Exelon Generation
4300 Winfield Road
Warrenville, IL 60555

www.exeloncorp.com

Exelon Nuclear

RS-09-178

10 CFR 50.90

December 16, 2009

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

Braidwood Station, Units 1 and 2
Facility Operating License Nos. NPF-72 and NPF-77
NRC Docket Nos. STN 50-456 and STN 50-457

Byron Station, Units 1 and 2
Facility Operating License Nos. NPF-37 and NPF-66
NRC Docket Nos. STN 50-454 and STN 50-455

Subject: License Amendment Request Regarding Large Break Loss-of-Coolant Accident Analysis Methodology

References:

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

2. Letter from H. N. Berkow (U.S. NRC) to J. A. Gresham (Westinghouse Electric Company), "Final Safety Evaluation for WCAP-16009-P, Revision 0, 'Realistic Large Break LOCA Evaluation Methodology Using Automated Statistical Treatment of Uncertainty Method (ASTRUM)' (TAC No. MB9483)," dated November 5, 2004

In accordance with 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit," Exelon Generation Company, LLC (EGC) requests an amendment to Facility Operating License Nos. NPF-72 and NPF-77 for Braidwood Station, Units 1 and 2, and Facility Operating License Nos. NPF-37 and NPF-66 for Byron Station, Units 1 and 2. The proposed change revises Technical Specifications (TS) Section 5.6.5, "Core Operating Limits Report (COLR)," to replace the existing large break loss-of-coolant accident (LOCA) analysis methodology. Specifically, the proposed change adds a reference to WCAP-16009-P-A, "Realistic Large Break LOCA Evaluation Methodology Using Automated Statistical Treatment of Uncertainty Method (ASTRUM)" (i.e., Reference 1) to TS 5.6.5.b. The NRC approved WCAP-16009-P-A in Reference 2.

This request is subdivided as follows.

- Attachment 1 provides a description and evaluation of the proposed change.
- Attachment 2 provides a markup of the affected TS page for Braidwood Station.
- Attachment 3 provides a markup of the affected TS page for Byron Station.
- Attachments 4 and 5 provide a markup of the affected TS Bases page for Braidwood Station and Byron Station, respectively. The TS Bases page is provided for information only and does not require NRC approval.

The proposed change has been reviewed by the Braidwood Station and Byron Station Plant Operations Review Committees and approved by the Nuclear Safety Review Board in accordance with the requirements of the EGC Quality Assurance Program.

EGC requests approval of the proposed change by December 16, 2010. Once approved, the amendment will be implemented within 60 days. This implementation period will provide adequate time for the affected station documents to be revised using the appropriate change control mechanisms.

In accordance with 10 CFR 50.91, "Notice for public comment; State consultation," paragraph (b), EGC is notifying the State of Illinois of this application for license amendment by transmitting a copy of this letter and its attachments to the designated State Official.

There are no regulatory commitments contained in this letter. Should you have any questions concerning this letter, please contact Mr. Kenneth M. Nicely at (630) 657-2803.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 16th day of December 2009.

Respectfully,

Patrick R. Simpson
Manager – Licensing

Attachments:
1. Evaluation of Proposed Change
2. Markup of Proposed Technical Specifications Page for Braidwood Station
3. Markup of Proposed Technical Specifications Page for Byron Station
4. Markup of Proposed Technical Specifications Bases Page for Braidwood Station
5. Markup of Proposed Technical Specifications Bases Page for Byron Station

cc: NRC Regional Administrator, Region III
NRC Senior Resident Inspector – Braidwood Station
NRC Senior Resident Inspector – Byron Station
Illinois Emergency Management Agency – Division of Nuclear Safety

ATTACHMENT 1
Evaluation of Proposed Change

1.0 SUMMARY DESCRIPTION

2.0 DETAILED DESCRIPTION

3.0 TECHNICAL EVALUATION

- 3.1 Methodology Background
- 3.2 Description of a LBLOCA Transient
- 3.3 Realistic LBLOCA Analyses Results
- 3.4 Conclusions

4.0 REGULATORY EVALUATION

- 4.1 Applicable Regulatory Requirements/Criteria
- 4.2 No Significant Hazards Consideration
- 4.3 Conclusions

5.0 ENVIRONMENTAL CONSIDERATION

6.0 REFERENCES

1.0 SUMMARY DESCRIPTION

In accordance with 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit," Exelon Generation Company, LLC (EGC) requests an amendment to Facility Operating License Nos. NPF-72 and NPF-77 for Braidwood Station, Units 1 and 2, and Facility Operating License Nos. NPF-37 and NPF-66 for Byron Station, Units 1 and 2. The proposed change revises Technical Specifications (TS) Section 5.6.5, "Core Operating Limits Report (COLR)," to replace the existing large break loss-of-coolant accident (LOCA) analysis methodology. Specifically, the proposed change adds a reference to WCAP-16009-P-A, "Realistic Large Break LOCA Evaluation Methodology Using Automated Statistical Treatment of Uncertainty Method (ASTRUM)" (i.e., Reference 1) to TS 5.6.5.b. The NRC approved WCAP-16009-P-A in Reference 2.

2.0 DETAILED DESCRIPTION

TS Section 5.6.5.a requires core operating limits to be established and documented in the COLR prior to each reload cycle, or prior to any remaining portion of a reload cycle. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, as listed in TS Section 5.6.5.b. The current methodology used for development of core operating limits related to the large break LOCA (LBLOCA) is listed in TS Section 5.6.5.b.6, which states:

6. 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.

The proposed change replaces the reference to WCAP-12945-P-A with a reference to WCAP-16009-P-A, as follows:

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

3.0 TECHNICAL EVALUATION

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 appended to the topical report (i.e., Reference 3).

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 (i.e., Reference 3) and follows the steps in the Code Scaling Applicability and Uncertainty (CSAU) methodology. However, the uncertainty analysis (i.e., 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 (i.e., Reference 1).

A best estimate LBLOCA analysis was completed for Braidwood/Byron Unit 1, and a separate best estimate LBLOCA analysis was completed for Braidwood/Byron Unit 2. Separate analyses were necessary due to differences in steam generator design. The application of the Westinghouse ASTRUM best estimate LOCA (BELOCA) evaluation model for the LBLOCA analyses is summarized below. Table 1 lists the major plant parameter assumptions used in the BELOCA analyses. Both EGC and the analysis vendor (i.e., 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.

Table 1: Major Plant Parameter Assumptions Used in the BELOCA Analyses

Parameter	Value
Plant Physical Description	
• Steam Generator Tube Plugging	≤ 5% (Byron/Braidwood Unit 1) ≤ 10% (Byron/Braidwood Unit 2)
Plant Initial Operating Conditions	
• Reactor Power	≤ 3658.33 MWt (+ 0% uncertainties)
• Peaking Factors	$F_Q \leq 2.6$ $F_{\Delta H} \leq 1.70$
• Axial Power Distribution	See Figures 19-1 and 19-2
Fluid Conditions	
• T_{AVG}	575.0 - 10.0 °F ≤ T_{AVG} ≤ 588.0 + 10.0 °F
• Pressurizer Pressure	2250 - 43 psia ≤ P_{RCS} ≤ 2250 + 43 psia
• Reactor Coolant Flow	≥ 92,000 gpm per loop
• Accumulator Temperature	60 °F ≤ T_{ACC} ≤ 130 °F
• Accumulator Pressure	587 psia ≤ P_{ACC} ≤ 692 psia [Note]
• Accumulator Water Volume	920 ft^3 ≤ V_{ACC} ≤ 980 ft^3
• Accumulator Boron Concentration	≥ 2200 ppm
Accident Boundary Conditions	
• Single Failure Assumptions	Loss of one Emergency Core Cooling System (ECCS) train
• Safety Injection Flow	Minimum
• Safety Injection Temperature	32 °F ≤ T_{SI} ≤ 120 °F [Note]
• Safety Injection Initiation Delay Time	≤ 27 sec (with offsite power) ≤ 40 sec (without offsite power)
• Containment Pressure	Bounded (minimum)

Note: The ranges for these values have been expanded to provide additional operating margin.

3.1 Methodology Background

When the final acceptance criteria (FAC) governing the LOCA for light water reactors was issued in 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors" (i.e., Reference 4), both the NRC and the industry recognized that stipulations of 10 CFR 50, Appendix K, "ECCS Evaluation Models," were highly conservative. That is, using the then accepted analysis methods, the performance of the ECCS would be conservatively underestimated, resulting in predicted 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 (i.e., 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 amended the requirements of 10 CFR 50.46 and Appendix K 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 (i.e., Reference 6).

To demonstrate use of the revised ECCS rule, the NRC and its consultants developed the CSAU evaluation methodology (i.e., 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 (i.e., Reference 3) for three and four loop pressurized water reactor (PWR) plants based on the revised 10 CFR 50.46 rule was developed by Westinghouse with the support of the Electric Power Research Institute (EPRI) and Consolidated Edison, and has been approved by the NRC.

More recently, Westinghouse developed an alternative uncertainty methodology (i.e., Reference 1) called ASTRUM. This method is still based on the CQD methodology and follows the steps in the CSAU methodology. However, the uncertainty analysis (i.e., 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 was approved by the NRC, as documented in the safety evaluation appended to Reference 1. The ASTRUM methodology remains applicable to three and four loop PWRs.

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

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

This analysis is in accordance with the applicability limits and usage conditions defined in Section 13-3 of WCAP-16009-P-A as applicable to the ASTRUM methodology, and the conditions and limitations discussed in Section 4.0 of Reference 2. Section 13-3 of WCAP-16009-P-A 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 appended to this WCAP. A best estimate LBLOCA analysis, and associated model, was completed for Braidwood/Byron Unit 1. A separate best estimate LBLOCA analysis, and associated model, was completed for Braidwood/Byron Unit 2.

3.2 Description of a LBLOCA Transient

Before the break occurs, the Reactor Coolant System (RCS) is assumed to be operating normally at full power in an equilibrium condition (i.e., 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 legs. 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 Realistic LBLOCA Analyses Results

3.3.1 ASTRUM Analyses Results

The results of the ASTRUM analyses are summarized in Tables 2-1 and 2-2. Tables 3-1 and 3-2 contain a sequence of events for the limiting PCT transient.

Table 2-1: Byron/Braidwood Unit 1 Best Estimate LBLOCA Results

10 CFR 50.46 Requirement	Value	Criteria
95/95 PCT (°F)	1913	< 2200
95/95 LMO (%)	5.51	< 17
95/95 CWO (%)	0.25	< 1

Table 2-2: Byron/Braidwood Unit 2 Best Estimate LBLOCA Results

10 CFR 50.46 Requirement	Value	Criteria
95/95 PCT (°F)	2041	< 2200
95/95 LMO (%)	8.27	< 17
95/95 CWO (%)	0.33	< 1

Table 3-1: Byron/Braidwood Unit 1 Best Estimate LBLOCA Sequence of Events for the Limiting PCT Case

Event	Time (sec)
Start of Transient	0.0
Safety Injection Signal	5.8
Accumulator Injection Begins	14.0
End of Blowdown	25.5
Bottom of Core Recovery	36.0
Safety Injection Begins	45.8
Accumulator Empty	46.0
PCT Occurs	~102
End of Transient	645.0

Table 3-2: Byron/Braidwood Unit 2 Best Estimate LBLOCA Sequence of Events for the Limiting PCT Case

Event	Time (sec)
Start of Transient	0.0
Safety Injection Signal	5.7
Accumulator Injection Begins	11.5
End of Blowdown	23.5
Bottom of Core Recovery	33.0
Safety Injection Begins	45.7
Accumulator Empty	56.0
PCT Occurs	~96
End of Transient	645.0

The scatter plots presented in Figures 1-1 and 1-2 show the effect of the effective break area on the analysis 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 1-1 and 1-2 are

provided to show the break area is a significant contributor to the variation in PCT.

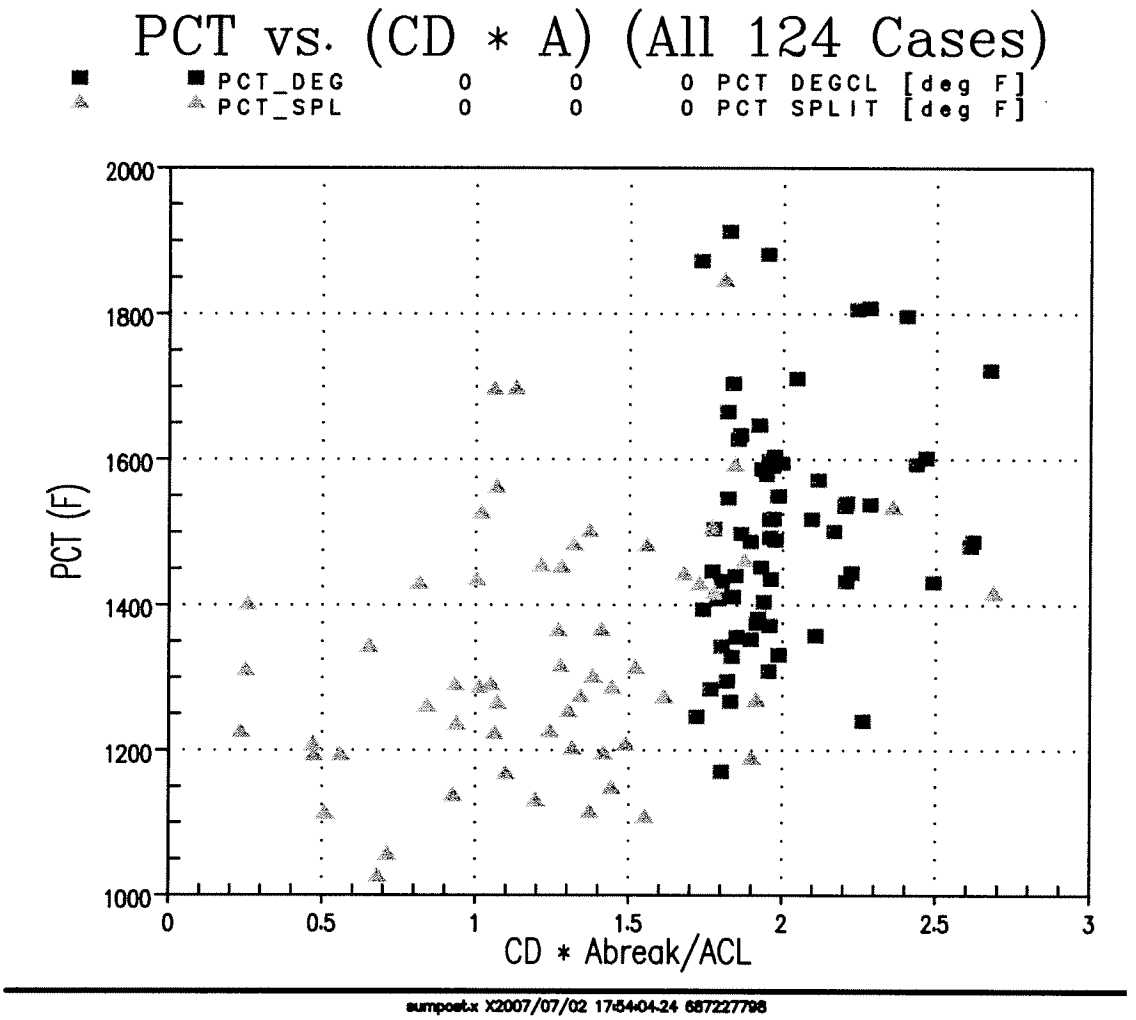


Figure 1-1: HOTSPOT PCT Versus Effective Break Area Scatter Plot for Byron/Braidwood Unit 1

CD = Discharge Coefficient
Abreak = Break Area
ACL = Cold Leg Area
PCT_DEG = PCT for Double Ended Guillotine Break
PCT_SPL = PCT for Split Break

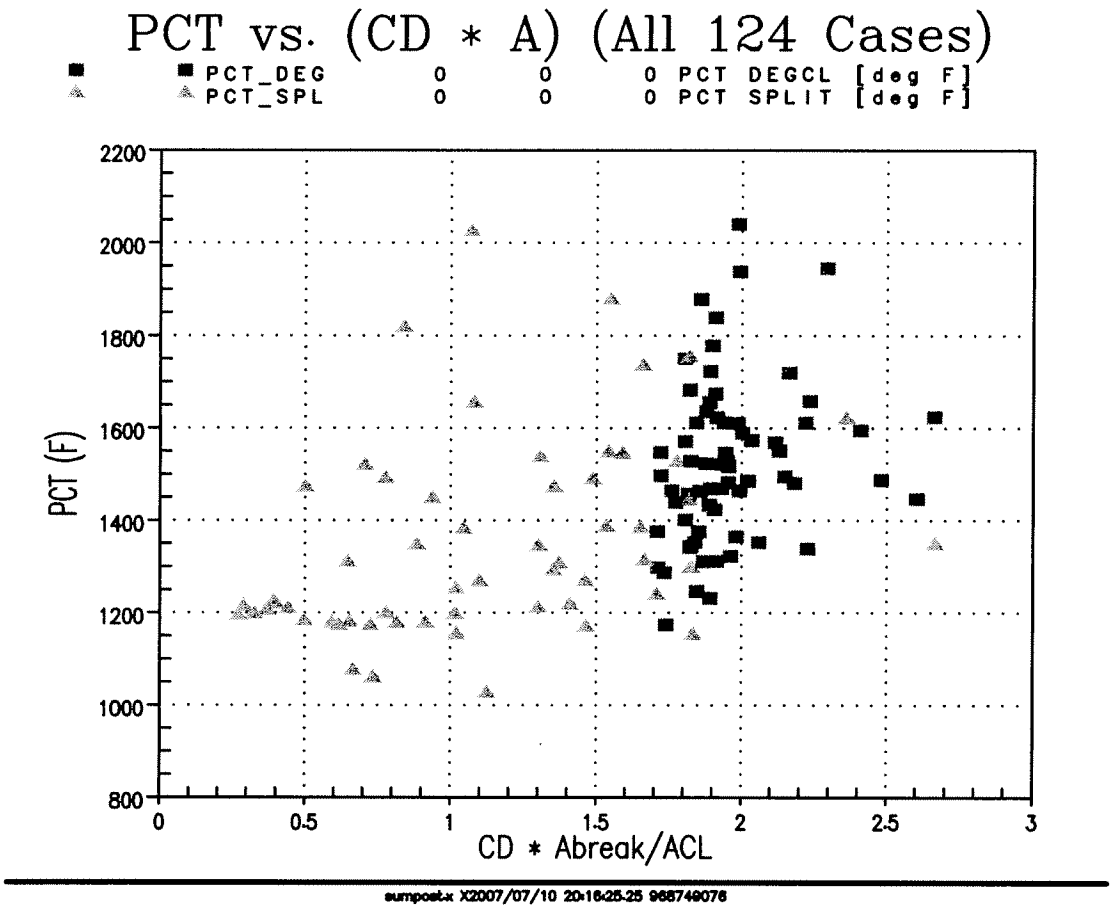


Figure 1-2: HOTSPOT PCT Versus Effective Break Area Scatter Plot for Byron/Braidwood Unit 2

CD = Discharge Coefficient
Abreak = Break Area
ACL = Cold Leg Area
PCT_DEG = PCT for Double Ended Guillotine Break
PCT_SPL = PCT for Split Break

From the 124 calculations performed as part of the ASTRUM analyses, different cases proved to be the limiting PCT and limiting LMO transient. Figures 2-1 and 2-2 show the predicted clad temperature transient at the PCT limiting elevation for the limiting PCT case. Figures 3-1 and 3-2 present the clad temperature transient predicted at the LMO elevation for the limiting LMO case.

