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**Indiana Michigan Power**
Cook Nuclear Plant
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Bridgman, MI 49106
AEP.com

December 27, 2007

AEP:NRC:7565-01
10 CFR 50.90

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Stop O-P1-17
Washington, DC 20555-0001

SUBJECT: Donald C. Cook Nuclear Plant Unit 1
Docket No. 50-315
License Amendment Request Regarding Large Break Loss-of-Coolant Accident Analysis Methodology

Reference: Letter from H. N. Berkow, U. S. Nuclear Regulatory Commission, 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 (ADAMS Accession Number ML043100073).

Dear Sir or Madam:

Pursuant to 10 CFR 50.90, Indiana Michigan Power Company (I&M), the licensee for Donald C. Cook Nuclear Plant Unit 1, proposes to amend Facility Operating License DPR-58. I&M proposes to modify Technical Specification (TS) 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits," and TS 5.6.5, "Core Operating Limits Report (COLR)," in support of a new analysis of a large break loss-of-coolant accident (LBLOCA).

In addition, I&M is requesting Nuclear Regulatory Commission (NRC) approval of a new Unit 1 LBLOCA analysis using a plant-specific adaptation of topical report WCAP-16009-P-A, "Realistic Large-Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)." The NRC approved the topical report methodology in the referenced letter. I&M intends to use an adaptation that employs an increased number of circumferential noding stacks in the downcomer.

Enclosure 1 provides an affirmation statement pertaining to this letter. Enclosure 2 provides I&M's evaluation of the proposed change. Enclosure 3 provides a proprietary document supporting the plant-specific adaptation using an increased number of circumferential noding stacks in the downcomer. Enclosure 3 is proprietary to Westinghouse Electric Company LLC. Enclosure 4 provides the non-proprietary version of Enclosure 3. Attachment 1 provides the Unit 1 TS pages

A001
NRR

marked to show changes. Attachment 2 provides the TS pages with the proposed changes incorporated. Attachment 3 provides the regulatory commitment made in support of this license amendment request.

The proprietary information in Enclosure 3 is supported by an affidavit signed by Westinghouse, the owner of the proprietary information. The affidavit sets forth the basis on which the information may be withheld from public disclosure by the NRC and addresses with specificity the considerations listed in paragraph (b)(4) of Section 2.390 of the NRC's regulations. Accordingly, it is respectfully requested that the information which is proprietary to Westinghouse be withheld from public disclosure in accordance with 10 CFR 2.390 of the NRC's regulations. Attachment 4 contains the Westinghouse authorization letter, CAW-07-2344, accompanying affidavit, Proprietary Information Notice, and Copyright Notice for Enclosure 3.

I&M requests approval of the proposed amendment in accordance with the normal NRC review schedule. The proposed changes to the Unit 1 TS will be implemented within 60 days of approval.

Copies of this letter and its attachments are being transmitted to the Michigan Public Service Commission and Michigan Department of Environmental Quality, in accordance with the requirements of 10 CFR 50.91.

Should you have any questions, please contact Mr. James M. Petro, Regulatory Affairs Manager, at (269) 466-2491.

Sincerely,

Joseph N. Jensen
Site Vice President

KAS/rdw

Enclosures:

1. Affirmation
2. Indiana Michigan Power Company's Evaluation
3. WCOBRA/TRAC Validation with Revised Downcomer Noding for D.C. Cook Units 1 and 2 (proprietary)
4. WCOBRA/TRAC Validation with Revised Downcomer Noding for D.C. Cook Units 1 and 2 (nonproprietary)

Attachments:

1. Donald C. Cook Nuclear Plant Unit 1 Technical Specification Pages Marked To Show Changes
2. Donald C. Cook Nuclear Plant Unit 1 Technical Specification Pages with the Proposed Changes Incorporated
3. Regulatory Commitment
4. Application For Withholding Proprietary Information From Public Disclosure

c: J. L. Caldwell, NRC Region III
K. D. Curry, Ft. Wayne AEP, w/o enclosures/attachments
J. T. King, MPSC
MDEQ – WHMD/RPMWS
NRC Resident Inspector
P. S. Tam, NRC Washington, DC

**Enclosure 1 to AEP:NRC:7565-01**

**AFFIRMATION**

I, Joseph N. Jensen, being duly sworn, state that I am Site Vice President of Indiana Michigan Power Company (I&M), that I am authorized to sign and file this request with the Nuclear Regulatory Commission on behalf of I&M, and that the statements made and the matters set forth herein pertaining to I&M are true and correct to the best of my knowledge, information, and belief.

Indiana Michigan Power Company

Joseph N. Jensen
Site Vice President

SWORN TO AND SUBSCRIBED BEFORE ME

THIS 27th DAY OF December, 2007

Regan D Wendzel
Notary Public

My Commission Expires

REGAN D. WENDZEL
Notary Public, Berrien County, MI
My Commission Expires Jan. 21, 2009

**Enclosure 2 to AEP:NRC:7565-01**

# INDIANA MICHIGAN POWER COMPANY'S EVALUATION

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


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4.0 TECHNICAL ANALYSIS

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## 1.0 DESCRIPTION

This letter is a request by Indiana Michigan Power Company (I&M) to amend Facility Operating License DPR-58 for Donald C. Cook Nuclear Plant (CNP) Unit 1. The proposed change modifies Technical Specification (TS) 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits," to reflect a higher minimum reactor coolant system (RCS) total flow rate and TS 5.6.5, "Core Operating Limits Report (COLR)," to replace the current large break loss-of-coolant accident (LBLOCA) analysis methodology with a plant-specific adaptation of WCAP-16009-P-A, "Realistic Large-Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)," (Reference 1) as the analytical method for determining core operating limits for Unit 1.

In addition, I&M is requesting Nuclear Regulatory Commission (NRC) approval of the Unit 1 LBLOCA analysis using a plant-specific adaptation of WCAP-16009-P-A. The plant-specific adaptation deviates from the NRC-approved methodology by use of an increased number of circumferential noding stacks in the downcomer.

## 2.0 PROPOSED CHANGE

TS Limiting Condition for Operation (LCO) 3.4.1.c and Surveillance Requirements 3.4.1.3 and 3.4.1.4 would be revised to reflect a higher minimum RCS total flow rate of 354,000 gallons per minute (gpm).

TS 5.6.5 would be revised by replacing the existing LBLOCA analysis methodology in TS 5.6.5.b.4 with the following:

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

I&M also requests NRC approval of the plant-specific adaptation of WCAP-16009-P-A using twelve circumferential noding stacks in the downcomer as compared to the four stacks used for four-loop plants in the WCAP-16009-P-A, Revision 0 methodology. NRC approval is needed because the Safety Evaluation (SE) enclosed in Reference 1 found the topical report (TR) methodology described in WCAP-16009-P-A acceptable for referencing in licensing applications and required license amendment requests that deviate from the TR be subject to a plant-specific review in accordance with applicable review standards.

Changes to TS Bases are required to reflect descriptive information on the LBLOCA methodology. These changes will be made in accordance with the CNP Technical Specification Bases Control Program.

## 3.0 BACKGROUND

TS 3.4.1 provides limits on RCS pressure, temperature, and flow rate to ensure that the minimum departure from nucleate boiling ratio will be met for each of the safety analysis transients analyzed. RCS flow rate normally remains constant during an operational fuel cycle with all reactor coolant pumps running. The minimum RCS flow limit corresponds to that assumed for DNB analyses. Flow rate indications are averaged to determine a value for comparison to the limit. A higher RCS flow rate will provide more margin to DNB limits. The new LBLOCA analysis uses a higher RCS flow rate value, which requires the proposed changes to TS 3.4.1.

TS 5.6.5.a states that core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle. TS 5.6.5.a requires the core operating limits to be documented in the COLR for the items listed in TS 5.6.5.a.1 through TS 5.6.5.a.11.

TS 5.6.5.b states that the analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, and specifically lists the analytical methods that may be used.

The regulations specified in 10 CFR 50.46(a)(1) identify calculation methodology requirements for nuclear power plant loss-of-coolant accident (LOCA) methodologies. Federal Regulation 10 CFR 50.46(c) identifies the types of processes which are required to assure that LOCA analyses performed for a given plant actually represent the plant. Section 50.46(a)(3)(i) and (ii) specify criteria to be applied and actions to be taken when significant changes or errors in parts of the plant-specific LOCA methodology, defined in accordance with 10 CFR 50.46(a)(1) and (c), are found to have accumulated.

The current CNP Unit 1 LBLOCA analysis-of-record was performed in 2000 using the BASH evaluation model methodology documented in WCAP-10266-P-A, "The 1981 Version of Westinghouse ECCS Evaluation Model Using the BASH Code." By Reference 2, I&M committed to provide the NRC a new LBLOCA analysis for CNP Unit 1 due to a cumulation of changes and errors requiring a scheduled reanalysis in accordance with 10 CFR 50.46(a)(3)(ii). The best estimate LBLOCA (BELOCA) analysis using a plant-specific adaptation of the methodology documented in WCAP-16009-P-A and the associated TS changes proposed in this license amendment request are presented in fulfillment of this commitment to the NRC. The ASTRUM methodology has received NRC approval for referencing in licensing applications by the SE in WCAP-16009-P-A.

## 4.0 TECHNICAL ANALYSIS

WCAP-16009-P-A describes a realistic (or best estimate) emergency core cooling system (ECCS) evaluation model for demonstrating plant compliance with 10 CFR 50.46 for postulated plant-specific LBLOCA transients. WCAP-16009-P-A uses a statistical approach in developing the peak cladding temperature (PCT), local maximum oxidation (LMO), and core wide oxidation

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

Westinghouse analyzed the CNP Unit 1 LBLOCA using a plant-specific adaptation of the ASTRUM methodology. The analysis was performed in compliance with all of the conditions and limitations identified in the NRC SE approving ASTRUM. This request for NRC review and approval is intended to meet the following limitation placed on WCAP-16009-P-A as stated in the NRC SE:

> The methodology described in WCAP-16009-P-A, Revision 0, is a separate and unique methodology. Any other version derived from this TR, such as designated by a new revision number, amendment number, addendum number or other equivalent designation, would constitute a definition of a new methodology requiring NRC review and acceptance prior to generic application and prior to any specific plant licensing application of a new methodology derived from ASTRUM.

Preliminary results with the as-approved ASTRUM method applied to CNP Unit 1 yielded unexpected results that have been attributed to overly conservative aspects of the model. A combination of CNP design features contribute to the as-approved WCOBRA/TRAC computer code predicting a significant loss of liquid inventory from the downcomer during reflood, due to extensive boiling by heat release from the vessel wall, core barrel, and thermal shield. Consequently, a plant-specific adaptation of ASTRUM has been used for the CNP Unit 1 analysis to better model the downcomer region by increasing the number of circumferential noding stacks from four to twelve. This finer nodalization has been assessed against experimental data, as described in Enclosure 3. Upon approval and implementation, the CNP Unit 1 LBLOCA plant-specific adaptation of ASTRUM will be annotated in the COLR as "Plant-specific adaptation of WCAP-16009-P-A, Revision 1, as approved by Safety Evaluation dated [DATE]."

