PEAKING FACTOR LIMIT REPORT (Continued)

Factor Limit Report, the Peaking Factor Limit Report shall be provided to the NRC Document Control desk with copies to the Regional Administrator and the Resident Inspector within 30 days of their implementation, unless otherwise approved by the Commission.

The analytical methods used to generate the Peaking Factor limits shall be those previously reviewed and approved by the NRC. If changes to these methods are deemed necessary they will be evaluated in accordance with 10 CFR 50.59 and submitted to the NRC for review and approval prior to their use if the change is determined to involve an unreviewed safety question or if such a change would require amendment of previously submitted documentation.

CORE OPERATING LIMITS REPORT

6.9.1.7 Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT (COLR) before each reload cycle or any remaining part of a reload cycle for the following:

1. Axial Flux Difference for Specifications 3.2.1.
2. Control Rod Insertion Limits for Specification 3.1.3.6.
3. Heat Flux Hot Channel Factor - $F_Q(Z)$ for Specification 3/4.2.2.

The analytical methods used to determine the AFD limits shall be those previously reviewed and approved by the NRC in:

1. WCAP-10216-P-A, "RELAXATION OF CONSTANT AXIAL OFFSET CONTROL F_Q SURVEILLANCE TECHNICAL SPECIFICATION," June 1983.
2. WCAP-8385, "POWER DISTRIBUTION CONTROL AND LOAD FOLLOWING PROCEDURES - TOPICAL REPORT," September 1974.

The analytical methods used to determine the K(Z) curve shall be those previously reviewed and approved by the NRC in:

1. WCAP-9220-P-A, Rev. 1, "Westinghouse ECCS Evaluation Model - 1981 Version," February 1982.
2. WCAP-9561-P-A, ADD. 3, Rev. 1, "BART A-1: A Computer Code for the Best Estimate Analysis of Reflood Transients - Special Report: Thimble Modeling W ECCS Evaluation Model."

3. +4. INSERT A

The analytical methods used to determine the Rod Bank Insertion Limits shall be those previously reviewed and approved by the NRC in:

1. WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985.

The ability to calculate the COLR nuclear design parameters are demonstrated in:

1. Florida Power & Light Company Topical Report NF-TR-95-01, "Nuclear Physics Methodology for Reload Design of Turkey Point & St. Lucie Nuclear Plants".

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Attachment 3

INSERT A

3. "WCAP-10054-P-A, (proprietary) and WCAP-10081-NP-A, (non-proprietary), "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code", October, 1985.".

4. WCAP-10054-P-A Addendum 2, (proprietary), "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation Model", August, 1994."

ATTACHMENT 4

LICENSING REPORT

SMALL BREAK LOSS-OF-COOLANT ACCIDENT ANALYSIS FOR THE TURKEY POINT UNITS 3 AND 4 UPRATING PROGRAM

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SMALL BREAK LOCA ACCIDENT ANALYSIS

1.0 INTRODUCTION

This report contains information regarding the small break Loss-of-Coolant Accident (LOCA) analysis and evaluations performed in support of the uprating program for Turkey Point Units 3 and 4. The purpose of analyzing the small break LOCA is to demonstrate conformance with the 10 CFR 50.46 (Reference 1) requirements for the conditions associated with the uprating. Important input assumptions, as well as analytical models and analysis methodology for the small break LOCA, are contained in subsequent sections. Analysis results are provided in the form of Tables and Figures, as well as a more detailed description of the limiting transient. It was determined that no design or regulatory limit related to the small break LOCA would be exceeded due to the uprated power and assumed plant parameters.

2.0 INPUT PARAMETERS AND ASSUMPTIONS

The important plant conditions and features are listed in Table 1. Several additional considerations that are not identified in Table 1 are discussed below:

Figure 1 depicts the hot rod axial power shape modeled in the small break LOCA analysis. This shape was chosen because it represents a distribution with power concentrated in the upper regions of the core (the axial offset is + 20%). Such a distribution is limiting for small break LOCA since it minimizes coolant swell while maximizing vapor superheating and fuel rod heat generation at the uncovered elevations. The chosen power shape has been conservatively scaled to a flat K(Z) envelope based on the peaking factors given in table 1.

