VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

November 16, 2006

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

Serial No. 06-936
SPS-LIC/CGL R0
Docket Nos. 50-280/281
License Nos. DPR-32/37

VIRGINIA ELECTRIC AND POWER COMPANY
SURRY POWER STATION UNITS 1 AND 2
PROPOSED TECHNICAL SPECIFICATIONS CHANGE
ADDITION OF ASTRUM METHODOLOGY TO CORE OPERATING LIMITS REPORT REFERENCES AND REVISED LARGE BREAK LOCA ANALYSIS

Pursuant to 10 CFR 50.90, Virginia Electric and Power Company (Dominion) requests amendments, in the form of changes to the Technical Specifications (TS) to Facility Operating License Numbers DPR-32 and DPR-37 for Surry Power Station Units 1 and 2, respectively. The proposed change will add a reference in Technical Specification 6.2.C, "Core Operating Limits Report (COLR)," to permit the use of the Westinghouse Best-Estimate Large Break Loss of Coolant Accident (BE-LBLOCA) analysis methodology using the Automated Statistical Treatment of Uncertainty Method (ASTRUM) for the analysis of LBLOCA.

The BE-LBLOCA analysis using ASTRUM was performed for Surry Units 1 and 2 in compliance with the NRC conditions and limitations identified in WCAP-16009-P-A and meets the commitment that Dominion made in a letter dated January 3, 2006 (Serial No. 05-828) to complete the LBLOCA reanalysis by September 30, 2006. Based on the analysis results, it is concluded that Surry Units 1 and 2 continue to maintain a margin of safety to the limits prescribed by 10 CFR 50.46.

A discussion of the proposed TS change is provided in Attachment 1. Attachment 1 also presents the technical basis for the TS change, including the BE-LBLOCA analysis results. The marked-up and proposed TS pages reflecting the proposed change are provided in Attachments 2 and 3, respectively.

This letter requests NRC review and approval of the following:

- The TS change request to add WCAP-16009-P-A to the TS 6.2.C list of NRC-approved methodologies used to determine core operating limits (i.e., the reference list of the Surry COLR).

- The analysis of the Surry LBLOCA which employs the Westinghouse BE-LBLOCA analysis methodology using ASTRUM.

An NRC letter dated October 12, 2006, issued TS Amendments 250/249 approving a Surry TS change request to revise the method for starting the inside and outside recirculation spray (RS) pumps as part of the resolution to GSI-191, *Assessment of Debris Accumulation on PWR Sump Performance*. This change is scheduled for implementation at Surry Power Station Units 1 and 2 during the fall 2007 and fall 2006 refueling outages, respectively. The supporting analyses for the implementation of the Westinghouse BE-LBLOCA analysis methodology using ASTRUM credits these changes to the Engineered Safety Features setpoints. Thus, coordinated implementation of the BE-LBLOCA analysis with the GSI-191 pump start change is planned prior to power operation during startup from the Unit 1 fall 2007 refueling outage. To support this planned implementation, approval of this TS change and the Surry BE-LBLOCA analysis is requested by August 30, 2007.

There are no further TS changes in process that will affect or be affected by this change request. Changes to UFSAR Chapter 14 associated with the BE-LBLOCA analysis will be implemented following NRC approval of this TS change.

We have evaluated the proposed TS change and have determined that it does not involve a significant hazards consideration as defined in 10 CFR 50.92. The basis for that determination is included in Attachment 1. We have also determined that operation with the proposed change will not result in any significant increase in the amount of effluents that may be released offsite or in individual or cumulative occupational radiation exposure. Therefore, the proposed amendment is eligible for categorical exclusion as set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment is needed in connection with the approval of the proposed change. The basis for our determination that the change does not involve any significant increase in effluents or radiation exposure is also included in Attachment 1.

The proposed changes have been reviewed and approved by the Station Nuclear Safety and Operating Committee.

If you have any questions or require additional information, please contact Mr. Gary D. Miller at (804) 273-2771.

Very truly yours,

Gerald T. Bischof
Vice President – Nuclear Engineering

Attachments:
1. Discussion of Change
2. Marked-up Technical Specifications Page
3. Proposed Technical Specifications Page

Commitments made in this letter:
1. Related UFSAR Chapter 14 changes reflecting the BE-LBLOCA analysis will be implemented following NRC approval of this TS change.

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COMMONWEALTH OF VIRGINIA)
)
COUNTY OF HENRICO)

The foregoing document was acknowledged before me, in and for the County and Commonwealth aforesaid, today by Gerald T. Bischof, who is Vice President – Nuclear Engineering, of Virginia Electric and Power Company. He has affirmed before me that he is duly authorized to execute and file the foregoing document in behalf of that Company, and that the statements in the document are true to the best of his knowledge and belief.

Acknowledged before me the 16TH day of November, 2006.

My Commission Expires: May 31, 2010

Vicki L. Hull
Notary Public

(SEAL)

Attachment 1

Discussion of Change

Virginia Electric and Power Company
(Dominion)
Surry Power Station Units 1 and 2

DISCUSSION OF CHANGE

1.0 INTRODUCTION

Virginia Electric and Power Company (Dominion) proposes to add the Westinghouse Best-Estimate Large Break Loss of Coolant Accident (BE-LBLOCA) analysis methodology using the Automated Statistical Treatment of Uncertainty Method (ASTRUM) for the analysis of large break loss of coolant accident (LBLOCA) to the list of methodologies approved for reference in the Core Operating Limits Report (COLR) in Surry Power Station Units 1 and 2 Technical Specification (TS) 6.2.C.

The current methodology at Surry for the analysis of the Emergency Core Cooling System (ECCS) performance for the postulated LBLOCA is the Nuclear Regulatory Commission (NRC) approved version of the Westinghouse LOCA-ECCS evaluation model denoted as the 1981 model with BASH (Reference 1). The associated analytical techniques are in full compliance with 10 CFR 50, Appendix K. As required by Appendix K of 10 CFR 50, certain conservative assumptions were made for the LOCA-ECCS analysis. The assumptions pertain to the conditions of the reactor and associated safety system equipment at the time that the LOCA is assumed to occur and include such items as the core peaking factors, the containment pressure, and the performance of the ECCS.

In 1988, the NRC Staff amended the requirements of 10 CFR 50.46 and Appendix K, "ECCS Evaluation Models," to permit the use of a realistic evaluation model to analyze the performance of the ECCS during a hypothetical LOCA. To demonstrate use of the revised ECCS rule, the NRC and its consultants developed a method called the Code Scaling, Applicability, and Uncertainty (CSAU) evaluation methodology. This method outlined an approach for defining and qualifying a best-estimate thermal-hydraulic code and quantifying the uncertainties in a LOCA analysis. A LOCA evaluation methodology for three-loop and four-loop Pressurized Water Reactor (PWR) plants based on the revised 10 CFR 50.46 rule was developed by Westinghouse and approved by the NRC (Reference 2).

More recently, Westinghouse developed an alternative uncertainty methodology called ASTRUM, which stands for Automated Statistical Treatment of Uncertainty Method. This method is still based on the Code Qualification Document (CQD) technology (Reference 2) and follows the steps in the CSAU methodology. However, the uncertainty analysis (Element 3 in CSAU) is replaced by a technique based on order statistics. The ASTRUM methodology replaces the response surface technique with a statistical sampling method where the uncertainty parameters are simultaneously sampled for each case. The ASTRUM methodology has received NRC approval in WCAP-16009-P-A (Reference 3).

A BE-LBLOCA analysis using ASTRUM has been completed for Surry Units 1 and 2. Reference 4 provides the technical basis for the USNRC review and approval of the implementation of the Westinghouse BE-LBLOCA using ASTRUM for the Surry analysis of the LBLOCA event. The Surry analysis was performed in compliance with the NRC conditions and limitations identified in WCAP-16009-P-A (Reference 3). Based on the analysis results, it is concluded that Surry Units 1 and 2 continue to maintain a margin of safety to the limits prescribed by 10 CFR 50.46.

By letter dated January 31, 2006 (Reference 5), Dominion requested changes to the Surry Technical Specifications to support a change to the method for starting the inside and outside recirculation spray (RS) pumps as part of the resolution to Generic Safety Issue (GSI) 191. These changes are scheduled for implementation at Surry Power Station Units 1 and 2 during the fall 2007 and fall 2006 refueling outages, respectively. The supporting analyses for the implementation of the Westinghouse BE-LBLOCA analysis methodology using ASTRUM (Reference 4) credits these changes to the Engineered Safety Features setpoints.

This license amendment request (LAR) for operating license number DPR-32 for Surry Unit 1 and number DPR-37 for Surry Unit 2 requests approval to incorporate the Westinghouse BE-LBLOCA analysis methodology using ASTRUM to the list of methodologies approved for reference in the Core Operating Limits Report (COLR) in TS 6.2.C. Further, this LAR requests a simultaneous implementation for the BE-LBLOCA analysis amendments for Surry Units 1 and 2 prior to power operation during startup from the Unit 1 fall 2007 refueling outage.

By letter dated January 3, 2006 (Reference 6), Dominion committed to provide the NRC new LBLOCA analyses for Surry using the Westinghouse BE-LBLOCA methodology with ASTRUM. The analysis and associated TS changes proposed in this LAR are presented in fulfillment of that commitment.

2.0 PROPOSED TECHNICAL SPECIFICATIONS CHANGES

The current Surry TS 6.2.C, Core Operating Limits Report (COLR), contains references to the analytical methods used to determine the core operating limits. The specific proposed changes are provided below:

TS 6.2.C, CORE OPERATING LIMITS REPORT (COLR)

TS 6.2.C is revised to delete the current References 2a, 2b, and 2c, to add a new reference that reflects ASTRUM (Reference 2a), and to renumber References 2d, 2e, and 2f as References 2b, 2c, and 2d. The new TS 6.2.C.2.a is as follows:

2a. WCAP-16009-P-A, "Realistic Large Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)," (Westinghouse Proprietary).

3.0 SAFETY SIGNIFICANCE SUMMARY

The Westinghouse BE-LBLOCA analysis methodology using the statistical treatment of uncertainties methodology, ASTRUM, has been approved by the USNRC (Reference 3). In addition, the NRC has approved the use of ASTRUM for PWR safety analyses at several sites, including Beaver Valley, Farley, Ginna, Indian Point, and Summer. An analysis of the LBLOCA for Surry Units 1 and 2 has been performed with the Westinghouse BE-LBLOCA analysis methodology using ASTRUM and is documented in Section 4.0 of this Discussion of Change. The analysis was performed in compliance with all the NRC conditions and limitations identified in WCAP-16009-P-A (Reference 3). Based on the analysis results, it is concluded that the Surry Units 1 and 2 continue to maintain a margin of safety to the limits prescribed by 10 CFR 50.46.

4.0 LICENSING REPORT - TECHNICAL BASIS FOR THE PROPOSED TECHNICAL SPECIFICATIONS CHANGES

4.1 Background

A BE-LBLOCA analysis has been completed for the Surry Units 1 and 2. This license amendment request (LAR) for operating license number DPR-32 for Surry Unit 1 and number DPR-37 for Surry Unit 2 requests approval to apply the Westinghouse BE-LBLOCA analysis methodology.

Westinghouse obtained generic NRC approval of its original topical report describing BE-LBLOCA methodology in 1996. NRC approval of the methodology is documented in the NRC safety evaluation report appended to the topical report (Reference 2).

Westinghouse recently underwent a program to revise the statistical approach used to develop the Peak Cladding Temperature (PCT) and oxidation results at the 95th percentile. This method is still based on the CQD technology (Reference 2) and follows the steps in the CSAU methodology (Reference 10). However, the uncertainty analysis (Element 3 in CSAU) is replaced by a technique based on order statistics. The ASTRUM methodology replaces the response surface technique with a statistical sampling method where the uncertainty parameters are simultaneously sampled for each case. The approved ASTRUM evaluation model is documented in WCAP-16009-P-A (Reference 3).

This licensing report summarizes the application of the Westinghouse ASTRUM BE-LBLOCA evaluation model to the Surry Units 1 and 2 analysis of the LBLOCA event. The analysis was performed in compliance with the NRC conditions and limitations identified in WCAP-16009-P-A (Reference 3). Because there are no consequential differences between Surry Units 1 and 2, a single WCOBRA/TRAC geometric model was developed and is applicable to both Surry units. Table 1 lists the major plant parameter assumptions used in the BE-LBLOCA analysis for Surry Units 1 and 2.

The ASTRUM methodology requires the execution of 124 transients to determine a bounding estimate of the 95^{th} percentile of the PCT, Local Maximum Oxidation (LMO), and Core Wide Oxidation (CWO) with a 95% confidence level. These results must satisfy the 10 CFR 50.46 criteria with regard to PCT, LMO, and CWO.