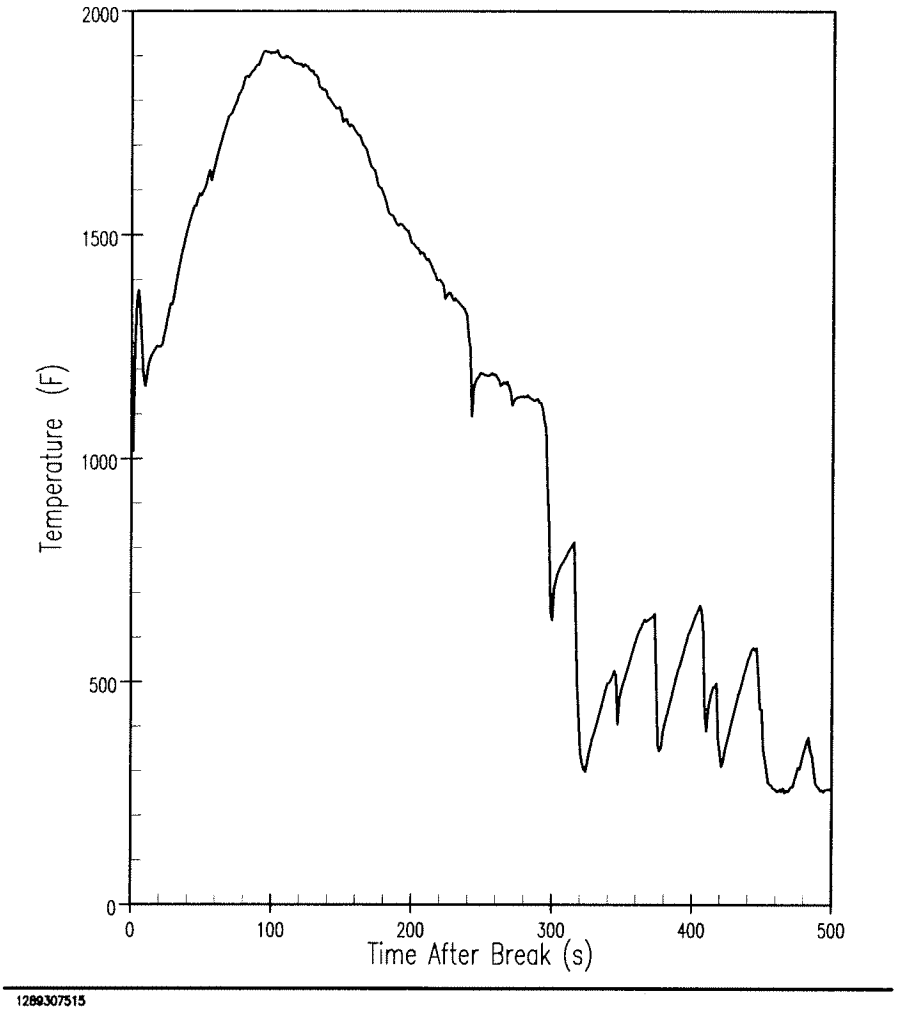


Figure 2-1: HOTSPOT Clad Temperature Transient at the Limiting Elevation for the Limiting PCT Case for Byron/Braidwood Unit 1

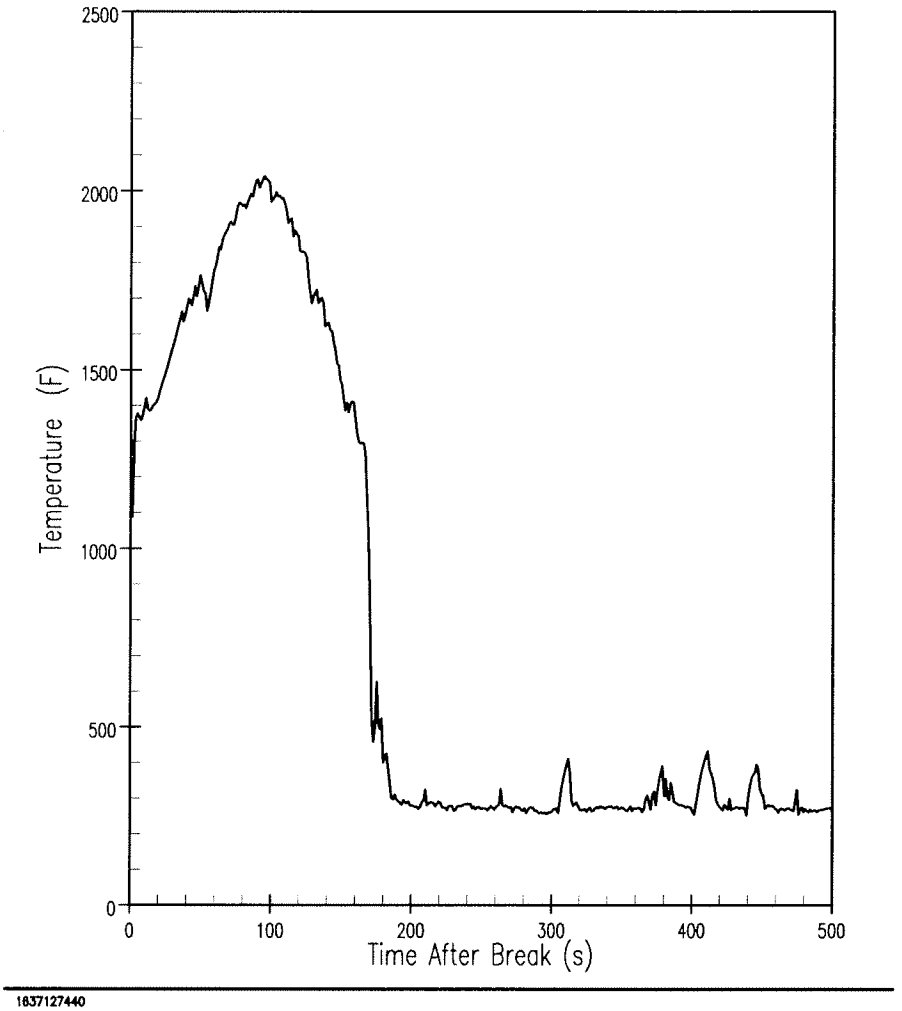


Figure 2-2: HOTSPOT Clad Temperature Transient at the Limiting Elevation for the Limiting PCT Case for Byron/Braidwood Unit 2

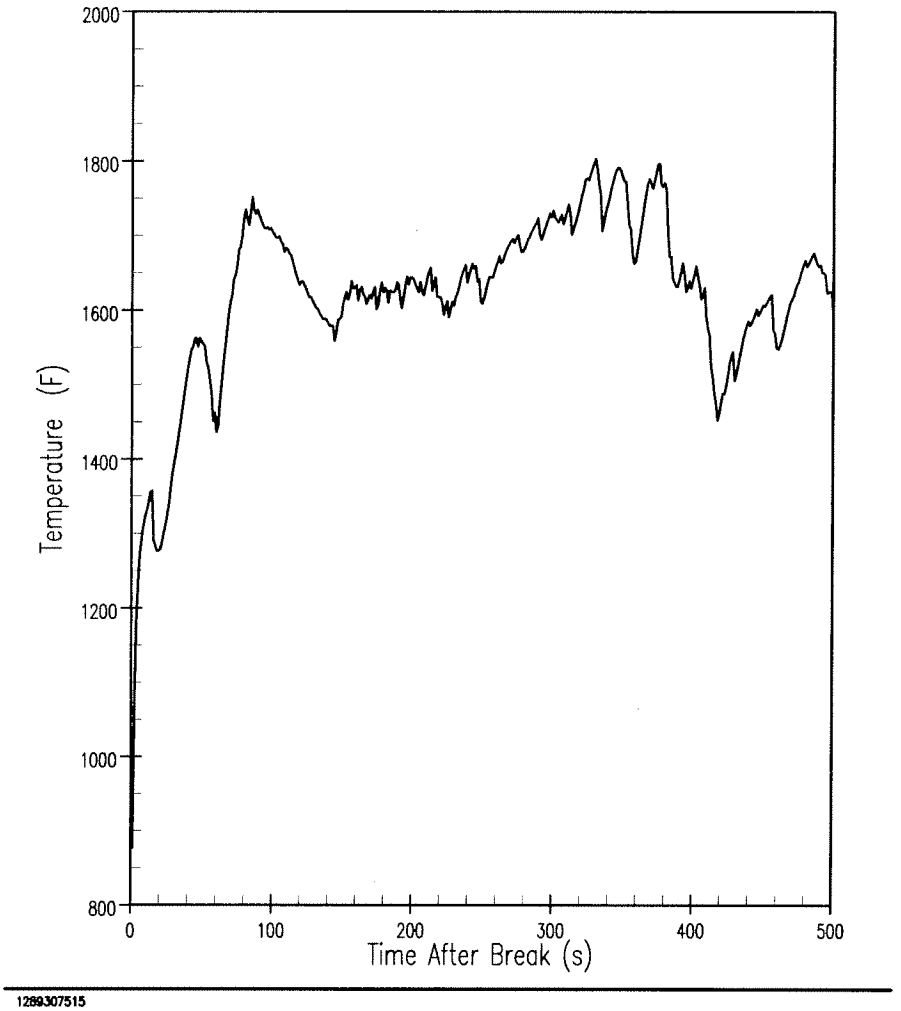


Figure 3-1: HOTSPOT Clad Temperature Transient at the Limiting Elevation for the Limiting LMO Case for Byron/Braidwood Unit 1

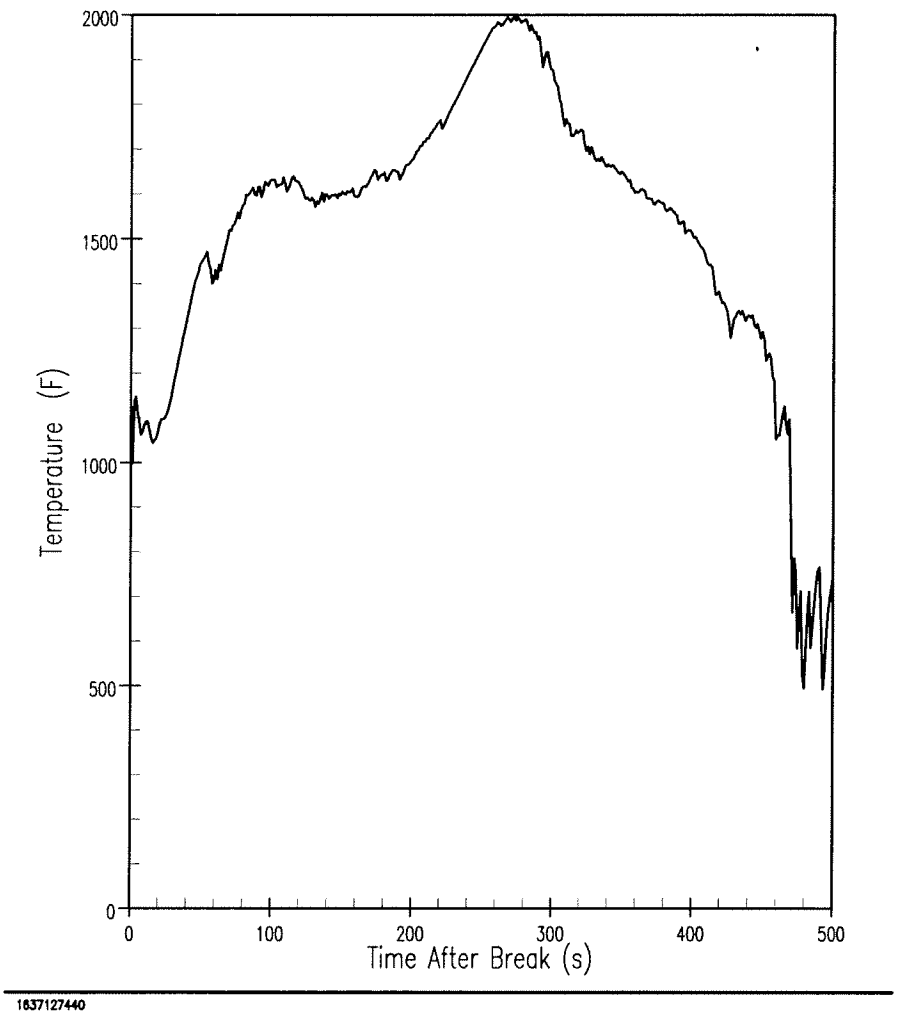
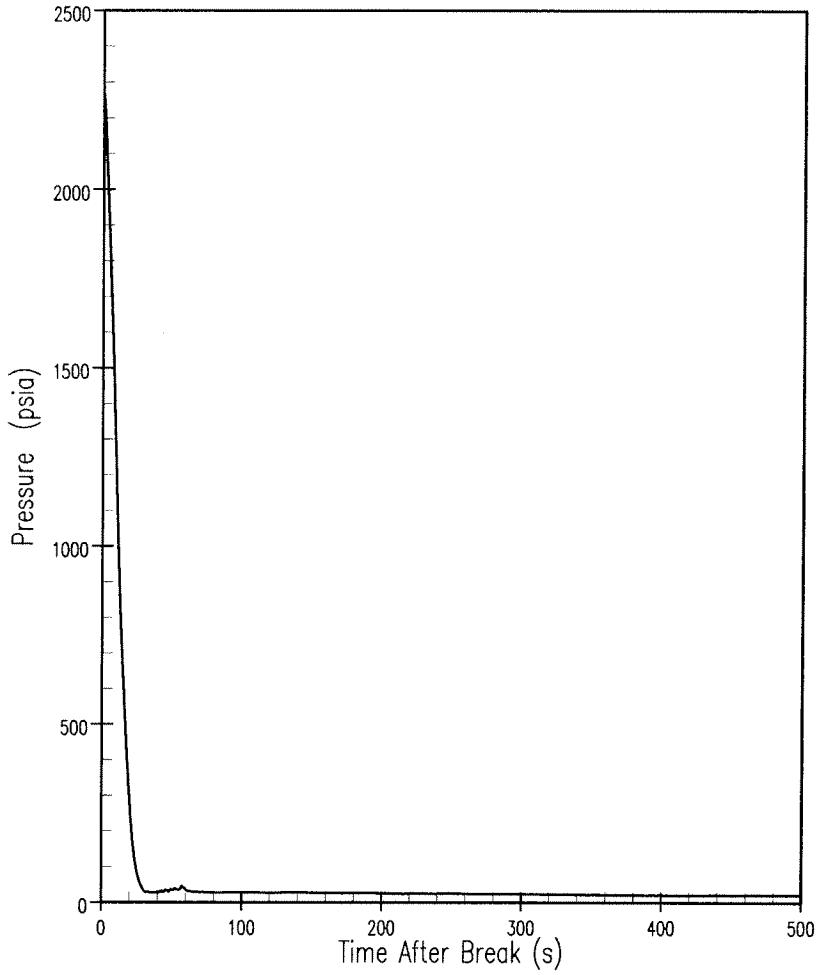


Figure 3-2: HOTSPOT Clad Temperature Transient at the Limiting Elevation for the Limiting LMO Case for Byron/Braidwood Unit 2

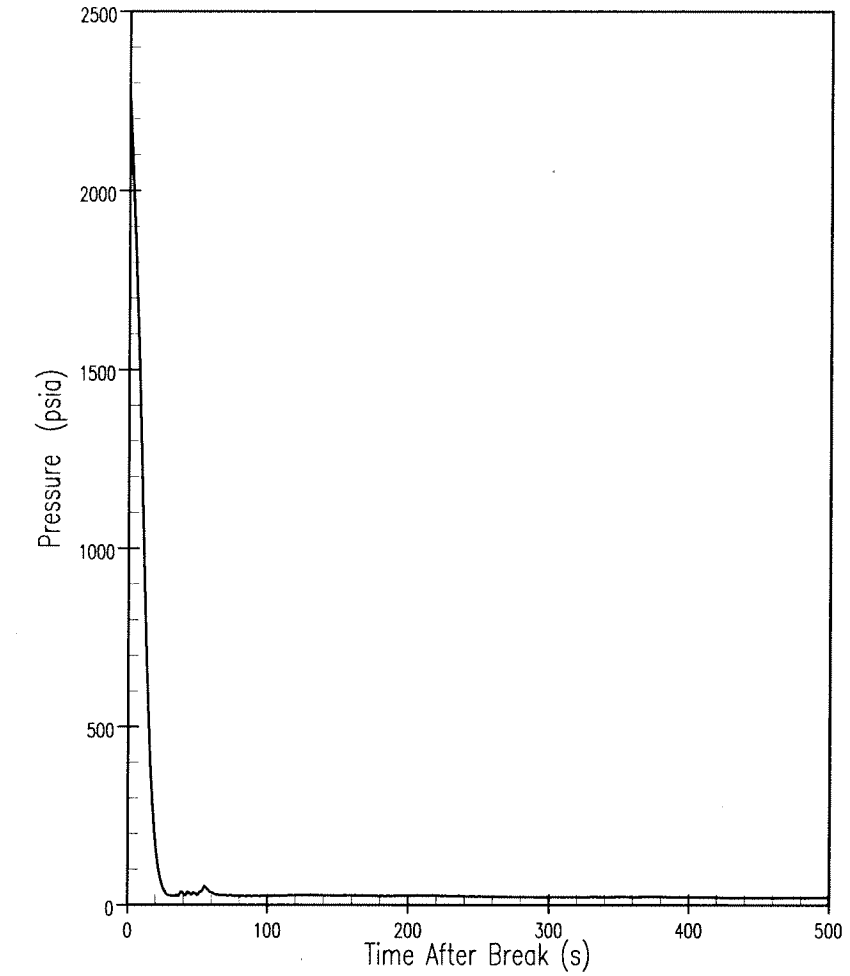
Figures 4-1 through 17-1 for Byron/Braidwood Unit 1, and Figures 4-2 through 17-2 for Byron/Braidwood Unit 2, illustrate the key major response parameters for the limiting PCT transient. The reference point for the lower plenum liquid level presented in Figures 11-1 and 11-2 is the bottom of the vessel. The reference point for the downcomer liquid level presented in Figures 12-1 and 12-2 is the point at which the outside of the core barrel, if extended downward, intersects with the vessel wall. The reference point for the core collapsed liquid levels presented in Figures 13-1, 13-2, 16-1, and 16-2 is the bottom of the active fuel.

The containment backpressure utilized for the LBLOCA analyses compared to the calculated containment backpressure is provided in Figures 18-1 and 18-2. The worst single failure for the LBLOCA analyses is the loss of one train of ECCS injection, consistent with Reference 1. However, all containment systems which would reduce containment pressure are modeled for the LBLOCA containment backpressure calculation.



