRUM analysis has demonstrated that there is a high level of probability that the following criteria set forth in 10 CFR 50.46 are met:

(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 2161 °F for BVPS-1, the analysis confirms that 10 CFR 50.46 acceptance criterion (b)(1), "Peak Clad Temperature less than 2200 °F," is demonstrated. The result is 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. Since the resulting LMO for the limiting case is 9.22 percent for BVPS-1, the analysis confirms that 10 CFR 50.46 acceptance criterion (b)(2), "Local Maximum Oxidation of the cladding less than 17 percent," is demonstrated. The result is shown in Table 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 0.94 percent for BVPS-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), "Core Wide Oxidation less than 1 percent," is demonstrated. The result is shown in Table 2.

(b)(4) 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 the PCT and maximum cladding oxidation limits remain in effect for best-estimate LOCA applications. The grid crush calculations currently in place for BVPS-1 remain unchanged with the application of the ASTRUM analysis; therefore, acceptance criterion (b)(4) is satisfied.

(b)(5) 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 BVPS-1 to maintain long-term cooling remain unchanged with the application of the ASTRUM analysis; therefore, acceptance criterion (b)(5) is satisfied.

Based on the ASTRUM analysis results (see Table 2), the margin of safety to the limits prescribed by 10 CFR 50.46 is maintained at BVPS-1.

Therefore, FENOC has concluded that adopting the ASTRUM best-estimate LBLOCA methodology for BVPS-1 and making the proposed TS changes would not adversely affect the health and safety of the public.

4.0 REGULATORY EVALUATION

The proposed change would revise the Technical Specifications to apply the Westinghouse Automated Statistical Treatment of Uncertainty Method best-estimate large break loss of coolant accident analysis methodology to Beaver Valley Power Station Unit No. 1.

4.1 Significant Hazards Consideration

FirstEnergy Nuclear Operating 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:

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

   No. No physical changes are required as a result of implementing the ASTRUM best-estimate large break loss of coolant accident (LOCA) methodology and associated technical specification changes. The plant conditions assumed in the analysis are bounded by the design conditions for all equipment in the plant. Therefore, there will be no increase in the probability of a loss of coolant accident. The consequences of a LOCA are not being increased, since it is shown that the emergency core cooling system is designed so that its calculated cooling performance conforms to the criteria contained in 10 CFR 50.46, Paragraph (b). No other accident is potentially affected by this change.

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

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

   No. There are no physical changes being made to the plant. No new modes of plant operation are being introduced. The parameters assumed in the analysis are within the design limits of the existing plant equipment. All plant systems will perform as designed during the response to a potential accident.

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

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

   No. The methodology used in the analysis would more realistically describe the expected behavior of plant systems during a postulated loss of coolant accident. Uncertainties have been accounted for as required by 10 CFR 50.46. A sufficient number of loss of coolant accidents with different break sizes, different locations and other variations in properties are analyzed to provide assurance that the most severe postulated loss of coolant accidents are calculated. As described in Section 3.3, there is a high level of probability that all criteria contained in 10 CFR 50.46, Paragraph (b) are met.

4.2 Applicable Regulatory Requirements/Criteria

The proposed changes have been evaluated to determine whether applicable regulations and requirements would continue to be met. FENOC has determined that the proposed changes do not require any exemptions or relief from regulatory requirements, other than the TS, and do not affect conformance with any General Design Criteria differently than described in the BVPS-1 UFSAR. Section 3.3 of this submittal demonstrates that the proposed change is consistent with 10 CFR 50.46.

4.3 Precedent

References 4 through 12 are license amendments that approved the use of the ASTRUM best-estimate large break LOCA analysis methodology.

All of the referenced plants submitted similar requests to implement ASTRUM. Three of the referenced plants (References 4, 10 and 11) included a request to permit the use of another approved methodology for small break LOCA analyses. References 10 and 11 also requested other Technical Specification revisions. Otherwise, there are no significant differences between this request and those of the referenced plants.

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 will 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.
2. 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.
3. 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.
4. U.S. Nuclear Regulatory Commission letter to Mrs. Mary G. Korsnick, Vice President R.E. Ginna Nuclear Power Plant, LLC, "R.E. Ginna Nuclear Power Plant - Amendment RE: Revised Loss Of-Coolant Accident Analyses," (ADAMS Accession No. ML061180353) dated May 31, 2006.
5. U.S. Nuclear Regulatory Commission letter to Mr. L. M. Stinson, Vice President - Farley Project, Southern Nuclear Operating Company, Inc. "Joseph M. Farley Nuclear Plant, Units 1 AND 2 — Issuance of Amendments for Best Estimate Loss-Of-Coolant Accident (LOCA) Analyses Using ASTRUM," (ADAMS Accession No. ML061810306) dated July 11, 2006.
6. U.S. Nuclear Regulatory Commission letter to Mr. Michael Kansler, Entergy Nuclear Operations, Inc. "Indian Point Nuclear Generating Unit No. 2 - Issuance of Amendment Re: Large Break Loss-Of-Coolant Accident (LBLOCA) Analysis Methodology," (ADAMS Accession No. ML061710291) dated July 24, 2006.
7. U.S. Nuclear Regulatory Commission letter to Mr. John S. Keenan, Senior Vice President and Chief Nuclear Officer, Pacific Gas and Electric Company, "Diablo Canyon Power Plant, Unit No. 2 - Issuance of Amendment RE: TS 5.6.5, Core Operating Limits Report (COLR)," (ADAMS Accession No.ML063380020) dated December 20, 2006.
8. U.S. Nuclear Regulatory Commission letter to Mr. Thomas J. Palmisano, Site Vice President, Entergy Nuclear Management Company, LLC, "Prairie Island Nuclear Generating Plant, Units 1 and 2 -Issuance of Amendments RE: Incorporate Large-Break Loss-Of Coolant Accident Analysis Using ASTRUM," (ADAMS Accession No. ML071230789) dated June 28, 2007.
9. U.S. Nuclear Regulatory Commission letter to Mr. David A. Christian, Senior Vice President and Chief Nuclear Officer, Virginia Electric and Power Company, "Surry Power Station, Unit Nos. 1 and 2, Issuance of Amendments Regarding Use Of Westinghouse Best-Estimate Large Break Loss-Of-Coolant Accident Analysis Methodology," (ADAMS Accession No. ML071430074) dated September 6, 2007.

10. U.S. Nuclear Regulatory Commission letter to Mr. M. R. Blevins, Executive Vice President and Chief Nuclear Officer, Luminant Generation Company LLC, "Comanche Peak Steam Electric Station, Units 1 and 2 – Issuance Of Amendments Re: Revision To Technical Specifications To Allow Use of Westinghouse-Developed/NRC-Approved Analytical Methods To Establish Core Operating Limits," (ADAMS Accession No. ML080500627) dated April 2, 2008.

11. U.S. Nuclear Regulatory Commission letter to Mr. David A. Christian, President and Chief Nuclear Officer, Virginia Electric and Power Company. "Millstone Power Station, Unit No. 3 - Issuance of Amendment RE: Stretch Power Uprate," (ADAMS Accession No. ML081640535) dated August 12, 2008.

12. U.S. Nuclear Regulatory Commission letter to Mr. Michael W. Rencheck, Senior Vice President and Chief Nuclear Officer, Indiana Michigan Power Company, "Donald C. Cook Nuclear Plant, Unit 1 -Issuance Of Amendment To Renewed Facility Operating License Regarding Use of The Westinghouse ASTRUM Large Break Loss-Of-Coolant Accident Analysis Methodology," (ADAMS Accession No. ML082670351) dated October 17, 2008.