4.2 Method of Thermal Analysis

When the Final Acceptance Criteria governing the LOCA for Light Water Reactors was issued in 10 CFR 50.46 (Reference 7), both the Nuclear Regulatory Commission (NRC) and the industry recognized that the stipulations of Appendix K 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 permitted 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 the overly conservative Appendix K requirements. In recognition of the Appendix K conservatism that was being quantified by the research programs, the NRC adopted an interim approach for evaluation methods. This interim approach is described in SECY-83-472 (Reference 8). The SECY-83-472 approach retained those features of Appendix K that were legal requirements, but permitted applicants to use best-estimate thermal-hydraulic models in their ECCS evaluation model. Thus, SECY-83-472 represented an important step in basing licensing decisions on realistic calculations, as opposed to those calculations prescribed by Appendix K.

In 1988, the NRC Staff amended the requirements of 10 CFR 50.46 and Appendix K, "ECCS Evaluation Models," to permit the use of a realistic evaluation model to analyze the performance of the ECCS during a hypothetical LOCA. This decision was based on an improved understanding of LOCA thermal-hydraulic phenomena gained by extensive research programs. Under the amended rules, best-estimate thermal-hydraulic models may be used in place of models with Appendix K features. The rule change also requires, as part of the LOCA analysis, an assessment of the uncertainty of the best-estimate calculations. It further requires that this analysis uncertainty be included when comparing the results of the calculations to the prescribed acceptance criteria of 10 CFR 50.46. Further guidance for the use of best-estimate codes is provided in Regulatory Guide 1.157 (Reference 9).

To demonstrate use of the revised ECCS rule, the NRC and its consultants developed a method called the CSAU evaluation methodology (Reference 10). This method outlined an approach for defining and qualifying a best-estimate thermal-hydraulic code and quantifying the uncertainties in a LOCA analysis.

A LOCA evaluation methodology for three-loop and four-loop PWR plants based on the revised 10 CFR 50.46 rules was developed by Westinghouse with the support of EPRI and Consolidated Edison and has been approved by the NRC (Reference 2).

More recently, Westinghouse developed an alternative uncertainty methodology called ASTRUM (Reference 3). This method is still based on the CQD methodology (Reference 2) and follows the steps in the CSAU methodology (Reference 10). However, the uncertainty analysis (Element 3 in the CSAU) is replaced by a technique based on order statistics. The ASTRUM methodology replaces the response surface technique with a statistical sampling method where the uncertainty parameters are simultaneously sampled for each case. The ASTRUM methodology has received NRC approval for reference in licensing calculations in WCAP-16009-P-A (Reference 3). The ASTRUM methodology remains applicable to three-loop and four-loop PWRs, as well as two-loop Westinghouse plants with upper plenum injection. This methodology was also extended to Combustion Engineering PWRs.

The three 10 CFR 50.46 criteria (peak clad temperature, maximum local oxidation, and core-wide oxidation) are satisfied by running a sufficient number of WCOBRA/TRAC calculations (sample size). In particular, the statistical theory predicts that 124 calculations are required to simultaneously bound the 95th percentile values of these three parameters with a 95-percent confidence level.

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

4.3 Description of a Large Break LOCA Transient

Before the break occurs, the RCS (Reactor Coolant System) 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. Traditionally, cold leg breaks have been limiting for large break LOCA. This location is the one where flow stagnation in the core appears most likely to occur. Scoping studies with WCOBRA/TRAC have confirmed that the cold leg remains the limiting break location (Reference 2).

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 accumulators begin 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 re-wets 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 and the broken loop hot leg, steam generators, and 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 ECCS water aids in the filling of the downcomer and subsequently supplies water to maintain downcomer water level and complete the reflood period.

4.4 ASTRUM Analysis Results for Surry Units 1 and 2

The Surry Units 1 and 2 PCT-limiting transient is a double-ended guillotine cold leg (DEGCL) break that considers the assumptions listed in Table 1.

Table 2 summarizes the results of the ASTRUM BE-LBLOCA analysis. The sequence of events following a nominal large DEGCL break LOCA is presented in Table 3.

The scatter plot presented in Figure 1 shows the impact 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 because the break area is a contributor to the variation in PCT.

Figures 2, 3, and 4 are presented to show the limiting cladding transient for each 10 CFR 50.46 criterion analyzed in the ASTRUM analysis. Figure 2 shows the HOTSPOT predicted clad temperature transient at the PCT limiting elevation for the limiting PCT case. Figure 3 presents the HOTSPOT clad temperature transient predicted at the LMO elevation for the limiting LMO case. Figure 4 shows the WCOBRA/TRAC predicted peak cladding temperature for the CWO limiting transient.

Figures 5 through 18 were generated using the limiting PCT case. The PCT-limiting case was chosen to illustrate a conservative representation of the response to a LBLOCA.

Figure 5 is a plot of the pressurizer pressure throughout the PCT-limiting transient. Figures 6 and 7 are plots of the mass flow rate through the break (vessel and loop side, respectively). Figure 8 presents the void fraction in both the intact and broken loop pumps; the dashed curve represents the broken loop pump. Figure 9 is a plot of the vapor flow rate at the top of the core above the hot assembly.

Figure 10 is a plot of an intact loop accumulator injection flow. Figure 11 is a plot of the safety injection flow into one of the intact cold legs. Figures 12, 13, and 14 are plots of the lower plenum, downcomer, and core average channel collapsed liquid levels, respectively. The reference point for the downcomer liquid level 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 is the bottom of the active fuel.

The vessel fluid inventory throughout the transient is plotted in Figure 15. Figure 16 is a plot of the PCT for all 5 rods modeled in WCOBRA/TRAC, and Figure 17 is a plot of the hot rod PCT elevation versus time. Note, the PCTs in Figure 16 are the WCOBRA/TRAC calculated temperatures, not the HOTSPOT calculated temperatures (Figures 2, 3, and 4 are HOTSPOT calculated temperatures).

The containment backpressure utilized in the BE-LBLOCA WCOBRA/TRAC analysis is shown in Figure 18.

4.5 10 CFR 50.46 Requirements

It must be demonstrated that there is a high level of probability that the limits set forth in 10 CFR 50.46 are met. The demonstration that these limits are met is as follows:

(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 2044°F, the analysis confirms that 10 CFR 50.46 acceptance criterion (b)(1), i.e., "Peak Clad Temperature less than 2200°F", is satisfied. 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 5.26 percent, the analysis confirms that 10 CFR 50.46 acceptance criterion (b)(2), i.e., "Local Maximum Oxidation of the cladding less than 17 percent," is satisfied. 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. The limiting Hot Assembly Rod (HAR) total maximum oxidation is 0.32 percent. A detailed CWO calculation takes advantage of the core power census that includes many lower power assemblies. Because there is significant margin to the regulatory limit, the CWO value can be conservatively chosen as that calculated for the limiting HAR. A detailed CWO calculation is not needed because the outcome will always be less than 0.32 percent. Therefore, the analysis confirms that 10 CFR 50.46 acceptance criterion (b)(3), i.e., "Core-Wide Oxidation less than 1 percent," is satisfied. The result is shown in Table 2.

(b)(4) 10 CFR 50.46 acceptance criterion (b)(4) requires that the calculated changes in core geometry are such that the core remains amenable to cooling. This criterion has historically been satisfied by adherence to criteria (b)(1) and (b)(2), and by assuring that fuel deformation due to combined LOCA and seismic loads is specifically addressed. It has been demonstrated that the PCT and maximum cladding oxidation limits have been satisfied. Furthermore, the approved methodology (Reference 2) specifies that effects of LOCA and seismic loads on core geometry do not need to be considered unless grid crushing extends beyond the 28 assemblies in the low-power channel. This situation is not calculated to occur for Surry Units 1 and 2; therefore, acceptance criterion (b)(4) is satisfied.

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

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

Table 1: Major Plant Parameter Assumptions Used in the Surry Units 1 and 2 Best-Estimate Large Break LOCA ASTRUM Analysis

Parameter	Value
Plant Physical Description	
• SG Tube Plugging	≤15%
Plant Initial Operating Conditions	
• Reactor Power	≤ 2597 MWt
• Peaking Factors	$F_Q \leq 2.6$ $F_{\Delta H} \leq 1.7$
• Axial Power Distribution	See Figure 19
Fluid Conditions	
• T_{AVG}	$573 - 7.1°F \leq T_{AVG} \leq 573 + 7.1°F$
• Pressurizer Pressure	$2250 - 60$ psia $\leq P_{RCS} \leq 2250 + 60$ psia
• Reactor Coolant Flow	≥ 88,500 gpm/loop
• Accumulator Temperature	$89°F \leq T_{ACC} \leq 110°F$
• Accumulator Pressure	580 psia $\leq P_{ACC} \leq$ 700 psia
• Accumulator Water Volume	970 $ft^3 \leq V_{ACC} \leq$ 1030 ft^3
• Accumulator Boron Concentration	≥ 2000 ppm
Accident Boundary Conditions	
• Single Failure Assumptions	Loss of one ECCS train
• Safety Injection Flow	Minimum
• Safety Injection Temperature	$37.5°F \leq T_{SI} \leq 62.5°F$
• Safety Injection Initiation Delay Time	≤ 25 sec (with offsite power) ≤ 40 sec (without offsite power)
• Containment Pressure	Bounded (minimum)

Table 2: Surry Units 1 and 2 Best-Estimate Large Break LOCA Results

10 CFR 50.46 Requirement	Value	Criteria
95/95 PCT[1] (°F)	2,044	< 2,200
95/95 LMO[2] (%)	5.26	< 17
95/95 CWO[3] (%)	0.32	< 1

Table 3: Surry Units 1 and 2 Best-Estimate Large Break LOCA Sequence of Events for Limiting PCT Case

Event	Time (sec)
Start of Transient	0.0
Safety Injection Signal	5.0
Accumulator Injection Begins	8.5
End of Blowdown	23.0
Accumulator Empty	30.0
Bottom of Core Recovery	31.0
Safety Injection Begins	45.0
PCT Occurs	245.0
Core Quenched	550.0
End of Transient	1450.0

[1] Peak Cladding Temperature
[2] Local Maximum Oxidation
[3] Core-Wide Oxidation