Table 1 lists the major plant parameter assumptions in the BELOCA analysis for CNP Unit 1 and Table 2 summarizes the results of the CNP Unit 1 ASTRUM analysis. Table 3 contains a sequence of events for the limiting PCT transient.

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

From the 124 calculations performed as part of the ASTRUM analysis, the same case proved to be the limiting PCT, LMO, and CWO transient for CNP Unit 1. Figure 2 shows the predicted

HOTSPOT cladding temperature transient at the PCT location for the limiting case. The HOTSPOT PCT plot includes local uncertainties applied to the Hot Rod. Figure 3 presents the WCOBRA/TRAC PCT transient predicted for the limiting case. This figure does not account for local uncertainties.

Figures 4 through 16 illustrate the key major response parameters for the limiting PCT transient. The reference point for the lower plenum liquid level presented in Figure 11 is the bottom of the vessel (10.1 feet below the bottom of the active fuel). The reference point for the downcomer liquid level presented in Figure 12 is the bottom of the vessel. The reference point for the core collapsed liquid levels presented in Figure 13 is the bottom of the active fuel.

The containment backpressure utilized for the LBLOCA analysis compared to the calculated containment backpressure is provided in Figure 17. The worst single failure for the LBLOCA analysis is the loss of one train of ECCS injection (consistent with the ASTRUM topical); however, all containment systems which would reduce containment pressure are modeled for the LBLOCA containment backpressure calculation.

Figure 18 provides the CNP Unit 1 BELOCA axial power distribution.

I&M and its analysis vendor (Westinghouse) have interface processes which identify plant configuration changes potentially impacting safety analyses. These interface processes, along with vendor internal processes for assessing evaluation model changes and errors, are used to identify the need for LOCA analyses impact assessments.

Implementation of the approved LBLOCA methodology will necessitate changes to TS 5.6.5.b to replace the reference to the current LBLOCA analysis-of-record with the plant-specific adaptation of WCAP-16009-P-A, as an approved LBLOCA analysis methodology for CNP Unit 1. Using the new methodology does not result in any new operating limits requiring a change to the COLR, other than a change to reflect the revised minimum RCS total flow rate.

The core power level is one of the key input parameters included in Table 1. The Unit 1 BELOCA analysis was performed modeling a bounding core power of 3315 megawatt-thermal ($MW_t$) for all cases. The CNP Unit 1 licensed maximum core power level is 3304 $MW_t$. Safety Injection delay time is another key parameter included in Table 1. A delay time of 54 seconds is assumed without offsite power available to account for an emergency diesel generator (DG) start time of 30 seconds. Current TS Surveillance Requirements verify the DG starts within 10 seconds. I&M is not requesting any core power level changes or changes to the DG start time as part of this license amendment request.

The WCOBRA/TRAC model used in the analysis assumes a residual heat removal (RHR) cross-tie valve modification has been installed. This modification is to be completed in accordance with 10 CFR 50.59 and is planned for installation during the Unit 1 Cycle 22 outage (Spring 2008). The modification would allow CNP Unit 1 to operate with RHR cross-tie valves

open, providing four-loop injection during a postulated LBLOCA. Upon installation CNP Unit 1 will operate with RHR cross-tie valves open to meet analysis assumptions. The previous design required RHR cross-tie valves to be closed to address concerns communicated by NRC Bulletin 88-04 regarding pump-to-pump interaction during miniflow operation.

Based on the results presented in Table 2, I&M concludes that CNP Unit 1 continues to maintain a margin of safety to the limits prescribed by 10 CFR 50.46.

**Table 1**
**Major Plant Parameter Assumptions Used in the CNP Unit 1 BELOCA Analysis**

| Parameter | Value |
|---|---|
| **Plant Physical Description** | |
| • Steam Generator Tube Plugging | ≤10 % |
| **Plant Initial Operating Conditions** | |
| • Reactor Power | 3315 MWt |
| • Peaking Factors | $F_Q \leq 2.15$ |
| | $F_{\Delta H} \leq 1.55$ |
| **Fluid Conditions** | |
| • Average Coolant Temperature (Tavg) | 553.7 – 4.1 °F ≤Tavg ≤575.4 + 5.1 °F |
| • Pressurizer Pressure ($P_{RCS}$) | 2100 – 67 psia $\leq P_{RCS} \leq$ 2100 + 67 psia |
| | 2250 – 67 psia $\leq P_{RCS} \leq$ 2250 + 67 psia |
| • Reactor Coolant Flow | 88,500 gpm per loop |
| • Accumulator Temperature ($T_{ACC}$) | 60 °F $\leq T_{ACC} \leq$ 120 °F |
| • Accumulator Pressure ($P_{ACC}$) | 599.7 psia $\leq P_{ACC} \leq$ 672.7 psia |
| • Accumulator Water Volume ($V_{ACC}$) | 921 $ft^3 \leq V_{ACC} \leq$ 971 $ft^3$ |
| • Minimum Accumulator Boron | 2228 parts per million |
| **Accident Boundary Conditions** | |
| • Safety Injection Flow | Minimum |
| • Safety Injection Temperature ($T_{SI}$) | 70 °F $\leq T_{SI} \leq$ 105 °F |
| • Safety Injection Delay Time | ≤27 seconds (with offsite power)<br>≤54 seconds (without offsite power) |
| • Containment Pressure | Bounded (minimum); See Figure 17 |
| • Single Failure Assumptions | Loss of one ECCS train |

**Table 2**
**Best Estimate LBLOCA Results**

| ASTRUM Result | Value | Criteria |
|---|---|---|
| 95/95 PCT (°F) | 2106 | < 2,200 |
| 95/95 LMO (%) | 10.0 | < 17 |
| 95/95 CWO (%) | 0.35 | < 1 |

**Table 3**
**CNP Unit 1 Best Estimate Sequence of Events for the Limiting PCT Case**

| Event | Time (seconds) |
|---|---|
| Break Initiation | 0.0 |
| Safety Injection Signal | 4.6 |
| Accumulator Injection Begins | 16.0 |
| End of Blowdown | 25.0 |
| Bottom of Core Recovery | 42.5 |
| Accumulator Empty | 49.3 |
| Safety Injection Begins | 58.6 |
| PCT Occurs | 247.0 |
| End of Calculation * | 500.0 |

* Quench is calculated to occur by the end of the calculation time.

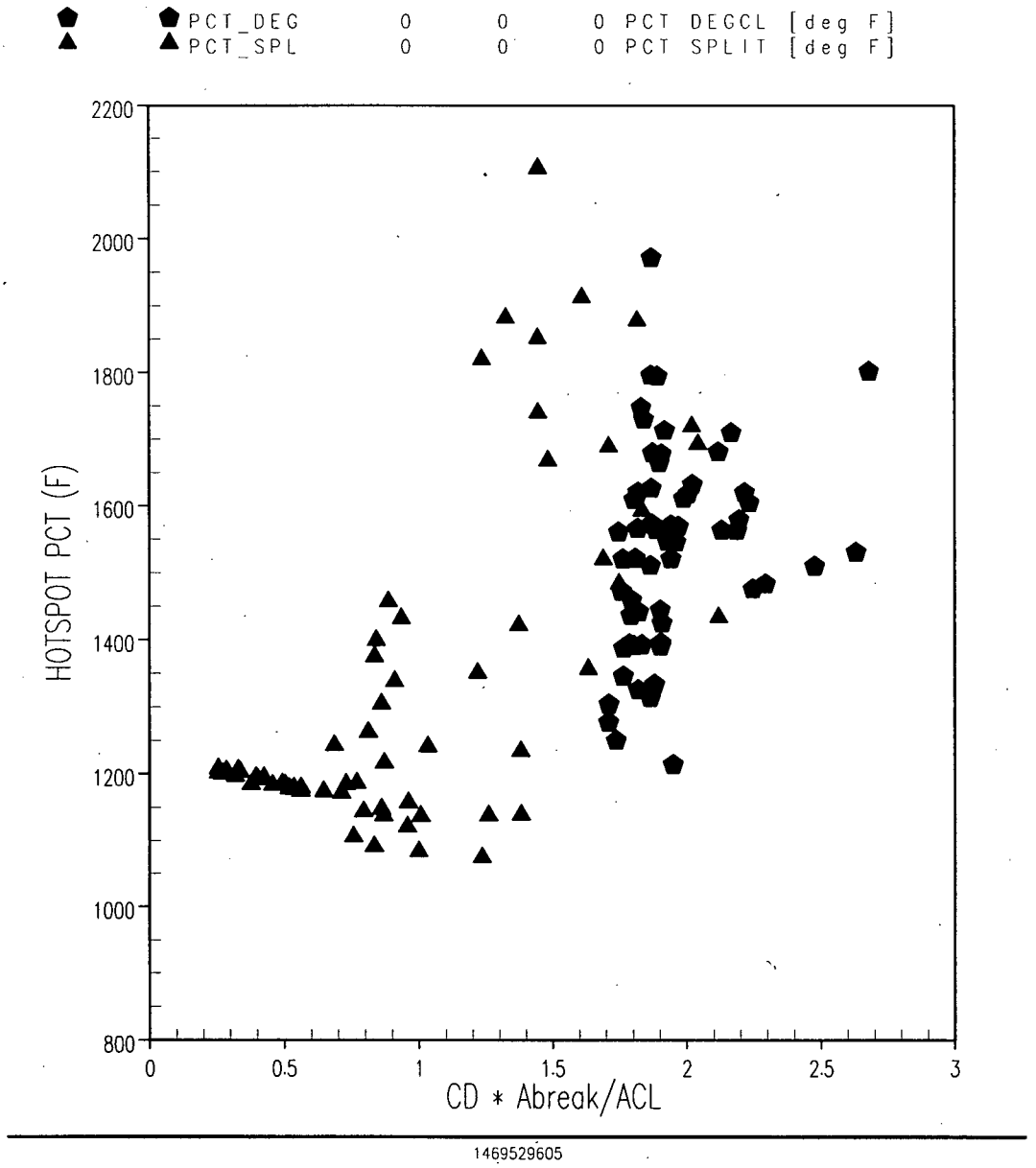


**Figure 1 – HOTSPOT PCT versus Effective Break Area Scatter Plot (CD = Discharge Coefficient, Abreak = Break Area, ACL = Cold Leg Area)**