Table 1 - Major Plant Parameter Assumptions Used in the BELOCA Analysis for BVPS - 1

Parameter	Value
Plant Physical Description	
• Steam Generator (SG) Tube Plugging	≤ 22%
Plant Initial Operating Conditions	
• Reactor Power	≤ 2900 Megawatt thermal (± 0.6% uncertainties)
• Peak Heat Flux Hot Channel Factor (F_Q)	$F_Q \leq 2.52$
• Peak Hot Rod Enthalpy Rise Hot Channel Factor ($F_{\Delta H}$)	$F_{\Delta H} \leq 1.75$
• Axial Power Distribution	See Figure 17
Fluid Conditions	
• Initial Average Fluid Temperature (T_{AVG})	566.2 – 4.0 °F ≤ T_{AVG} ≤ 580.0 + 4.0 °F
• Pressurizer Pressure (P_{RCS})	2250 - 50 psia ≤ P_{RCS} ≤ 2250 + 50 psia
• Reactor Coolant Flow	≥ 87,200 gpm (in each of three loops)
• Accumulator Temperature (T_{ACC})	70 °F ≤ T_{ACC} ≤ 108 °F
• Accumulator Pressure (P_{ACC})	575 psia ≤ P_{ACC} ≤ 716 psia
• Accumulator Water Volume (V_{ACC})	893 ft^3 ≤ V_{ACC} ≤ 1022 ft^3
• Accumulator Boron Concentration	≥ 2300 ppm
Accident Boundary Conditions	
• Single Failure Assumptions	Loss of one ECCS train
• Safety Injection Flow	Minimum (see Table 1A)
• Safety Injection Temperature (T_{SI})	45 °F ≤ T_{SI} ≤ 65 °F
• Safety Injection Initiation Delay Time	≤ 17 seconds (with offsite power) ≤ 27 seconds (without offsite power)
• Containment Pressure	Bounded (minimum); Figure 16

Table 1A: BVPS-1 Best-Estimate Large Break LOCA Total Minimum Injected SI Flow

HHSI and LHSI from Two Intact Loops

Reactor Coolant System (RCS) Pressure (psig)	Total Injected Flow Rate (gpm)
0	2433.0
10	2272.1
20	2106.1
50	1569.1
100	338.1
105	278.4
150	270.4
200	261.4
400	219.2
600	173.4

Table 2 - BVPS - 1 Best-Estimate Large Break LOCA Results

10 CFR 50.46 Requirement	Value	Criteria
95/95 Peak Cladding Temperature (°F)	2161	< 2200
95/95 Local Maximum Oxidation (%)	9.22	< 17
95/95 Core Wide Oxidation (%)	0.94	< 1

Table 3 - BVPS - 1 Best-Estimate Large Break Sequence of Events for the Limiting PCT Case in seconds

Start of Transient	0.0
Safety Injection Signal	4.5
Accumulator Injection Begins	9.5
Safety Injection Begins	21.5
End of Blowdown	25
Bottom of Core Recovery	32
Accumulator Empty	36.5
PCT Occurs	79
Quench Time	350
End of Transient	500

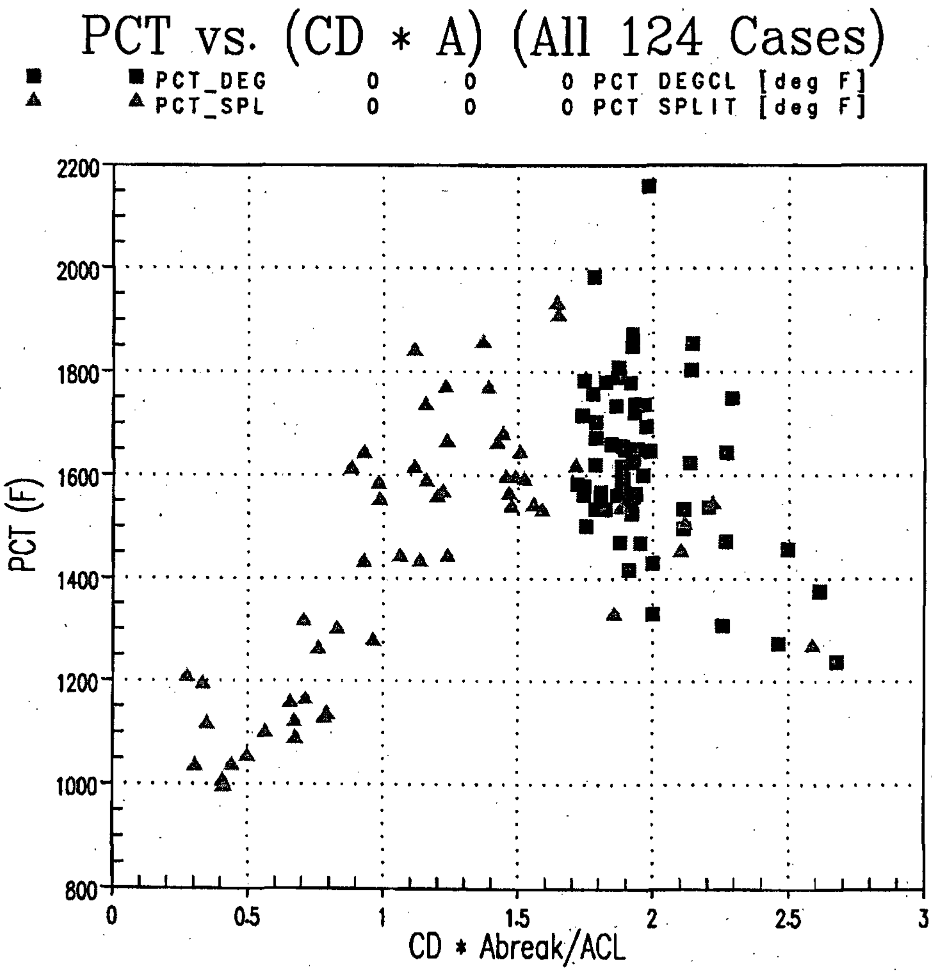

Figure 1 – HOTSPOT PCT versus Effective Break Area Scatter Plot (CD = Discharge Coefficient, Abreak = Break Area, ACL = Cold Leg Area, A = Abreak/ACL)

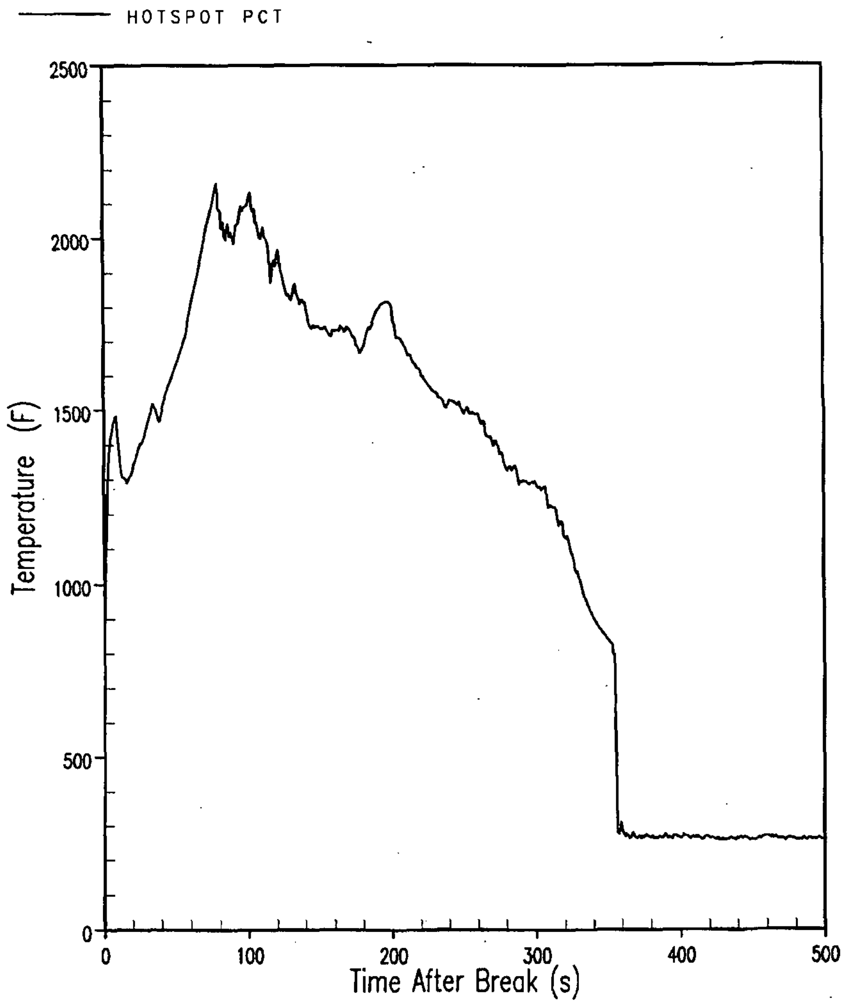

Figure 2 – HOTSPOT Clad Temperature Transient at the Limiting Elevation for the Limiting Case