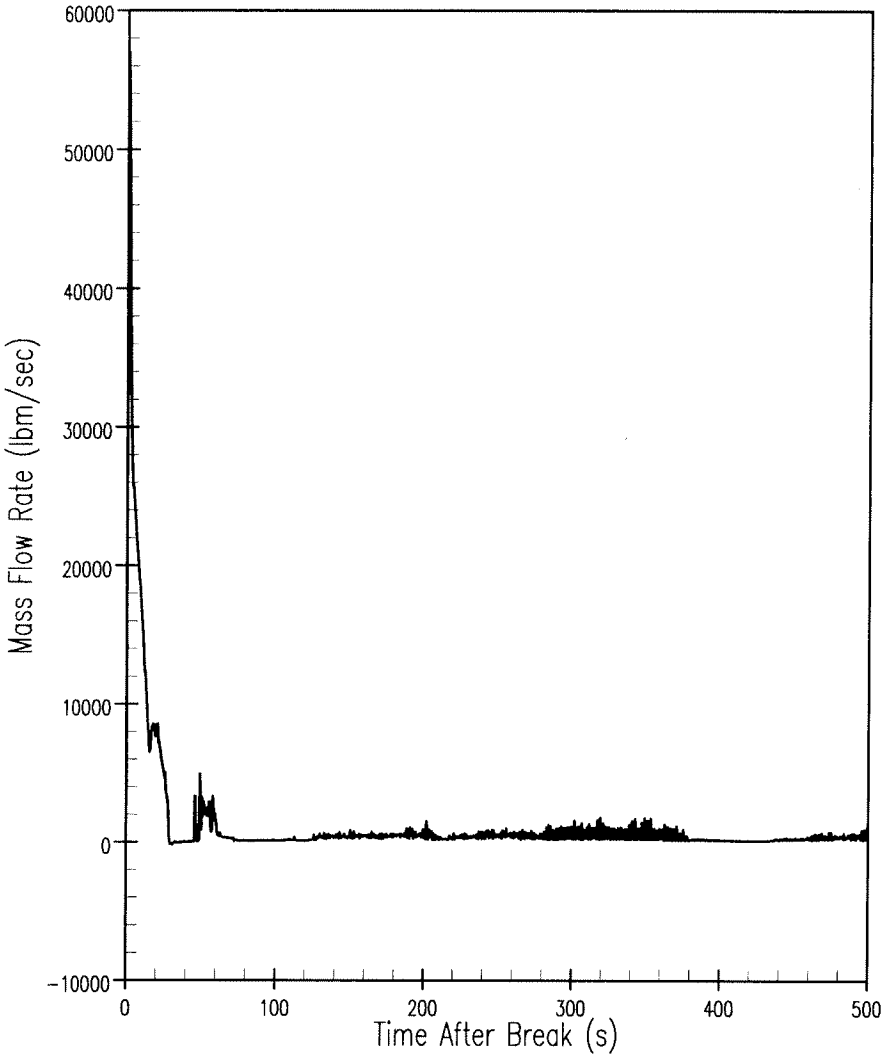
1289307515

Figure 4-1: Pressurizer Pressure for the Limiting PCT Case for Byron/Braidwood Unit 1



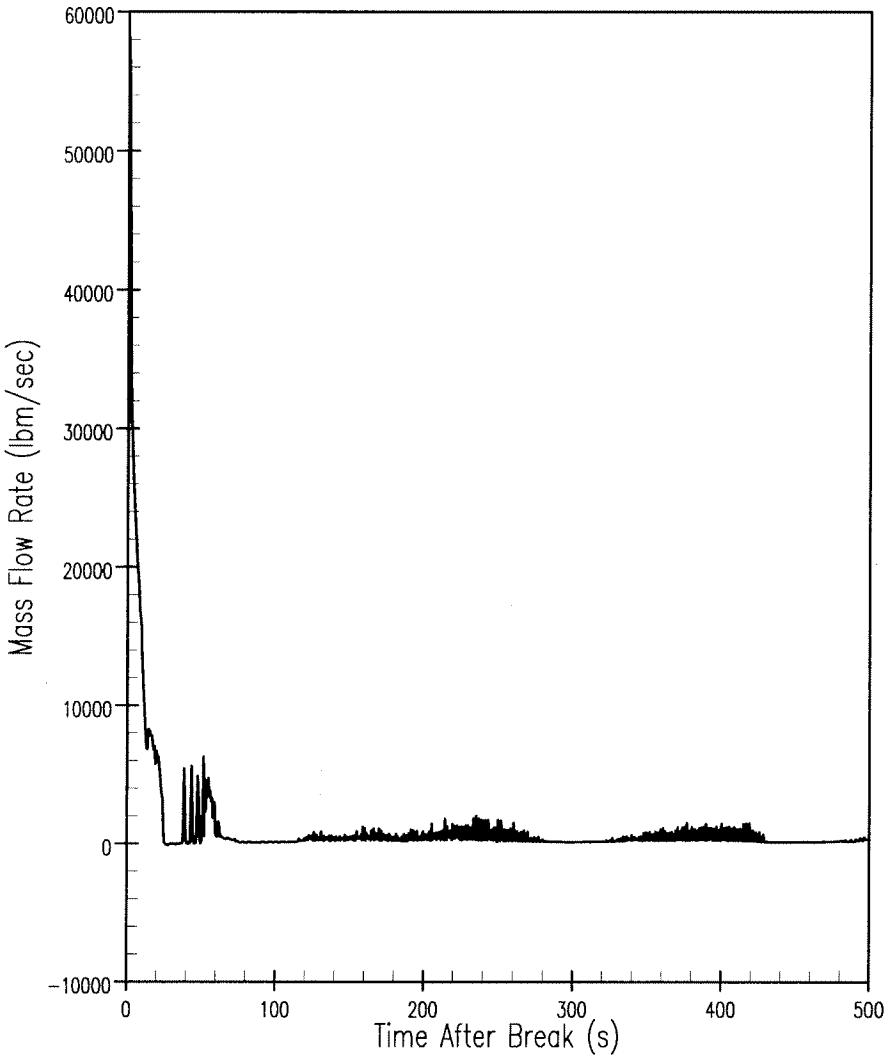
1637127440

Figure 4-2: Pressurizer Pressure for the Limiting PCT Case for Byron/Braidwood Unit 2



1289307515

Figure 5-1: Vessel Side Break Flow for the Limiting PCT Case for Byron/Braidwood Unit 1



1637127440

Figure 5-2: Vessel Side Break Flow for the Limiting PCT Case for Byron/Braidwood Unit 2

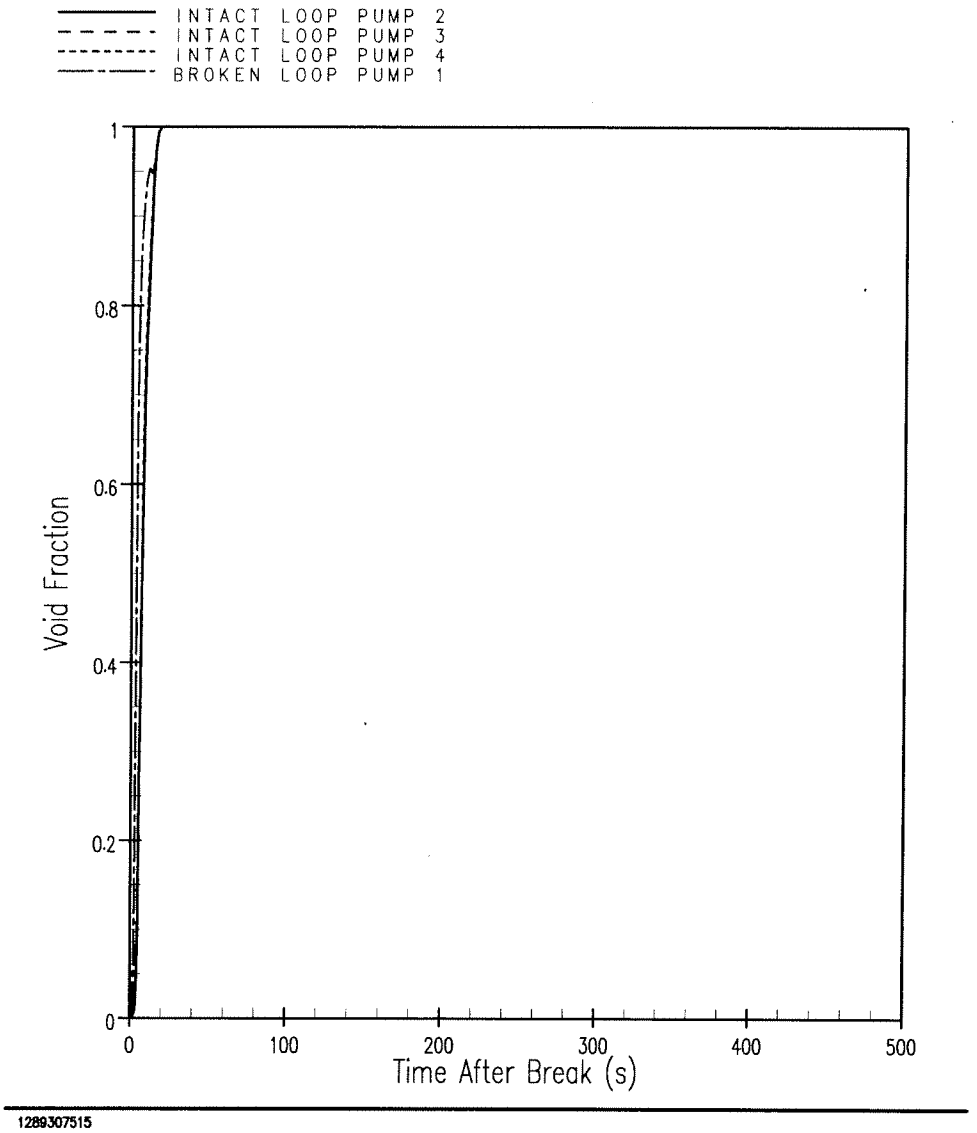


Figure 6-1: Void Fraction in Pumps for the Limiting PCT Case for Byron/Braidwood Unit 1

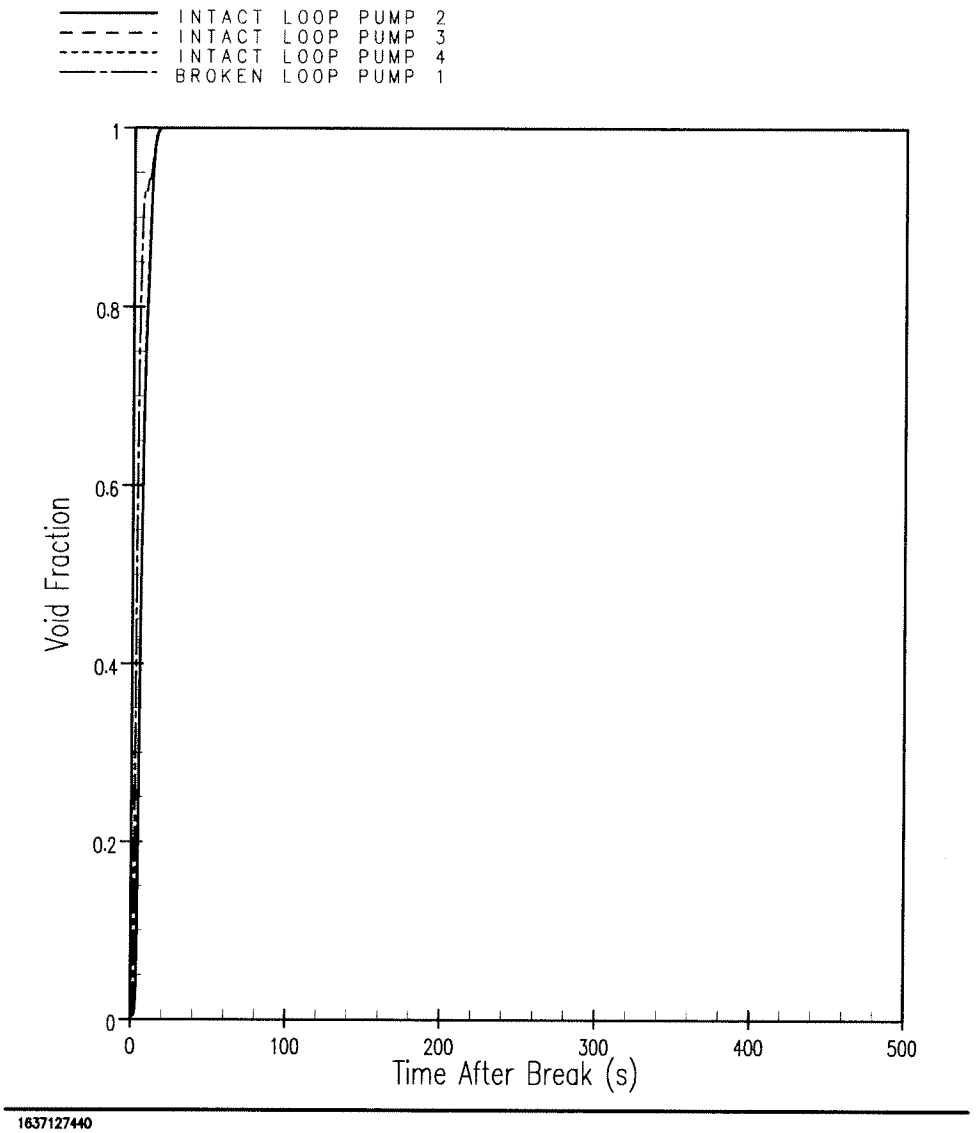
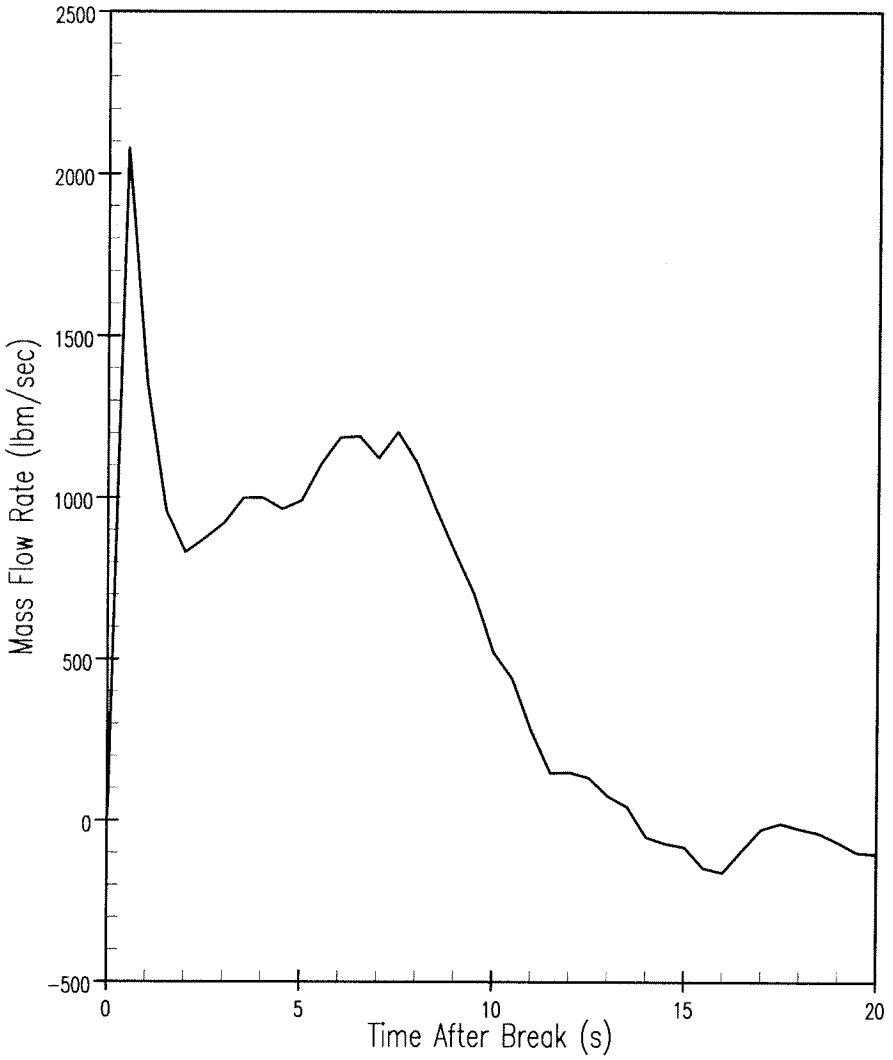
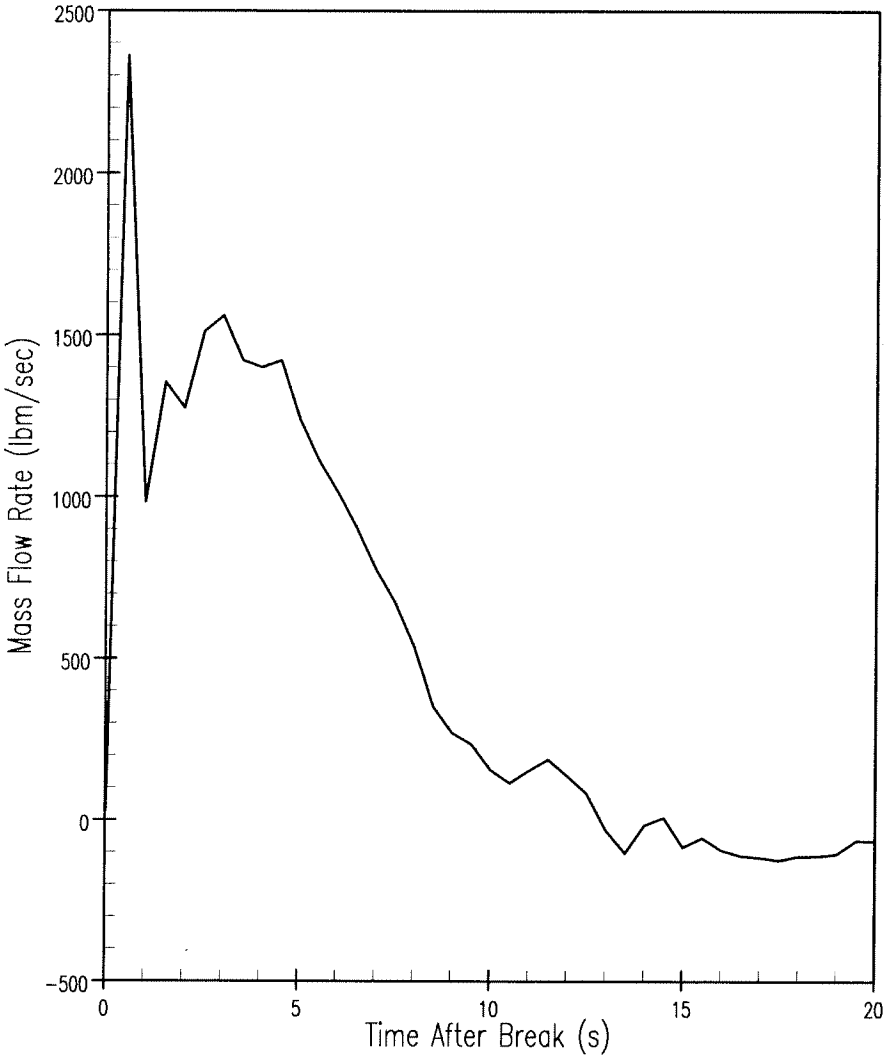


Figure 6-2: Void Fraction in Pumps for the Limiting PCT Case for Byron/Braidwood Unit 2