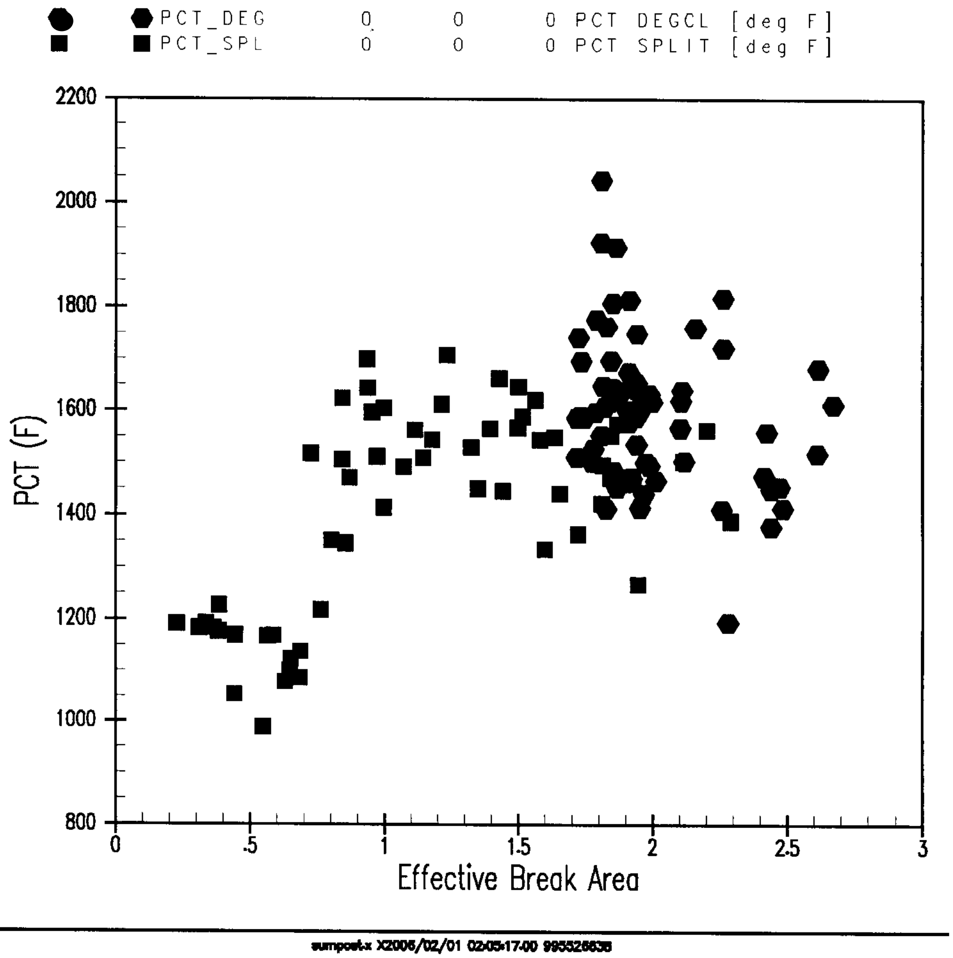


Figure 1: Surry Units 1 and 2 PCT vs. Effective Break Area

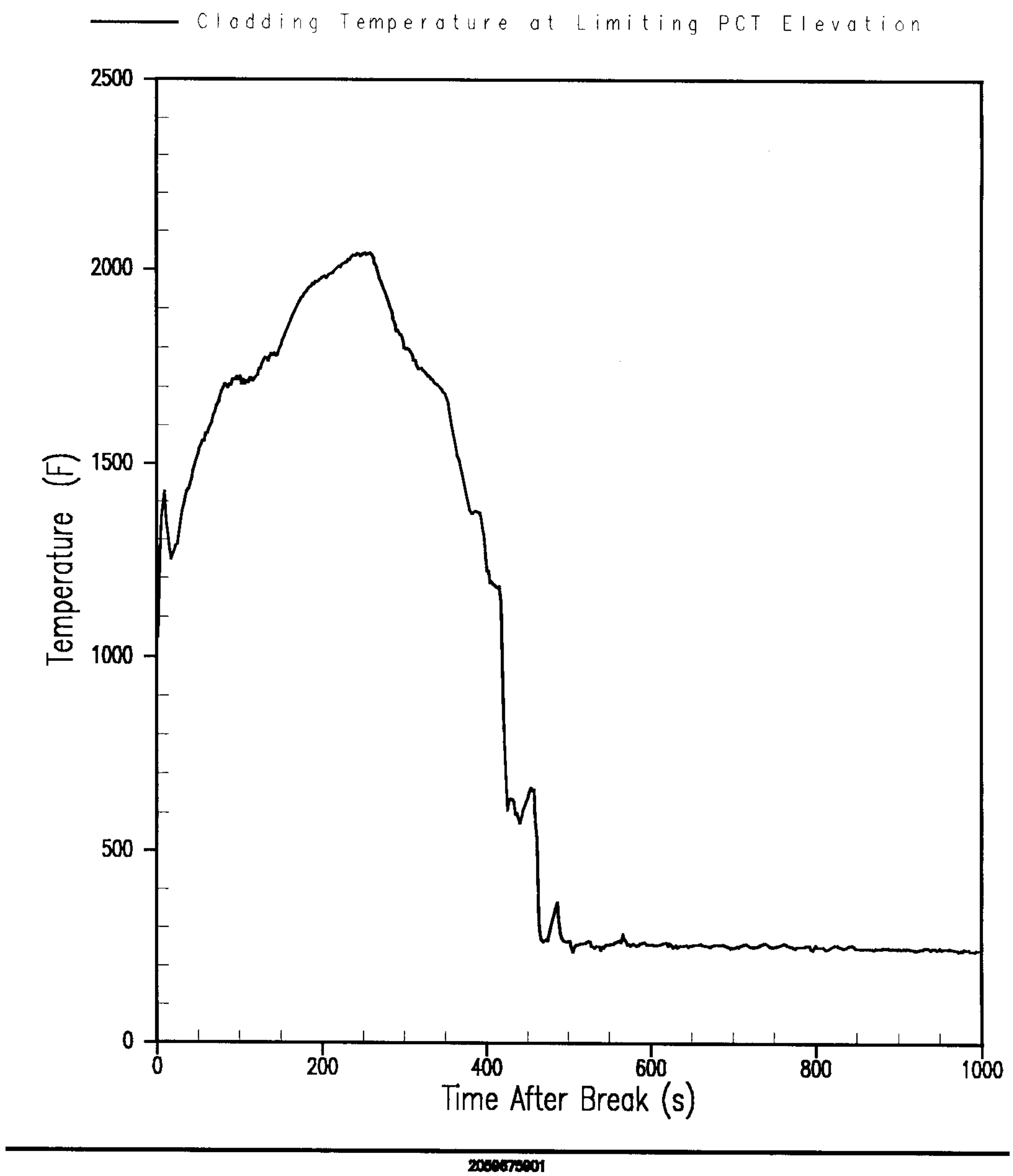


Figure 2: Surry Units 1 and 2 BE-LBLOCA Analysis HOTSPOT Clad Temperature Transient at the Limiting Elevation for the Limiting PCT Case

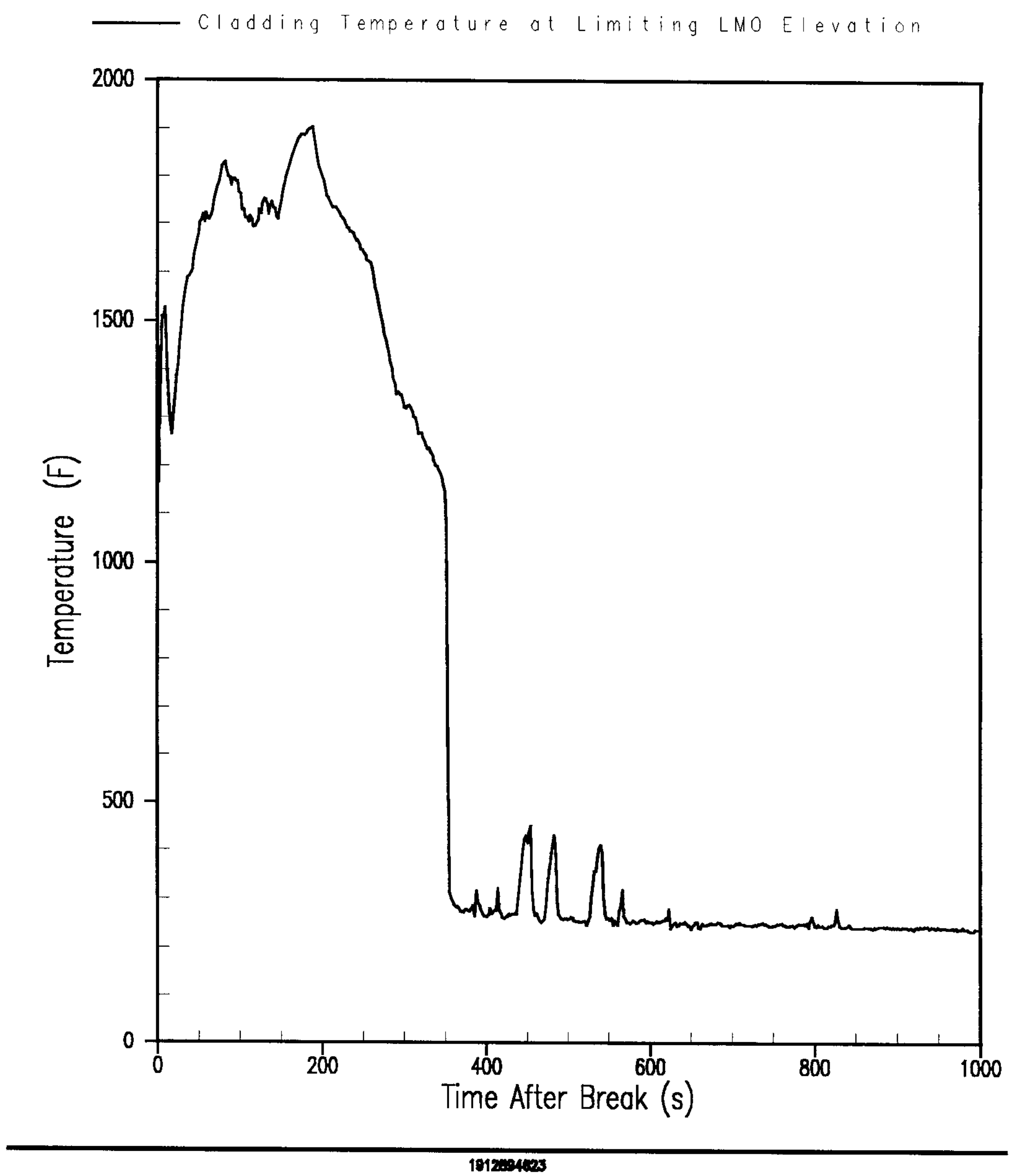


Figure 3: Surry Units 1 and 2 BE-LBLOCA Analysis HOTSPOT Clad Temperature Transient at the Limiting Elevation for the Limiting LMO Case

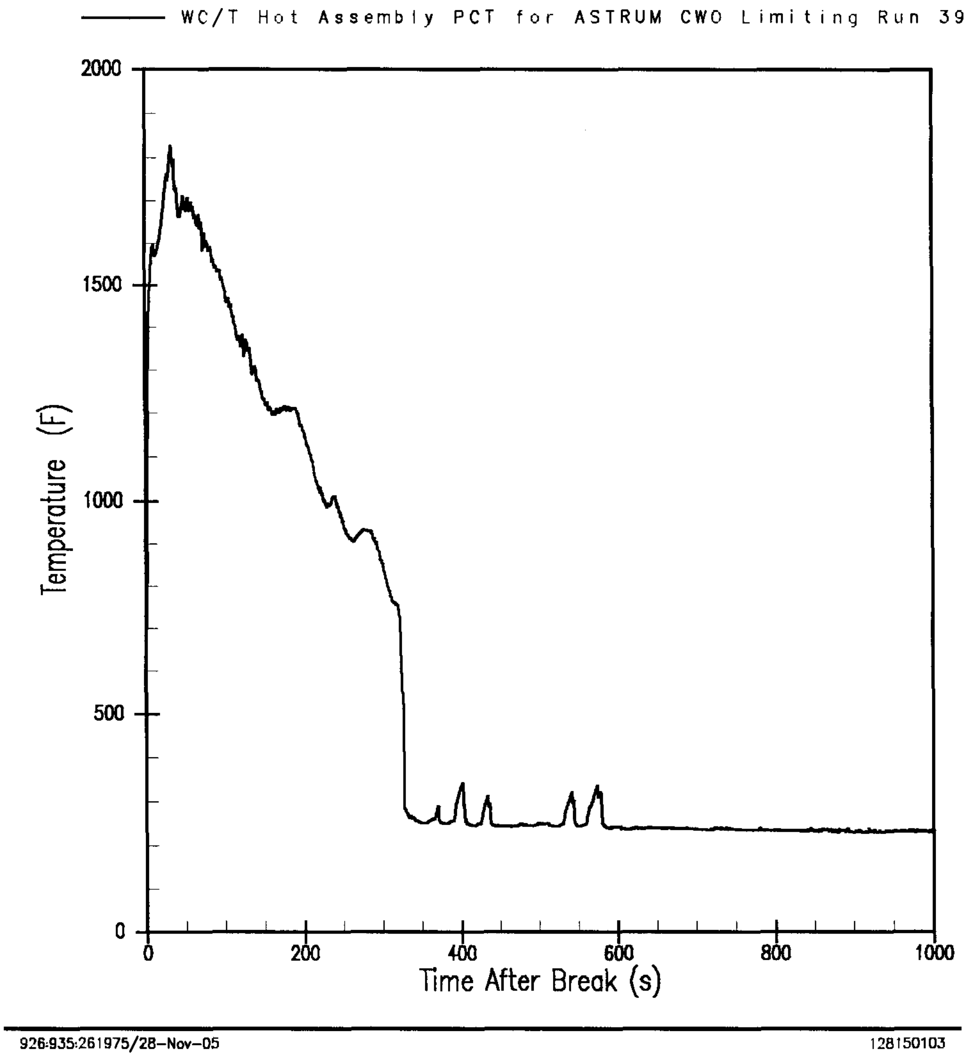


Figure 4: Surry Units 1 and 2 BE-LBLOCA Analysis WCOBRA/TRAC Hot Assembly PCT Transient for the Limiting CWO Case