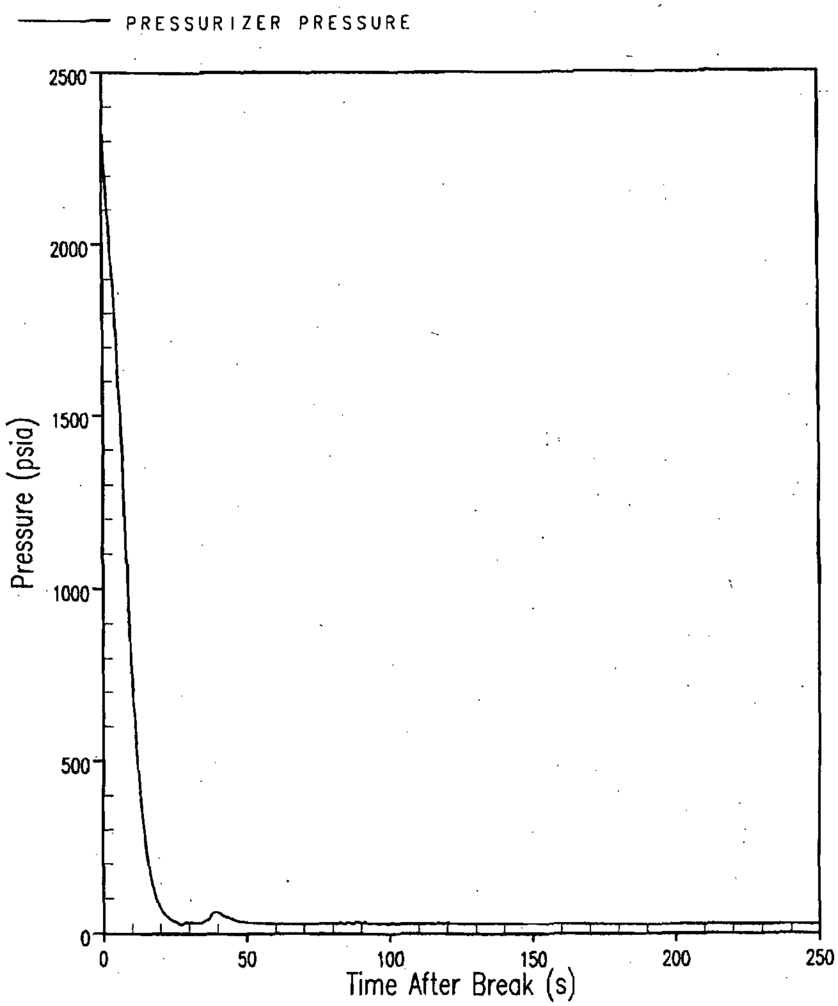

Figure 3 – Pressurizer Pressure for the Limiting PCT Case

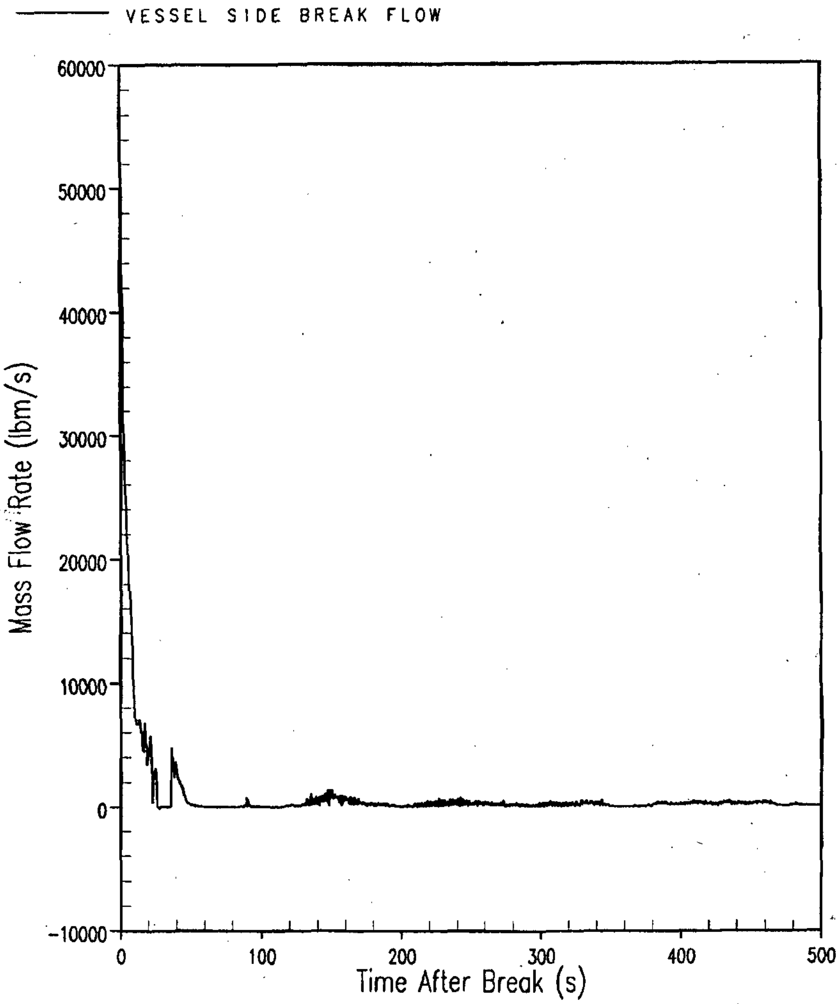

Figure 4 – Vessel Side Break Flow for the Limiting PCT Case

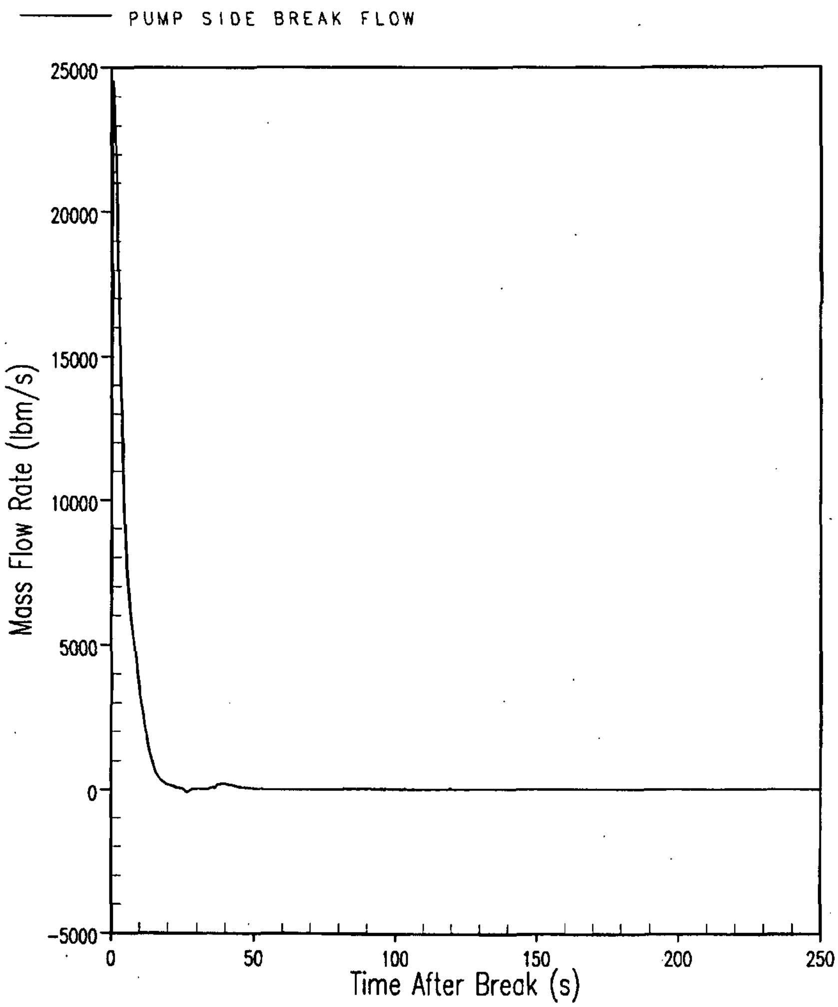

Figure 5 – Pump Side Break Flow for the Limiting PCT Case

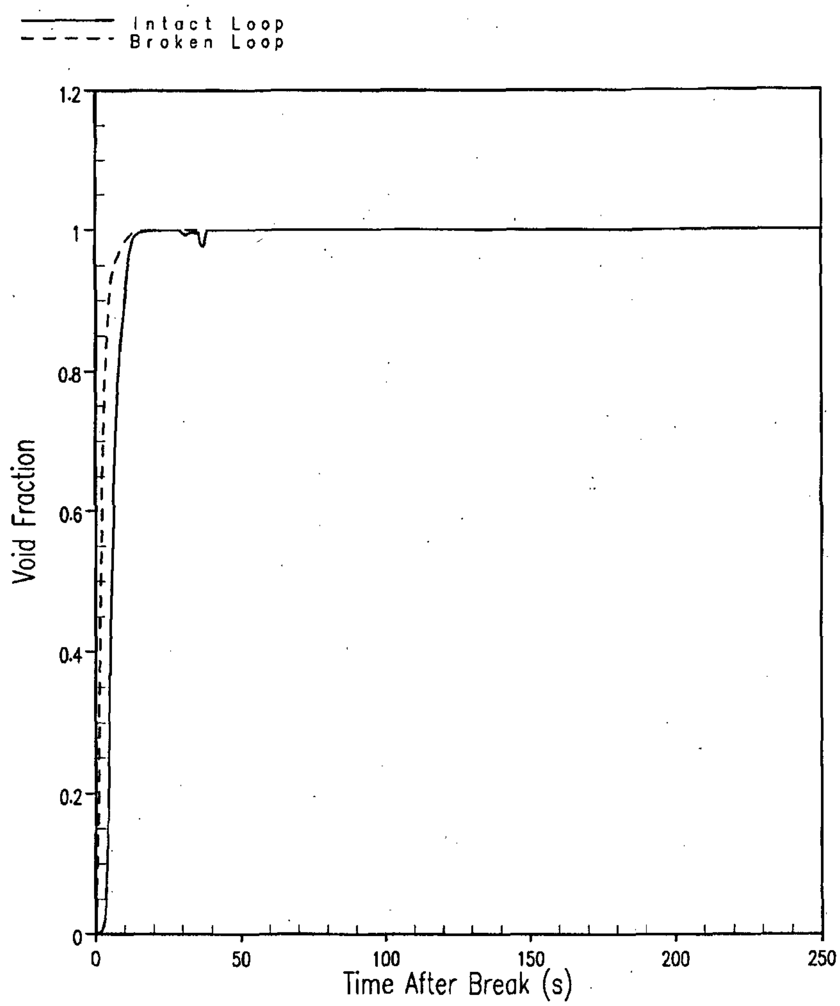