Figure 2 provides the degraded High Head Safety Injection (HHSI) flow versus pressure curve modeled in the small break LOCA analysis. The flow from one HHSI pump only is assumed in this analysis.

3.0 DESCRIPTION OF ANALYSES / EVALUATIONS PERFORMED

Analytical Model

For small breaks, the NOTRUMP computer code (References 2 and 3) is employed to calculate the transient depressurization of the reactor coolant system (RCS), as well as to describe the mass and energy of the fluid flow through the break. The NOTRUMP computer code is a one-dimensional general network code incorporating a number of advanced features. Among these advanced features are: calculation of thermal non-equilibrium in all fluid volumes, flow regime-dependent drift flux calculations with counter-current flooding limitations, mixture level tracking logic in multiple-stacked fluid nodes, regime-dependent drift flux calculations in multiple-stacked fluid nodes and regime-dependent heat transfer correlations. The NOTRUMP small break LOCA Emergency Core Cooling System (ECCS) Evaluation Model was developed to determine the RCS response to design basis small break LOCAs, and to address NRC concerns expressed in NUREG-0611 (Reference 4).

The RCS model is nodalized into volumes interconnected by flow paths. The broken loop is modeled explicitly, while the intact loops are lumped together into a second loop. Transient behavior of the system is determined from the governing conservation equations of mass, energy, and momentum. The multi-node capability of the program enables explicit, detailed spatial representation of various system components which, among other capabilities, enables a proper calculation of the behavior of the loop seal during a small break LOCA. The reactor core is represented as heated control volumes with associated phase separation models to permit transient mixture height calculations.

Fuel cladding thermal analyses are performed with a version of the LOCTA-IV code (Reference 5) using the NOTRUMP calculated core pressure, fuel rod power history, uncovered core steam flow and mixture heights as boundary conditions (see Figure 3).

Analysis

A spectrum of 2-inch, 3-inch, and 4-inch equivalent diameter cold leg breaks, was performed using the analytical model described above. A sensitivity of the limiting transient to the RCS vessel average temperature was also performed.

The most limiting single active failure assumed for a small break LOCA is that of an emergency power train failure which results in the loss of one complete train of ECCS components. In addition, a Loss-of-Offsite Power (LOOP) is assumed to occur coincident with reactor trip. This means that credit may be taken for at most two high head safety injection (SI) pumps and one low head, or residual heat removal (RHR), pump. However, in the analysis of the small break LOCA presented here, only the minimum delivered ECCS flow from a single high head SI pump with degraded flow was assumed.

The small break LOCA analysis performed for the Turkey Point Units 3 and 4 uprating program utilizes the NRC-approved NOTRUMP Evaluation Model (References 2 and 3), with appropriate modifications to model pumped SI and accumulator injection in the broken loop as well as an improved condensation model (COSI) for the pumped SI into the broken and intact loops (Reference 6 and 7).

The small break LOCA analysis performed for the Turkey Point uprating program assumes SI is delivered to both the intact and broken loops at the RCS backpressure.

Prior to break initiation, the plant is assumed to be in a full uprated power (102%) equilibrium condition, i.e., the heat generated in the core is being removed via the secondary system. Other initial plant conditions assumed in the analysis are given in Section 2.0 and Table 1. Subsequent to the break opening, a period of reactor coolant system blowdown ensues in which the heat from fission product decay, the hot reactor internals, and the reactor vessel continues to be transferred to the RCS fluid. The heat transfer between the RCS and the secondary system may be in either direction and is a function of the relative temperatures of the primary and secondary. In the case of continuous heat addition to the secondary during a period of quasi-equilibrium, an increase in the secondary system pressure results in steam relief via the steam generator safety valves, which were modeled with 3 percent accumulation and 3 percent tolerance.