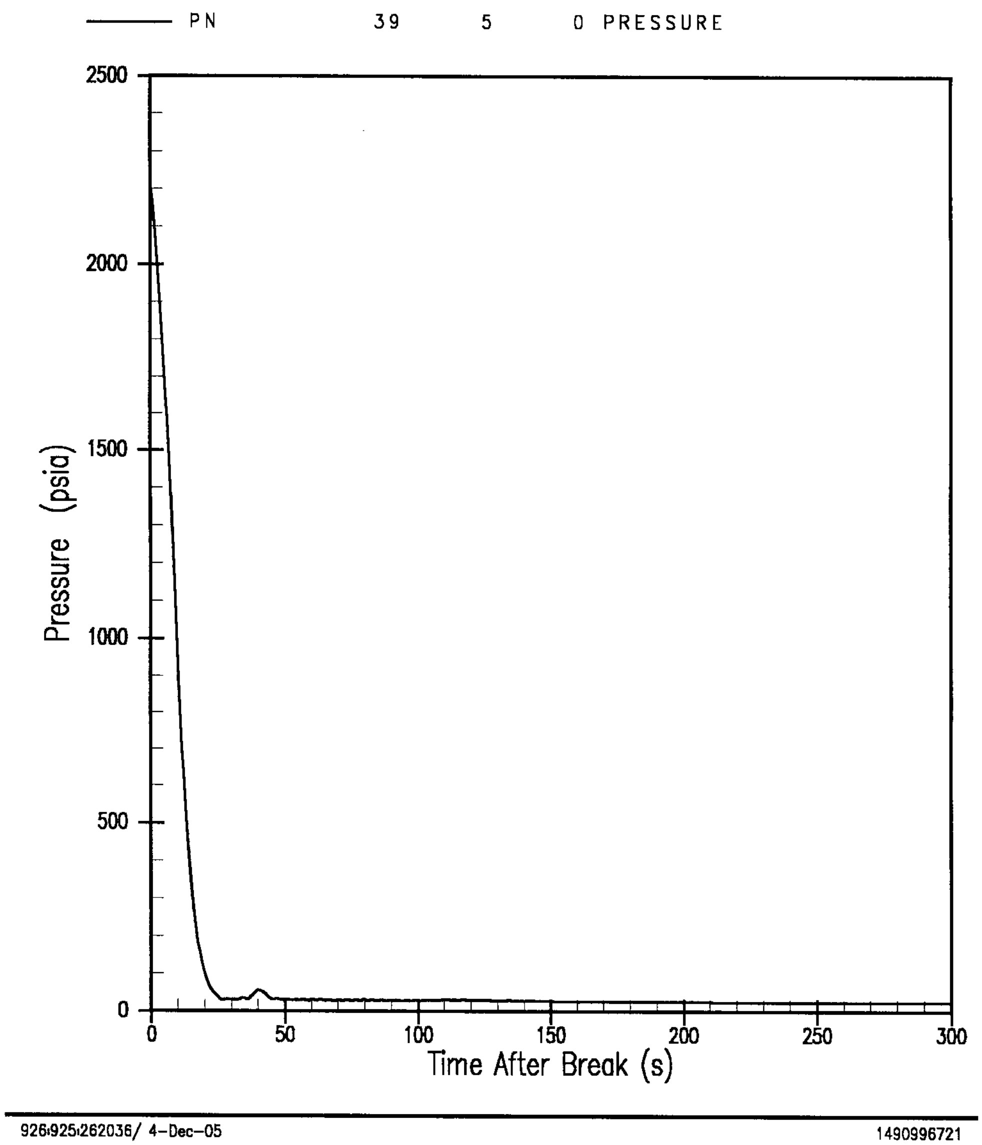


Figure 5: Surry Units 1 and 2 Limiting PCT Case Pressurizer Pressure

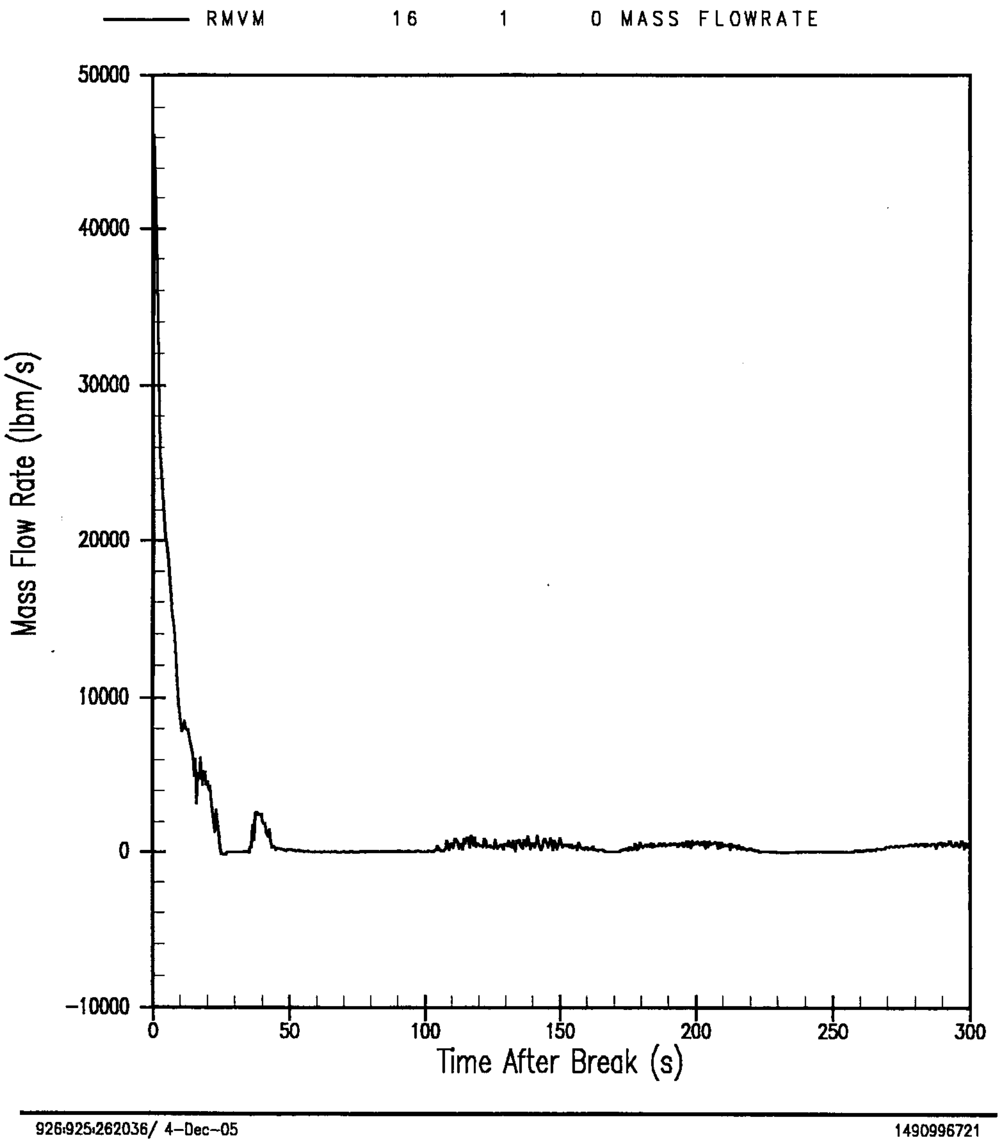


Figure 6: Surry Units 1 and 2 Limiting PCT Case Vessel Side Break Flow

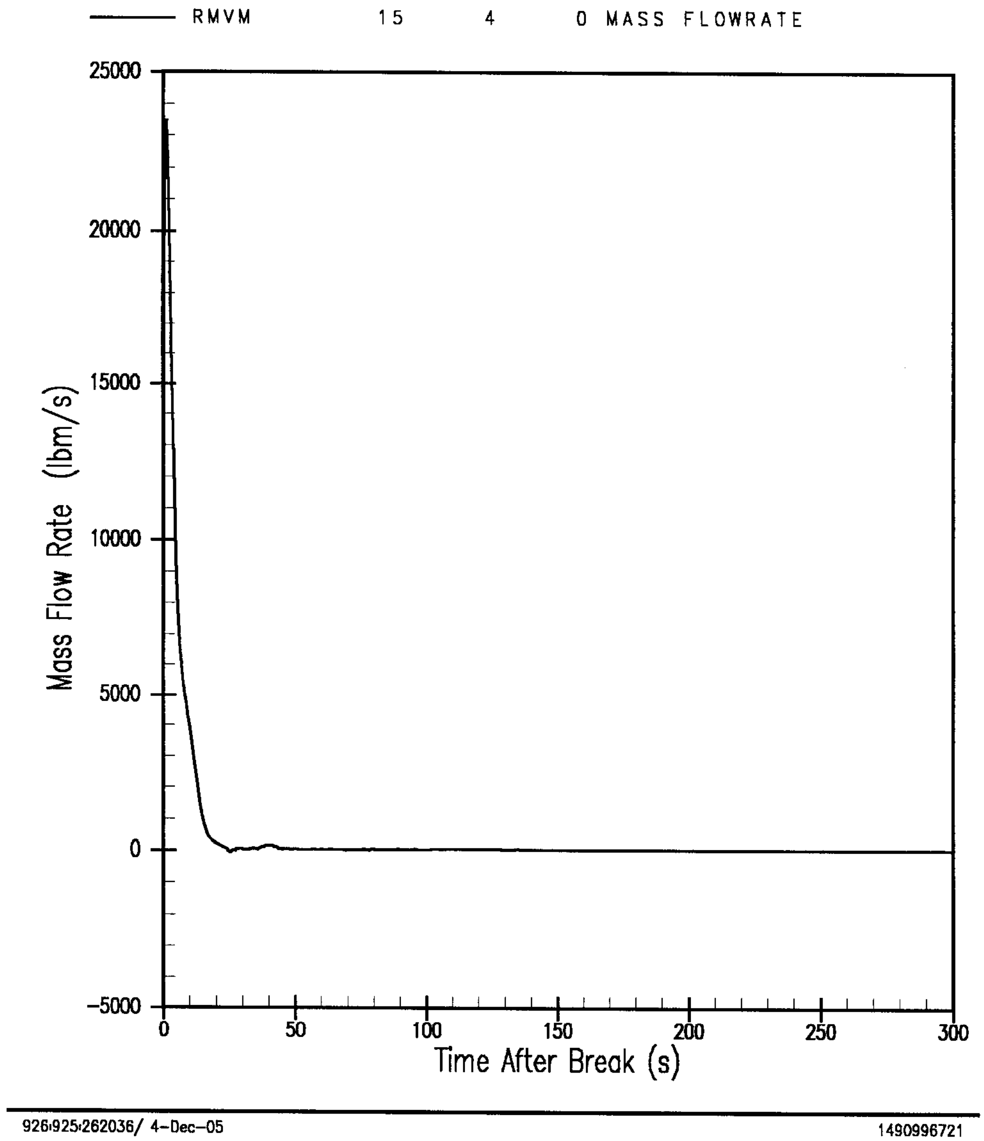


Figure 7: Surry Units 1 and 2 Limiting PCT Case Loop Side Break Flow

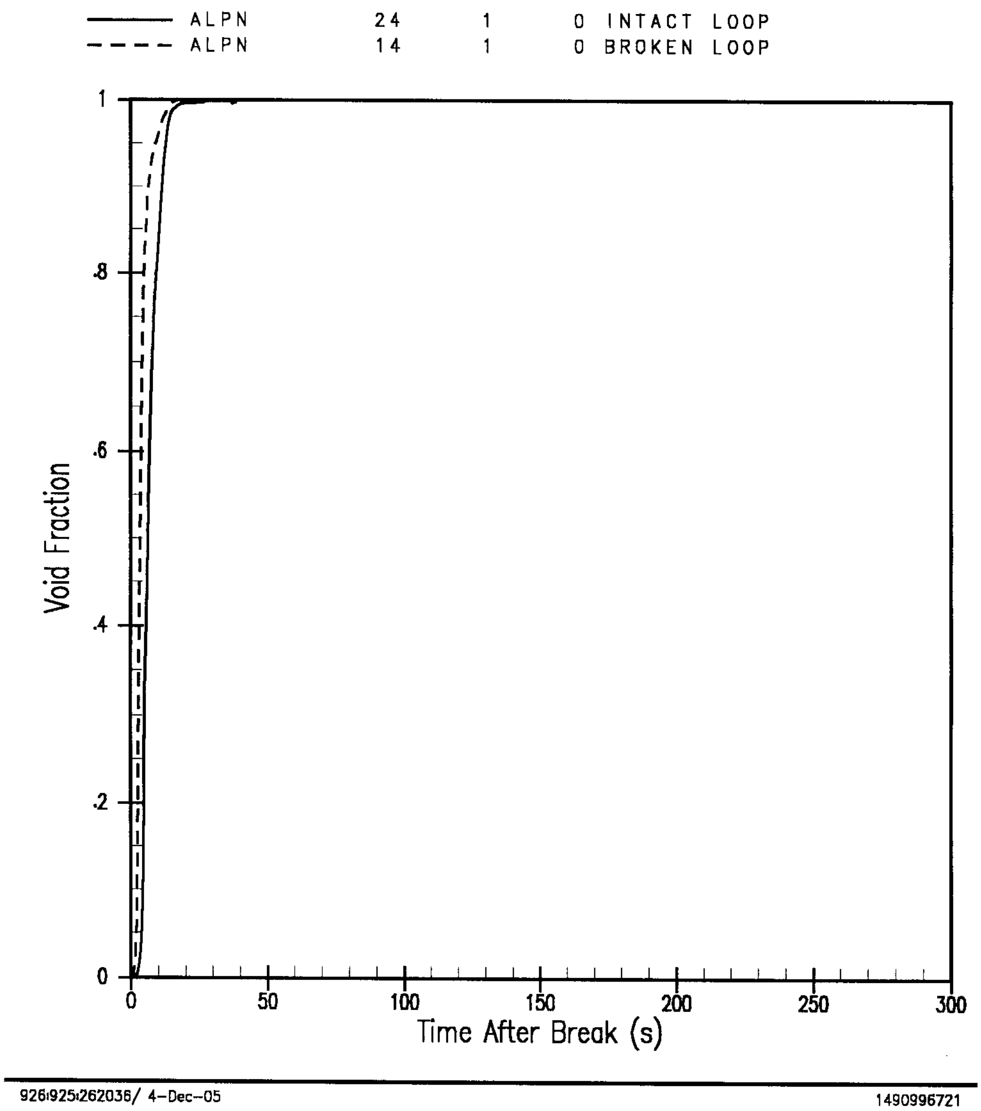


Figure 8: Surry Units 1 and 2 Limiting PCT Case Broken and Intact Loop Pump Void Fraction