Figure 6 – Void Fraction in Pumps for the Limiting PCT Case

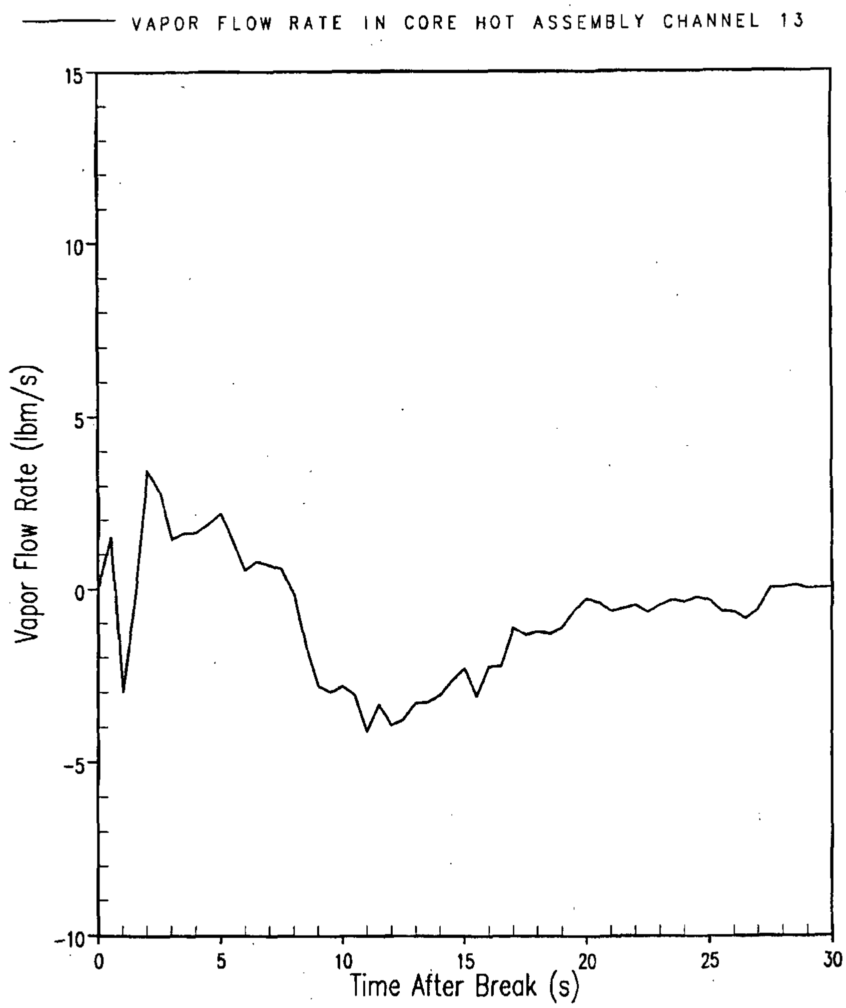

Figure 7 – Vapor Flow in the Hot Assembly Channel for the Limiting PCT Case

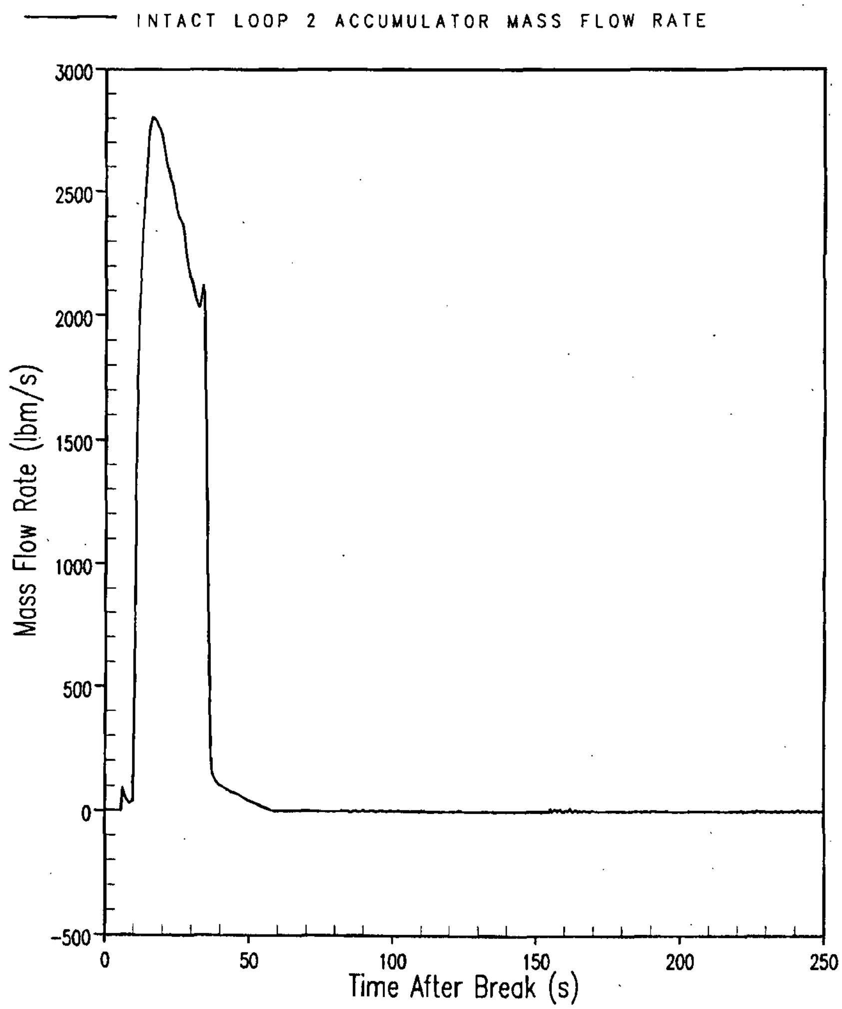

Figure 8 – Accumulator Injection Flow for the Limiting PCT Case

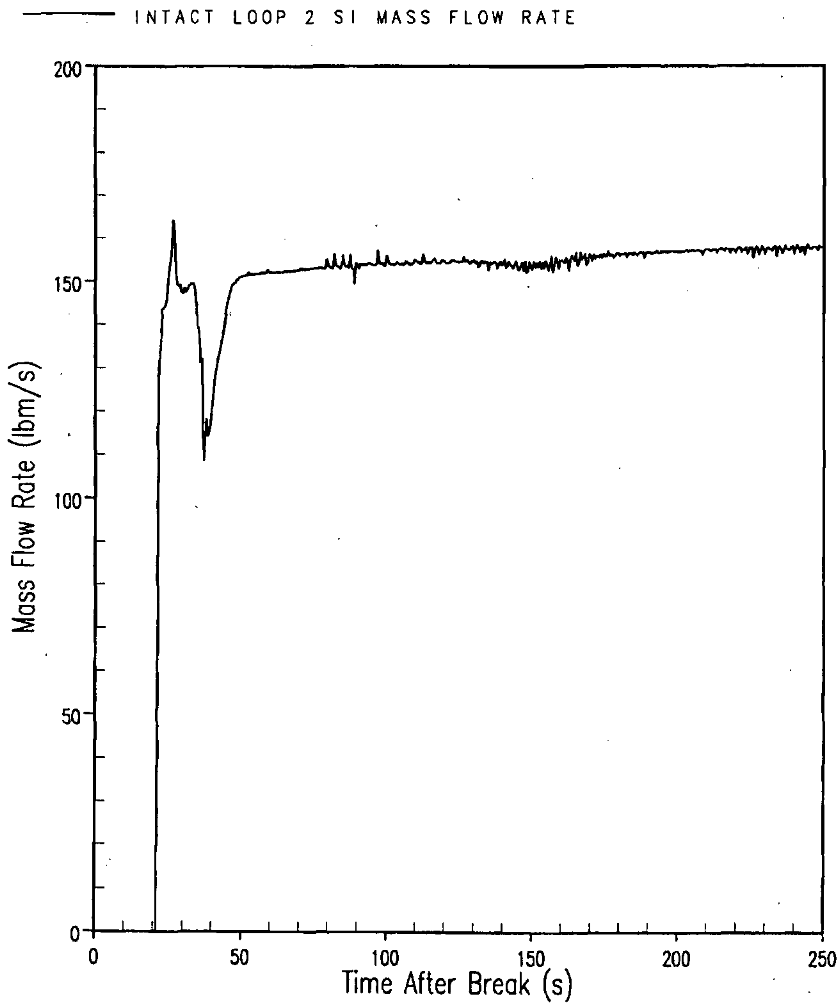

Figure 9 – Safety Injection Flow for the Limiting PCT Case