Should a small break LOCA occur, depressurization of the RCS causes fluid to flow into the loops from the pressurizer resulting in a pressure and level decrease in the pressurizer. The reactor trip signal

subsequently occurs when the pressurizer low-pressure reactor trip setpoint, conservatively modeled as 1805 psia (including uncertainties), is reached. LOOP is assumed to occur coincident with reactor trip. A safety injection signal is generated when the pressurizer low-pressure safety injection setpoint, conservatively modeled as 1615 psia (including uncertainties), is reached. Safety injection is delayed 35 seconds after the occurrence of the low pressure condition. This delay accounts for signal initiation, diesel generator start up and emergency power bus loading consistent with the assumed loss of offsite power coincident with reactor trip, as well as the time involved in aligning the valves and bringing the HHSI pump up to full speed. These countermeasures limit the consequences of the accident in two ways:

1). Reactor trip and borated water injection supplement void formation in causing a rapid reduction of nuclear power to a residual level corresponding to the delayed fission and fission product decay. No credit is taken in the LOCA analysis for the boron content of the injection water. (However, an average RCS/sump mixed boron concentration is calculated to ensure that the post-LOCA core remains subcritical). In addition, credit is taken in the small break LOCA analysis for the insertion of Rod Cluster Control Assemblies (RCCAs) subsequent to the reactor trip signal, while assuming the most reactive RCCA is stuck in the full out position. A rod drop time of 3 seconds was assumed while also considering an additional 2 seconds for the signal processing delay time. Therefore, a total delay time of 5 seconds from the time of reactor trip signal to full rod insertion was used in the small break LOCA analysis.

2). Injection of borated water ensures sufficient flooding of the core to prevent excessive cladding temperatures.

During the earlier part of the small break transient prior to the assumed loss-of-offsite power coincident with reactor trip, the loss of flow through the break is not sufficient enough to overcome the positive core flow maintained by the reactor coolant pumps. During this period, upward flow through the core is maintained. However, following the reactor coolant pump trip (due to a LOOP) and subsequent pump coastdown, a partial period of core uncovery occurs. Ultimately, the small break transient analysis is terminated when the ECCS flow provided to the RCS exceeds the break flow rate.

The core heat removal mechanisms associated with the small break transient include not only the break itself and the injected ECCS water, but also that heat transferred from the RCS to the steam generator secondary side. Main Feedwater (MFW) is assumed to be isolated coincident with the safety injection signal, and the MFW pumps coast down to 0% flow in 10 seconds. A continuous supply of makeup water is also provided to the secondary using the auxiliary feedwater (AFW) system. An AFW actuation signal occurs coincident with the safety injection signal, resulting in the assumed delivery of full AFW system flow 120 seconds following the signal. The heat transferred to the secondary side of the steam generator aids in the reduction of the RCS pressure.

Should the RCS depressurize to approximately 600 psig, as in the case of the limiting 3-inch break and the 4-inch break, the cold leg accumulators begin to inject borated water into the reactor coolant loops. In the case of the 2-inch break however, the vessel mixture level is recovered without the aid of accumulator injection.

Evaluations

Upon completion of the small break LOCA analysis, an evaluation was performed for automatic containment spray actuation during small break LOCA. This evaluation accounts for the fact that Turkey Point Units 3 and 4 may be subject to SI interruption for up to 2 minutes while switching over to cold leg recirculation. The results of this evaluation are discussed in Section 5.0.

4.0 ACCEPTANCE CRITERIA FOR ANALYSES / EVALUATIONS

The Acceptance Criteria for the LOCA are described in 10 CFR 50.46 (Reference 1) as follows:

1). The calculated maximum fuel element cladding temperature shall not exceed 2200°F,

2). The calculated total oxidation of the cladding shall 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 shall 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). Calculated changes in core geometry shall be such that the core remains amenable to cooling,

5). After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

Criteria 1 through 3 are explicitly covered by the small break LOCA analysis at uprated conditions.

For criterion 4), the appropriate core geometry was modeled in the analysis. The results based on this geometry satisfy the PCT criterion of 10 CFR 50.46 and consequently, demonstrate the core remains amenable to cooling.

For criterion 5), Long-Term Core Cooling (LTCC) considerations are not directly applicable to the small break LOCA analysis.