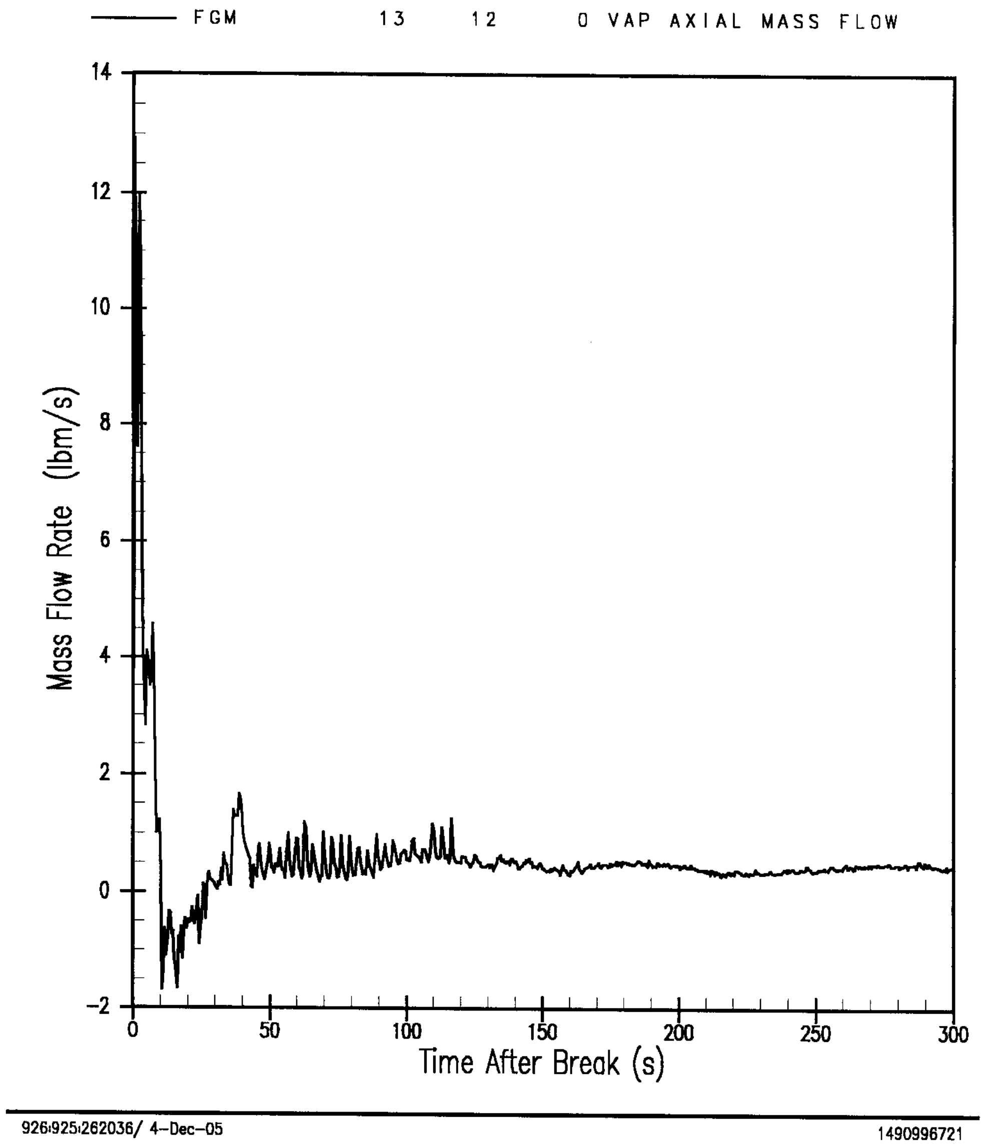


Figure 9: Surry Units 1 and 2 Limiting PCT Case Hot Assembly Top of Core Vapor Flow