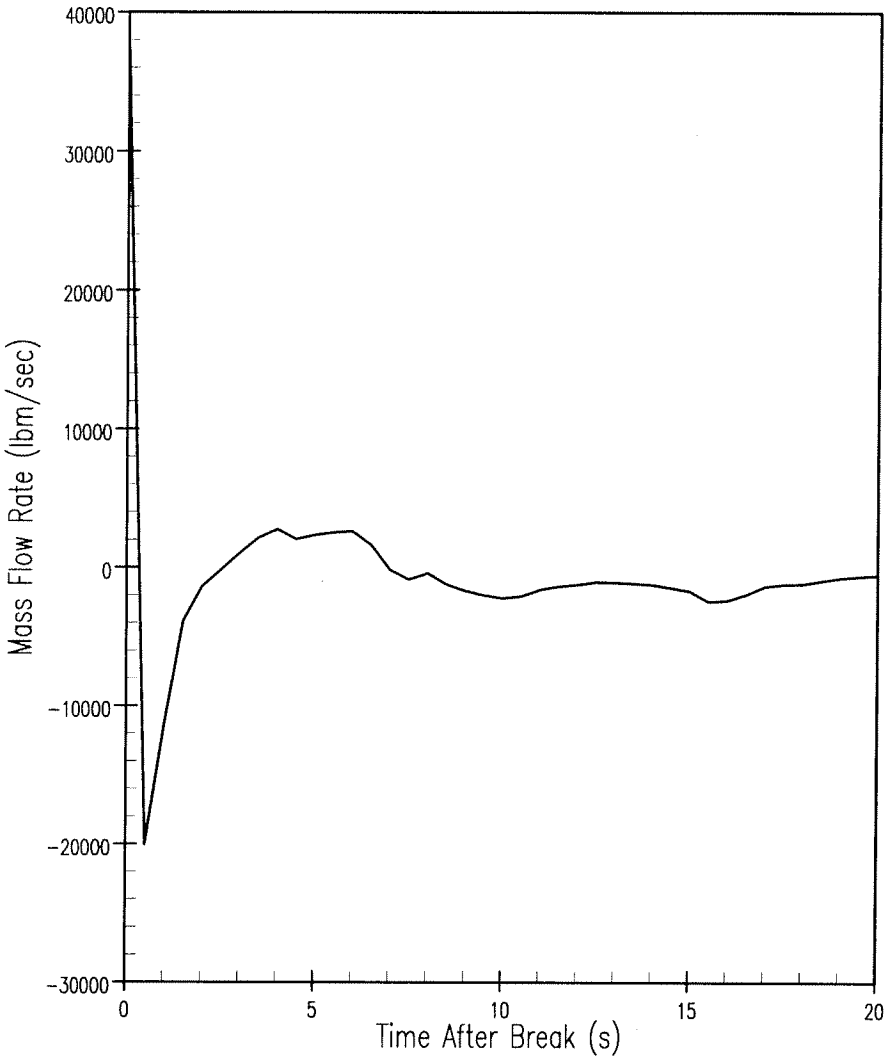
1289307515

Figure 7-1: Vapor Flow at Top of Core for the Limiting PCT Case for Byron/Braidwood Unit 1



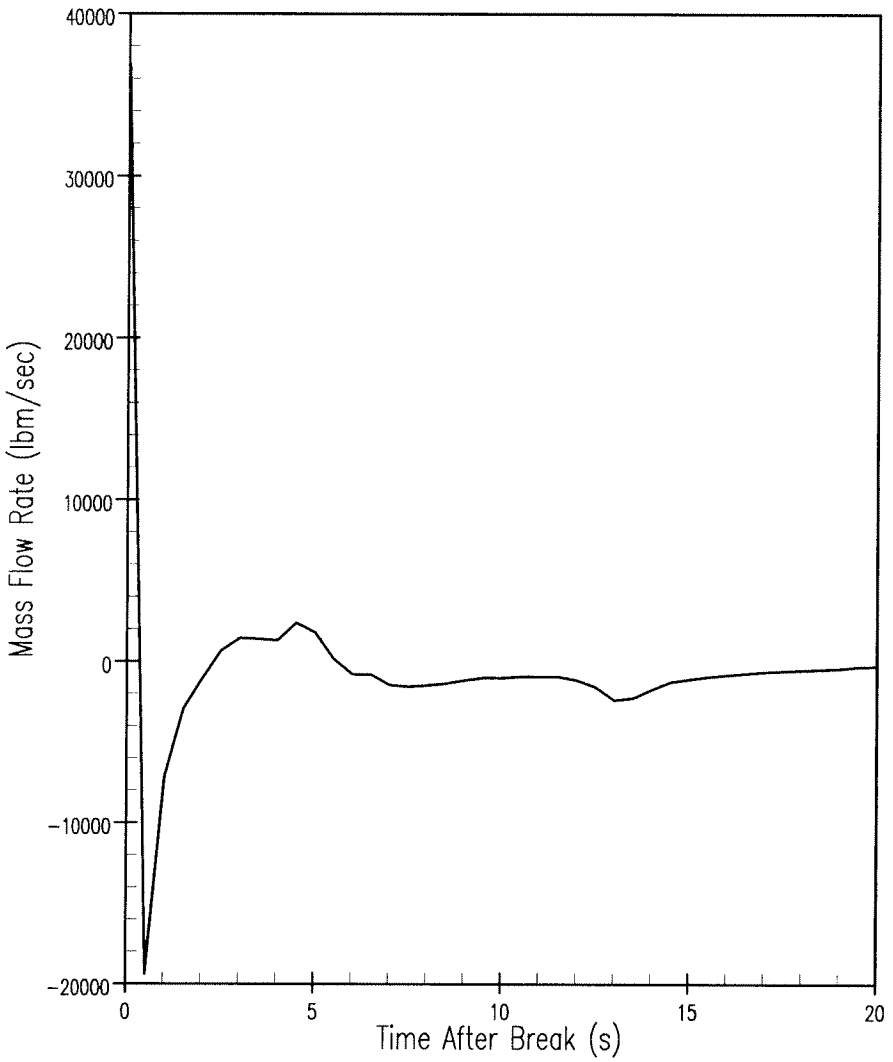
1637127440

Figure 7-2: Vapor Flow at Top of Core for the Limiting PCT Case for Byron/Braidwood Unit 2



1289307515

Figure 8-1: Total Flow at Bottom of Core for the Limiting PCT Case for Byron/Braidwood Unit 1



1637127440

Figure 8-2: Total Flow at Bottom of Core for the Limiting PCT Case for Byron/Braidwood Unit 2

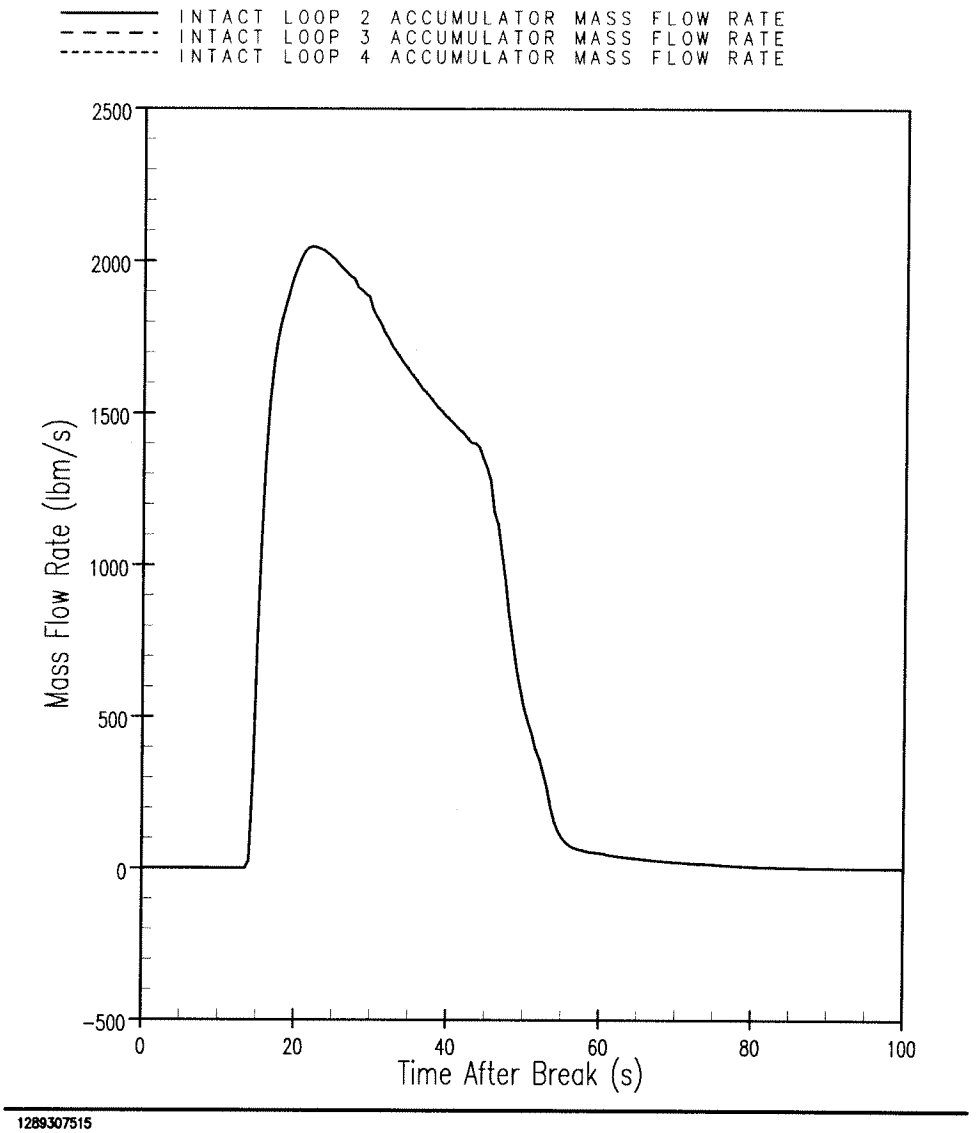


Figure 9-1: Accumulator Injection Flow for the Limiting PCT Case for Byron/Braidwood Unit 1

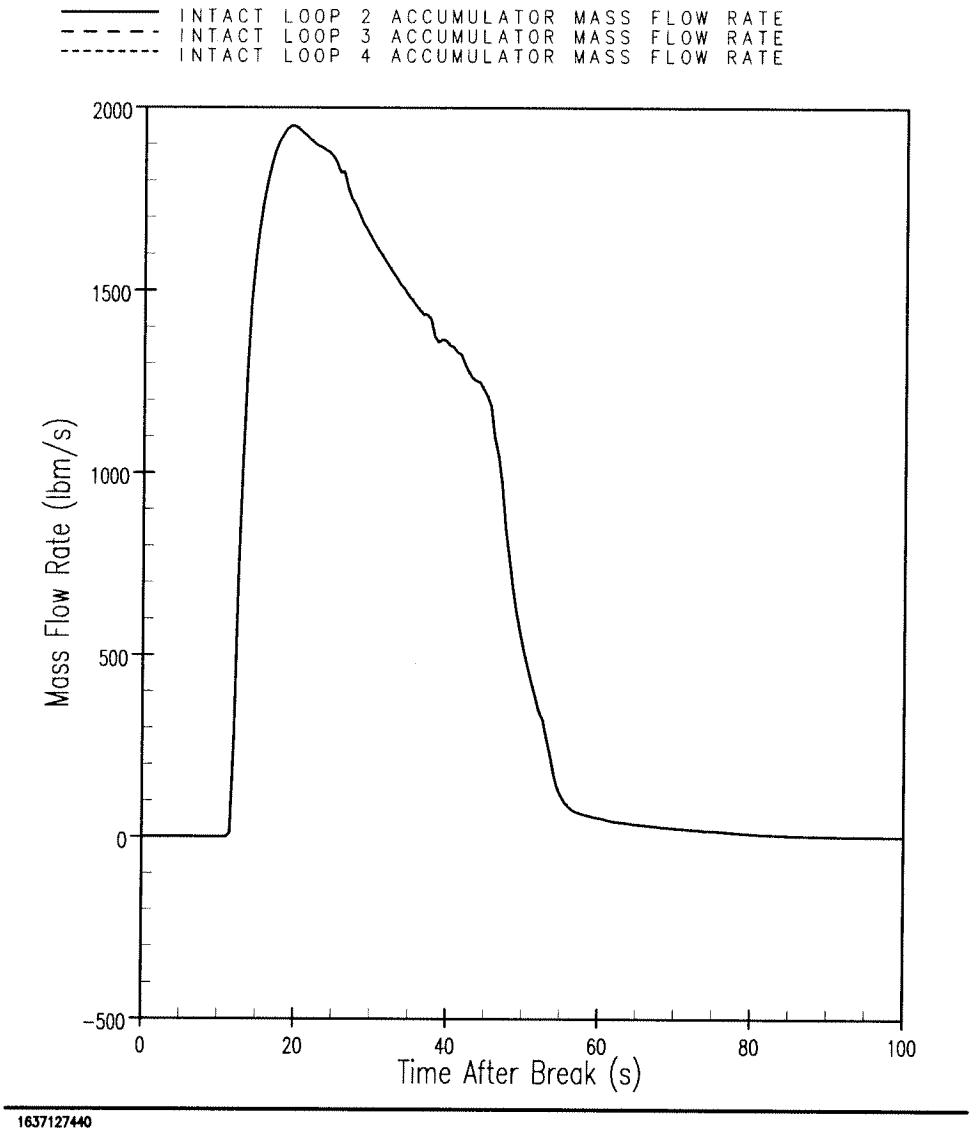


Figure 9-2: Accumulator Injection Flow for the Limiting PCT Case for Byron/Braidwood Unit 2

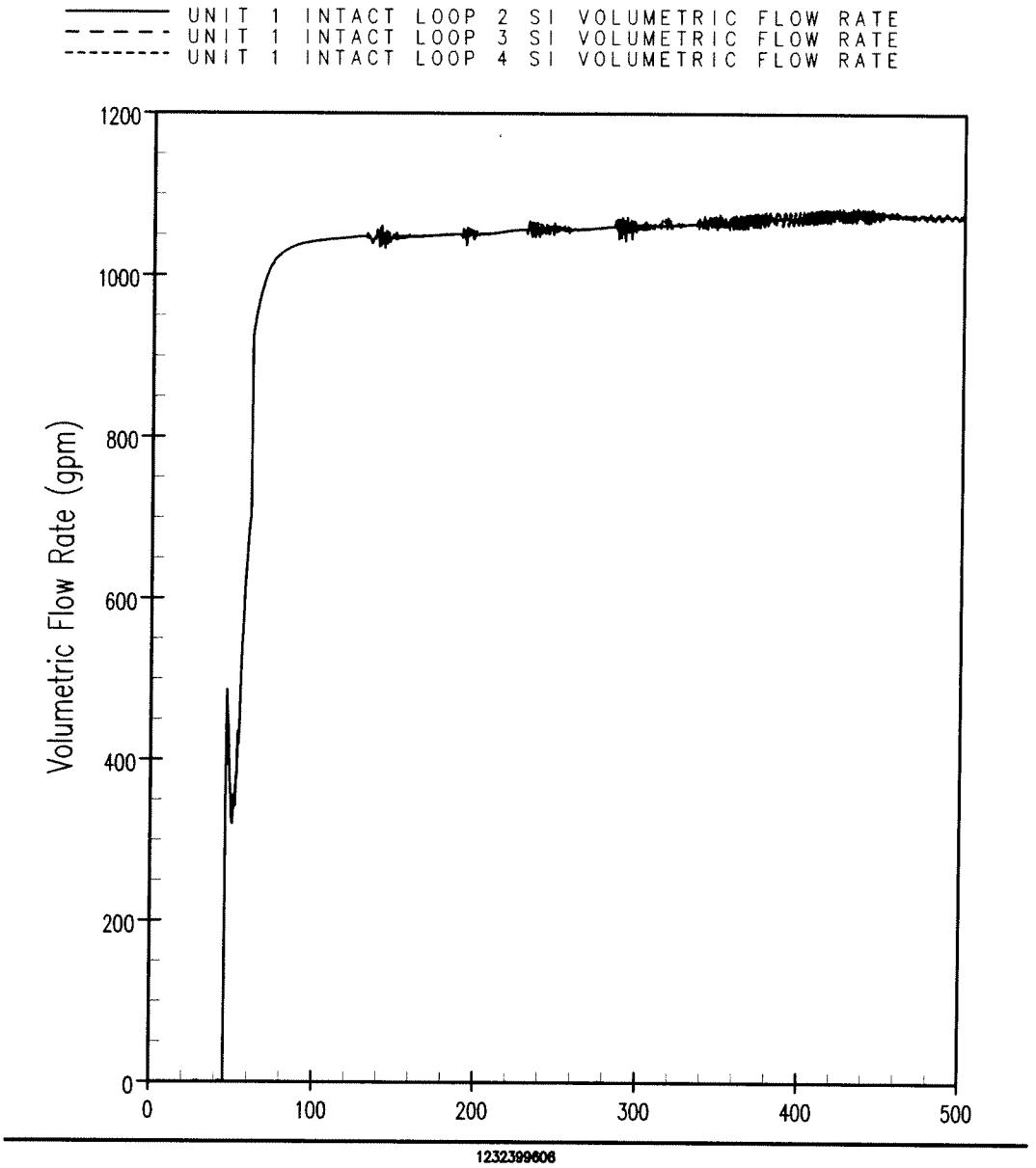


Figure 10-1: Safety Injection Flow for the Limiting PCT Case for Byron/Braidwood Unit 1

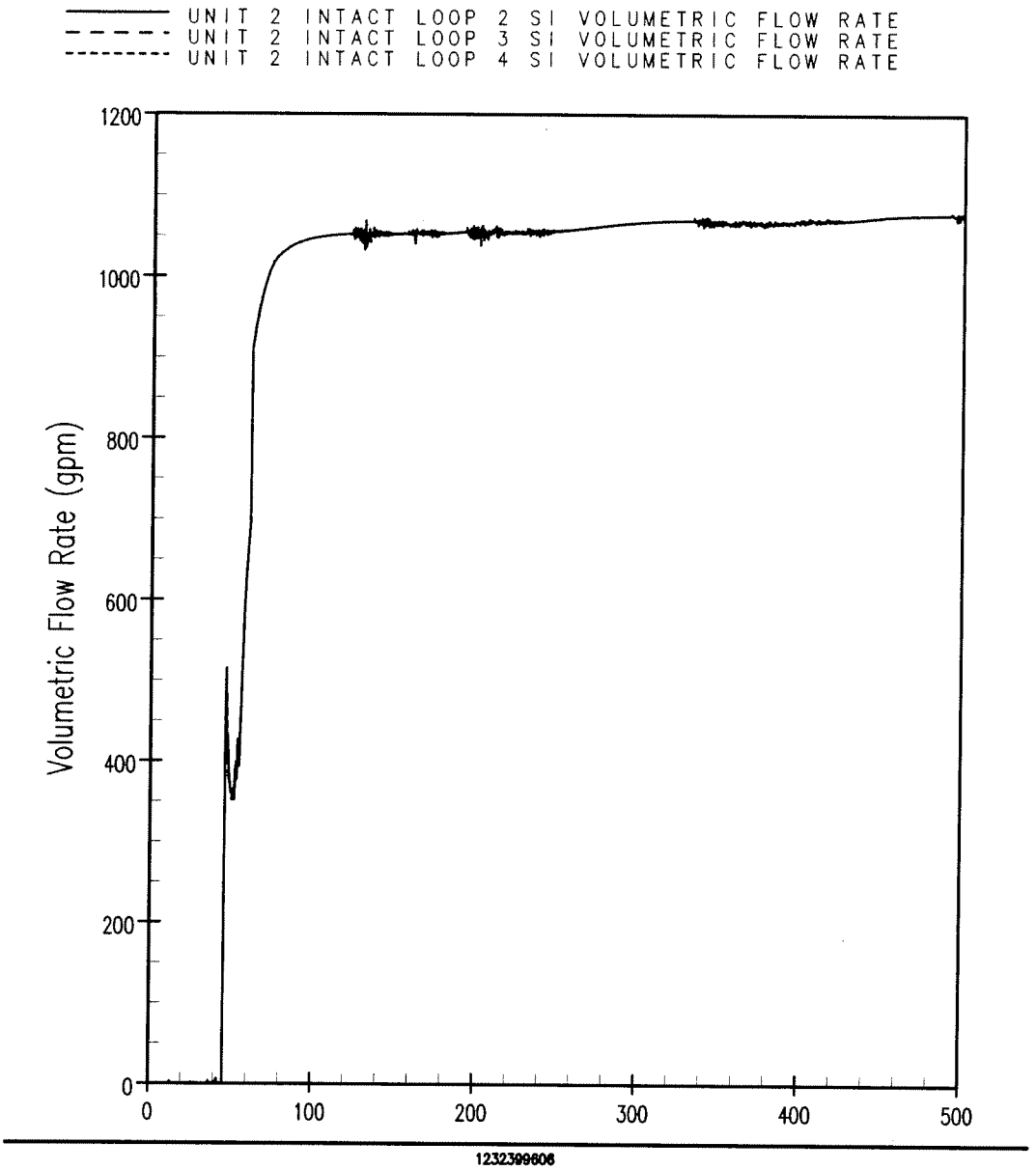


Figure 10-2: Safety Injection Flow for the Limiting PCT Case for Byron/Braidwood Unit 2