The criteria were established to provide a significant margin in emergency core cooling system (ECCS) performance following a LOCA.

5.0 RESULTS

In order to determine the conditions that produced the most limiting small break LOCA case (as determined by the highest calculated peak cladding temperature), a total of four cases were examined. These cases included the investigation of variables including break size and RCS temperature to ensure that the most severe postulated small break LOCA event was analyzed. The following discussions

provide insight into the analyzed conditions.

First, a break spectrum based on high RCS T_{AVG} was performed, as this was expected to yield more limiting PCT results than low RCS T_{AVG}. The limiting break for the Turkey Point Units was found to be a 3-inch diameter cold leg break. The results of Reference 8 demonstrate that the cold leg break location is limiting with respect to postulated cold leg, hot leg and pump suction leg break locations. The PCT attained during the transient was 1688°F (refer to Table 2). While Table 3 provides the key transient event times.

A summary of the transient response for the limiting high T_{AVG} 3-inch break case is shown in Figures 4 through 12 These figures present the response of the following parameters:

1). RCS Pressure Transient, (Figure 4)

2). Core Mixture Level, (Figure 5)

3). Peak Cladding Temperature, (Figure 6)

4). Top Core Node Vapor Temperature, (Figure 7)

5). Safety Injection Mass Flow Rate for the Intact and Broken Loops, (Figures 8 and 9)

6). Cold Leg Break Mass Flow Rate, (Figure 10)

7). Hot Rod Surface Heat Transfer Coefficient at the Hot Spot, (Figure 11)

8). Fluid Temperature at the Hot Spot, (Figure 12)

Upon initiation of the limiting 3-inch break, there is a slow depressurization of the RCS (see Figure 4). During the initial period of the small break transient, the effect of the break flow rate is not sufficient to overcome the flow rate maintained by the reactor coolant pumps as they coast down. As such, normal upward flow is maintained through the core and core heat is adequately removed. Following reactor trip, the removal of the heat generated as a result of fission products decay is accomplished via a two-phase mixture level covering the core. From the core mixture level and cladding temperature transient plots for the 3-inch break calculations given in Figures 5 and 6, respectively, it is seen that the peak cladding temperature occurs near the time when the core is most deeply uncovered and the top of the core is being cooled by steam. This time is characterized by the highest vapor superheating above the mixture level (refer to Figure 7).

A comparison of the flow provided by the safety injection system to the intact and broken loops to the total break mass flow rate at the end of the transient (as given in Figures 8, 9 and 10, respectively), shows that at the time the transient was terminated, the total safety injection flow rate that was delivered to the intact and broken loops exceeds the mass flow rate out the break (70.1 lbm/sec versus 61.2 lbm/sec). In addition, the inner vessel core mixture level has recovered the top of the core (Figure 5).

Figures 11 and 12 provide additional information on the hot rod surface heat transfer coefficient at the hot spot and fluid temperature at the hot spot, respectively.

After this point in the transient, there is no longer a concern of exceeding the 10 CFR 50.46 criteria as described in Section 4.0 since:

1). The RCS pressure is gradually decaying, and

2). The net mass inventory is increasing.

As the RCS inventory continues to gradually increase, the core mixture level will continue to increase and the fuel cladding temperatures will continue to decline. The 3-inch high T_{AVG} small break LOCA transient is terminated.

Additional Break Cases

Studies documented in Reference 8 have determined that the limiting small break LOCA transient occurs for breaks of less than 10 inches in diameter. To ensure that the 3-inch diameter break was the most limiting, calculations were also performed with break equivalent diameters of 2 inches and 4 inches. The results of each of these cases are given in Tables 2 and 3. Plots of the following parameters for each case are also given in Figures 13 through 15 for the 2-inch break case and Figures 16 through 18 for the 4-inch break.

1). RCS Pressure Transient,

2). Core Mixture Level, and

3). Peak Cladding Temperature.

The PCTs for the 2-inch and 4-inch breaks were 1656°F and 1583°F, respectively (see Table 2). The PCTs for each of these cases was calculated to be less than that for the 3-inch break case based on high T_{AVG} conditions.