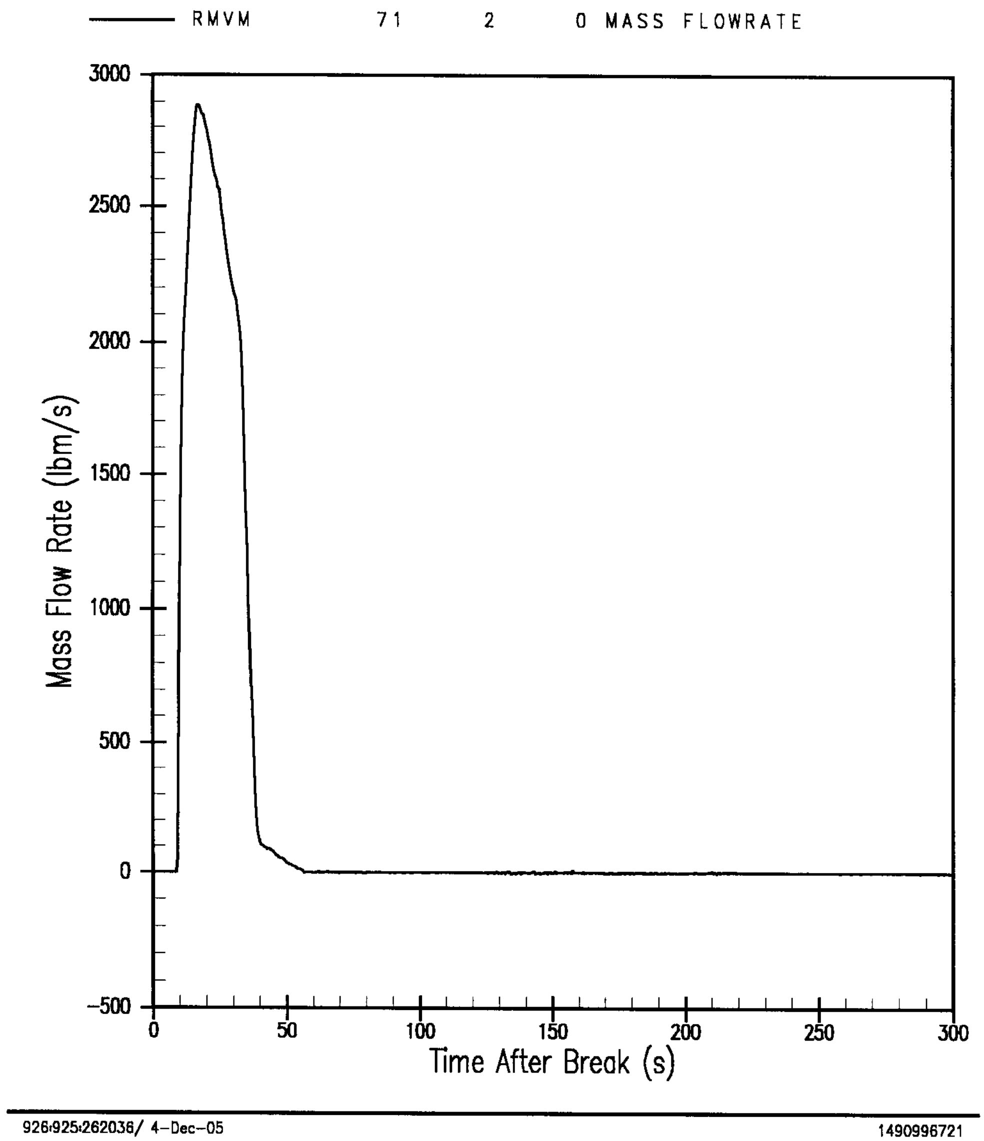


Figure 10: Surry Units 1 and 2 Limiting PCT Case Loop 2 Accumulator Flow

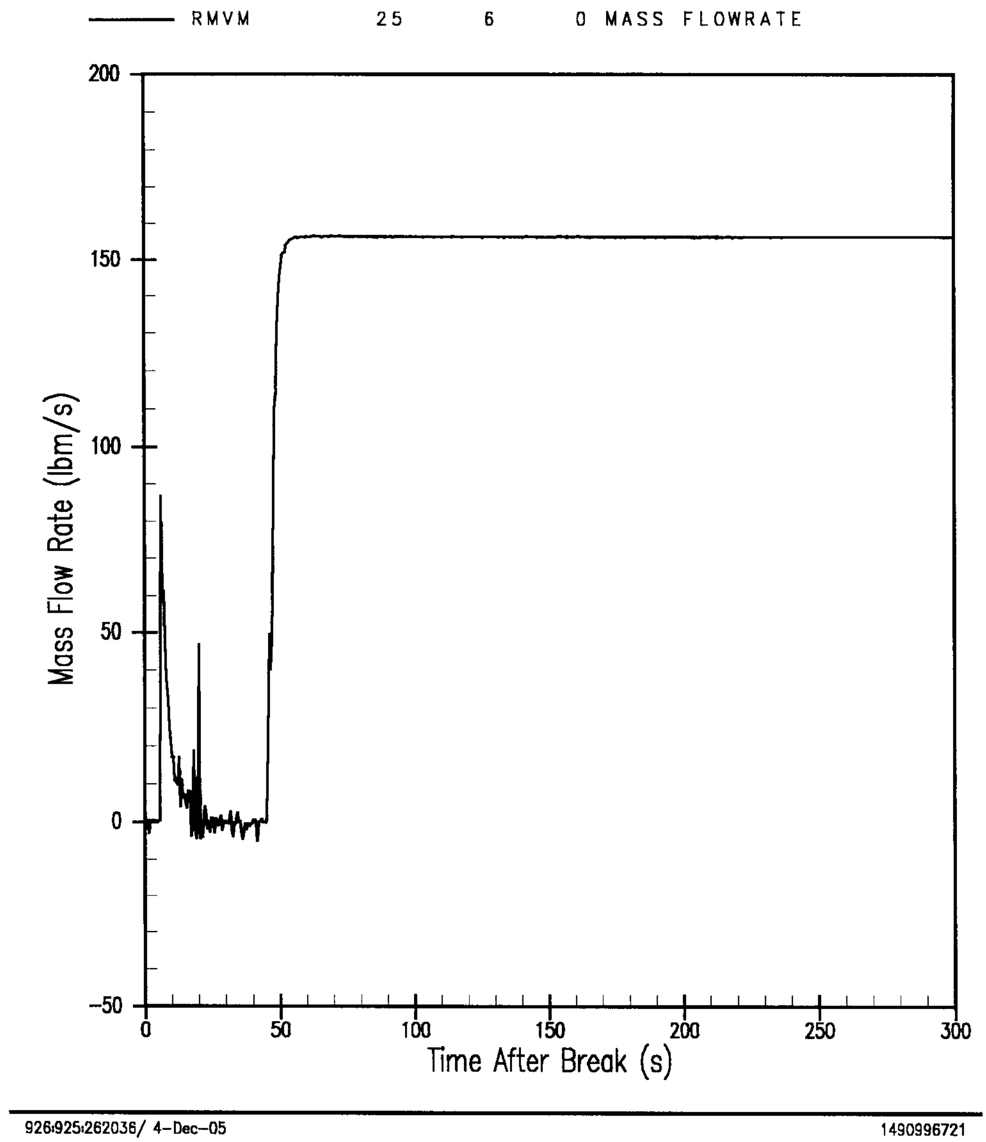


Figure 11: Surry Units 1 and 2 Limiting PCT Case Loop 2 Safety Injection Flow

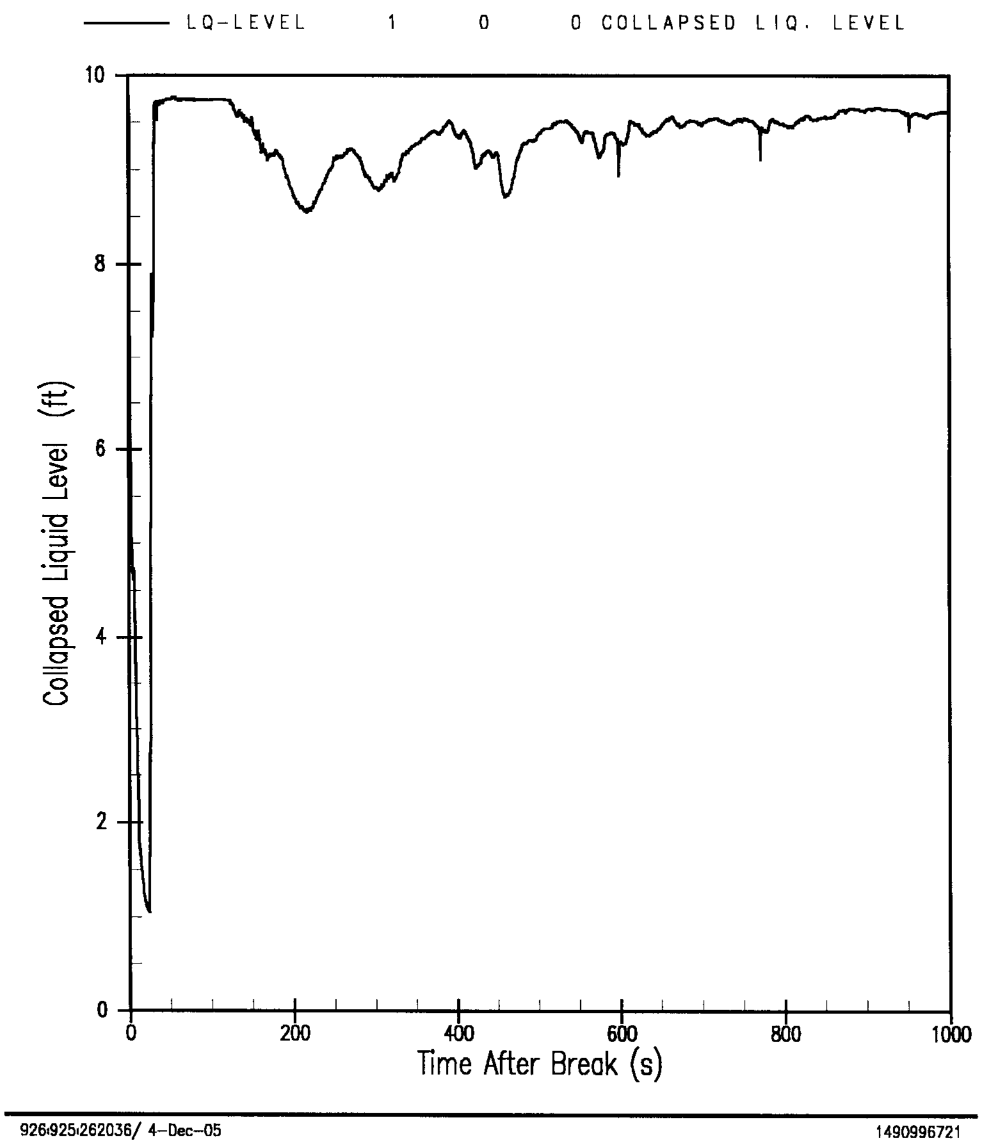


Figure 12: Surry Units 1 and 2 Limiting PCT Case
Lower Plenum Collapsed Liquid Level

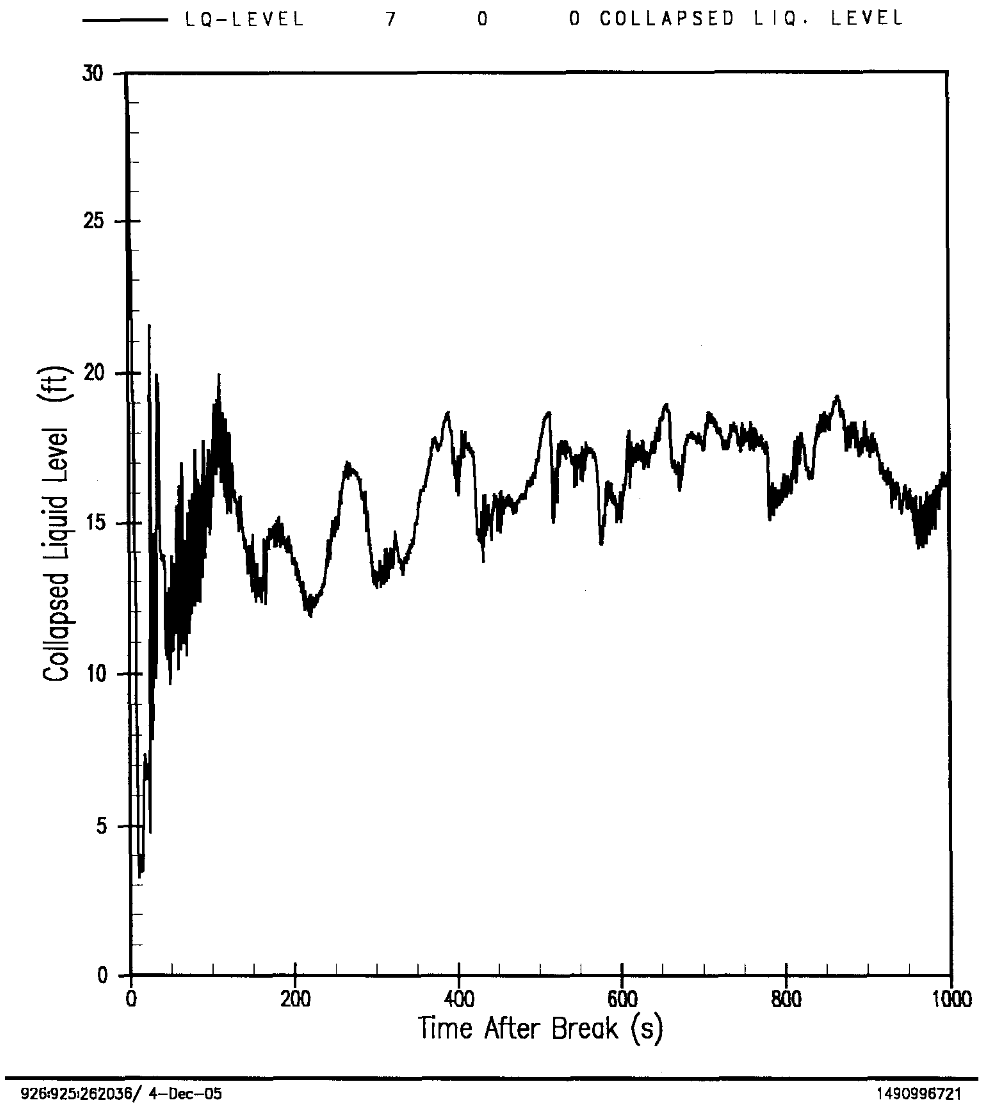


Figure 13: Surry Units 1 and 2 Limiting PCT Case Loop 2 Downcomer Collapsed Liquid Level

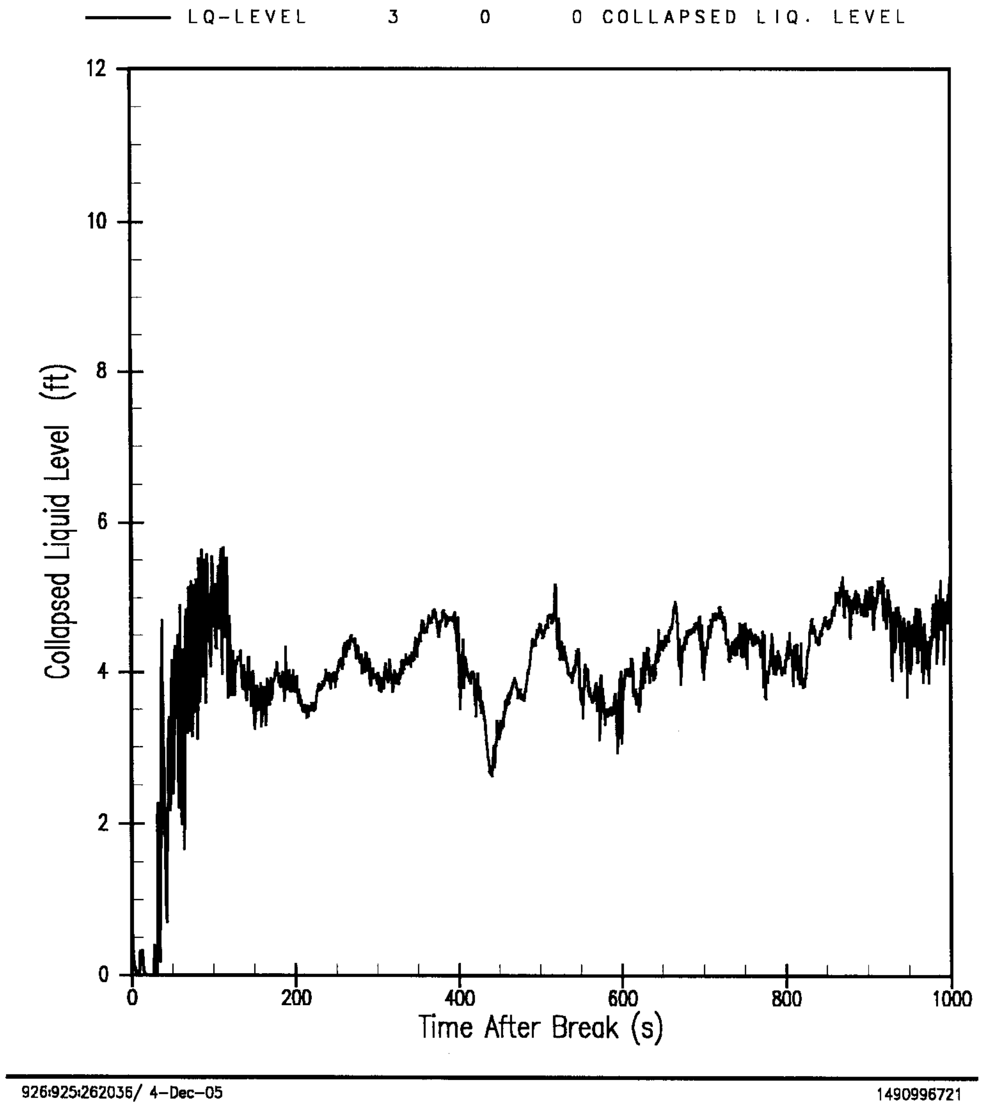


Figure 14: Surry Units 1 and 2 Limiting PCT Case Core Average Channel Collapsed Liquid Level