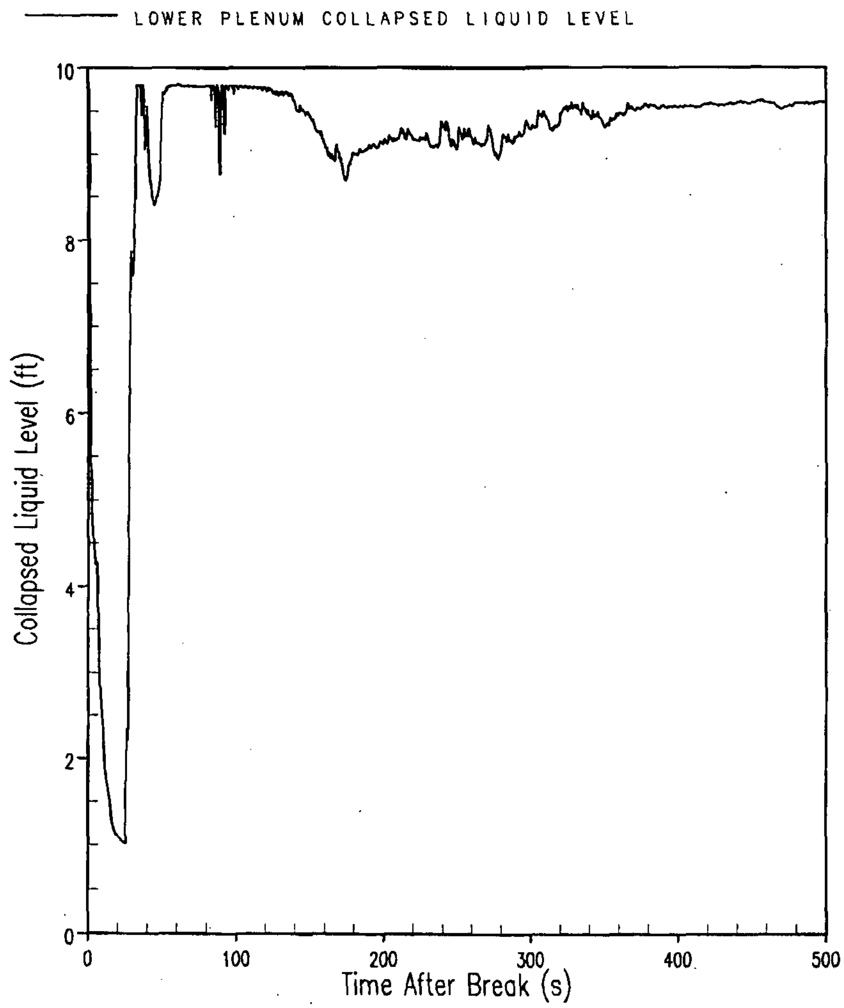

Figure 10 – Lower Plenum Collapsed Liquid Level for the Limiting PCT Case

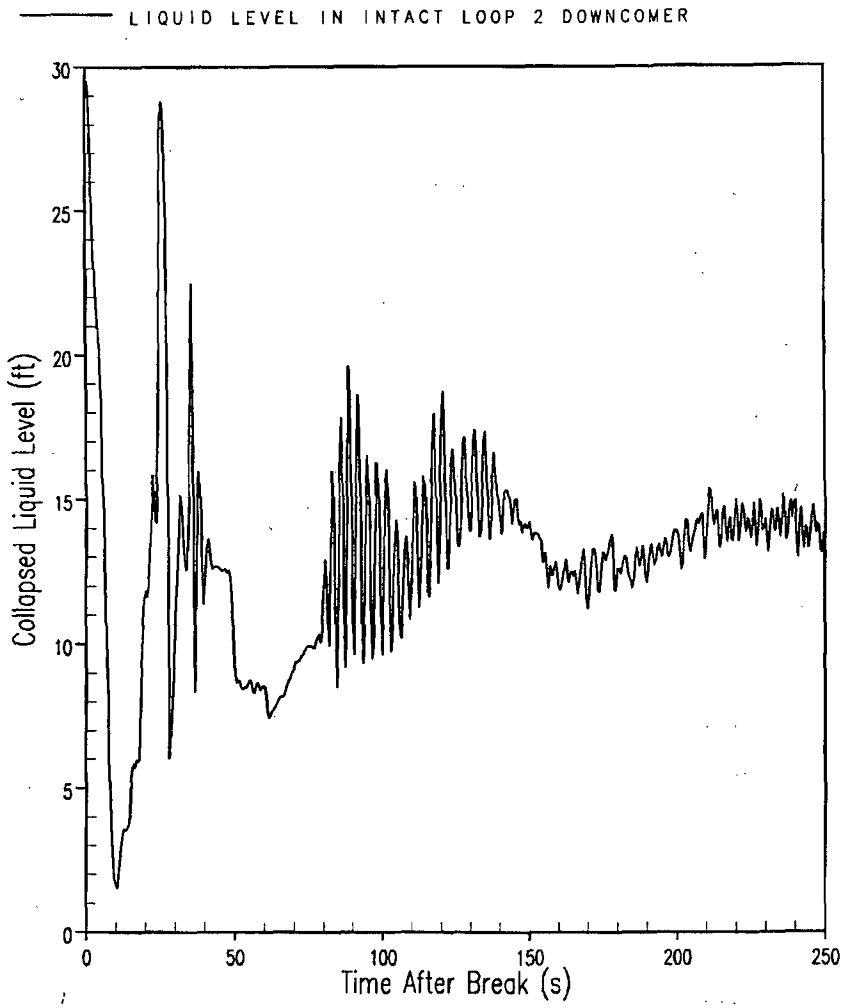

Figure 11 – Downcomer Collapsed Liquid Level for the Limiting PCT Case

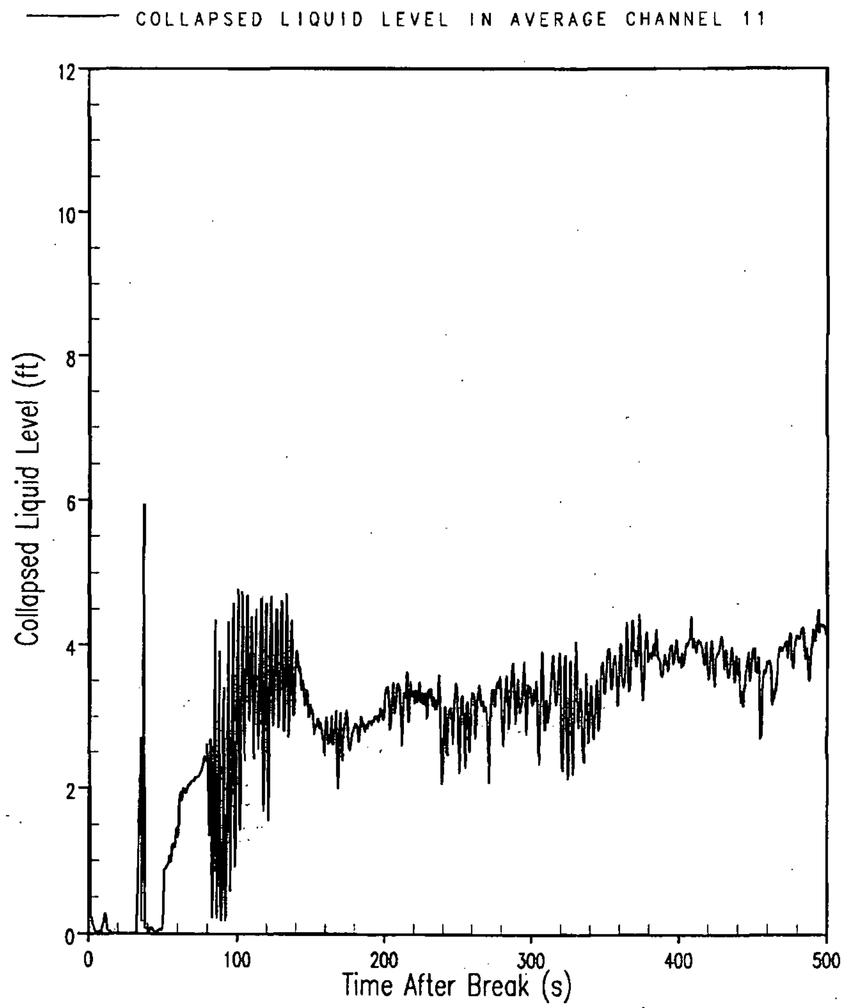

Figure 12 – Core Collapsed Liquid Level in Core Average Channel for the Limiting PCT Case