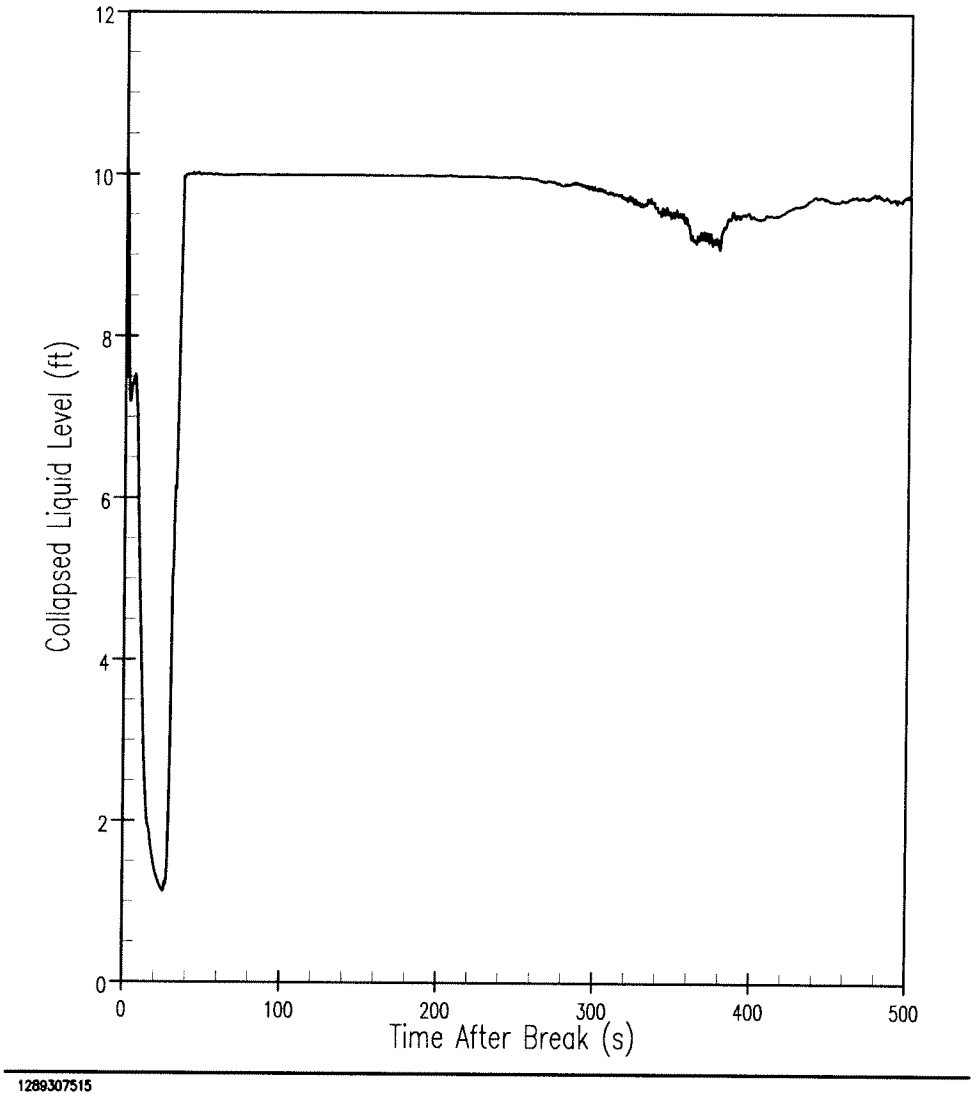
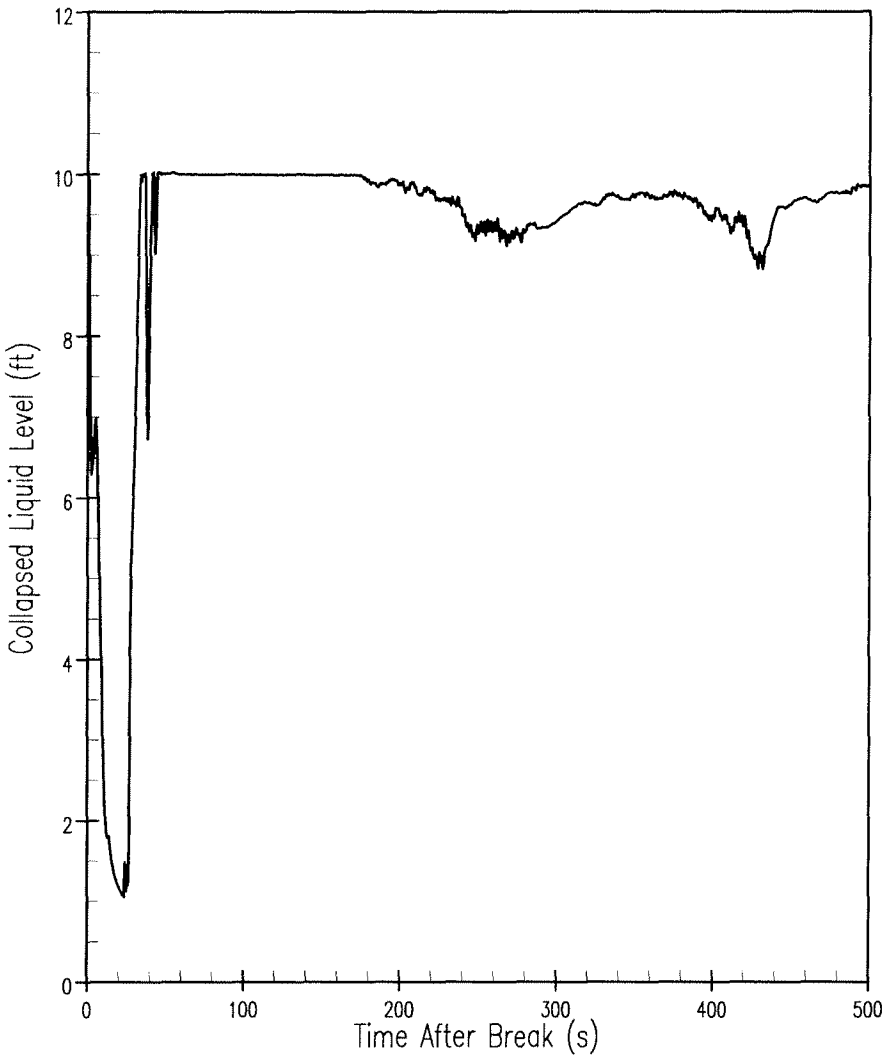


Figure 11-1: Lower Plenum Collapsed Liquid Level for the Limiting PCT Case for Byron/Braidwood Unit 1



1637127440

Figure 11-2: Lower Plenum Collapsed Liquid Level for the Limiting PCT Case for Byron/Braidwood Unit 2

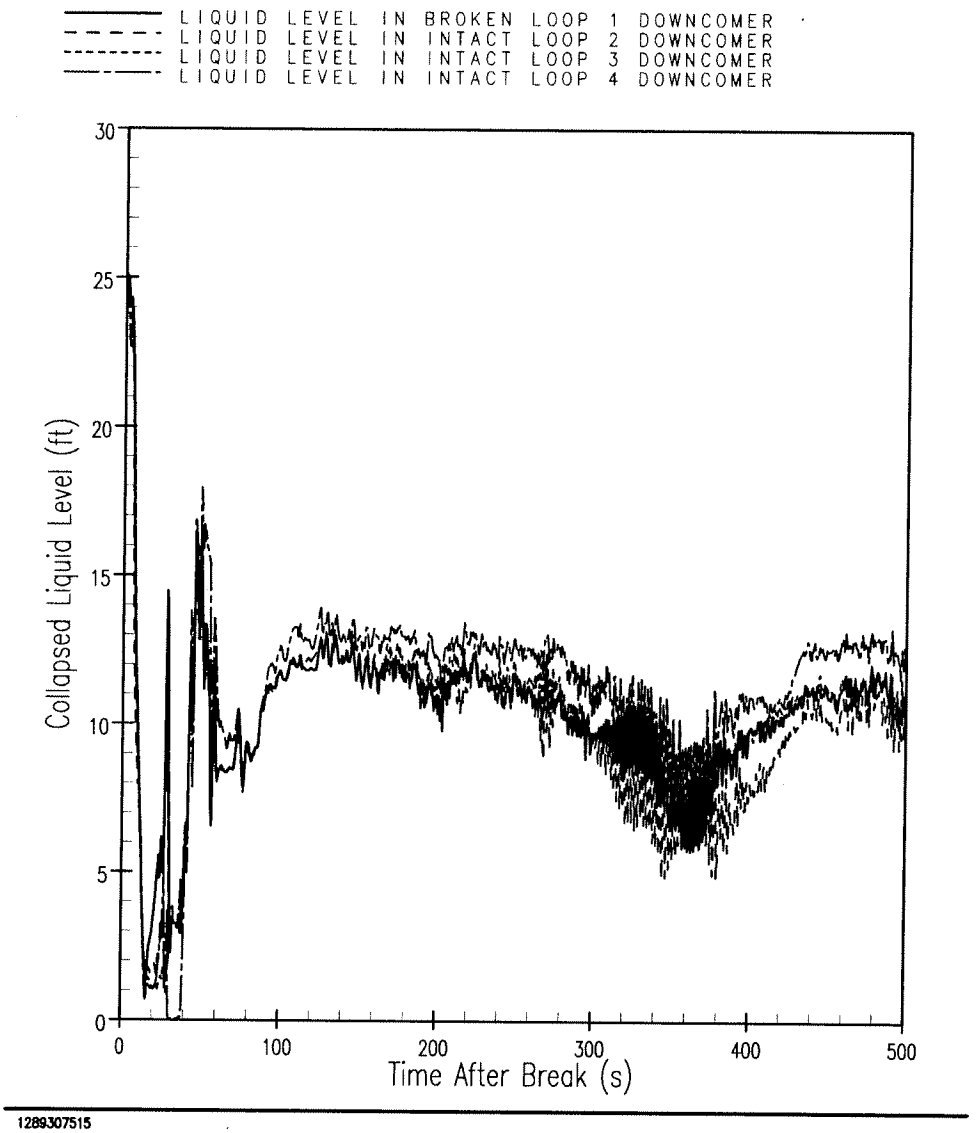


Figure 12-1: Downcomer Collapsed Liquid Levels for the Limiting PCT Case for Byron/Braidwood Unit 1

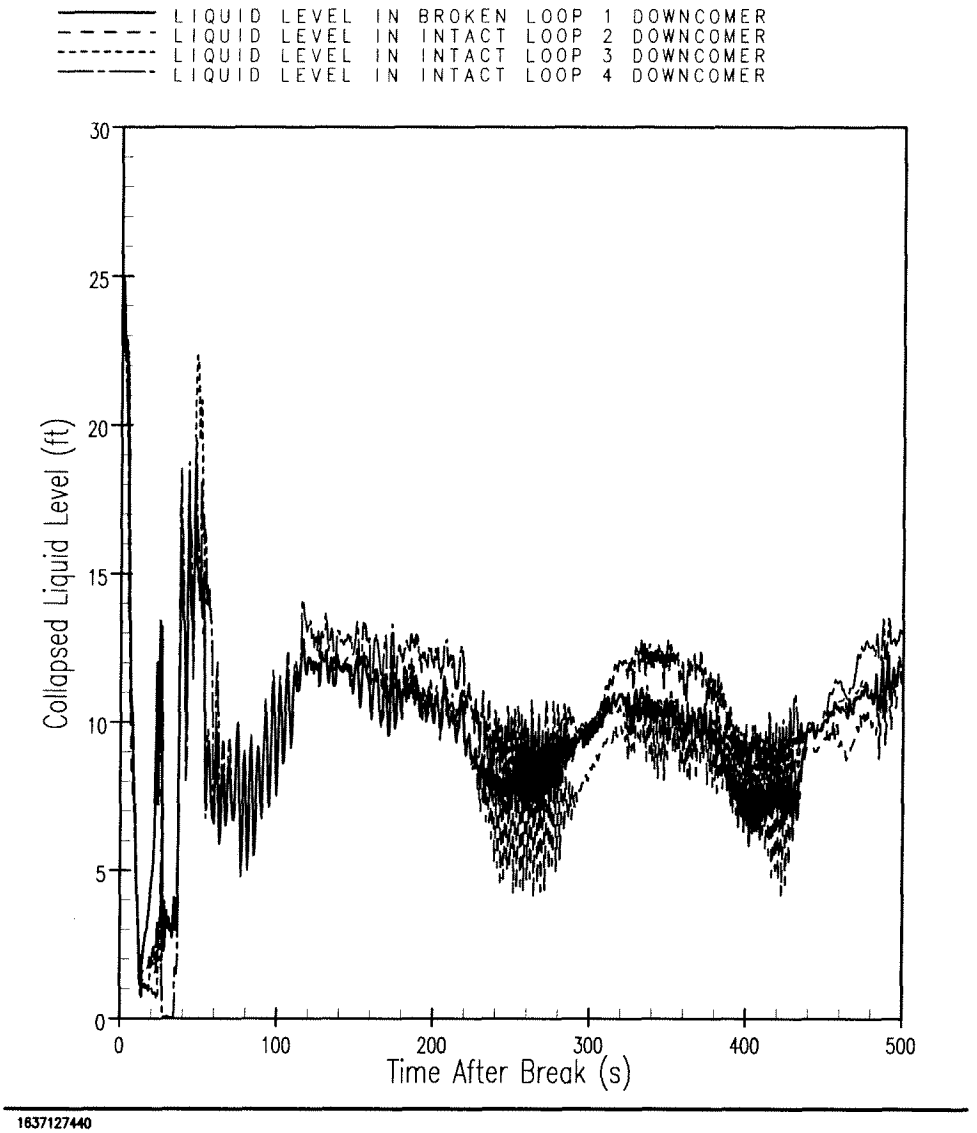
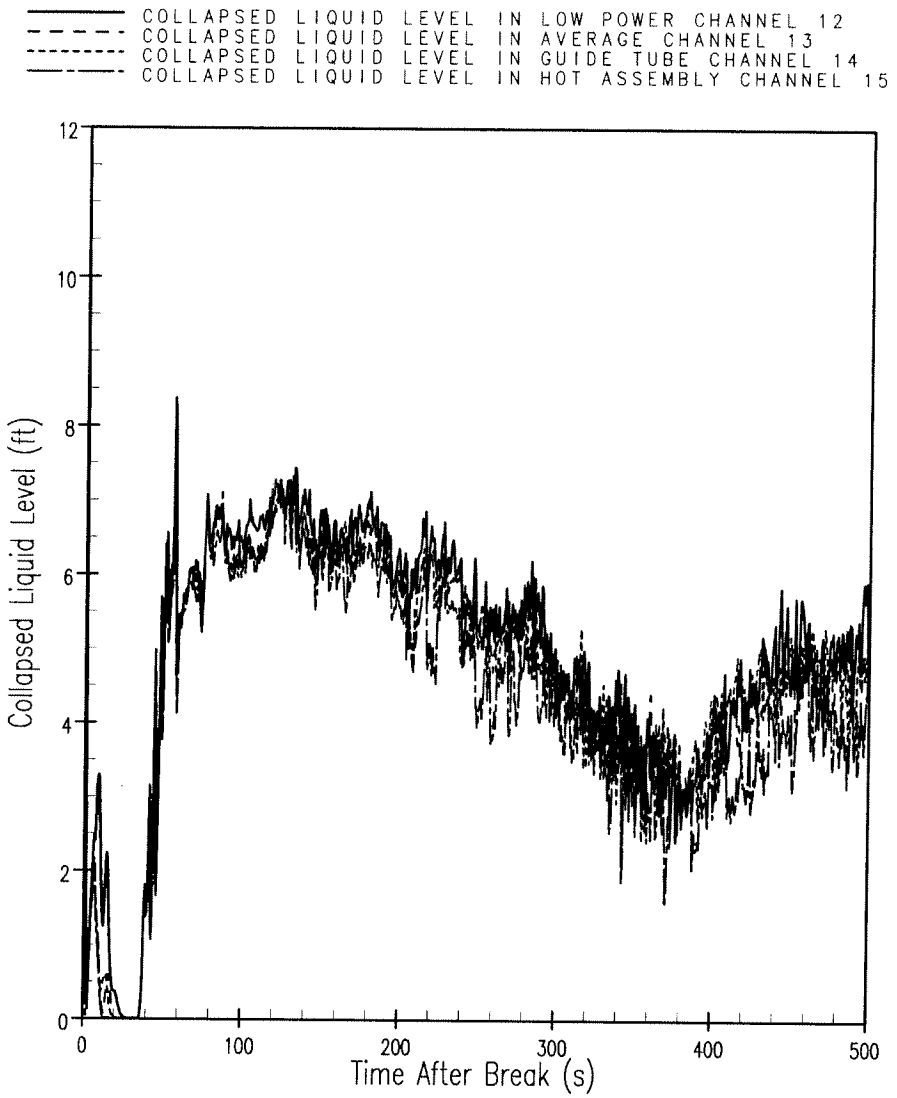
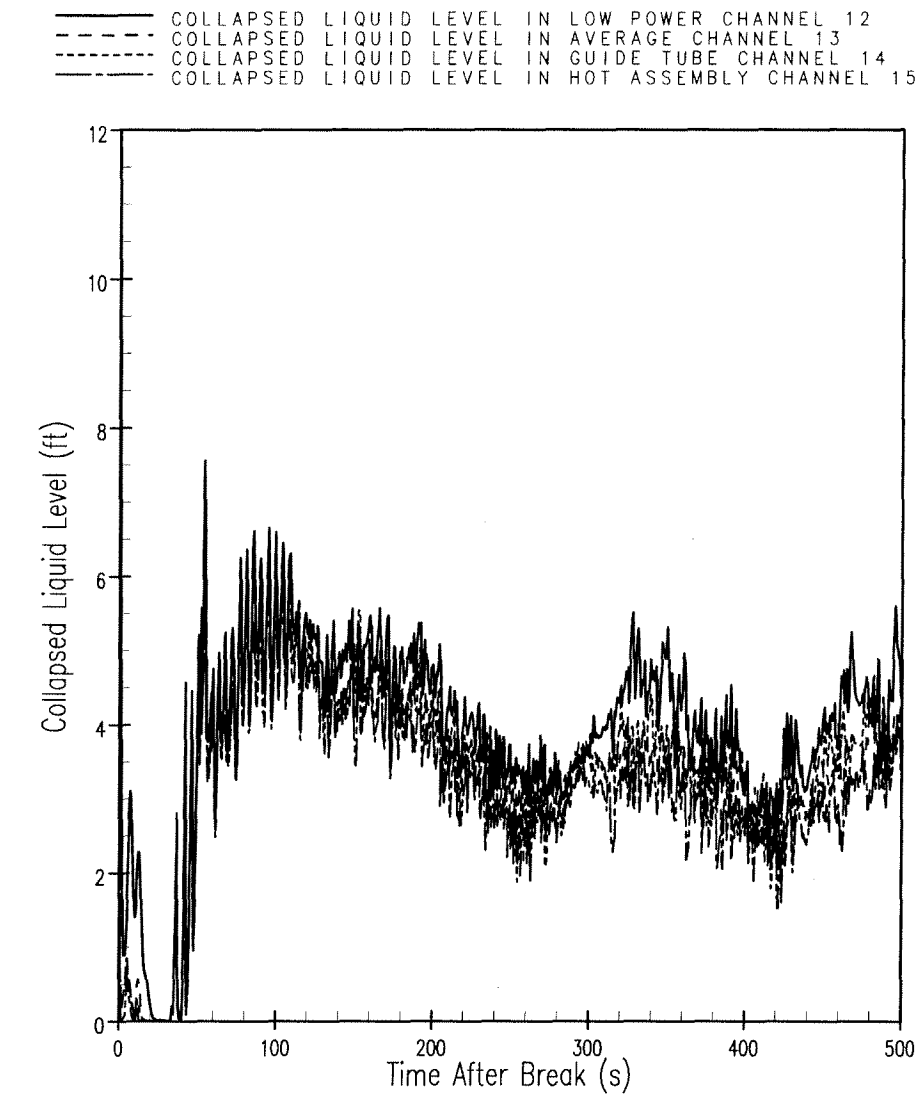