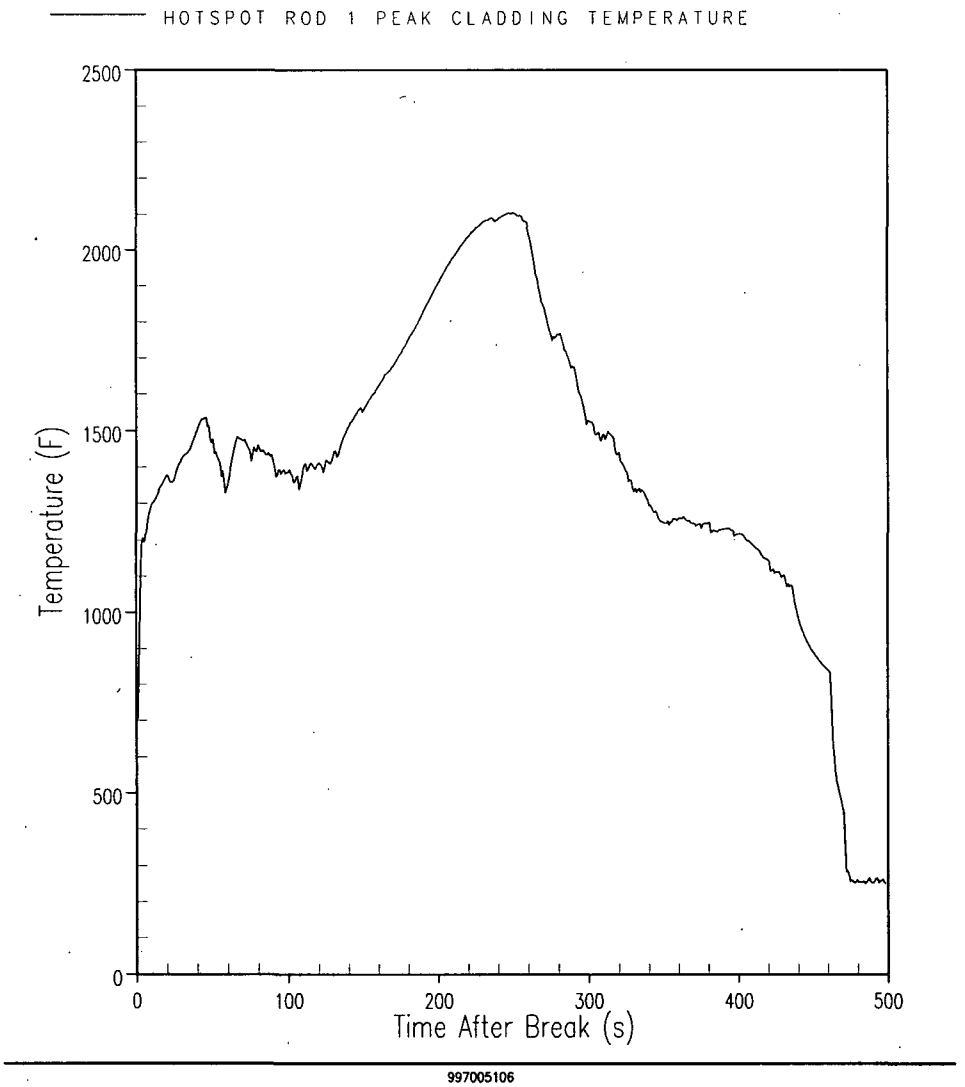


**Figure 2 – HOTSPOT Cladding Temperature Transient for the Limiting Case**

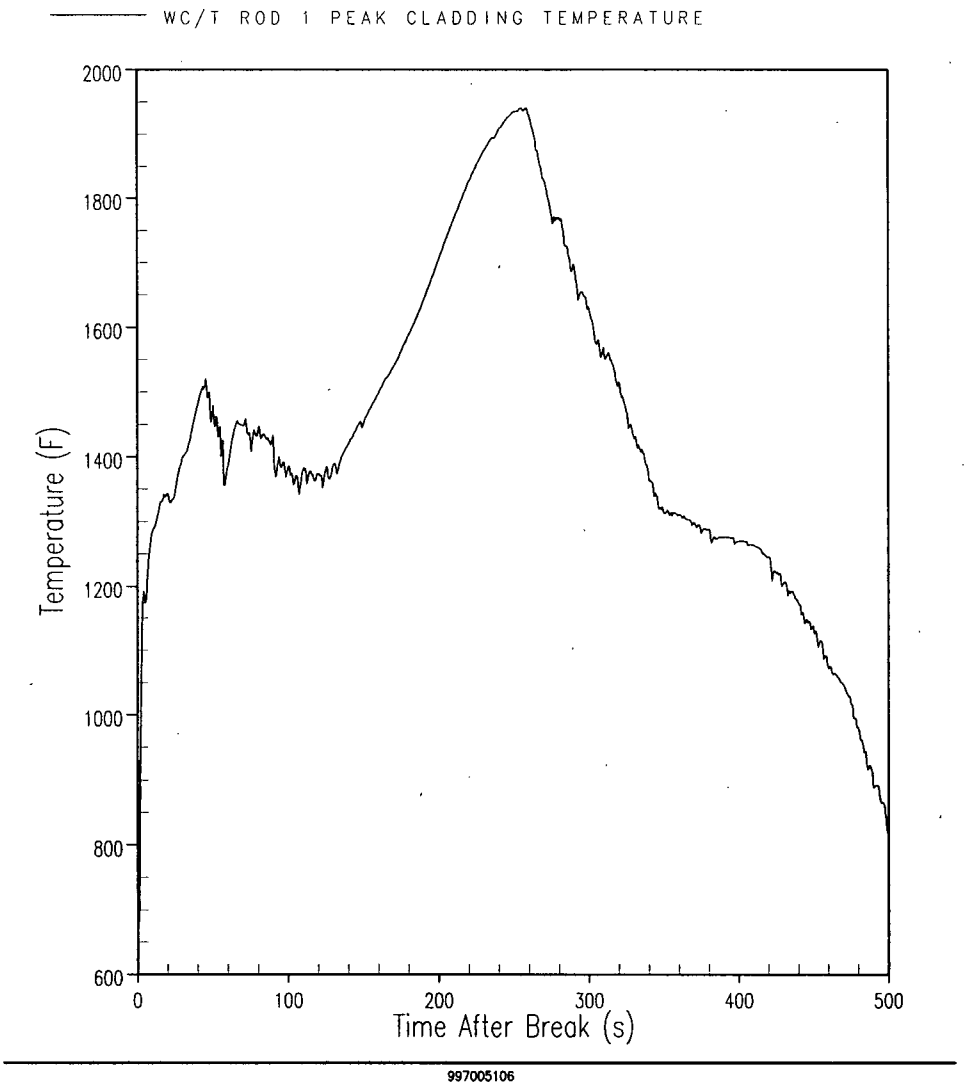


**Figure 3 – WCOBRA/TRAC Cladding Temperature Transient for the Limiting Case**

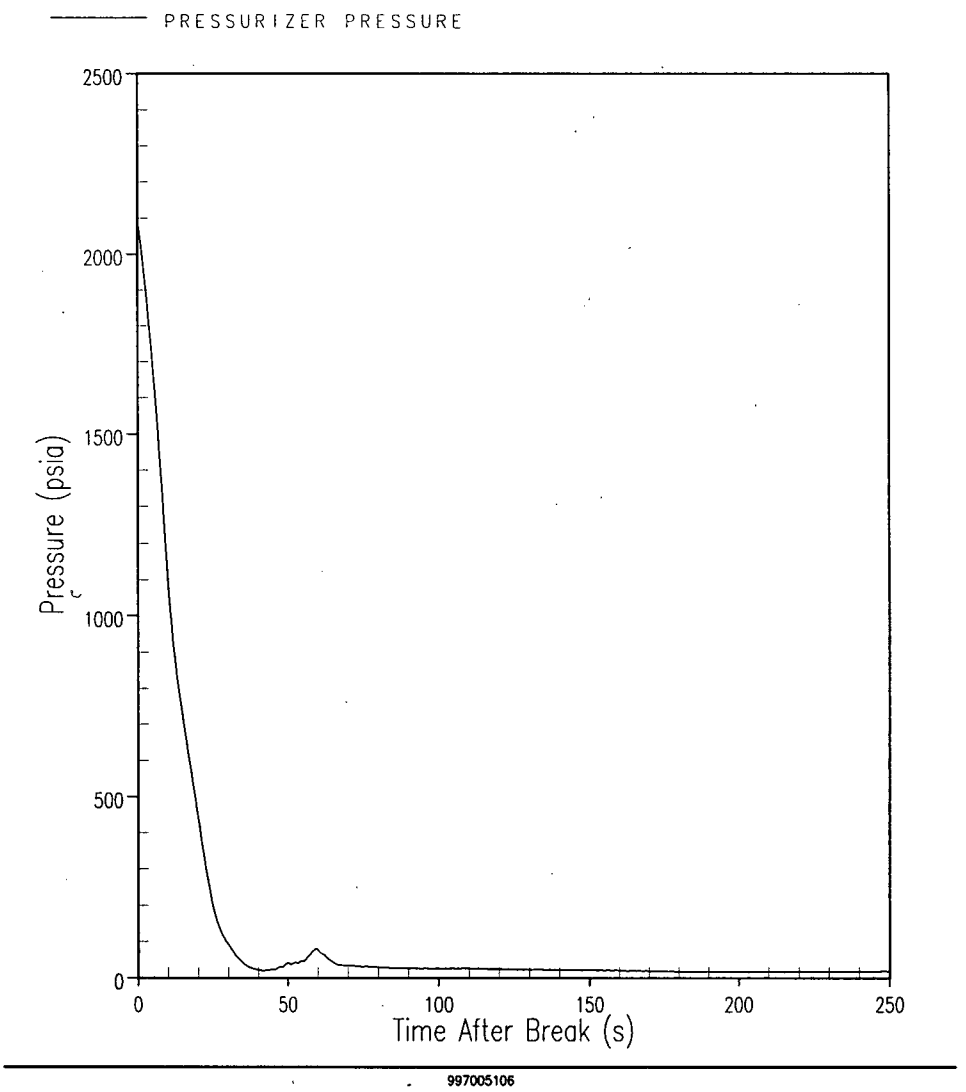


**Figure 4 – Pressurizer Pressure for the Limiting Case**

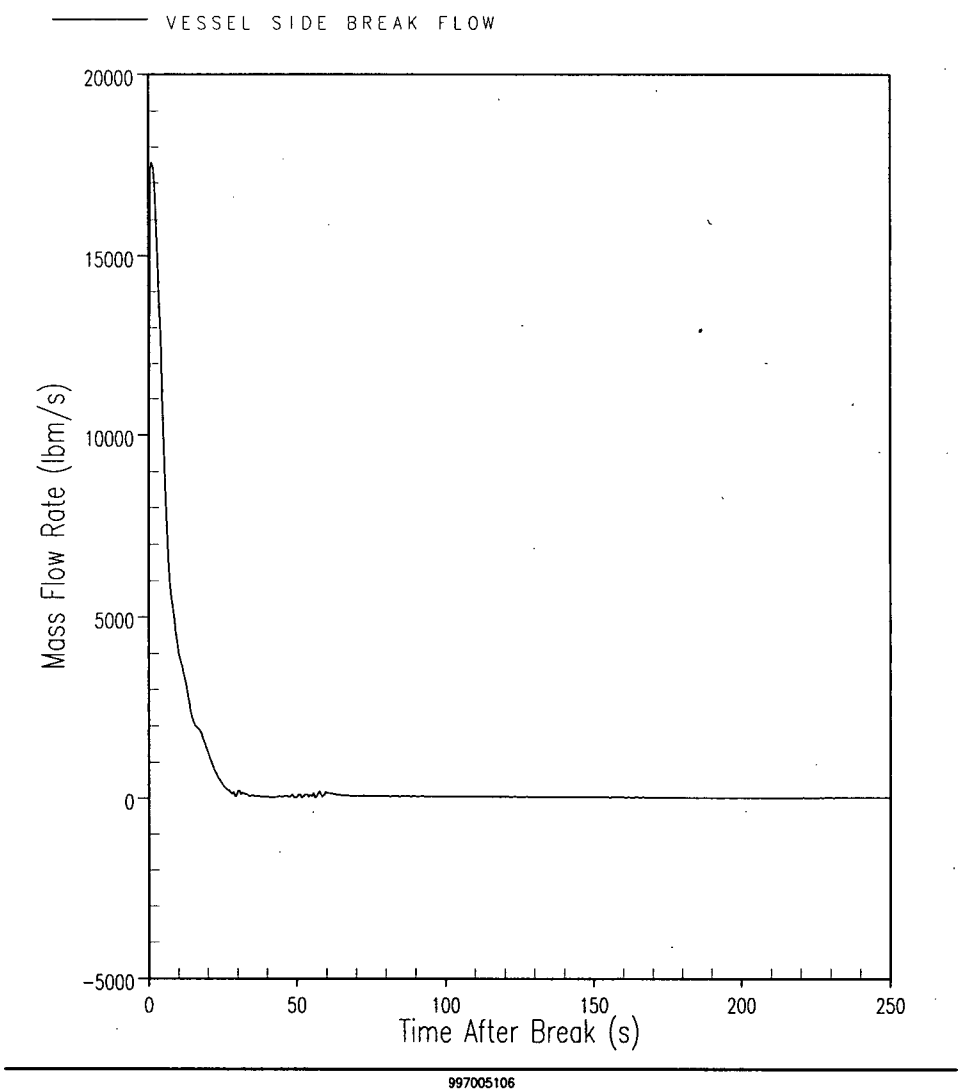