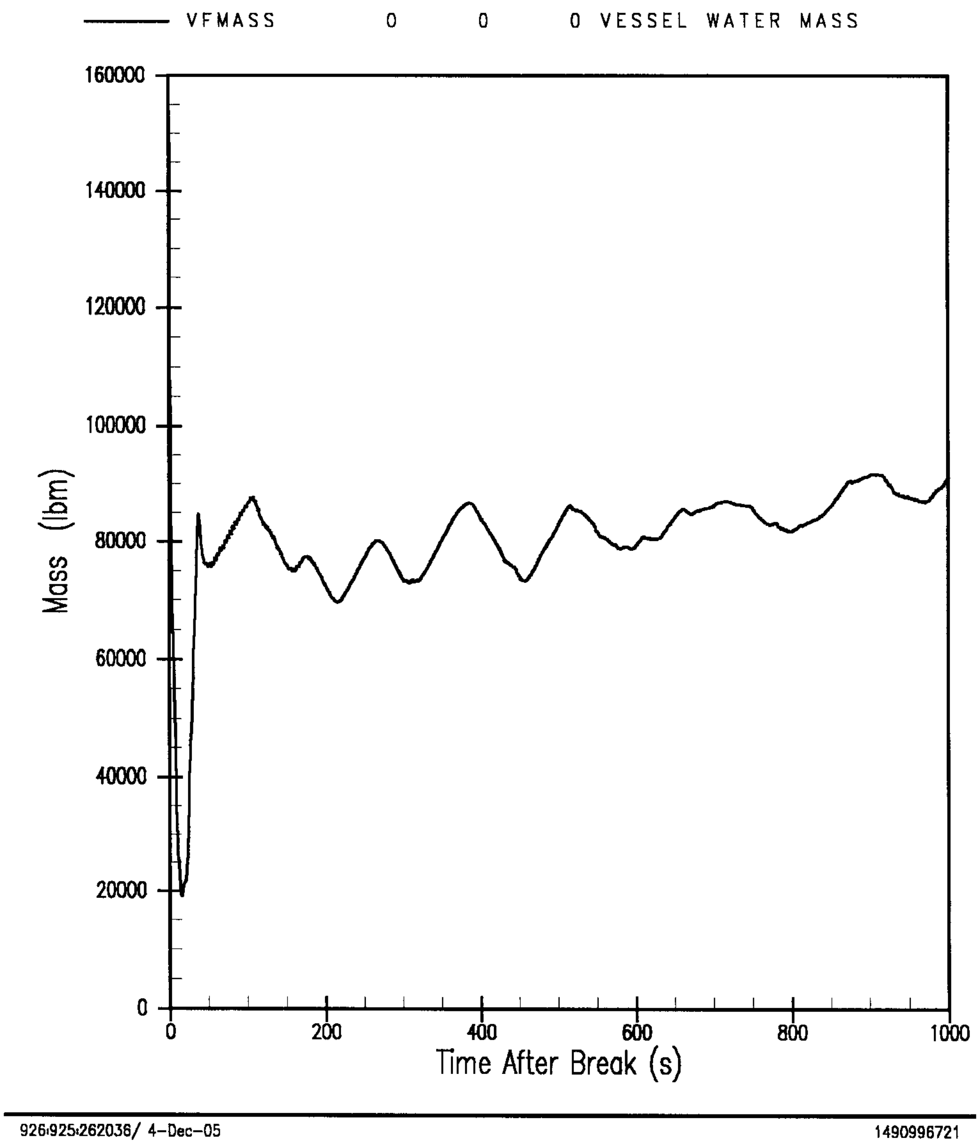


Figure 15: Surry Units 1 and 2 Limiting PCT Case Vessel Water Mass

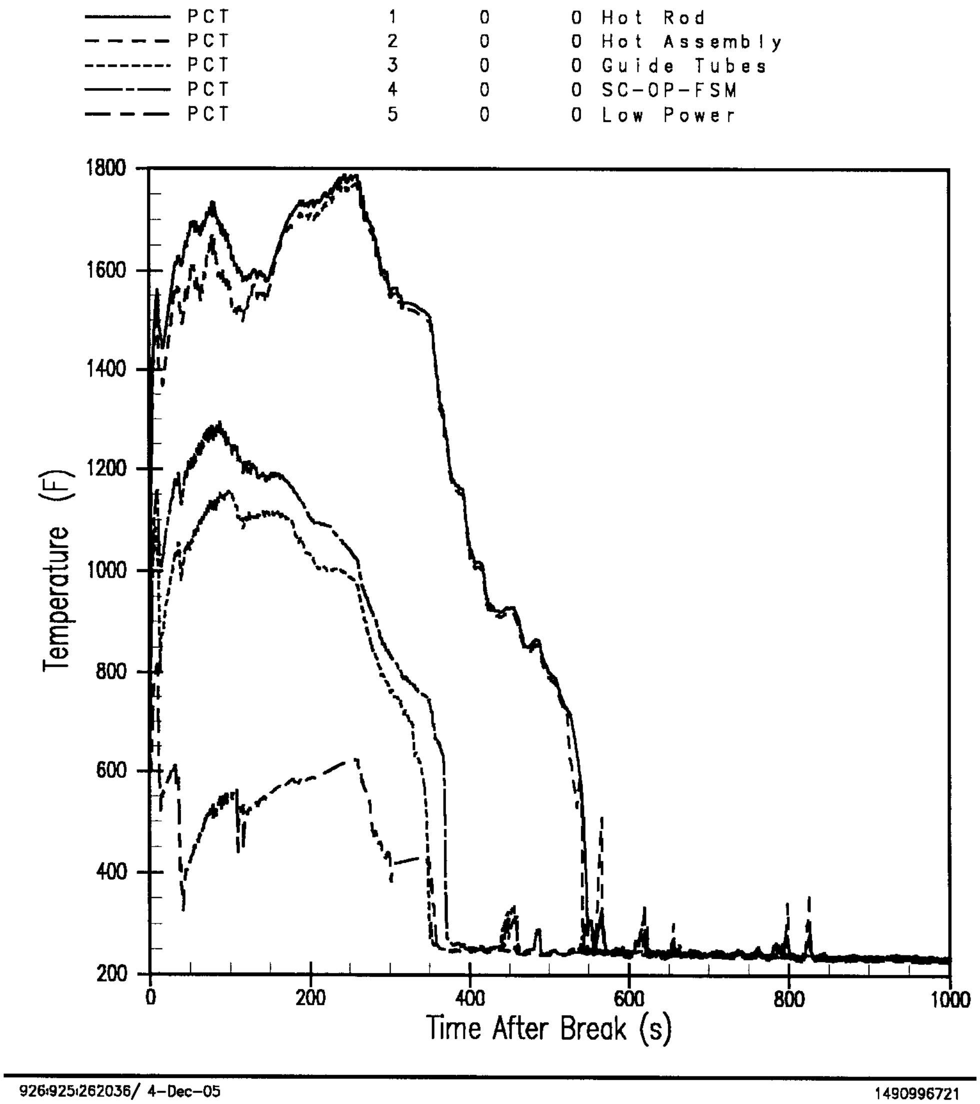


Figure 16: Surry Units 1 and 2 Peak Cladding Temperature for All 5 Rods for the Limiting PCT Case

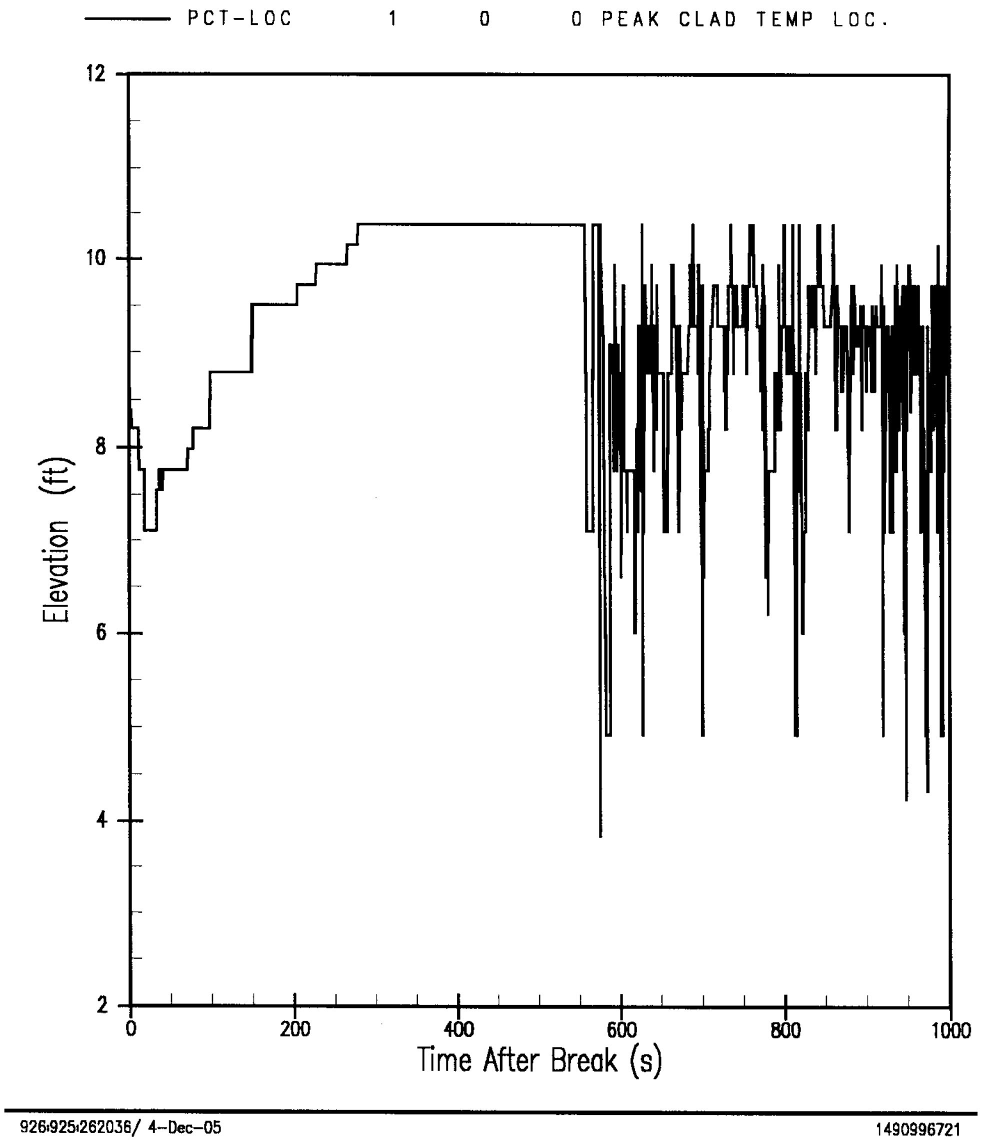


Figure 17: Surry Units 1 and 2 PCT Location for Limiting PCT Case

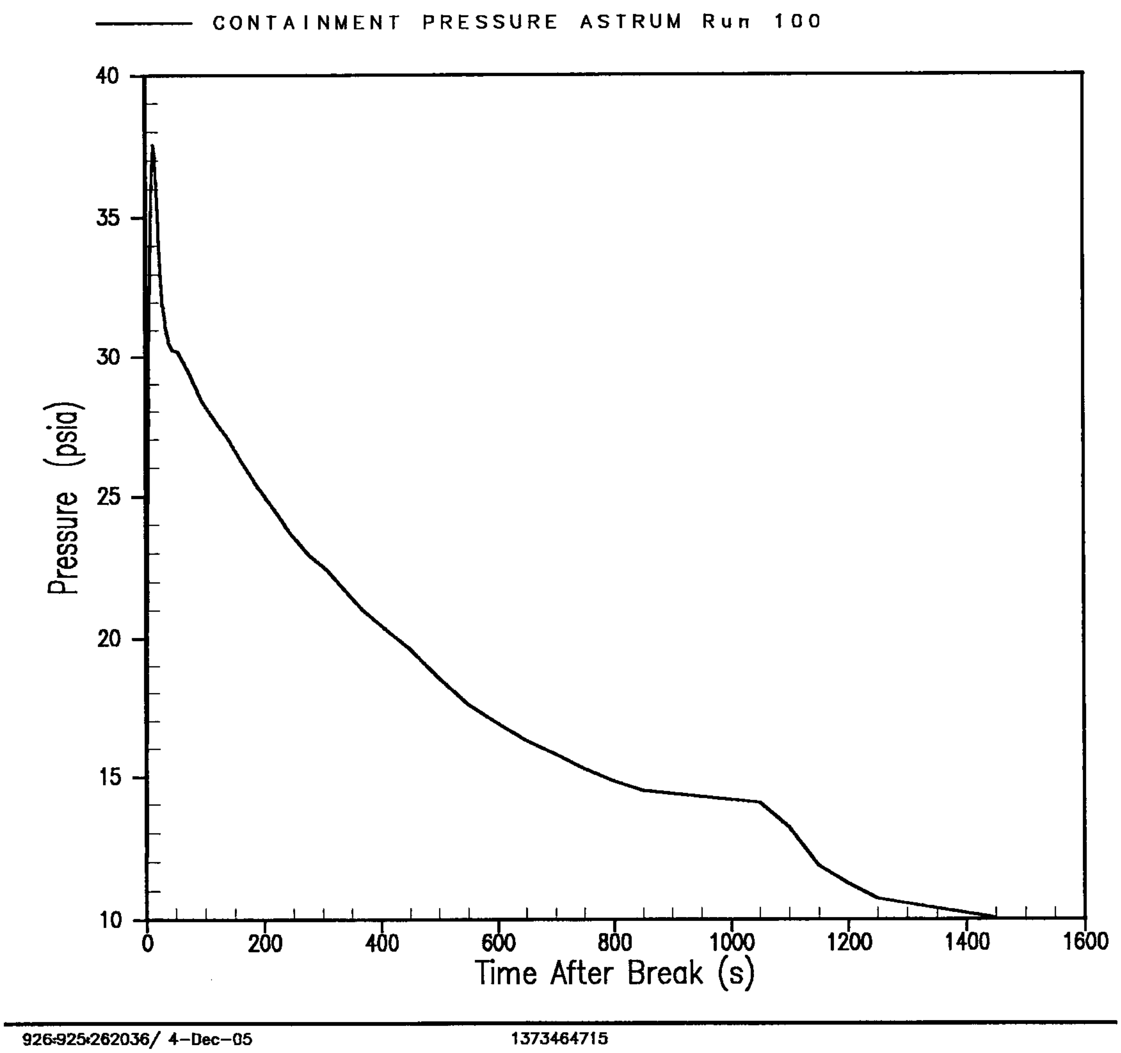
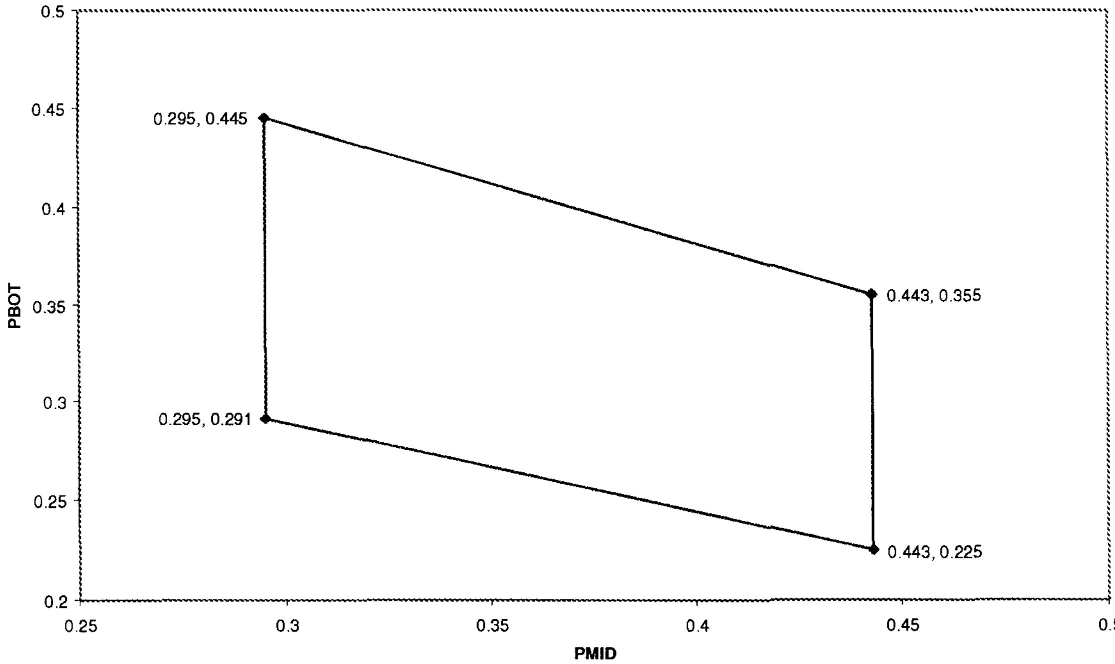