Figure 12-2: Downcomer Collapsed Liquid Levels for the Limiting PCT Case for Byron/Braidwood Unit 2



1289307515

Figure 13-1: Core Collapsed Liquid Levels for the Limiting PCT Case for Byron/Braidwood Unit 1



1637127440

Figure 13-2: Core Collapsed Liquid Levels for the Limiting PCT Case for Byron/Braidwood Unit 2

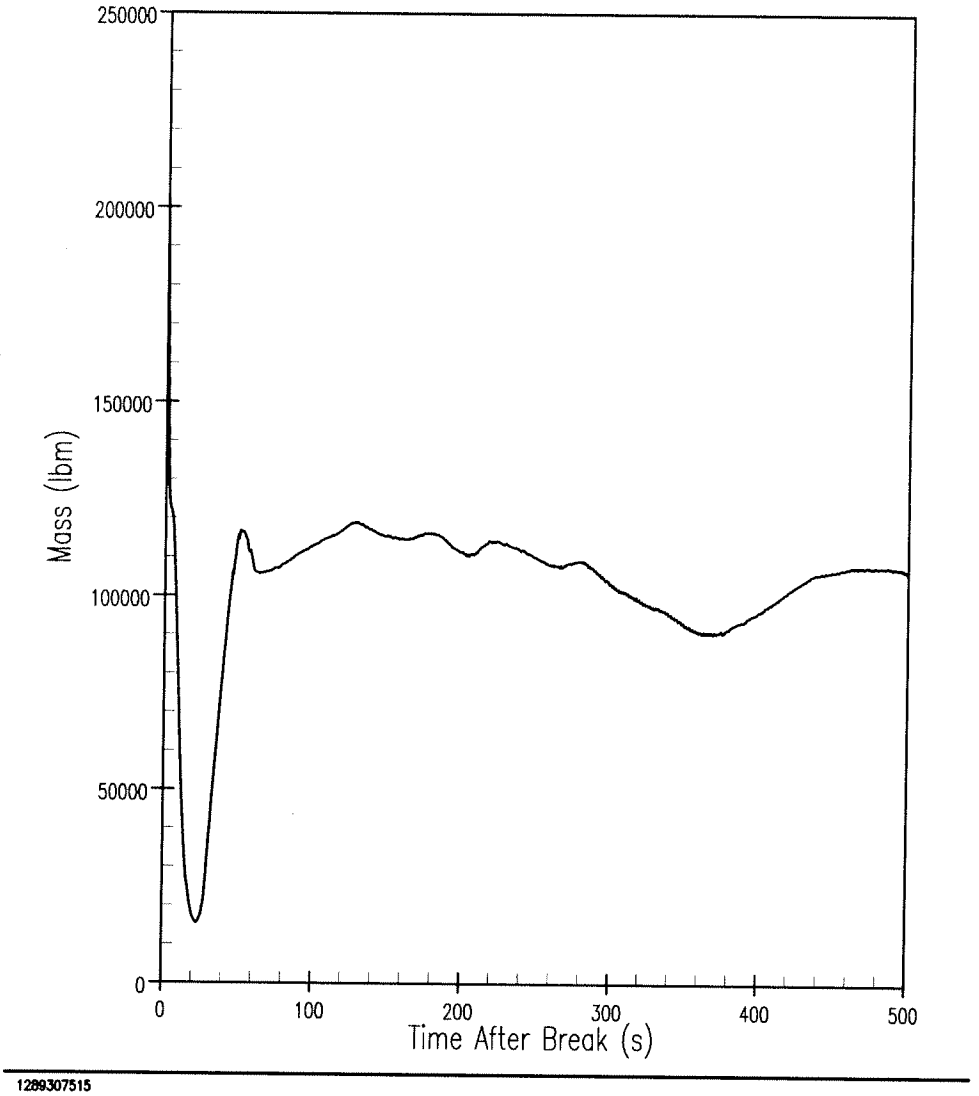


Figure 14-1: Vessel Fluid Mass for the Limiting PCT Case for Byron/Braidwood Unit 1

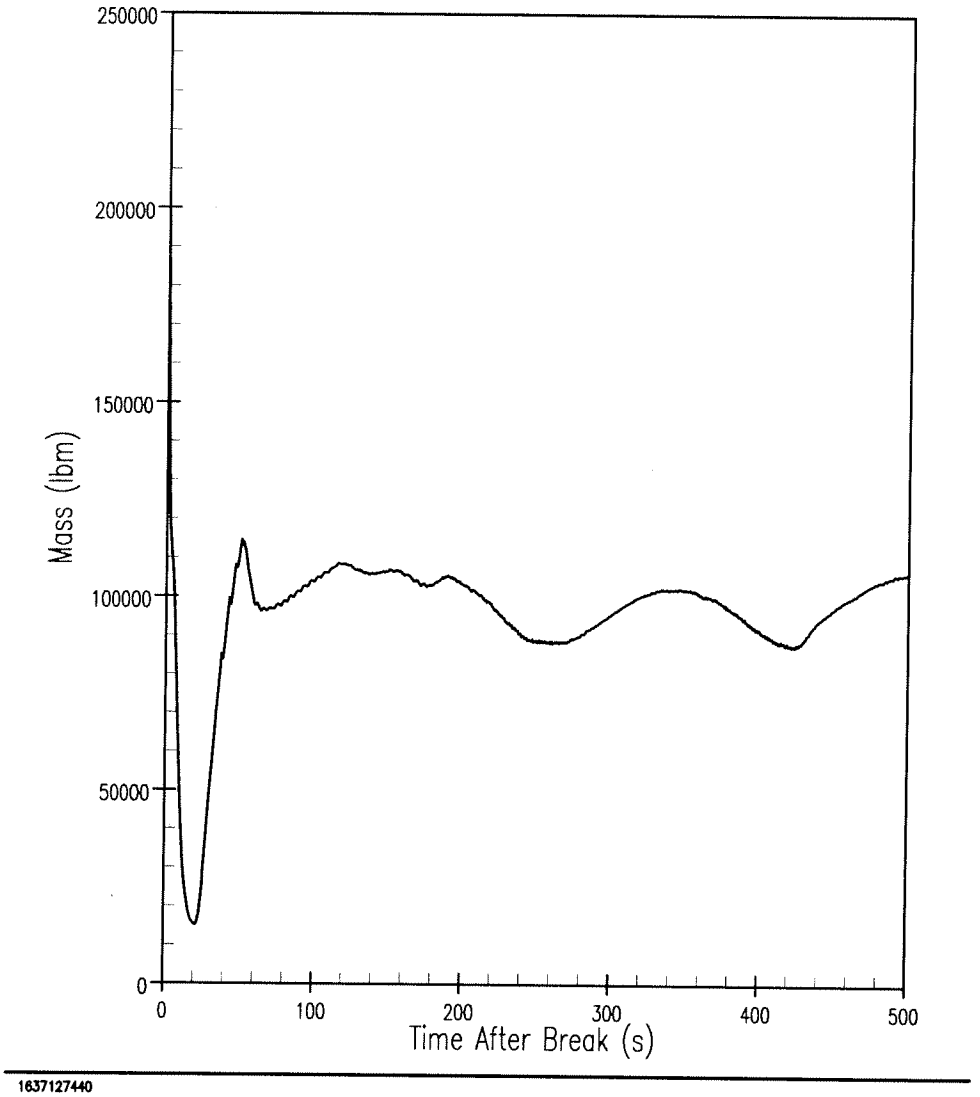
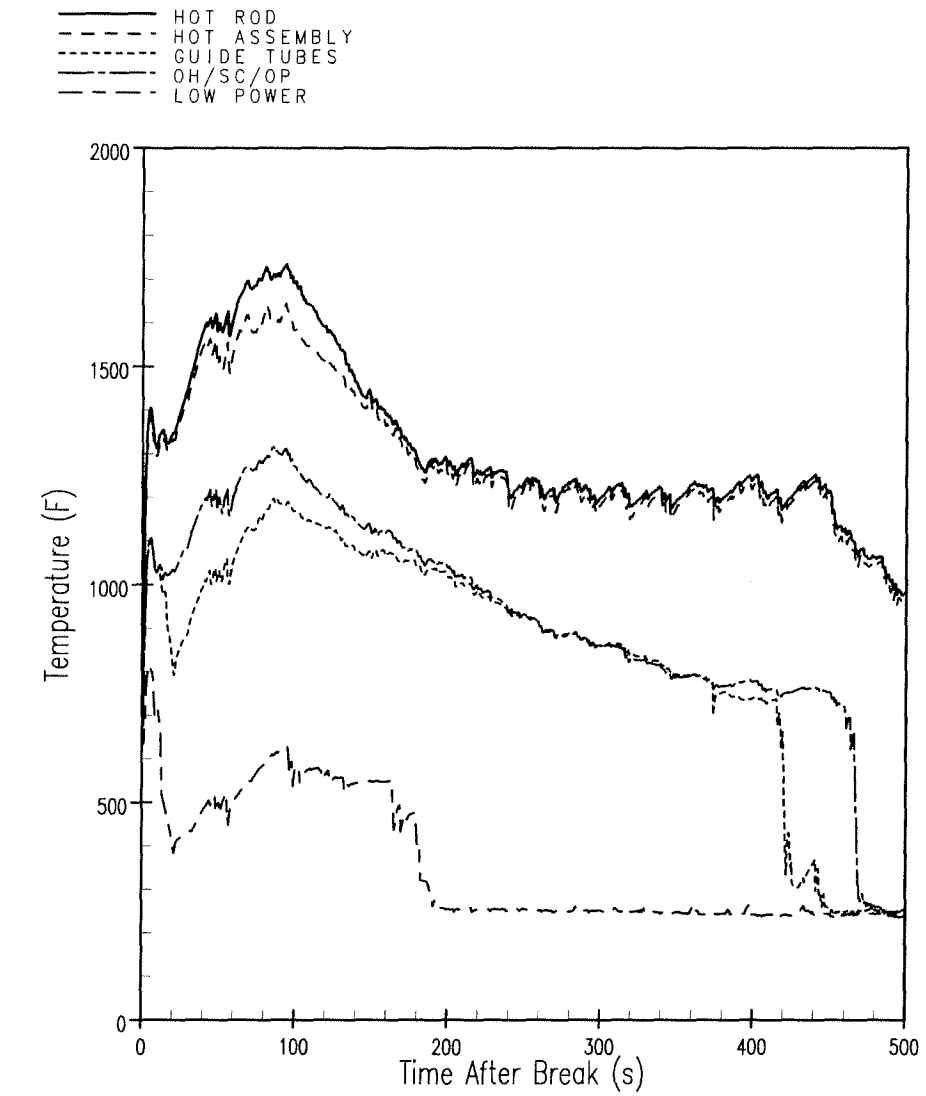


Figure 14-2: Vessel Fluid Mass for the Limiting PCT Case for Byron/Braidwood Unit 2



1289307515

Figure 15-1: WCOBRA/TRAC PCT for All Five Rod Groups for the Limiting PCT Case for Byron/Braidwood Unit 1

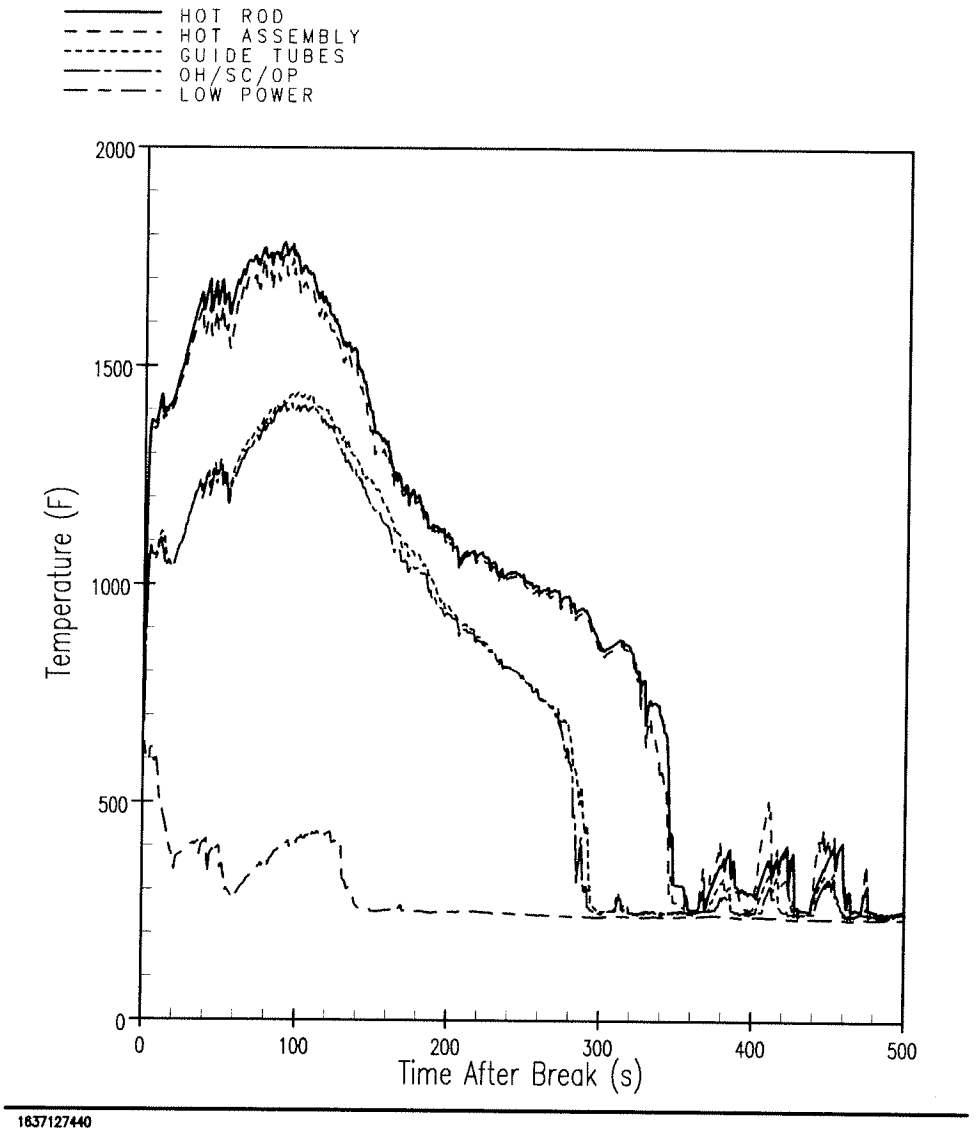
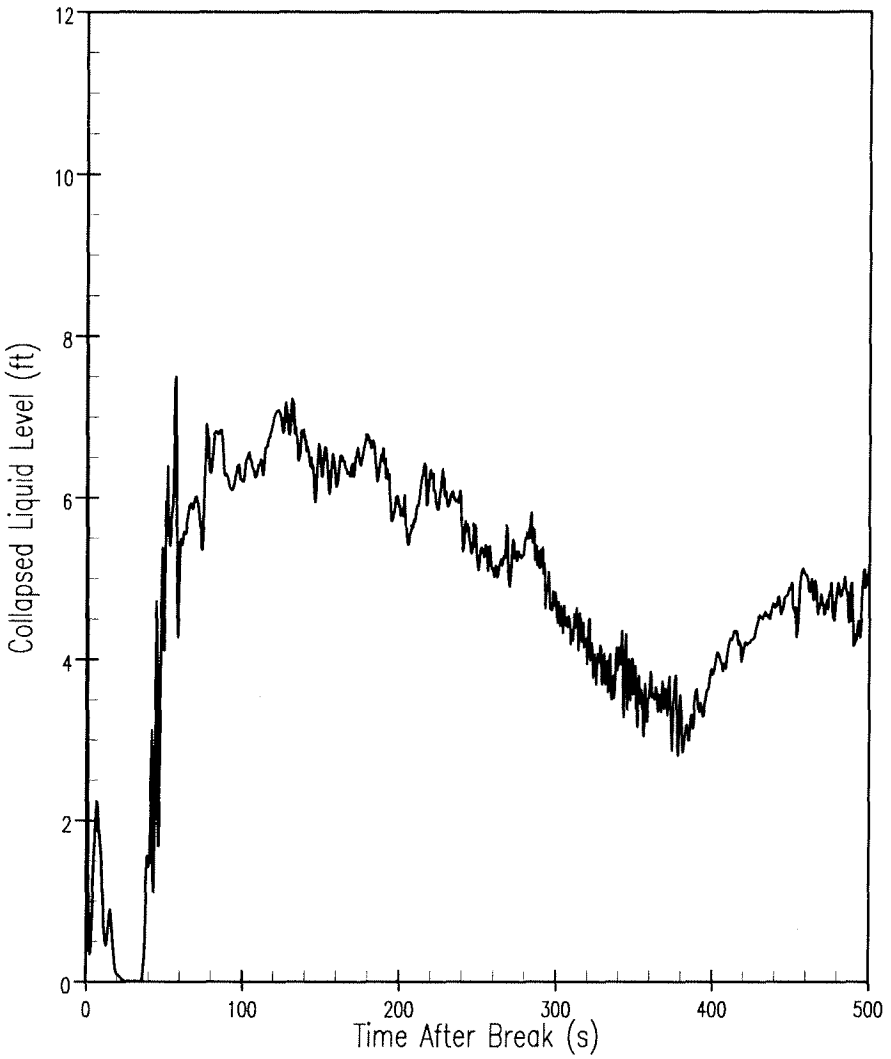
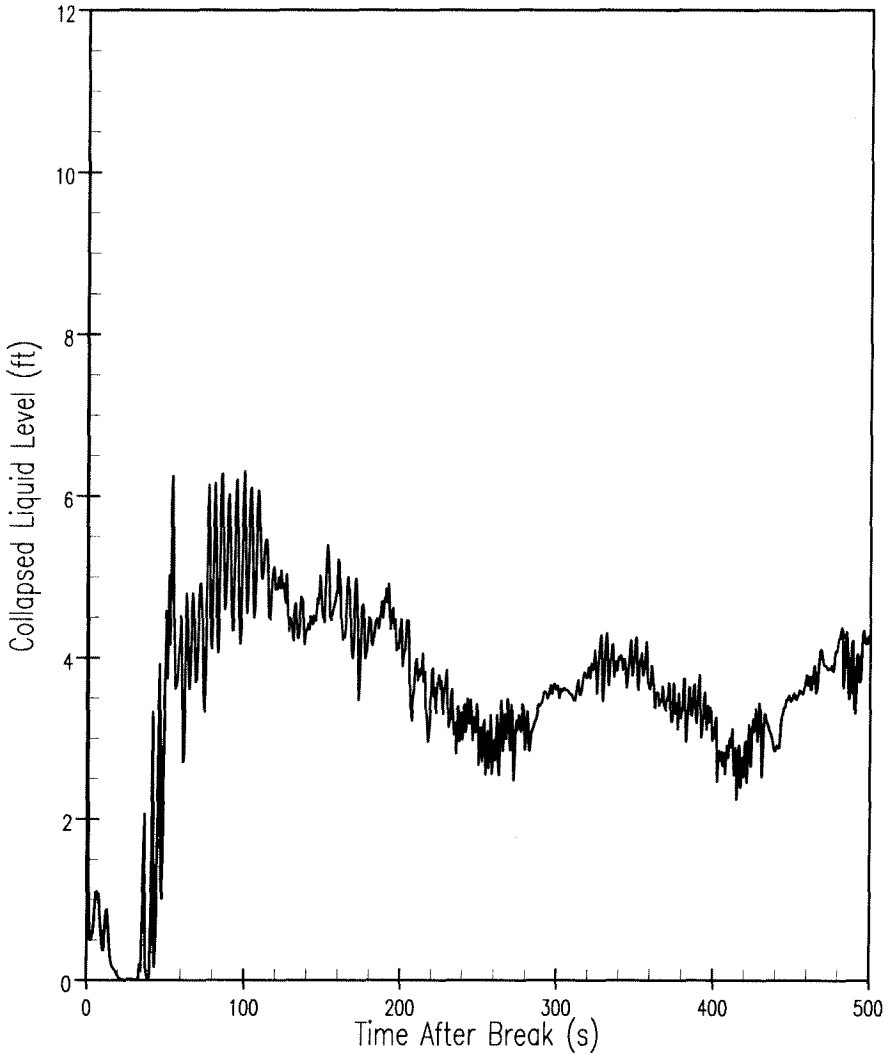