**Figure 5 – Vessel Side Break Flow for the Limiting Case**

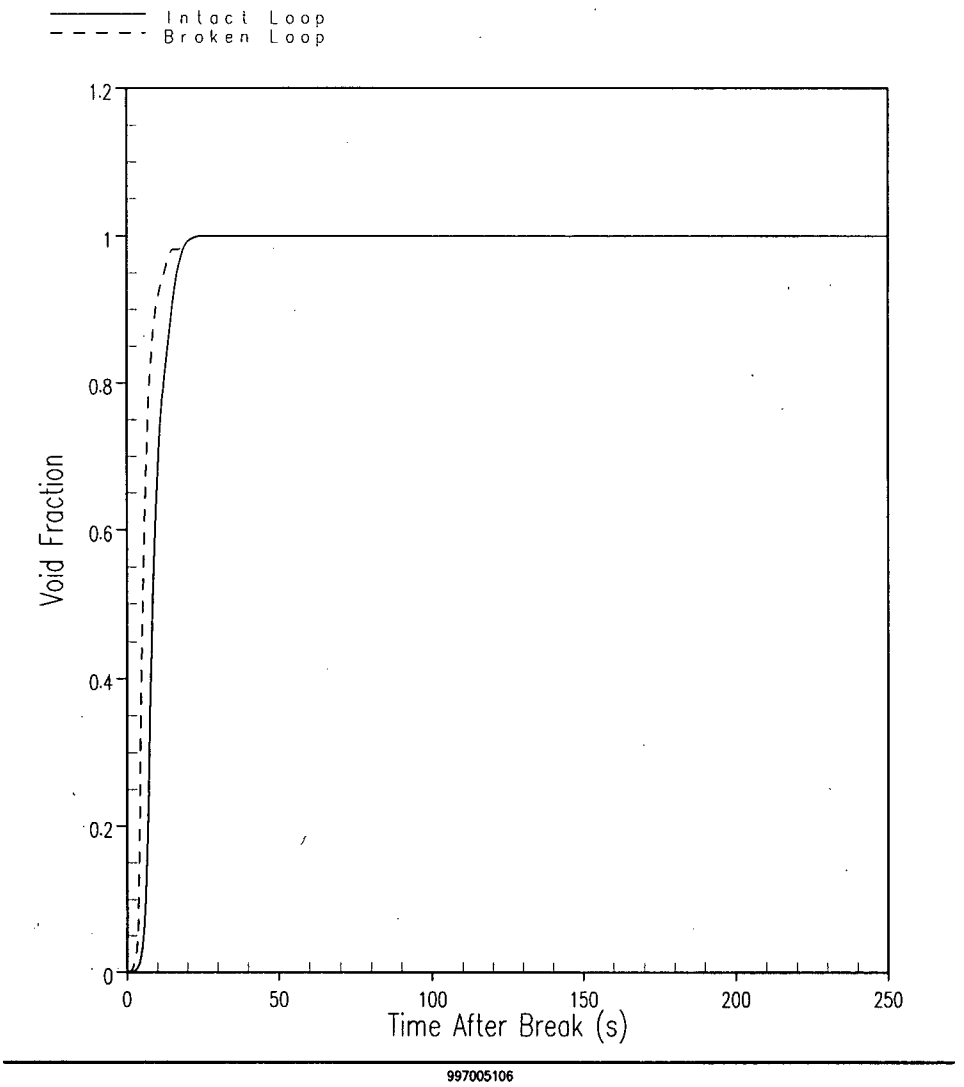


**Figure 6 – Void Fraction in Pumps for the Limiting Case**

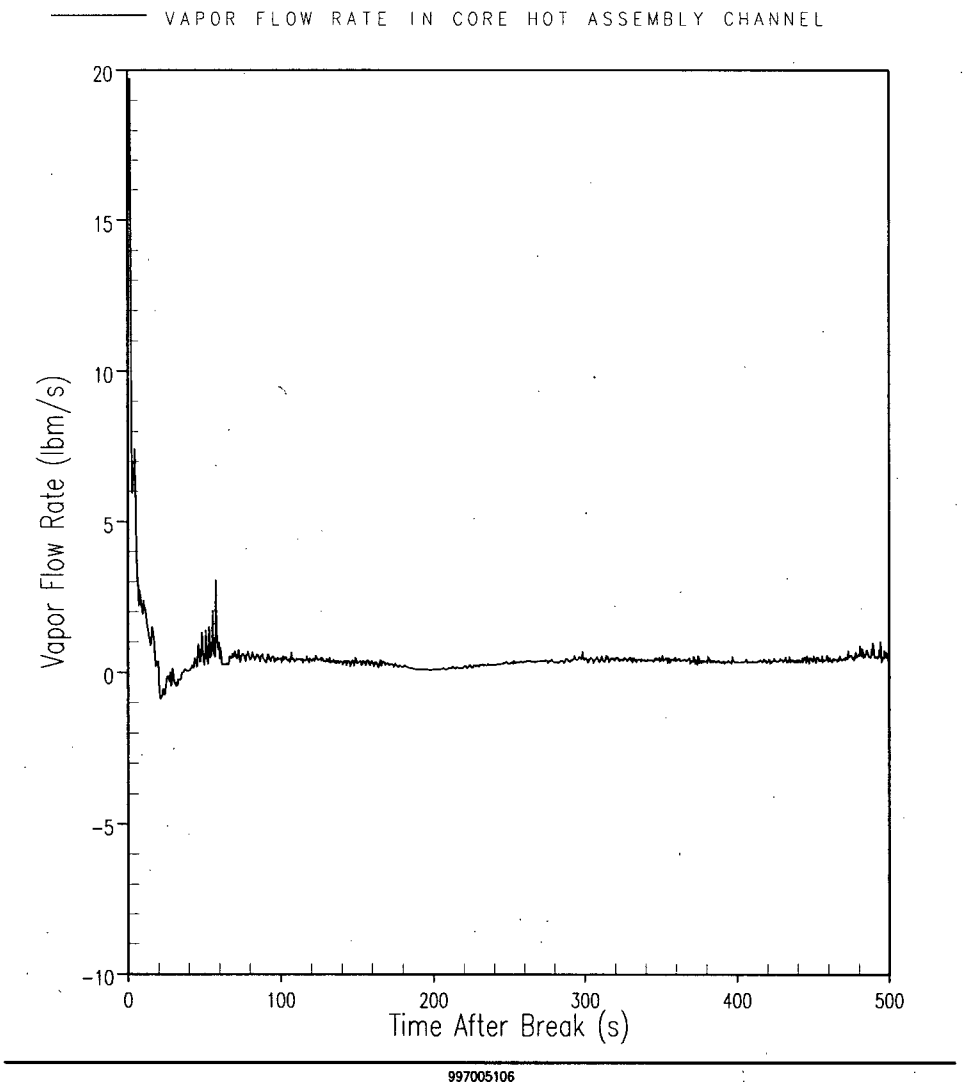


**Figure 7 – Vapor Flow at Top of Core Hot Assembly Channel for the Limiting Case**

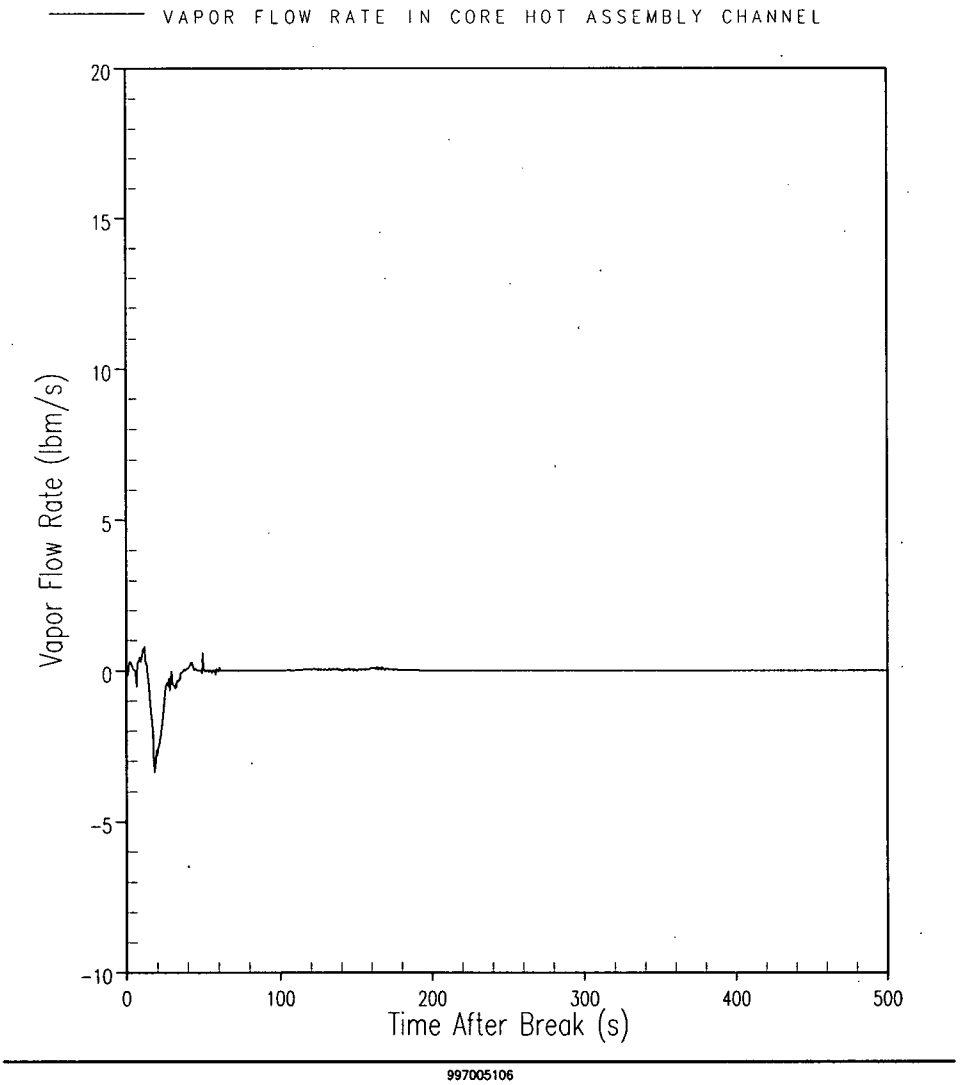


**Figure 8 – Vapor Flow at Bottom of Core Hot Assembly Channel for the Limiting Case**

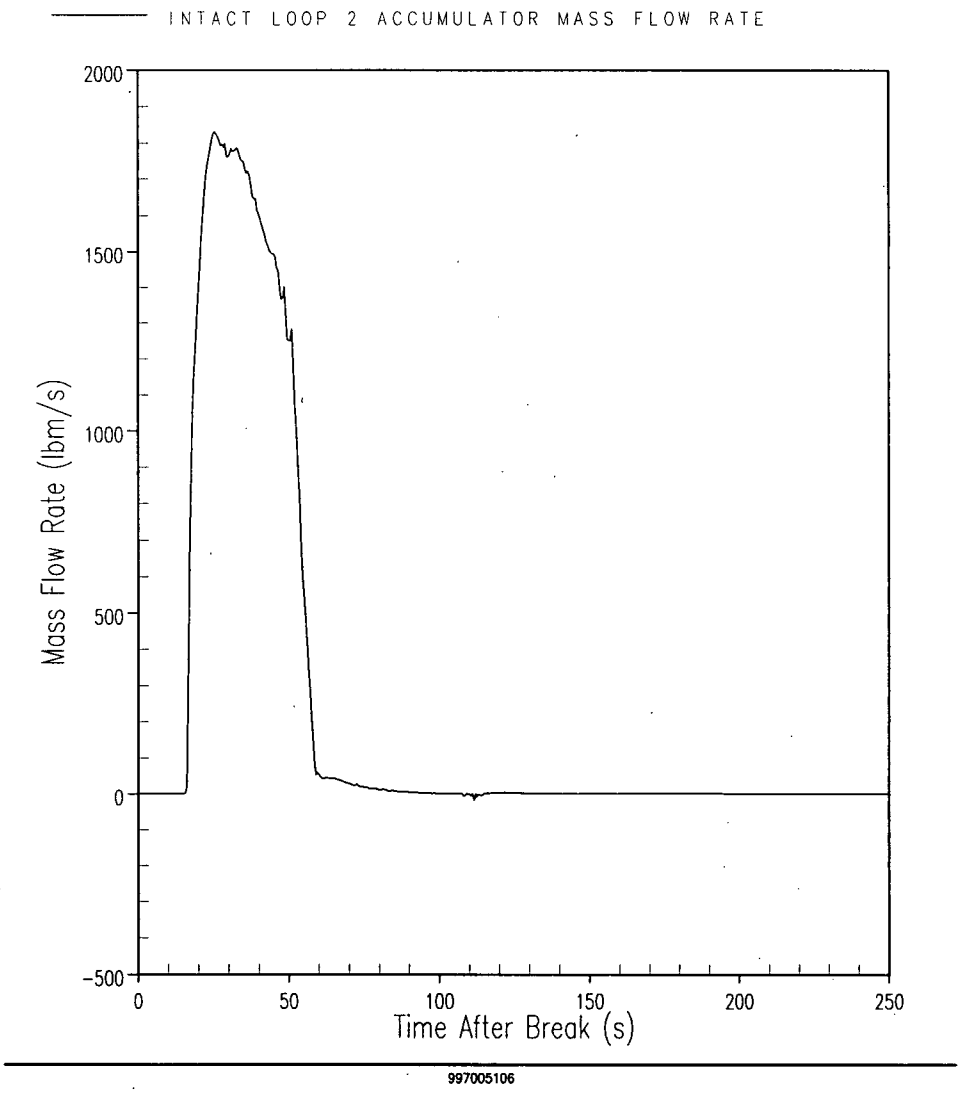