Limiting Temperature Conditions

Reduced operating temperature typically results in a PCT benefit for the small break LOCA. However, due to competing effects and the complex nature of small break LOCA transients, there have been some instances where more limiting results have been observed for the reduced operating temperature case. For this reason, a small break LOCA transient based on a lower bound RCS vessel average temperature was performed.

The temperature window analyzed was based on a nominal vessel average temperature of 574.2°F, with ± 3°F for an operating window and ± 8.5°F to bound uncertainties. The break spectrum was performed at the high vessel average temperature (585.7°F), as this case was expected to yield limiting results. Then, a sensitivity analysis for the low vessel average temperature (562.7°F) was performed based on the limiting 3-inch break case from the break spectrum analyses previously described.

Plots of the following parameters are given in Figures 19 through 21 for the 3-inch break case at low T_{AVG} conditions:

1). RCS Pressure Transient,

2). Core Mixture Level, and

3). Peak Cladding Temperature.

The PCT for the 3-inch break case based on low vessel average temperature was 1619°F (see Table 2). Therefore, the PCT for this case was calculated to be less than that for the 3-inch break case with high vessel average temperature conditions.

Evaluations

The evaluation for containment spray actuation in small break LOCA resulted in no change to the predicted small break LOCA PCT for the various cases analyzed. The DRFA fuel stack height above the lower core plate was explicitly modeled for the various cases analyzed.

6.0 CONCLUSIONS

A full break spectrum Small Break LOCA analysis supporting the uprated Turkey Point core with the high nominal vessel average temperature, $T_{AVG} = 585.7°F$, was performed. Peak cladding temperatures of 1656°F, 1688°F, and 1583°F were calculated for the 2-inch, 3-inch, and 4-inch cold leg breaks, respectively, thus identifying the 3-inch equivalent diameter break as limiting. A sensitivity to low nominal vessel average temperature, $T_{AVG} = 562.7°F$, was performed. The calculated peak cladding temperature was 1619°F, identifying the 3-inch equivalent diameter cold leg break, high nominal vessel average temperature, as the limiting case.

The analyses presented in this section show that the high head safety injection subsystems of the Emergency Core Cooling System, together with the heat removal capability of the steam generator, provide sufficient core heat removal capability to maintain the calculated peak cladding temperatures below the required limit of 10 CFR 50.46 which is defined in Section 4.0.

7.0 REFERENCES

1. "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 1974, as amended in Federal Register, Volume 53, September 1988.

2. Meyer, P.E., "NOTRUMP - A Nodal Transient Small Break and General Network Code,"WCAP-10079-P-A, (proprietary) and WCAP-10080-NP-A (non-proprietary), August 1985.

3. Lee, N. et al., "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," WCAP-10054-P-A (proprietary) and WCAP-10081-NP-A (non-proprietary), August 1985.

4. "Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in Westinghouse - Designed Operating Plant," NUREG-0611, January 1980.

5. Bordelon, F. M. et al., "LOCTA-IV Program: Loss-of-Coolant Transient Analysis", WCAP-8301 (proprietary) and WCAP-8305 (non-proprietary), June 1974.

6. Thompson, C. M, et al., "Addendum to the Westinghouse Small Break LOCA Evaluation Model Using the NOTRUMP Code: Safety Injection Into the Broken Loop and the COSI Condensation Model", WCAP-10054-P, Addendum 2 (proprietary) and WCAP-10081-NP (non-proprietary), August 1994.

7. Shimeck, D. J., "COSI SI/Steam Condensation Experiment Analysis", WCAP-11767-P-A (proprietary), March 1988.

8. Rupprecht, S. D. et al, "Westinghouse Small Break LOCA ECCS Evaluation Model Generic Study with the NOTRUMP Code", WCAP-11145-P-A (proprietary) and WCAP-11372-NP-A (non-proprietary), October 1986.