Figure 15-2: WCOBRA/TRAC PCT for All Five Rod Groups for the Limiting PCT Case for Byron/Braidwood Unit 2



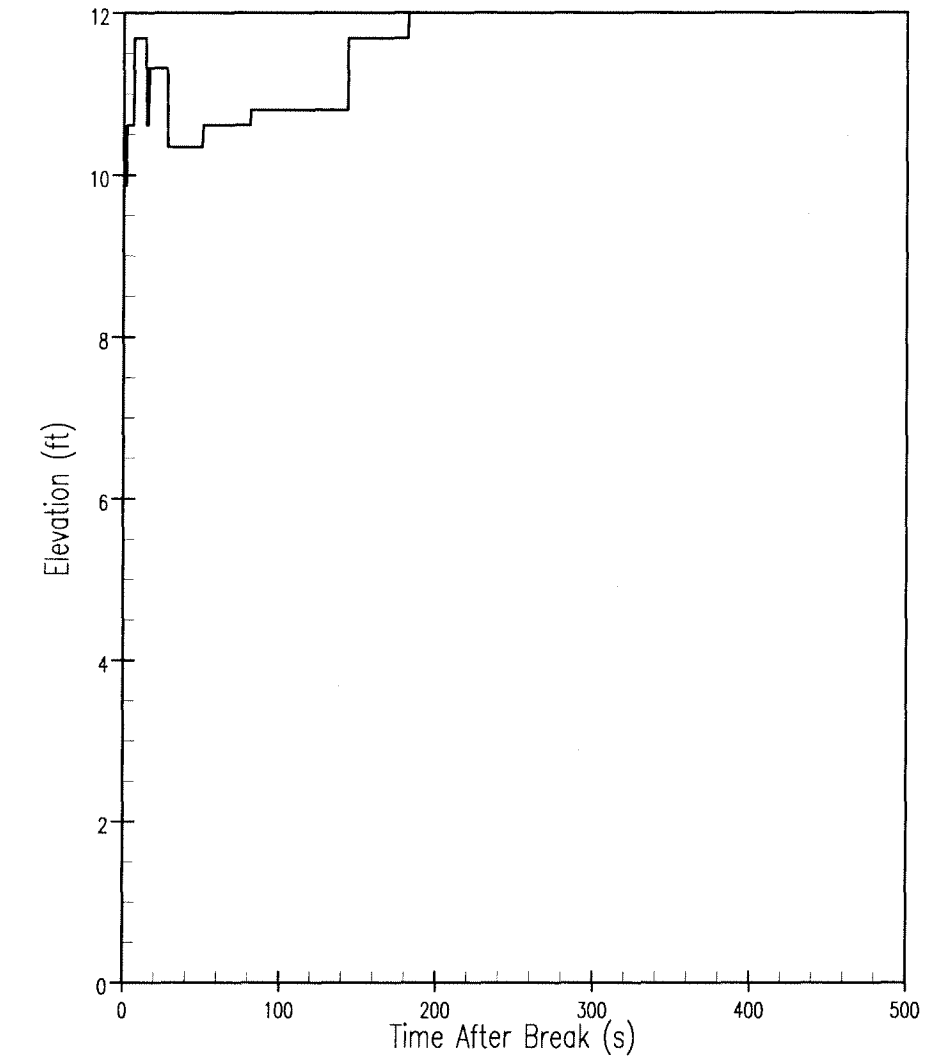
1289307515

Figure 16-1: Average Core Collapsed Liquid Level per Assembly for the Limiting PCT Case for Byron/Braidwood Unit 1



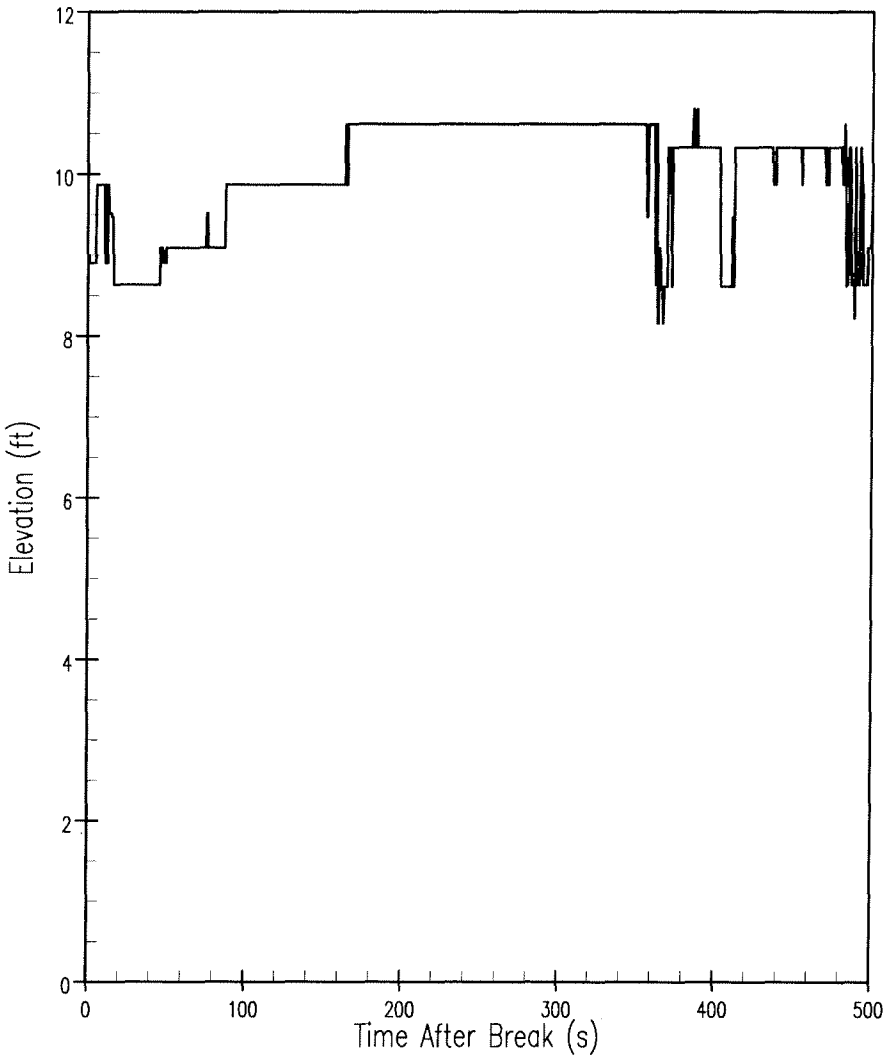
1637127440

Figure 16-2: Average Core Collapsed Liquid Level per Assembly for the Limiting PCT Case for Byron/Braidwood Unit 2



1289307515

Figure 17-1: PCT Elevation for the Hot Rod for the Limiting PCT Case for Byron/Braidwood Unit 1



1637127440

Figure 17-2: PCT Elevation for the Hot Rod for the Limiting PCT Case for Byron/Braidwood Unit 2

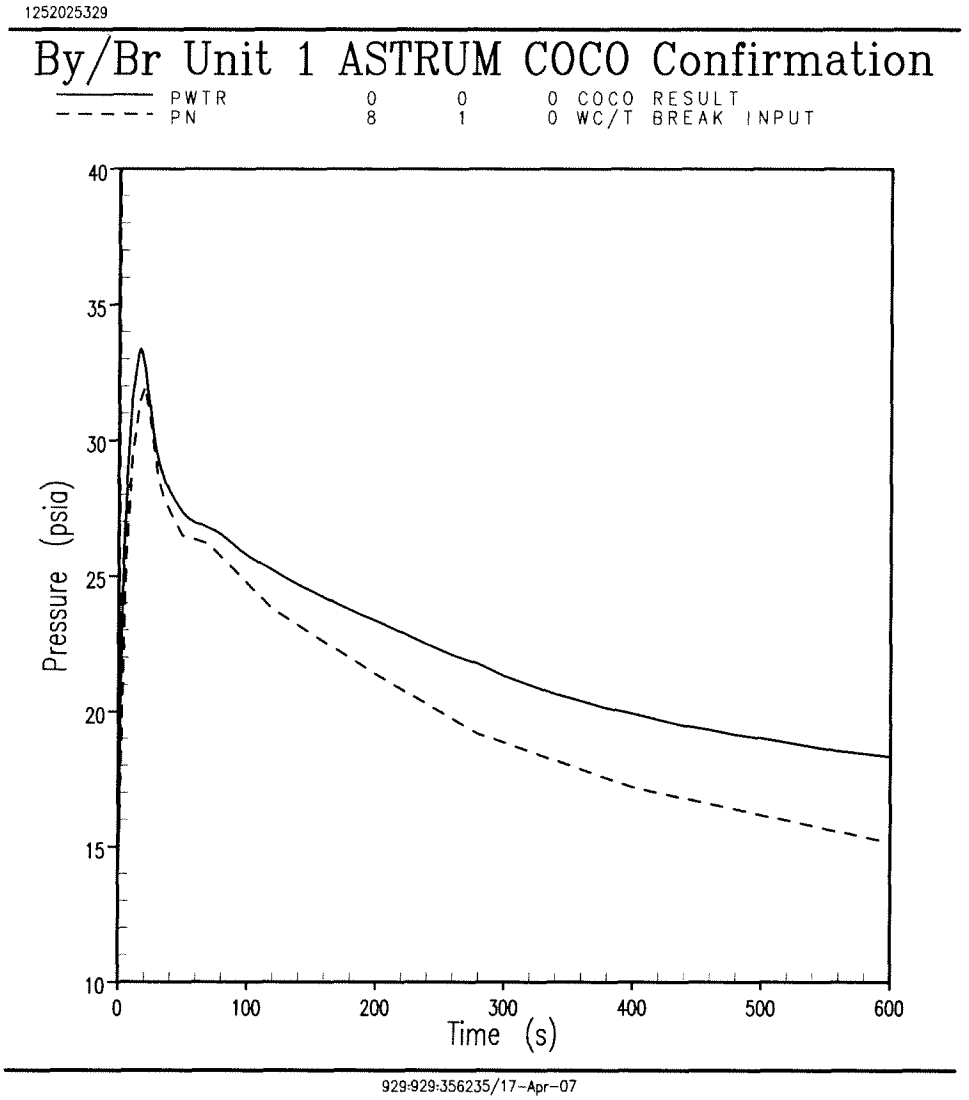


Figure 18-1: Analysis Versus Calculated Containment Backpressure for Byron/Braidwood Unit 1

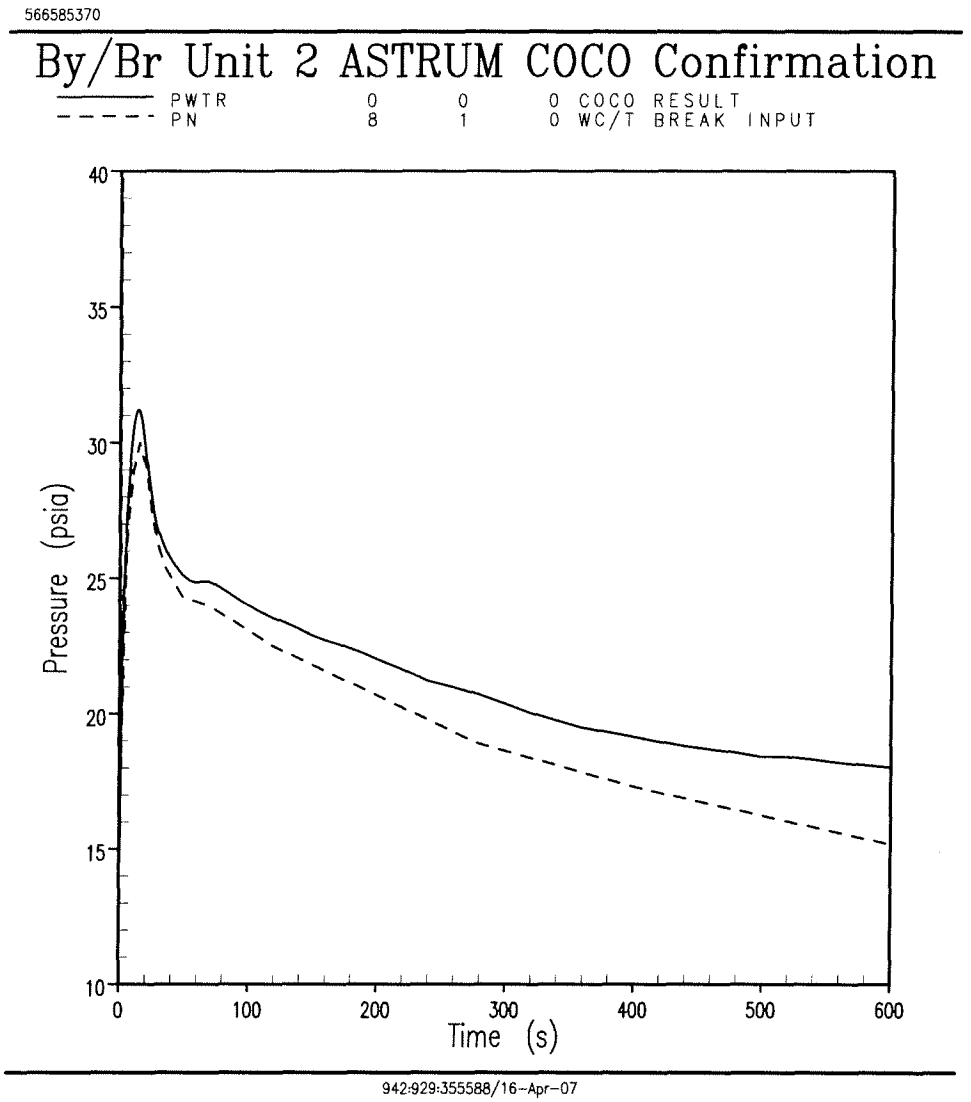


Figure 18-2: Analysis Versus Calculated Containment Backpressure for Byron/Braidwood Unit 2

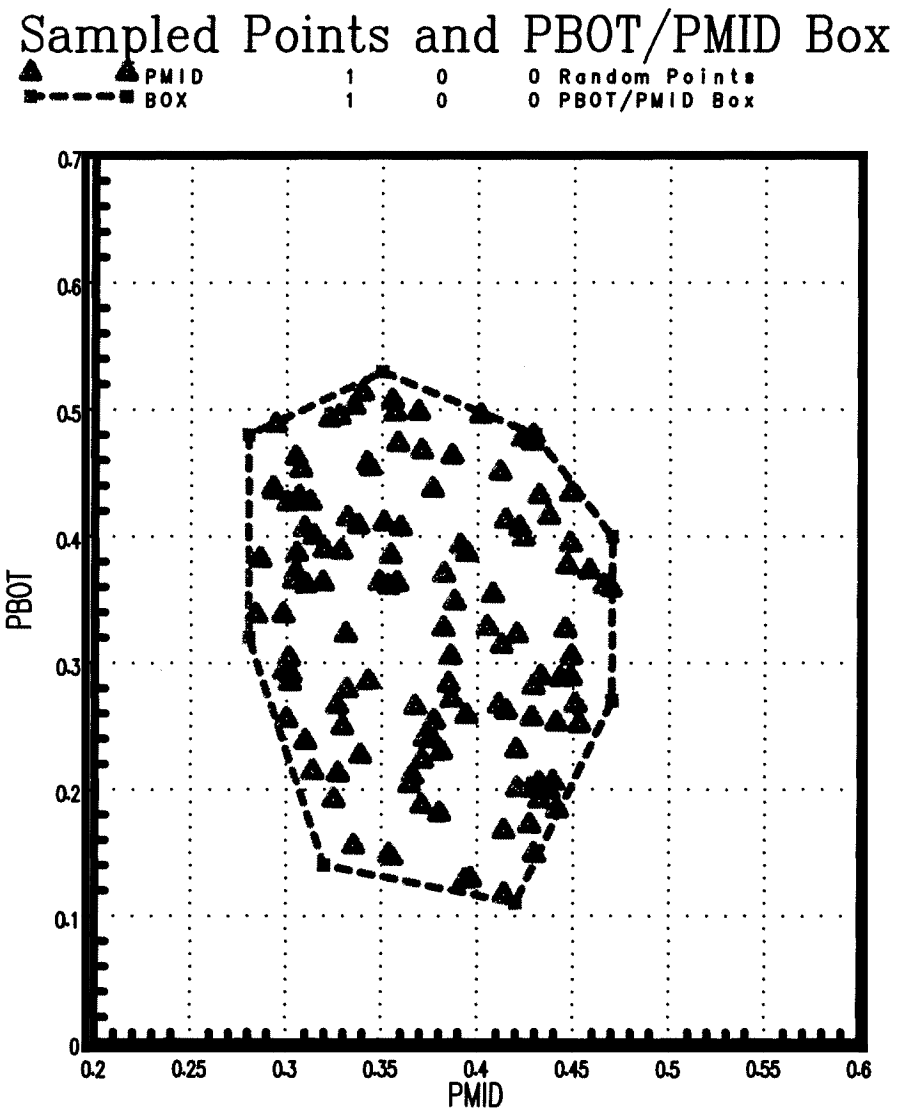


Figure 19-1: BELOCA Analysis Axial Power Shape Operating Space Envelope for Byron/Braidwood Unit 1