**Figure 9 – Accumulator Injection Flow for the Limiting Case**

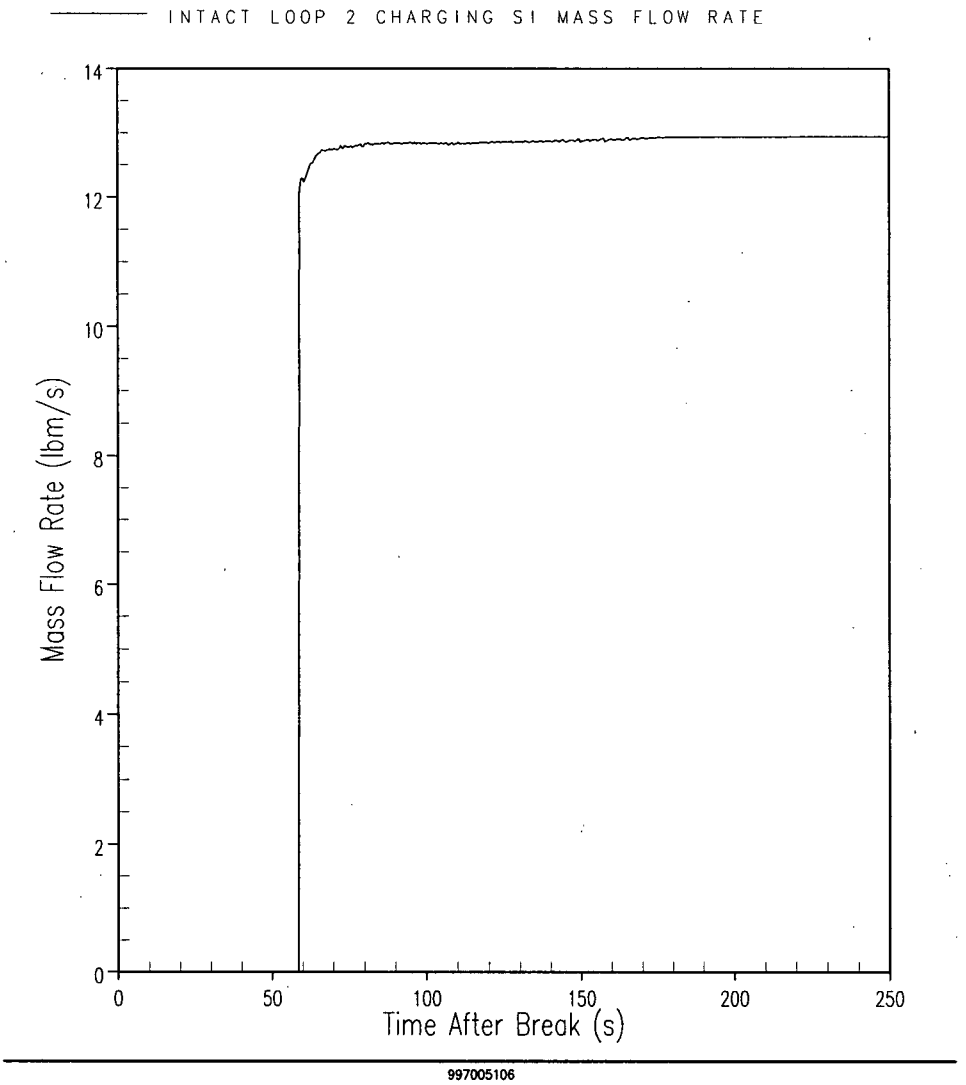


**Figure 10A – Charging Safety Injection (SI) Flow for the Limiting Case**

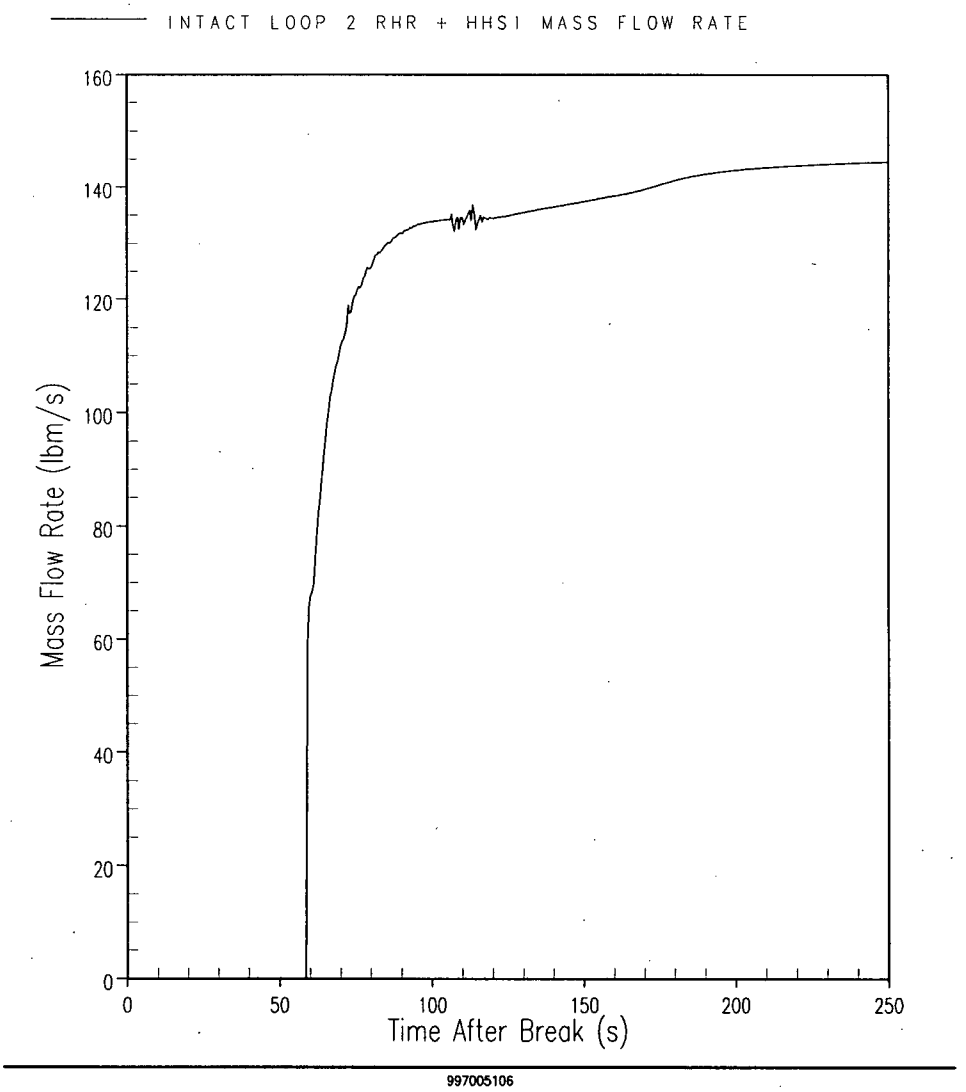


**Figure 10B – Residual Heat Removal (RHR) + High Head Safety Injection (HHSI) Flow for the Limiting Case**

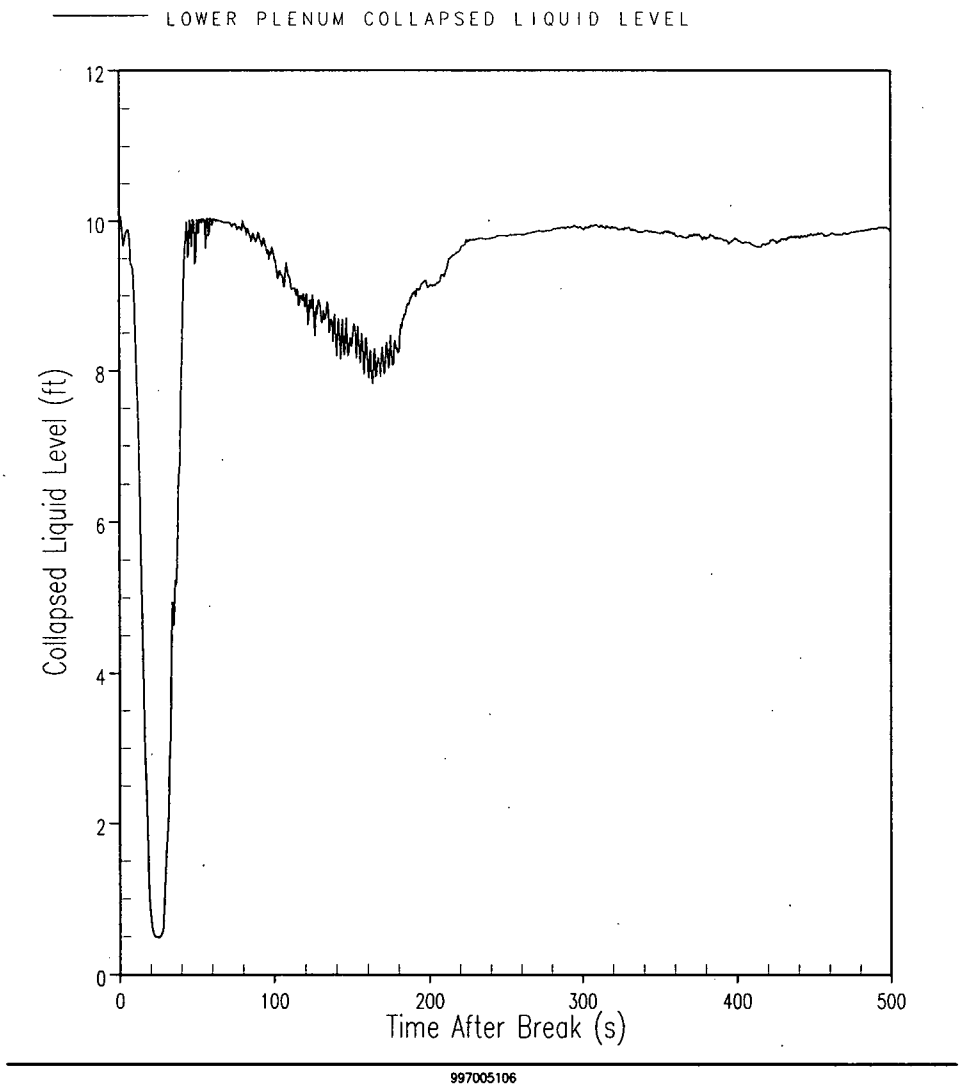