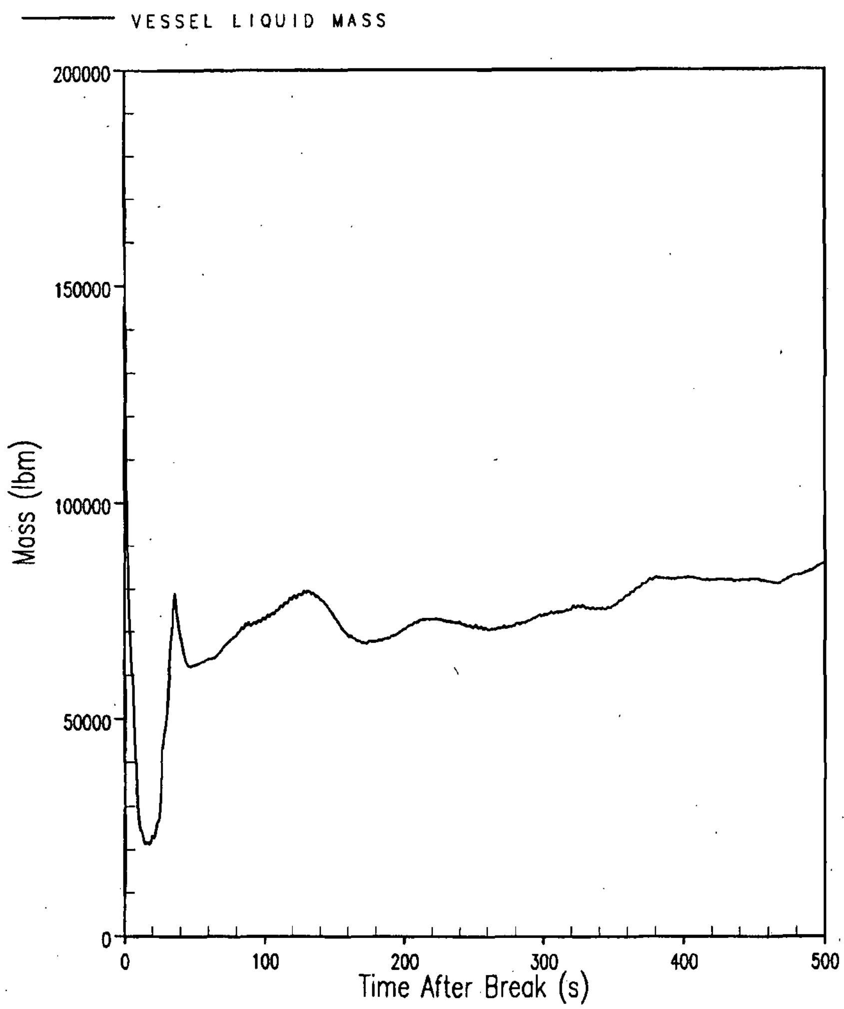

Figure 13 – Vessel Liquid Mass for the Limiting PCT Case

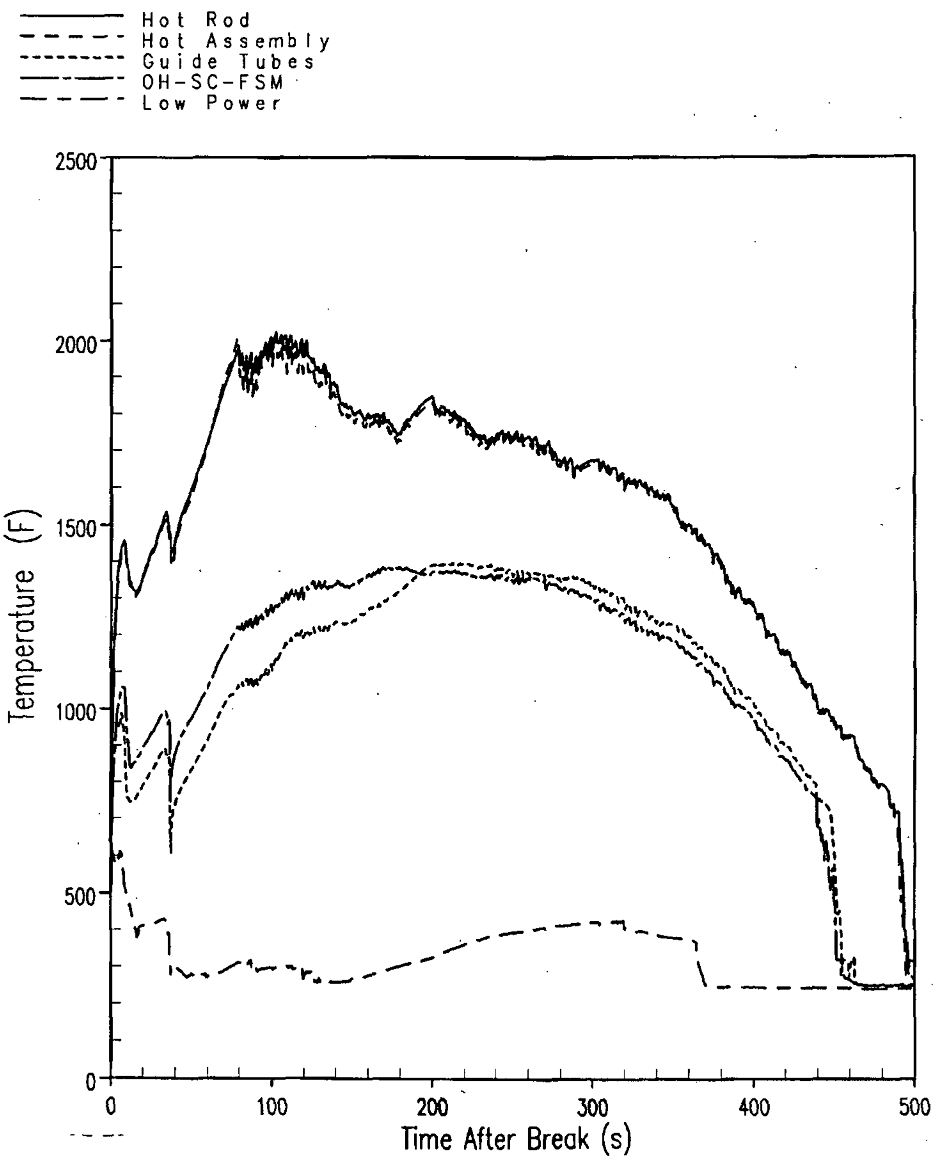

Figure 14 – WCOBRA/TRAC Peak Clad Temperature for all 5 Rod Groups for the Limiting PCT Case