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

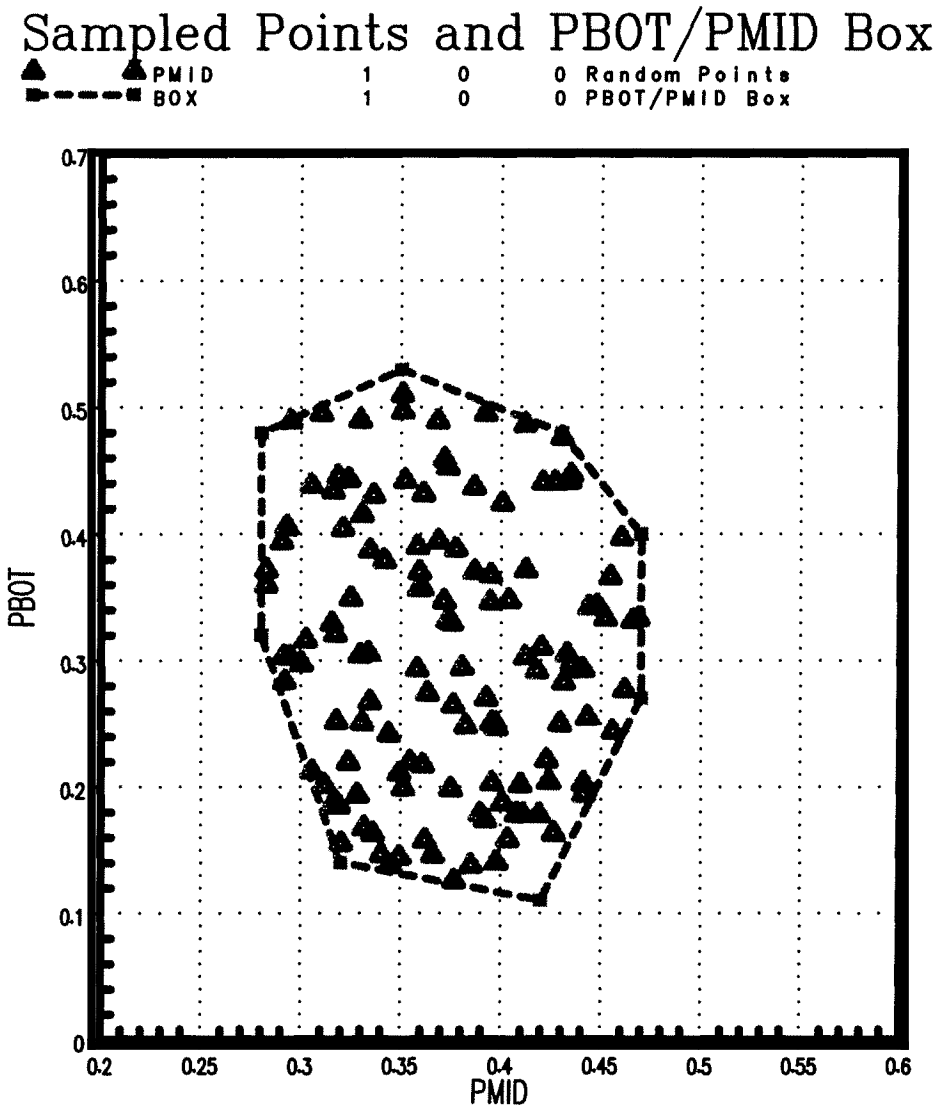


Figure 19-2: BELOCA Analysis Axial Power Shape Operating Space Envelope for Byron/Braidwood Unit 2

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

3.3.2 10 CFR 50.46 Requirements

It must be demonstrated that there is a high level of probability that the following limits set forth in 10 CFR 50.46 are met.

10 CFR 50.46(b)(1)

The limiting PCT corresponds to a bounding estimate of the 95th percentile PCT at the 95% confidence level. Since the resulting PCTs for the limiting case is 1913°F for Byron/Braidwood Unit 1, and 2041°F for Byron/Braidwood Unit 2, the analyses confirm that 10 CFR 50.46 acceptance criterion (b)(1) (i.e., PCT less than 2200°F) is demonstrated.

10 CFR 50.46(b)(2)

The maximum cladding oxidation corresponds to a bounding estimate of the 95th percentile LMO at the 95% confidence level. Since the resulting LMOs for the limiting case is 5.51 percent for Byron/Braidwood Unit 1, and 8.27 percent for Byron/Braidwood Unit 2, the analyses confirm that 10 CFR 50.46 acceptance criterion (b)(2) (i.e., LMO of the cladding less than 17 percent) is demonstrated.

The maximum expected total of the normal operation (i.e., pre-transient) and LOCA transient oxidation for any time in life was also considered. The pre-transient oxidation increases with burnup, from a low value at beginning of life to a maximum value at the discharge of the fuel (i.e., end of life). The transient oxidation decreases with burnup when considering consistent peaking factor burndown credit. It has been demonstrated that the sum of the pre-transient plus transient oxidation remains below 17 percent at all times in life.

10 CFR 50.46(b)(3)

The limiting CWO corresponds to a bounding estimate of the 95th percentile CWO at the 95% 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 0.25 percent for Byron/Braidwood Unit 1, and 0.33 percent for Byron/Braidwood Unit 2. Thus, a detailed CWO calculation is not needed because the calculations would include many lower power assemblies and the outcome would always be less than the limiting HAR total maximum oxidation. Therefore, the analyses confirm that 10 CFR 50.46 acceptance criterion (b)(3) (i.e., CWO less than 1 percent) is demonstrated.

10 CFR 50.46(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 the PCT and maximum cladding oxidation limits remain in effect for BELOCA applications. The grid crush calculations currently in place, which include combined LOCA and seismic loads, remain unchanged with the application of the ASTRUM methodology; therefore, acceptance criterion (b)(4) is satisfied.

10 CFR 50.46(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 to maintain long-term cooling remain unchanged with the application of the ASTRUM methodology.

Based on the ASTRUM analyses results presented in Tables 2-1 and 2-2, it is concluded that the Byron/Braidwood Unit 1, and Byron/Braidwood Unit 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 the 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 four loop plants. Braidwood Station and Byron Station are Westinghouse four loop plants.

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

The proposed change incorporates the best estimate LBLOCA analyses using ASTRUM into the Braidwood Station and Byron Station licensing bases, and revises TS Section 5.6.5.b to add WCAP-16009-P-A to the list of NRC-approved methods for establishing core operating limits.

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

10 CFR 50.46 includes requirements and acceptance criteria pertaining to the evaluation of post accident ECCS 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 cooling performance is compared to the criteria ... there is a high level of probability that the criteria would not be exceeded."

The proposed change requests NRC approval to use the ASTRUM methodology described in WCAP-16009-P-A 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 10 CFR 50.46. The NRC has reviewed WCAP-16009-P-A and found it acceptable for referencing in licensing applications for Westinghouse designed four loop PWRs. WCAP-16009-P-A is applicable to Braidwood Station and Byron Station, and the plant-specific application of the ASTRUM methodology to the LBLOCA analyses have been performed in accordance with the conditions and limitations of the topical report and the associated NRC safety evaluation. The plant-specific analyses demonstrate that the requirements of 10 CFR 50.46 will continue to be met, thus ensuring continued safe plant operation.

4.2 No Significant Hazards Consideration

In accordance with 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit," Exelon Generation Company, LLC (EGC) requests an amendment to Facility Operating License Nos. NPF-72 and NPF-77 for Braidwood Station, Units 1 and 2, and Facility Operating License Nos. NPF-37 and NPF-66 for Byron Station, Units 1 and 2. The proposed change revises Technical Specifications (TS) Section 5.6.5, "Core Operating Limits Report (COLR)," to replace the existing large break loss-of-coolant accident (LOCA) analysis methodology. Specifically, the proposed change adds a reference to WCAP-16009-P-A, "Realistic Large Break LOCA Evaluation Methodology Using Automated Statistical Treatment of Uncertainty Method (ASTRUM)" to TS 5.6.5.b. The NRC approved WCAP-16009-P-A in a safety evaluation dated November 5, 2004.

According to 10 CFR 50.92, "Issuance of amendment," paragraph (c), a proposed amendment to an operating license involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not:

(1) Involve a significant increase in the probability or consequences of any accident previously evaluated; or

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or

(3) Involve a significant reduction in a margin of safety.

EGC has evaluated the proposed change, using the criteria in 10 CFR 50.92, and has determined that the proposed change does not involve a significant hazards consideration. The following information is provided to support a finding of no significant hazards consideration.

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

   Response: No

   The proposed change revises TS Section 5.6.5 to incorporate a new large break LOCA analysis methodology. Specifically, the proposed change adds WCAP-16009-P-A to TS 5.6.5.b as a method used for establishing core operating limits.

   Accident analyses are not accident initiators; therefore, the proposed 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. Large break LOCA analyses performed consistent with the methodology in NRC approved WCAP-16009-P-A, including applicable assumptions, limitations and conditions, demonstrate that 10 CFR 50.46 acceptance criteria are met; thus, this change does not involve a significant increase in the consequences of an accident. No physical changes to the plant are associated with the proposed 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 accident previously evaluated?

   Response: No

   The proposed change revises TS Section 5.6.5 to incorporate a new large break LOCA analysis methodology. Specifically, the proposed change adds WCAP-16009-P-A to TS 5.6.5.b as a method used for establishing core operating limits. There are no physical changes being made to the plant as a result of using the Westinghouse ASTRUM analysis methodology in WCAP-16009-P-A for performance of the large break LOCA analyses. Large break LOCA analyses performed consistent with the methodology in NRC approved WCAP-16009-P-A, including applicable assumptions, limitations and conditions, demonstrate that 10 CFR 50.46 acceptance criteria are met. No new modes of plant operation are being introduced. The configuration, operation, and accident response of the structures or components are unchanged by use 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 analyses are within the design limits of existing plant equipment.

In addition, employing the Westinghouse ASTRUM large break LOCA analysis methodology does not create any new failure modes that could lead to a different kind of accident. The design of 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 instrumentation actuation setpoints. Adding the reference to WCAP-16009-P-A in TS Section 5.6.5.b is an administrative change that does not create the possibility of a new or different kind of accident.

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

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

Response: No

The proposed change revises TS Section 5.6.5 to incorporate a new large break LOCA analysis methodology. Specifically, the proposed change adds WCAP-16009-P-A to TS 5.6.5.b as a method used for establishing core operating limits. The analyses using ASTRUM demonstrated that the applicable acceptance criteria in 10 CFR 50.46 are met. Margins of safety for large break LOCAs include quantitative limits for fuel performance established in 10 CFR 50.46. These acceptance criteria are not being changed by this proposed new methodology. Large break LOCA analyses performed consistent with the methodology in NRC approved WCAP-16009-P-A, including applicable assumptions, limitations and conditions, demonstrate that 10 CFR 50.46 acceptance criteria are met; thus, this change does not involve a significant reduction in a margin of safety.

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

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

4.3 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 the health and safety of the public.

5.0 ENVIRONMENTAL CONSIDERATION

EGC 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, "Standards for Protection Against Radiation." 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, "Criterion for categorical exclusion; identification of licensing and regulatory actions eligible for categorical exclusion or otherwise not requiring environmental review," paragraph (c)(9). Therefore, pursuant to 10 CFR 51.22, paragraph (b), no environmental impact statement or environmental assessment needs to be prepared in connection with the proposed amendment.

6.0 REFERENCES

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

2. Letter from H. N. Berkow (U.S. NRC) to J. A. Gresham (Westinghouse Electric Company), "Final Safety Evaluation for WCAP-16009-P, Revision 0, 'Realistic Large Break LOCA Evaluation Methodology Using Automated Statistical Treatment of Uncertainty Method (ASTRUM)' (TAC No. MB9483)," dated November 5, 2004

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

4. Federal Register, Volume 39, Number 3, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water-Cooled Nuclear Power Reactors," dated January 4, 1974

5. SECY-83-472, "Emergency Core Cooling System Analysis Methods," dated November 17, 1983

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

7. NUREG/CR-5249, "Quantifying Reactor Safety Margins: Application of Code Scaling, Applicability, and Uncertainty Evaluation Methodology to a Large-Break, Loss-of-Coolant Accident," dated December 1989

ATTACHMENT 2
Markup of Proposed Technical Specifications Page for Braidwood Station

Braidwood Station, Units 1 and 2

Facility Operating License Nos. NPF-72 and NPF-77

REVISED TECHNICAL SPECIFICATIONS PAGE

5.6-4

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

5. ComEd letter from D. Saccomando to the Office of Nuclear Reactor Regulation dated December 21, 1994, transmitting an attachment that documents applicable sections of WCAP-11992/11993 and ComEd application of the UET methodology addressed in "Additional Information Regarding Application for Amendment to Facility Operating Licenses-Reactivity Control Systems."

6. ~~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.~~

7. WCAP-10079-P-A, "NOTRUMP, A Nodal Transient Small Break and General Network Code," August 1985.

8. WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model using NOTRUMP Code," August 1985.

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

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

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; and

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

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

ATTACHMENT 3
Markup of Proposed Technical Specifications Page for Byron Station

Byron Station, Units 1 and 2

Facility Operating License Nos. NPF-37 and NPF-66

REVISED TECHNICAL SPECIFICATIONS PAGE

5.6-4

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

5. ComEd letter from D. Saccomando to the Office of Nuclear Reactor Regulation dated December 21, 1994, transmitting an attachment that documents applicable sections of WCAP-11992/11993 and ComEd application of the UET methodology addressed in "Additional Information Regarding Application for Amendment to Facility Operating Licenses-Reactivity Control Systems."

6. ~~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.~~

7. WCAP-10079-P-A, "NOTRUMP, A Nodal Transient Small Break and General Network Code," August 1985.

8. WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model using NOTRUMP Code," August 1985.

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

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

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; and

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

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

ATTACHMENT 4
Markup of Proposed Technical Specifications Bases Page for Braidwood Station

Braidwood Station, Units 1 and 2

Facility Operating License Nos. NPF-72 and NPF-77

REVISED TECHNICAL SPECIFICATIONS BASES PAGE

B 3.5.1-5

BASES

APPLICABLE SAFETY ANALYSES (continued)

Both the large and the small break LOCA analyses model the pipe water volume from the accumulator to the SI accumulator discharge header downstream cold leg injection check valve (SI8948). However, an evaluation was performed neglecting the pipe water volume between the SI accumulator discharge header upstream cold leg injection check valve (SI8956) to the SI accumulator discharge header downstream cold leg injection check valve (SI8948) to address gas accumulation. This evaluation determined that the impact on peak clad temperature was minimal for both the large break and the small break LOCA analyses. Since the range of the allowed accumulator volumes is relatively small and has a minimal effect on peak clad temperature, a nominal water volume is used in the small break LOCA analysis. The small break LOCA analysis assumes a nominal water volume of 7106 gallons based on the Technical Specification (TS) minimum and maximum limits of 6995 gallons (935 ft^3, 31% of indicated level) and 7217 gallons (965 ft^3, 63% of indicated level). The large break LOCA analysis assumes a water volume range of 6882 gallons (920 ft^3, 15% of indicated level) to 7331 gallons (980 ft^3, 79% of indicated level) which bounds the TS limits.

The minimum boron concentration setpoint is used in the post LOCA boron concentration calculation. The calculation is performed to assure reactor subcriticality in a post LOCA environment. Of particular interest is the large break LOCA, since no credit is taken for control rod assembly insertion. A reduction in the accumulator minimum boron concentration would produce a subsequent 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 analyses are performed at the minimum nitrogen cover pressure, since sensitivity analyses have demonstrated that higher nitrogen cover pressure results in a computed peak clad temperature benefit. The large break LOCA analyses are performed at a nitrogen cover pressure range of ~~602 psia to 677 psia~~ 587 psia to 692 psia. The maximum nitrogen cover pressure limit prevents accumulator relief valve actuation, and ultimately preserves accumulator integrity.

ATTACHMENT 5
Markup of Proposed Technical Specifications Bases Page for Byron Station

Byron Station, Units 1 and 2

Facility Operating License Nos. NPF-37 and NPF-66

REVISED TECHNICAL SPECIFICATIONS BASES PAGE

B 3.5.1-5

BASES

APPLICABLE SAFETY ANALYSES (continued)

Both the large and the small break LOCA analyses model the pipe water volume from the accumulator to the SI accumulator discharge header downstream cold leg injection check valve (SI8948). However, an evaluation was performed neglecting the pipe water volume between the SI accumulator discharge header upstream cold leg injection check valve (SI8956) to the SI accumulator discharge header downstream cold leg injection check valve (SI8948) to address gas accumulation. This evaluation determined that the impact on peak clad temperature was minimal for both the large break and the small break LOCA analyses. Since the range of the allowed accumulator volumes is relatively small and has a minimal effect on peak clad temperature, a nominal water volume is used in the small break LOCA analysis. The small break LOCA analysis assumes a nominal water volume of 7106 gallons based on the Technical Specification (TS) minimum and maximum limits of 6995 gallons (935 ft^3, 31% of indicated level) and 7217 gallons (965 ft^3, 63% of indicated level). The large break LOCA analysis assumes a water volume range of 6882 gallons (920 ft^3, 15% of indicated level) to 7331 gallons (980 ft^3, 79% of indicated level) which bounds the TS limits.

The minimum boron concentration setpoint is used in the post LOCA boron concentration calculation. The calculation is performed to assure reactor subcriticality in a post LOCA environment. Of particular interest is the large break LOCA, since no credit is taken for control rod assembly insertion. A reduction in the accumulator minimum boron concentration would produce a subsequent 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 analyses are performed at the minimum nitrogen cover pressure, since sensitivity analyses have demonstrated that higher nitrogen cover pressure results in a computed peak clad temperature benefit. The large break LOCA analyses are performed at a nitrogen cover pressure range of ~~602 psia to 677 psia~~ 587 psia to 692 psia. The maximum nitrogen cover pressure limit prevents accumulator relief valve actuation, and ultimately preserves accumulator integrity.