**Figure 11 – Lower Plenum Collapsed Liquid Level for the Limiting Case**

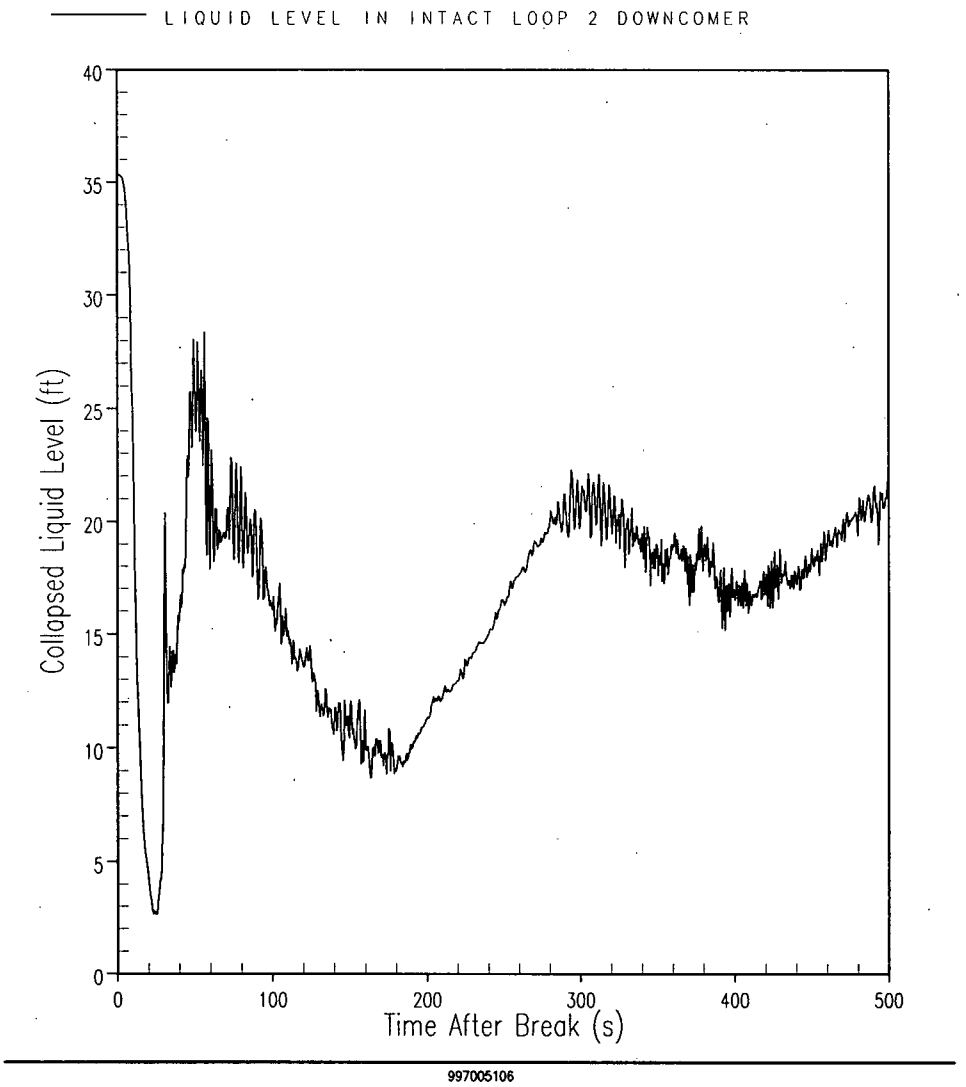


**Figure 12 – Downcomer Collapsed Liquid Level for the Limiting Case**

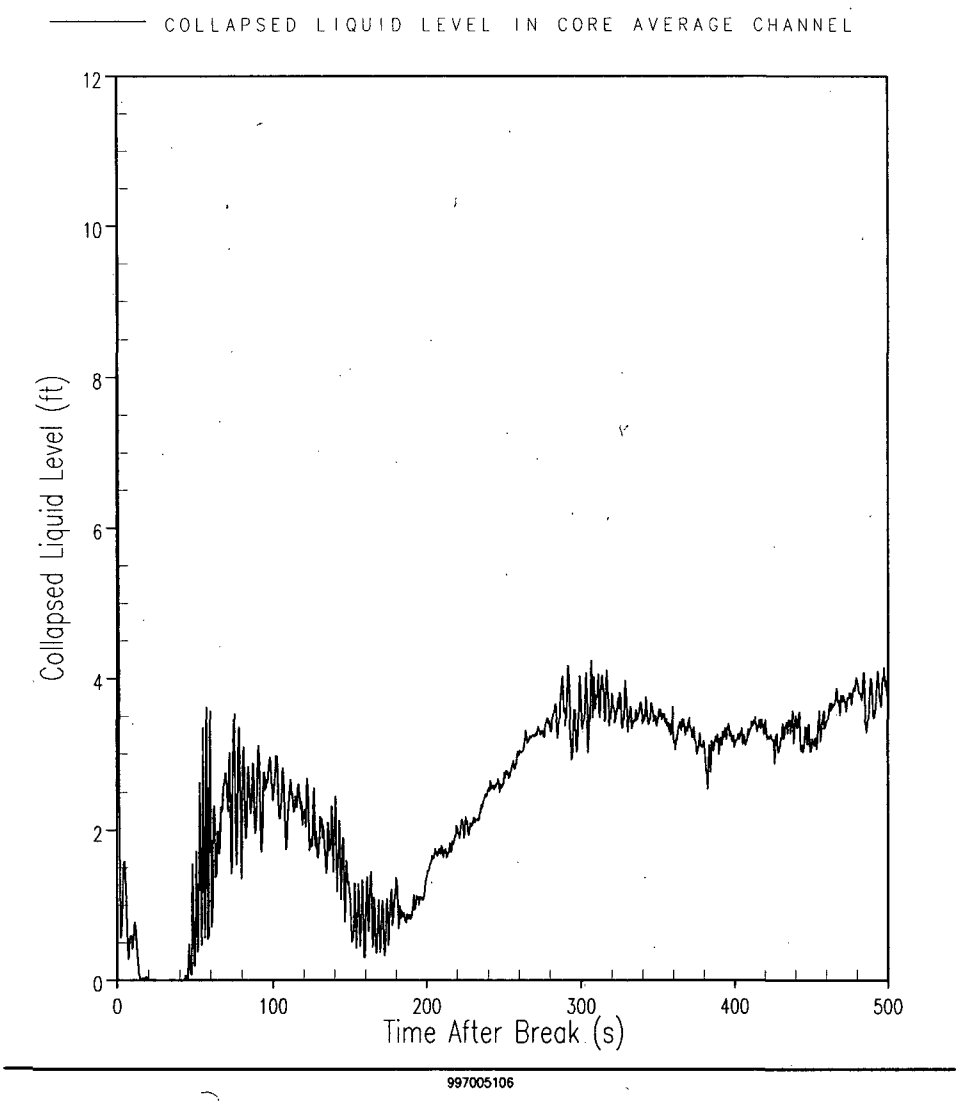


**Figure 13 – Core Average Channel Collapsed Liquid Level for the Limiting Case**

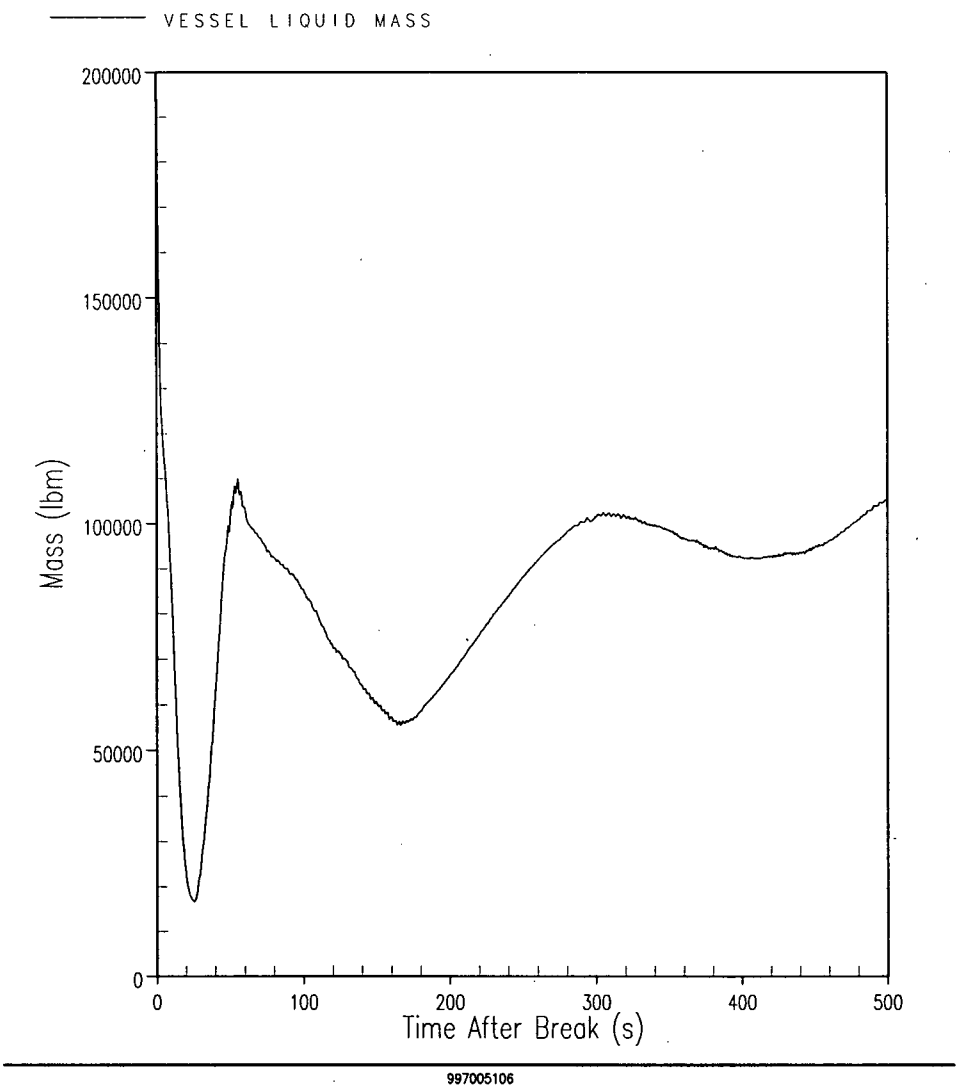