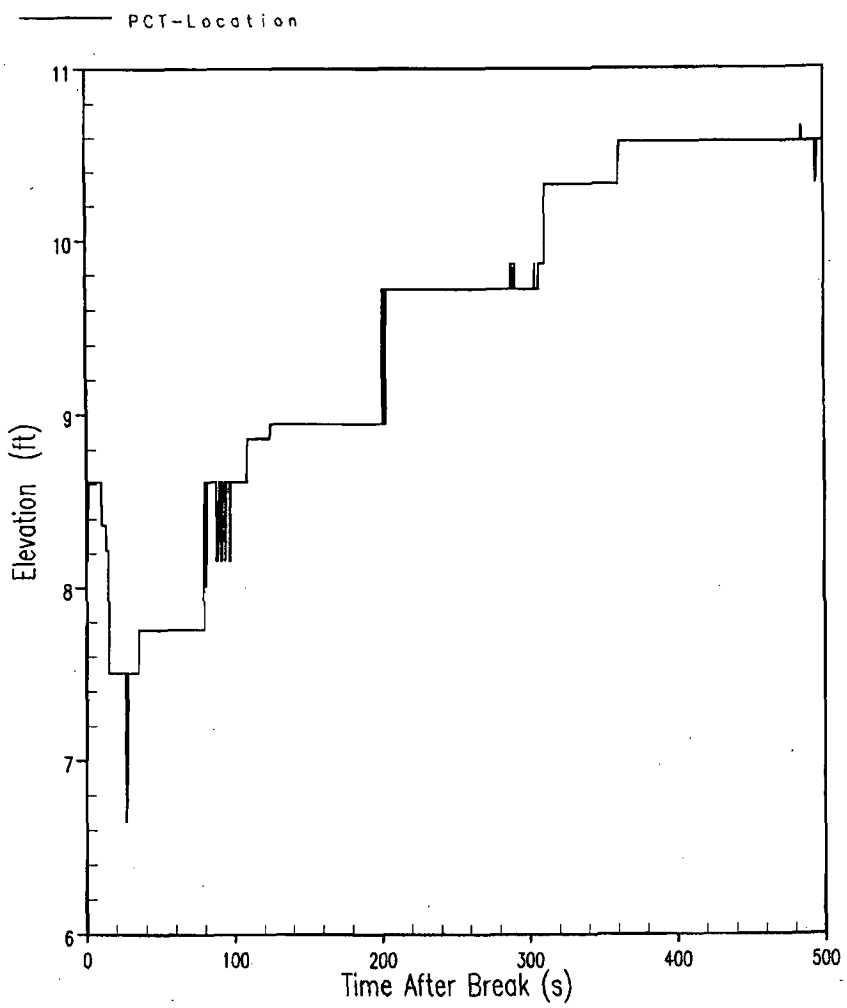

Figure 15 – Peak Clad Temperature Elevation for the Hot Rod for the Limiting PCT Case

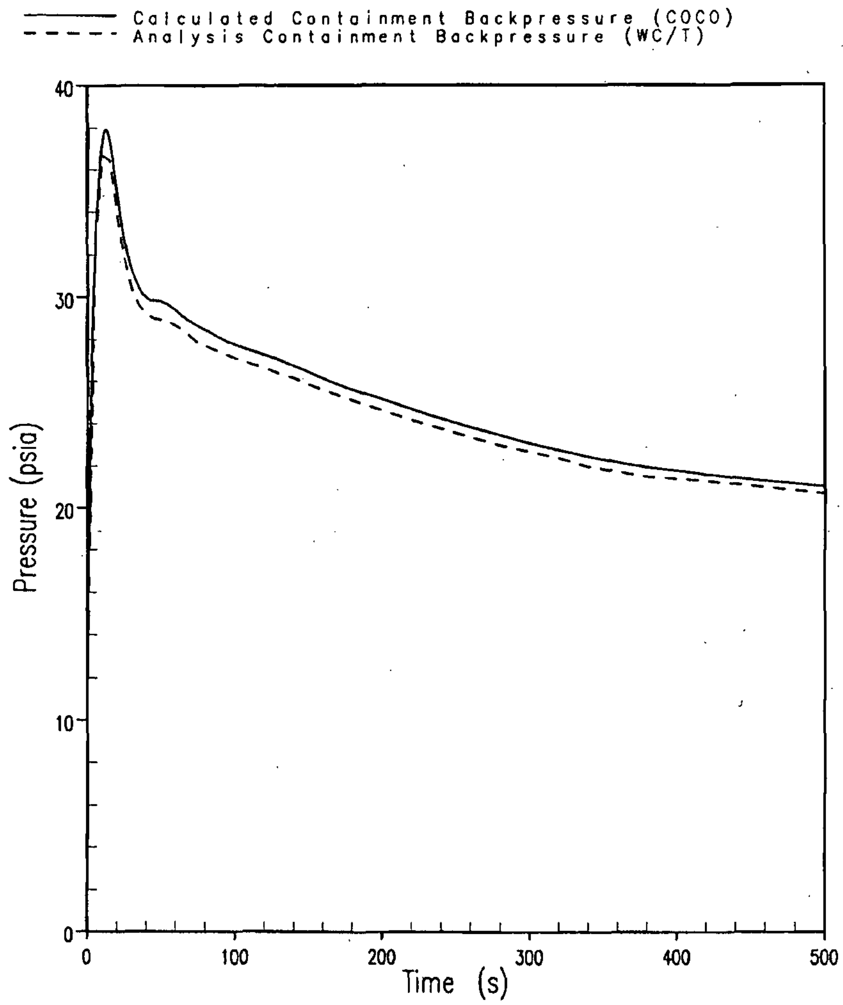

Figure 16 – WCOBRA/TRAC Assumed Backpressure versus Calculated Containment Backpressure

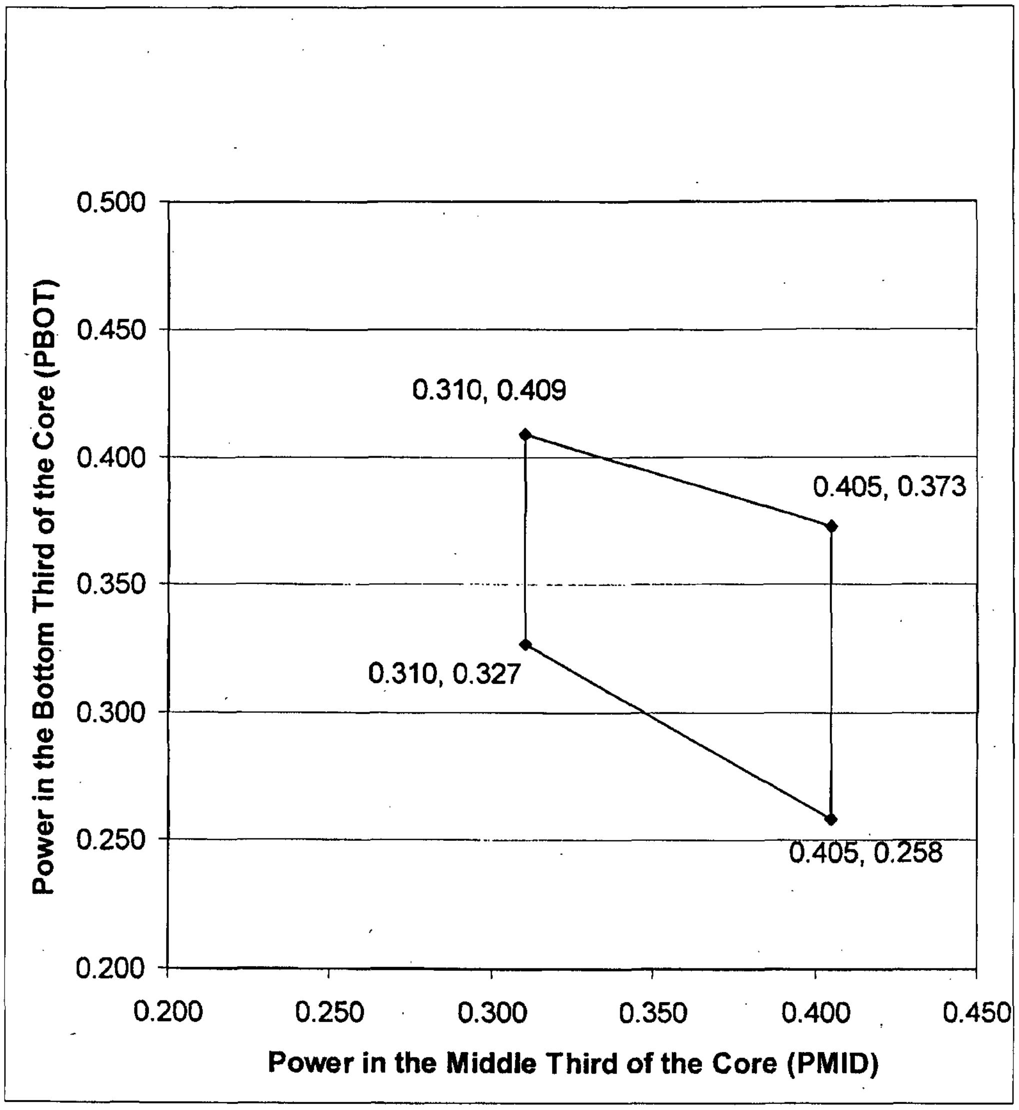

Figure 17 – BVPS - 1 BELOCA Analysis Axial Power Shape Operating Space Envelope

Attachment 1

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

License Amendment Request No. 09-004

The following is a list of the affected pages:

5.6-1 *
5.6-2
5.6-3 *

* No change. Page provided for context only.

5.0 ADMINISTRATIVE CONTROLS

5.6 Reporting Requirements

The following reports shall be submitted in accordance with 10 CFR 50.4.

5.6.1 Annual Radiological Environmental Operating Report

- NOTE -

A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station.

The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted by May 15 of each year. The report shall include summaries, interpretations, and analyses of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in the Offsite Dose Calculation Manual (ODCM), and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

5.6.2 Radioactive Effluent Release Report

- NOTE -

A single submittal may be made for a multiple unit station. The submittal shall combine sections common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.