Figure 18: Surry Units 1 and 2 WCOBRA/TRAC
Containment Pressure for Limiting PCT Case



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

Figure 19: Surry Units 1 and 2 BE-LBLOCA Analysis Axial Power Shape Operating Space Envelope

5.0 NO SIGNIFICANT HAZARDS CONSIDERATION

Virginia Electric and Power Company (Dominion) proposes to implement the Westinghouse Best-Estimate Large Break Loss of Coolant Accident (BE-LBLOCA) analysis methodology using the Automated Statistical Treatment of Uncertainty Method (ASTRUM) to perform analyses of the large break loss of coolant accident (LBLOCA) at Surry Units 1 and 2. The Westinghouse BE-LBLOCA analysis methodology using ASTRUM has been approved by the USNRC. An analysis of the LBLOCA for Surry Units 1 and 2 has been performed with the Westinghouse BE-LBLOCA analysis methodology using ASTRUM. The analysis was performed in compliance with the NRC conditions and limitations identified in WCAP-16009-P-A. Based on the analysis results, it is concluded that the Surry Units 1 and 2 continue to maintain a margin of safety to the limits prescribed by 10 CFR 50.46. Technical Specification 6.2.C is being revised to include a reference to ASTRUM. It is concluded that the proposed change does not involve a significant hazards consideration as defined in 10 CFR 50.92. The basis for this determination is delineated below:

1. The probability of occurrence or the consequences of an accident previously evaluated are not significantly increased.

No physical plant changes are being made as a result of using the Westinghouse Best Estimate Large Break LOCA (BE-LBLOCA) analysis methodology. The proposed TS change simply involves updating the references in TS 6.2.C, Core Operating Limits Report (COLR), to reference the Westinghouse BE-LBLOCA analysis methodology. The consequences of a LOCA are not being increased, since the analysis has shown that the Emergency Core Cooling System (ECCS) is designed such that its calculated cooling performance conforms to the criteria contained in 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors." No other accident consequence is potentially affected by this change.

All systems will continue to be operated in accordance with current design requirements under the new analysis, therefore no new components or system interactions have been identified that could lead to an increase in the probability of any accident previously evaluated in the Updated Final Safety Analysis Report (UFSAR). No changes were required to the Reactor Protection System (RPS) or Engineering Safety Features (ESF) setpoints because of the new analysis methodology.

An analysis of the LBLOCA accident for Surry Units 1 and 2 has been performed with the Westinghouse BE-LBLOCA analysis methodology using ASTRUM. The analysis was performed in compliance with all the NRC conditions and limitations as identified in WCAP-16009-P-A. Based on the analysis results, it is concluded that the Surry Units 1 and 2 continue to maintain a margin of safety to the limits prescribed by 10 CFR 50.46.

There are no changes to assumptions of the radiological dose calculations. Hence, there is no increase in the predicted radiological consequences of accidents postulated in the UFSAR.

Therefore, neither the probability of occurrence nor the consequences of an accident previously evaluated is significantly increased.

2. The possibility for a new or different type of accident from any accident previously evaluated is not created.

The use of the Westinghouse BE-LBLOCA analysis methodology with ASTRUM does not impact any of the applicable design criteria and all pertinent licensing basis criteria will continue to be met. Demonstrated adherence to the criteria in 10 CFR 50.46 precludes new challenges to components and systems that could introduce a new type of accident. Safety analysis evaluations have demonstrated that the use of Westinghouse BE-LBLOCA analysis methodology with ASTRUM is acceptable. All design and performance criteria will continue to be met and no new single failure mechanisms will be created. The use of the Westinghouse BE-LBLOCA analysis methodology with ASTRUM does not involve any alteration to plant equipment or procedures that would introduce any new or unique operational modes or accident precursors. Furthermore, no changes have been made to any RPS or ESF actuation setpoints. Based on this review, it is concluded that no new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of the proposed changes.

Therefore, the possibility for a new or different kind of accident from any accident previously evaluated is not created.

3. The margin of safety is not significantly reduced.

It has been shown that the analytical technique used in the Westinghouse BE-LBLOCA analysis methodology using ASTRUM realistically describes the expected behavior of the reactor system during a postulated LOCA. Uncertainties have been accounted for as required by 10 CFR 50.46. A sufficient number of LOCAs with different break sizes, different locations, and other variations in properties have been considered to provide assurance that the most severe postulated LOCAs have been evaluated. The analysis has demonstrated that all acceptance criteria contained in 10 CFR 50.46 continue to be satisfied.

Therefore, it is concluded that this change does not involve a significant reduction in the margin of safety.

Based on the above information, Dominion concludes that the proposed change will not involve a significant increase in the probability or consequences of an accident previously evaluated, create the possibility of a new or different kind of accident from any accident previously evaluated, or involve a significant reduction in a margin of safety. It is concluded that the proposed change meets the requirements of 10 CFR 50.92(c) and does not involve a significant hazards consideration.

6.0 ENVIRONMENTAL ASSESSMENT

The proposed TS change to allow the use of the Westinghouse BE-LBLOCA analysis methodology with ASTRUM to perform analyses of the LBLOCA at Surry Units 1 and 2 meets the eligibility criteria for categorical exclusion from an environmental assessment set forth in 10 CFR 51.22(c)(9), as discussed below:

(i) The license condition and associated exemptions from the Code of Federal Regulations involve No Significant Hazards Consideration.

As discussed in the attached evaluation of the No Significant Hazards Consideration, the use of the Westinghouse BE-LBLOCA analysis methodology with ASTRUM to perform analyses of the LBLOCA at Surry Units 1 and 2 does not involve a significant increase in the probability or consequences of an accident previously evaluated. The possibility of a new or different kind of accident from any accident previously evaluated is also not created, and the proposed change does not involve a significant reduction in a margin of safety. Therefore, the proposed change meets the requirements of 10 CFR 50.92(c) and does not involve a significant hazards consideration.

(ii) There is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite.

The use of the Westinghouse BE-LBLOCA analysis methodology with ASTRUM to perform analyses of the LBLOCA at Surry Units 1 and 2 does not affect the way in which the fuel is handled, operated, and stored. An analysis of the LBLOCA for Surry Units 1 and 2 has been performed with the Westinghouse BE-LBLOCA analysis methodology using ASTRUM. Based on the analysis results, it is concluded that Surry Units 1 and 2 continue to maintain a margin of safety to the limits prescribed by 10 CFR 50.46. The existing radiological consequences analyses for Surry Units 1 and 2 remain applicable. Therefore, the use of the Westinghouse BE-LBLOCA analysis methodology with ASTRUM to perform analyses of the LBLOCA at Surry Units 1 and 2 will not significantly change the types, or significantly increase the amounts, of effluents that may be released offsite.

(iii) There is no significant increase in individual or cumulative occupational radiation exposure.

The use of the Westinghouse BE-LBLOCA analysis methodology with ASTRUM to perform analyses of the LBLOCA at Surry Units 1 and 2 does not involve a significant increase in individual or cumulative occupational radiation exposure since the change only involves application of an analytical method and does not involve any change in the plant equipment or operation, nor does it significantly increase overall operations and maintenance requirements, nor is any different type of equipment required to be installed.

Dominion has reviewed the proposed change pursuant to 10 CFR 50.92 and determined that it does not involve a significant hazards consideration. In addition, there is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite and there is no significant increase in individual or cumulative occupational radiation exposure. Consequently, the proposed TS change qualifies for a categorical exclusion from an environmental assessment as set forth in 10 CFR 51.22(c)(9). Therefore, no environmental impact statement or environmental assessment is needed in connection with the approval of the proposed technical specification changes.

7.0 CONCLUSIONS

The use of the Westinghouse BE-LBLOCA analysis methodology with ASTRUM has been demonstrated to provide acceptable results for the analysis of the LBLOCA at Surry Power Station Units 1 and 2. The Westinghouse BE-LBLOCA analysis methodology using ASTRUM will be added to the list of methodologies approved for reference in the COLR in TS 6.2.C upon USNRC approval. The references to the Westinghouse BASH-EM will be simultaneously removed from the list of methodologies approved for reference in the COLR in TS 6.2.C upon USNRC approval.

8.0 REFERENCES

1. WCAP-10266-P-A, Revision 2, "The 1981 Version of Westinghouse ECCS Evaluation Model Using the BASH Code," March 1987.

2. 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.

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

4. WCAP-16547, Revision 0, "Best-Estimate Analysis of the Large-Break Loss-of-Coolant Accident for Surry Units 1 and 2 Nuclear Plant Using the ASTRUM Methodology," July 2006.

5. Letter from L. N. Hartz (Dominion) to US NRC Document Control Desk "Virginia Electric and Power Company, Surry Power Station Units 1 and 2, Proposed Technical Specification Change and Supporting Safety Analyses Revisions to Address Generic Safety Issue 191," Serial No. 06-014, dated January 31, 2006.

6. Letter from L. N. Hartz (Dominion) to US NRC Document Control Desk "Virginia Electric and Power Company, Surry Power Station, 30-Day Report of Emergency Core Cooling System (ECCS) Model Changes Pursuant to the Requirements of 10 CFR 50.46," Serial No. 05-828, dated January 3, 2006.

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

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

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

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

Attachment 2

Marked-Up Technical Specifications Page

**Virginia Electric and Power Company
(Dominion)
Surry Power Station Units 1 and 2**

The analytical methods used to determine the core operating limits identified above shall be those previously reviewed and approved by the NRC, and identified below. The COLR will contain the complete identification for each of the TS referenced topical reports used to prepare the COLR (i.e., report number, title, revision, date, and any supplements). The core operating limits shall be determined so that applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met. The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided for information for each reload cycle to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

<u>REFERENCES</u>

1. VEP-FRD-42-A, "Reload Nuclear Design Methodology"

Insert 1 →

~~2a. WCAP-9220-P-A, "Westinghouse ECCS Evaluation Model - 1981 Version," (W Proprietary)~~

~~2b. WCAP-9561-P-A, "BART A-1: A Computer Code for the Best Estimate Analysis of Reflood Transients-Special Report: Thimble Modeling in W ECCS Evaluation Model," (W Proprietary)~~

~~2c. WCAP-10266-P-A, "The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code," (W Proprietary)~~

2b. ~~2d.~~ WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," (W Proprietary)

2c. ~~2e.~~ WCAP-10079-P-A, "NOTRUMP, A Nodal Transient Small Break and General Network Code," (W Proprietary)

2d. ~~2f.~~ WCAP-12610-P-A, "VANTAGE+ Fuel Assembly Report," (Westinghouse Proprietary)

3a. VEP-NE-2-A, "Statistical DNBR Evaluation Methodology"

3b. VEP-NE-3-A, "Qualification of the WRB-1 CHF Correlation in the Virginia Power COBRA Code"

Insert 1 – insert as indicated on page TS 6.2-2:

2a. WCAP-16009-P-A, "Realistic Large Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)," (Westinghouse Proprietary).

Attachment 3

Proposed Technical Specifications Page

Virginia Electric and Power Company
(Dominion)
Surry Power Station Units 1 and 2

The analytical methods used to determine the core operating limits identified above shall be those previously reviewed and approved by the NRC, and identified below. The COLR will contain the complete identification for each of the TS referenced topical reports used to prepare the COLR (i.e., report number, title, revision, date, and any supplements). The core operating limits shall be determined so that applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met. The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided for information for each reload cycle to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

<u>REFERENCES</u>

1. VEP-FRD-42-A, "Reload Nuclear Design Methodology"

2a. WCAP-16009-P-A, "Realistic Large Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)," (Westinghouse Proprietary).

2b. WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," (W Proprietary)

2c. WCAP-10079-P-A, "NOTRUMP, A Nodal Transient Small Break and General Network Code," (W Proprietary)

2d. WCAP-12610-P-A, "VANTAGE+ Fuel Assembly Report," (Westinghouse Proprietary)

3a. VEP-NE-2-A, "Statistical DNBR Evaluation Methodology"

3b. VEP-NE-3-A, "Qualification of the WRB-1 CHF Correlation in the Virginia Power COBRA Code"