**Figure 14 – Vessel Fluid Mass for the Limiting Case**

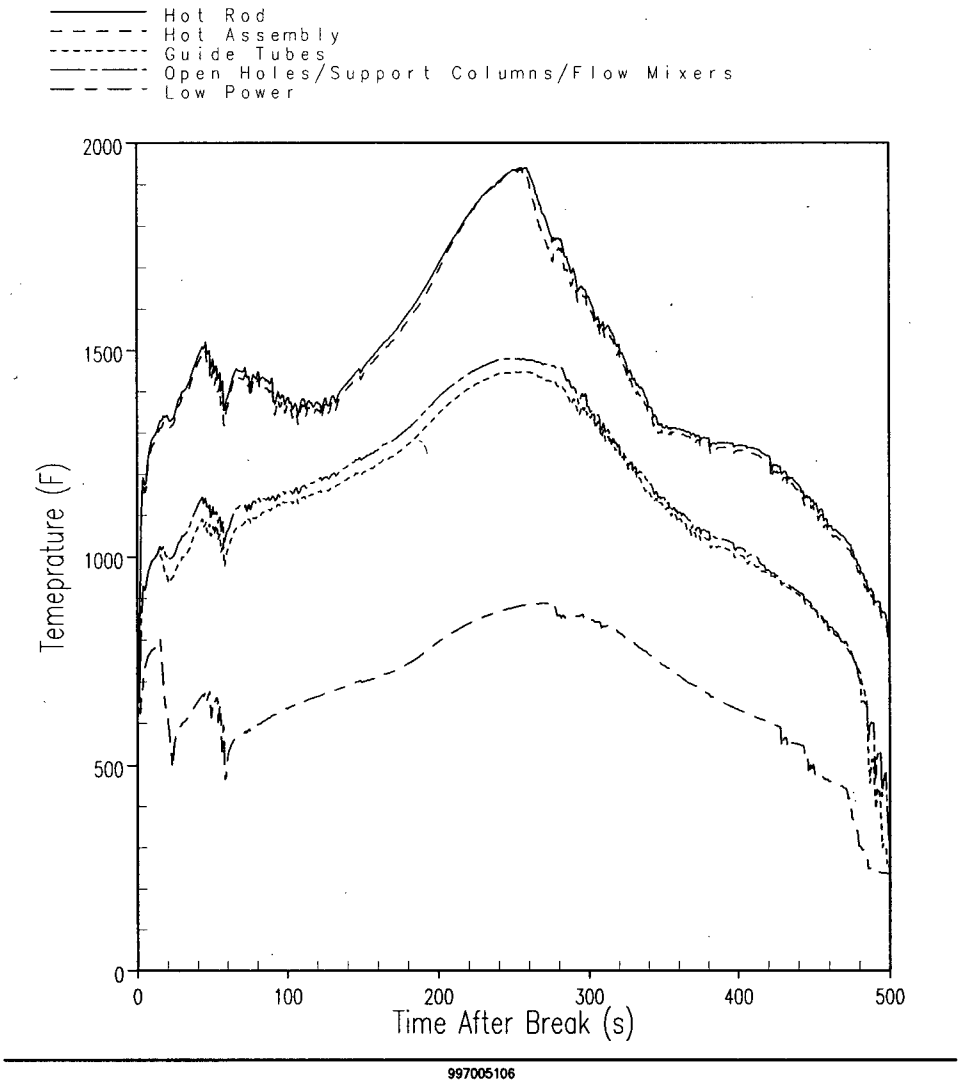


**Figure 15 – WCOBRA/TRAC Peak Cladding Temperature for all 5 Rod Groups for the Limiting Case**

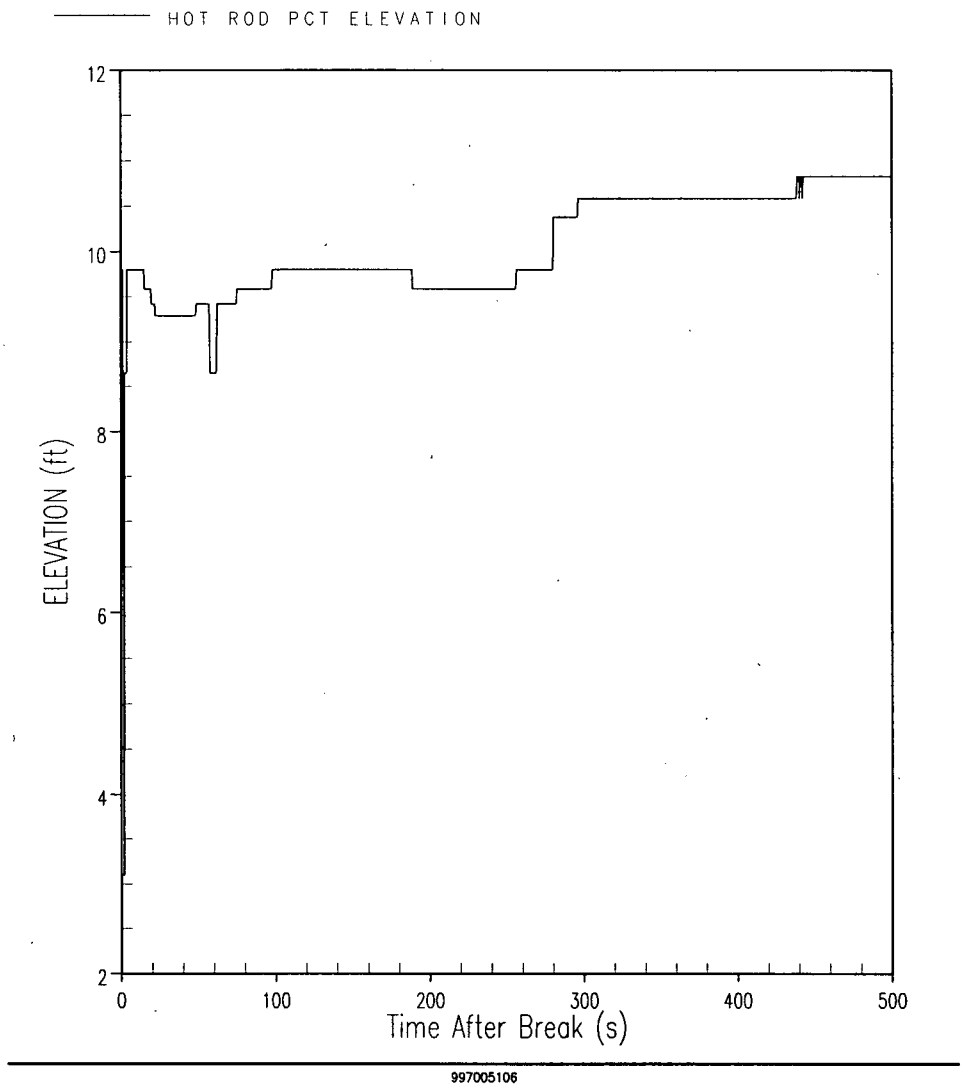


**Figure 16 – Peak Cladding Temperature Elevation for the Hot Rod for the Limiting Case***

*The PCT location is based on the core noding (approximately one node for every 1.8" of core elevation).