The Radioactive Effluent Release Report covering the operation of the unit in the previous year shall be submitted prior to May 1 of each year in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR Part 50, Appendix I, Section IV.B.1.

5.6.3 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

SL 2.1.1, "Reactor Core Safety Limits"

LCO 3.1.1, "SHUTDOWN MARGIN (SDM)"

LCO 3.1.3, "Moderator Temperature Coefficient (MTC)"

5.6 Reporting Requirements

5.6.3 CORE OPERATING LIMITS REPORT (COLR) (continued)

LCO 3.1.5, "Shutdown Bank Insertion Limits"

LCO 3.1.6, "Control Bank Insertion Limits"

LCO 3.2.1, "Heat Flux Hot Channel Factor ($F_Q(Z)$)"

LCO 3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)"

LCO 3.2.3, "Axial Flux Difference (AFD)"

LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation" - Overtemperature and Overpower ΔT Allowable Value parameter values

LCO 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits"

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, specifically those described in the following documents:

WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology,"

WCAP-8745-P-A, "Design Bases for the Thermal Overtemperature ΔT and Thermal Overpower ΔT Trip Functions,"

WCAP-12945-P-A, Volumes 1 through 5, "Code Qualification Document for Best Estimate LOCA Analysis,"

(For Unit 1 only) WCAP-16009-P-A, "Realistic Large Break LOCA Evaluation Methodology Using Automated Statistical Treatment of Uncertainty Method (ASTRUM),"

WCAP-10216-P-A, "Relaxation of Constant Axial Offset Control/ F_Q Surveillance Technical Specification,"

WCAP-14565-P-A, "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis,"

WCAP-12610-P-A, "VANTAGE+ Fuel Assembly Reference Core Report,"

WCAP-15025-P-A, "Modified WRB-2 Correlation, WRB-2M, for Predicating Critical Heat Flux in 17x17 Rod Bundles with Modified LPD Mixing Vane Grids."

As described in reference documents listed above, when an initial assumed power level of 102% of RATED THERMAL POWER is specified in a previously approved method, 100.6% of RATED THERMAL POWER may be used when input for reactor thermal power measurement of feedwater flow is by the leading edge flow meter (LEFM).

No change. Page included for context only.

5.6 Reporting Requirements

5.6.3 CORE OPERATING LIMITS REPORT (COLR) (continued)

Caldon, Inc. Engineering Report-80P, "Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM √ ™ System"

Caldon, Inc. Engineering Report-160P, "Supplement to Topical Report ER-80P: Basis for a Power Uprate with the LEFM √ ™ System"

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

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

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

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

LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," and

LCO 3.4.12, "Overpressure Protection System (OPPS)"

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

NRC Letter, "Beaver Valley Power Station, Units 1 and 2 – Acceptance of Methodology for Referencing Pressure and Temperature Limits Report (TAC Nos. MB3319 and MB3320)," dated October 8, 2002.

WCAP-14040-NP-A, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves."

The methodology listed in WCAP-14040-NP-A was used with two exceptions:

- ASME Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limits for Section XI, Division 1."

Attachment 2

Beaver Valley Power Station, Unit No. 1
Retyped Technical Specification Replacement Pages
License Amendment Request No. 09-004

The following is a list of the affected pages:

5.6-2

5.6 Reporting Requirements

5.6.3 CORE OPERATING LIMITS REPORT (COLR) (continued)

LCO 3.1.5, "Shutdown Bank Insertion Limits"

LCO 3.1.6, "Control Bank Insertion Limits"

LCO 3.2.1, "Heat Flux Hot Channel Factor ($F_Q(Z)$)"

LCO 3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)"

LCO 3.2.3, "Axial Flux Difference (AFD)"

LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation" - Overtemperature and Overpower ΔT Allowable Value parameter values

LCO 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits"

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, specifically those described in the following documents:

WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology,"

WCAP-8745-P-A, "Design Bases for the Thermal Overtemperature ΔT and Thermal Overpower ΔT Trip Functions,"

WCAP-12945-P-A, Volumes 1 through 5, "Code Qualification Document for Best Estimate LOCA Analysis,"

(For Unit 1 only) WCAP-16009-P-A, "Realistic Large Break LOCA Evaluation Methodology Using Automated Statistical Treatment of Uncertainty Method (ASTRUM),"

WCAP-10216-P-A, "Relaxation of Constant Axial Offset Control/ F_Q Surveillance Technical Specification,"

WCAP-14565-P-A, "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis,"

WCAP-12610-P-A, "VANTAGE+ Fuel Assembly Reference Core Report,"

WCAP-15025-P-A, "Modified WRB-2 Correlation, WRB-2M, for Predicating Critical Heat Flux in 17x17 Rod Bundles with Modified LPD Mixing Vane Grids."

As described in reference documents listed above, when an initial assumed power level of 102% of RATED THERMAL POWER is specified in a previously approved method, 100.6% of RATED THERMAL POWER may be used when input for reactor thermal power measurement of feedwater flow is by the leading edge flow meter (LEFM).

Attachment 3

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

License Amendment Request No. 09-004

The following is a list of the affected pages:

B 3.5.1-2
B 3.5.1-3

BASES

APPLICABLE SAFETY ANALYSES

The accumulators are assumed to be OPERABLE in both the large and small break LOCA analyses at full power and hot zero power (HZP) steam line break (SLB) analysis (Ref. 1). These are the Design Basis Accidents (DBAs) that establish the acceptance limits for the accumulators. Reference to the analyses for these DBAs is used to assess changes in the accumulators as they relate to the acceptance limits.

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

The limiting large break LOCA is ~~a split break in the cold leg (Unit 1) and~~ a double ended guillotine break in the cold leg ~~(Unit 2)~~ for both Units 1 and 2. During this event, the accumulators discharge to the RCS as soon as RCS pressure decreases to below accumulator pressure.

No credit is taken for ECCS pump flow in the analysis until full flow is available. If offsite power is not available, the analysis accounts for the diesels starting and the pumps being loaded and delivering full flow. 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 charging pumps both play a part in terminating the rise in clad temperature. As break size continues to decrease, the role of the accumulators continues to decrease until they are not required and the charging 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 (Ref. 2) will be met following a LOCA:

a. Maximum fuel element cladding temperature is ≤ 2200°F, APPLICABLE SAFETY ANALYSES (continued)

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

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

d. Core is maintained in a coolable geometry.

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

For both the large and small break LOCA analyses, a nominal contained accumulator water volume is used. The nominal water volume assumed in the analyses is within the range of accumulator volumes specified in Surveillance Requirement 3.5.1.2. The contained water volume is not the same as the usable volume of the accumulators, since the accumulators are not completely emptied after discharge. 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. Therefore, the large break LOCA analyses ~~also~~ use a range of accumulator volumes. The Unit 1 ASTRUM large break LOCA analysis statistically calculates the accumulator water volume over the range of accumulator volumes specified in Surveillance Requirement 3.5.1.2. ~~For Unit 1, the large break LOCA analysis assumes values of 6681 gallons and 7645 gallons for accumulator volume.~~ For Unit 2, the large break LOCA analysis assumes values of 6898 gallons and 8019 gallons for accumulator volume. The large break LOCA analyses also credit the line water volume from the accumulator to the check valve.

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

The small break LOCA analysis is performed at the minimum nitrogen cover pressure, since sensitivity analyses have demonstrated that a higher nitrogen cover pressure results in a computed peak clad

Attachment 4

Beaver Valley Power Station, Unit No. 1
Proposed Licensing Requirements Manual Changes

License Amendment Request No. 09-004

The following is a list of the affected pages:

5.1-7

5.1 Core Operating Limits Report

5.1.12 References

1. WCAP-9272-P-A, "WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY," July 1985 (Westinghouse Proprietary).

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

3. WCAP-12945-P-A, Volume 1 (Revision 2) and Volumes 2 through 5 (Revision 1), "Code Qualification Document for Best Estimate LOCA Analysis," March 1998 (Westinghouse Proprietary).

4. WCAP-10216-P-A, Revision 1A, "Relaxation of Constant Axial Offset Control-F_Q Surveillance Technical Specification," February 1994.

5. WCAP-14565-P-A, "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis," October 1999.

6. WCAP-12610-P-A, "VANTAGE+ Fuel Assembly Reference Core Report," April 1995 (Westinghouse Proprietary).

7. WCAP-15025-P-A, "Modified WRB-2 Correlation, WRB-2M, for Predicating Critical Heat Flux in 17x17 Rod Bundles with Modified LPD Mixing Vane Grids," April 1999.

8. Caldon, Inc. Engineering Report-80P, "Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM√™ System," Revision 0, March 1997.

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

10. WCAP-16009-P-A, "Realistic Large Break LOCA Evaluation Methodology Using Automated Statistical Treatment of Uncertainty Method (ASTRUM)," Revision 0, January 2005.