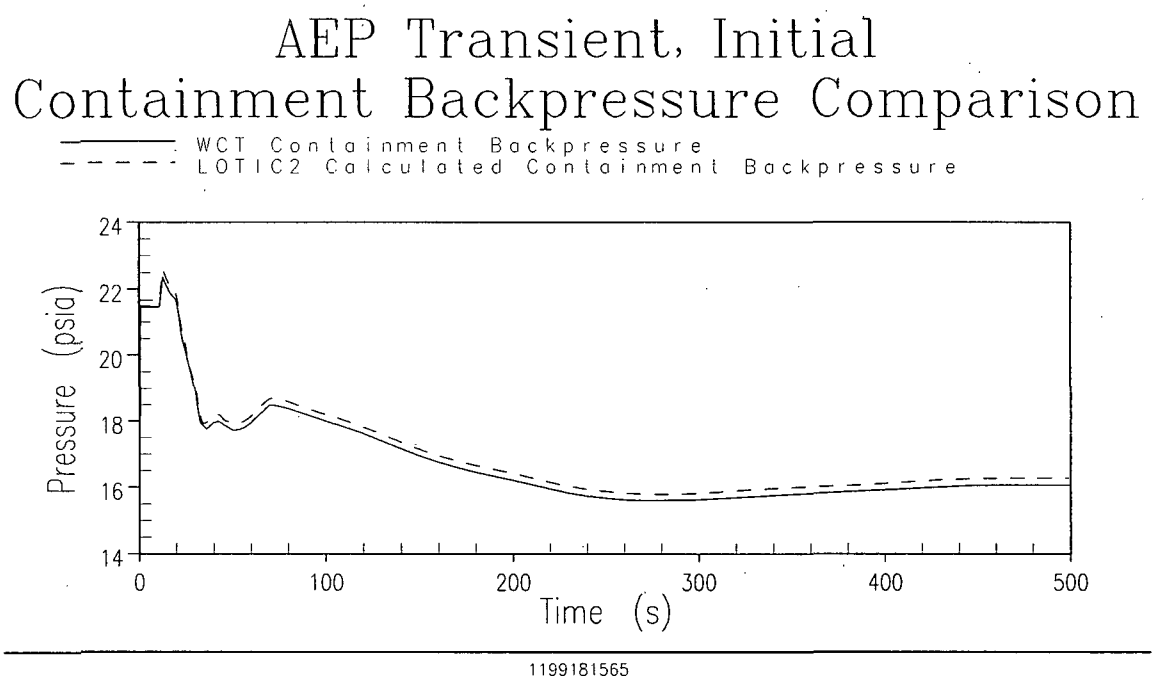


**Figure 17 – Analyzed Versus Calculated Containment Backpressure**

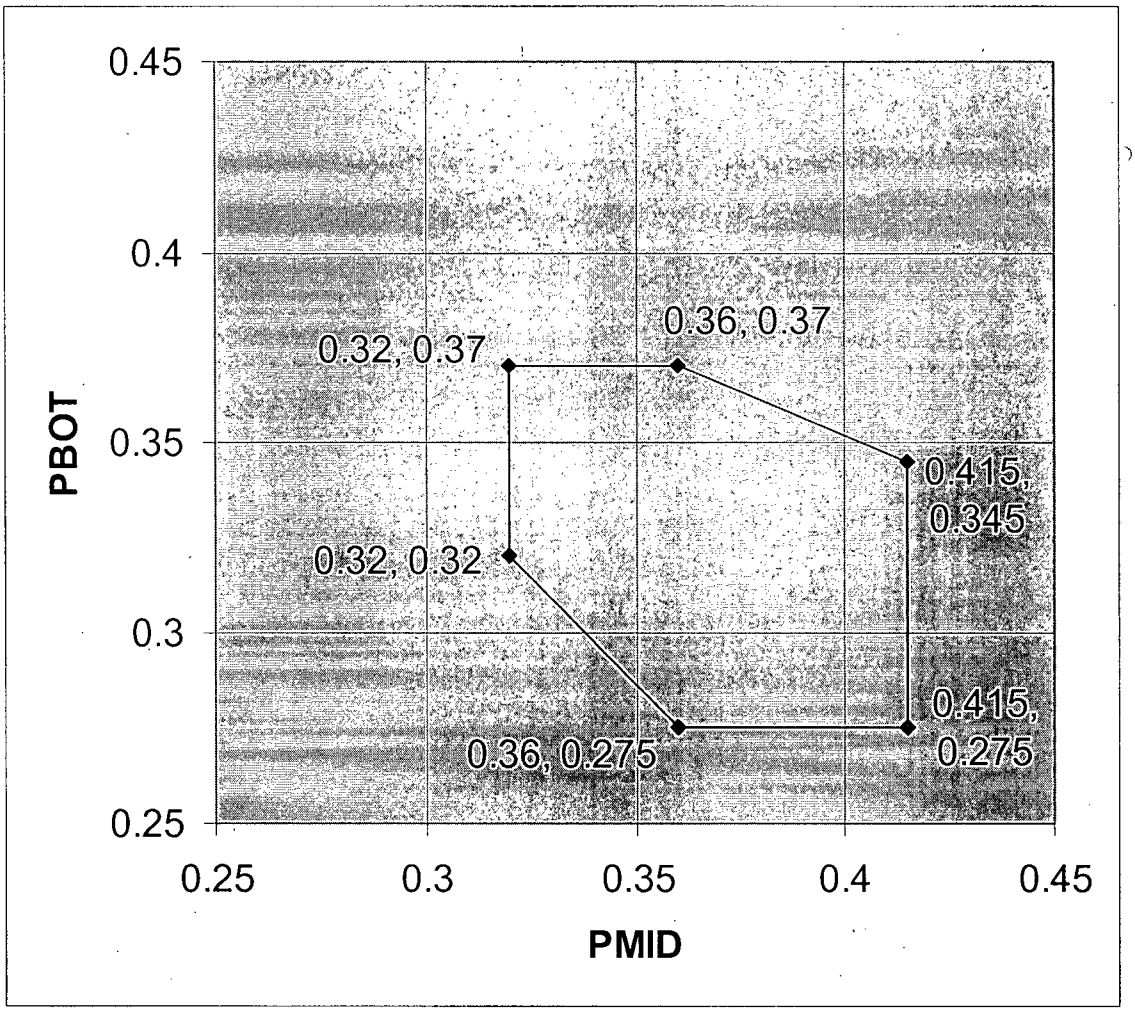


**Figure 18 – CNP Unit 1 BELOCA Analysis Axial Power Shape Operating Space Envelope**

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

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

## 5.0 REGULATORY SAFETY ANALYSIS

### 5.1 No Significant Hazards Consideration

Indiana Michigan Power Company (I&M) has evaluated whether a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

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

   Response: No

   The proposed Technical Specification (TS) change to allow the use of the best estimate large break loss-of-coolant accident (LOCA) analysis methodology using the automated statistical treatment of uncertainty methodology (ASTRUM) relies on a physical alteration of plant equipment to achieve four-loop low head safety injection. Therefore, there will be no increase in the probability of a LOCA. The consequences of a LOCA are not being increased.

   The plant conditions assumed in the analysis require the proposed TS change of a higher reactor coolant system (RCS) flow rate. The analysis assumes plant conditions that are bounded by the design conditions for all equipment in Unit 1. That is, it is shown that the emergency core cooling system is designed so that its calculated cooling performance conforms to the criteria contained in 10 CFR 50.46, Paragraph b. No other accident is potentially affected by this change.

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

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

   Response: No

   The proposed TS changes rely upon a modification permitting four-loop injection of low head safety injection. This modification to an accident mitigation system does not result in any new accident initiators. Analyses have confirmed that no transient event results in a new sequence of events that could lead to a new accident scenario. The parameters assumed in the analysis are within design limits of existing plant equipment.

In addition, employing the ASTRUM methodology does not create any new failure modes that could lead to a different kind of accident. No changes are being made to any reactor protection system or engineered safeguards features actuation system setpoints.

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

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

   Response: No

   It has been shown that the analytic technique used in the analysis realistically describes the expected behavior of the Donald C. Cook Nuclear Plant (CNP) Unit 1 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 analyzed to provide assurance that the most severe postulated LOCAs were analyzed. WCOBRA/TRAC validation with the revised downcomer noding has been found acceptable for application of the ASTRUM methodology, with no changes to the uncertainty treatment. The analysis has demonstrated that all acceptance criteria contained in 10 CFR 50.46, Paragraph b, continue to be satisfied. The higher RCS flow rate will provide more margin to departure from nucleate boiling limits.

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

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

## 5.2 Applicable Regulatory Requirements/Criteria

In accordance with 10 CFR 50.46, the conclusions of the best estimate large break LOCA analysis show that CNP Unit 1 continues to maintain a margin of safety to the limits prescribed by the following criteria:

1. The calculated maximum fuel element cladding temperature (i.e., peak cladding temperature) will not exceed 2,200°F.
2. The calculated total oxidation of the cladding (i.e. maximum cladding oxidation) will nowhere exceed 0.17 times the total cladding thickness before oxidation.
3. The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam (i.e., maximum hydrogen generation) will not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding

cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

4. The calculated changes in core geometry are such that the core remains amenable to cooling.
5. After successful initial operation of the emergency core cooling system, the core temperature will be maintained at an acceptably low value and decay heat will be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

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

## 6.0 ENVIRONMENTAL CONSIDERATIONS

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

## 7.0 REFERENCES

1. Letter from H. N. Berkow, 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 (ADAMS Accession Number ML043100073).

2. Letter from J. N. Jensen, I&M, to NRC Document Control Desk, "Donald C. Cook Nuclear Plant Unit 1, Thirty-Day Report for Loss-of-Coolant Accident Evaluation Model Changes," AEP:NRC:5046, dated April 29, 2005 (ADAMS Accession Number ML051310368).

## 8.0 PRECEDENT

The NRC has approved the following stations for use of the ASTRUM methodology:

Joseph M. Farley, Units 1 and 2 (Accession No. ML061810306)
Indian Point Nuclear Generating Unit 2 (Accession No. ML061710291)
Diablo Canyon Power Plant, Unit 2 (Accession No. ML063380020)
Prairie Island Nuclear Generating Plant, Units 1 and 2 (Accession No. ML071230789)

The CNP license amendment request differs from the NRC-approved amendments above by requesting NRC review of a plant-specific adaptation of the NRC-approved ASTRUM methodology. In addition, a TS change is requested to modify the minimum RCS total flow rate which the precedent plant approvals did not include in their license amendment requests. This change is necessary based on a change to input assumptions used in the CNP analysis.