Table 1

SMALL BREAK LOCA ACCIDENT ANALYSIS INPUT PARAMETERS FOR TURKEY POINT

Parameter	High T_{AVG}	(Low T_{AVG})
Reactor core rated thermal power[1], (MWt)	2300	
Peak linear power[1,2], (kw/ft)	14.9	
Total peaking factor (F_Q^T) at peak[2]	2.50	
Power shape[2]	See Figure 1	
$F_{\Delta H}$	1.70	
P_{HA}	1.515	
Fuel[3]	15x15 DRFA	
Accumulator water volume, nominal (ft^3/acc.)	892	
Accumulator tank volume, nominal (ft^3/acc.)	1200	
Accumulator gas pressure, minimum (psig)	600	
Pumped safety injection flow	See Figure 2	
Steam generator tube plugging level (%)[4]	20	
Thermal Design Flow/loop, (gpm)	85,000	
Vessel average temperature w/ uncertainties, (°F)	585.7	(562.7)
Reactor coolant pressure w/ uncertainties, (psia)	2320	
Min. aux. feedwater flowrate/loop, (lb/sec)[5]	9.26	

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[1] Two percent is added to this power to account for calorimetric error.
[2] This represents a power shape corresponding to a one-line segment peaking factor envelope, K(z), based on $F_Q^T = 2.50$.

[3] DRFA fuel type modeled in the small break LOCA analysis.

[4] Maximum plugging level in any one or all steam generators.

[5] Flowrates per steam generator.

Table 2

SMALL BREAK LOCA ANALYSIS FUEL CLADDING RESULTS

Break Spectrum, (High T_{AVG})

	2-inch	3-inch	4-inch
Peak Cladding Temperature (°F)	1656	1688	1583
Peak Cladding Temperature Location (ft)	11.75	11.75	11.50
Peak Cladding Temperature Time (sec)	2627	1188	668
Local Zr/H_2O Reaction, Max (%)	2.0188	1.5535	0.6679
Local Zr/H_2O Reaction Location (ft)	11.75	11.50	11.25
Total Zr/H_2O Reaction (%)	< 1.0	< 1.0	< 1.0
Hot Rod Burst Time (sec)	No Burst	No Burst	No Burst
Hot Rod Burst Location (ft)	N/A	N/A	N/A

Results for the limiting 3-inch break size

	High T_{AVG}	Low T_{AVG}
Peak Cladding Temperature (°F)	1688	1619
Peak Cladding Temperature Location (ft)	11.75	11.50
Peak Cladding Temperature Time (sec)	1188	1229
Local Zr/H_2O Reaction, Max (%)	1.5535	1.1034
Local Zr/H_2O Reaction Location (ft)	11.50	11.50
Total Zr/H_2O Reaction (%)	< 1.0	< 1.0
Hot Rod Burst Time (sec)	No Burst	No Burst
Hot Rod Burst Location (ft)	N/A	N/A

Table 3

SMALL BREAK LOCA ANALYSIS TIME SEQUENCE OF EVENTS

Break Spectrum, (High T_{AVG})

	2-inch	3-inch	4-inch
Break Occurs (sec)	0.0	0.0	0.0
Reactor Trip Signal (sec)	40.6	17.0	10.4
Safety Injection Signal (sec)	58.9	30.4	21.4
Top Of Core Uncovered (sec)	1402	482	278[1]
Accumulator Injection Begins (sec)	N/A	1040	525
Peak Cladding Temperature Occurs (sec)	2627	1188	668
Top Of Core Covered (sec)	4554	2363	965

Results for the limiting 3-inch break size

	High T_{AVG}	Low T_{AVG}
Break Occurs (sec)	0.0	0.0
Reactor Trip Signal (sec)	17.0	14.4
Safety Injection Signal (sec)	30.4	21.8
Top Of Core Uncovered (sec)	482	526
Accumulator Injection Begins (sec)	1040	1086
Peak Cladding Temperature Occurs (sec)	1188	1229
Top Of Core Covered (sec)	2363	2343

[1] Momentary core uncovery occurred at 213 seconds during prelude to loop seal clearing. The beginning of the subsequent extended core uncovery at 278 seconds is the time listed.

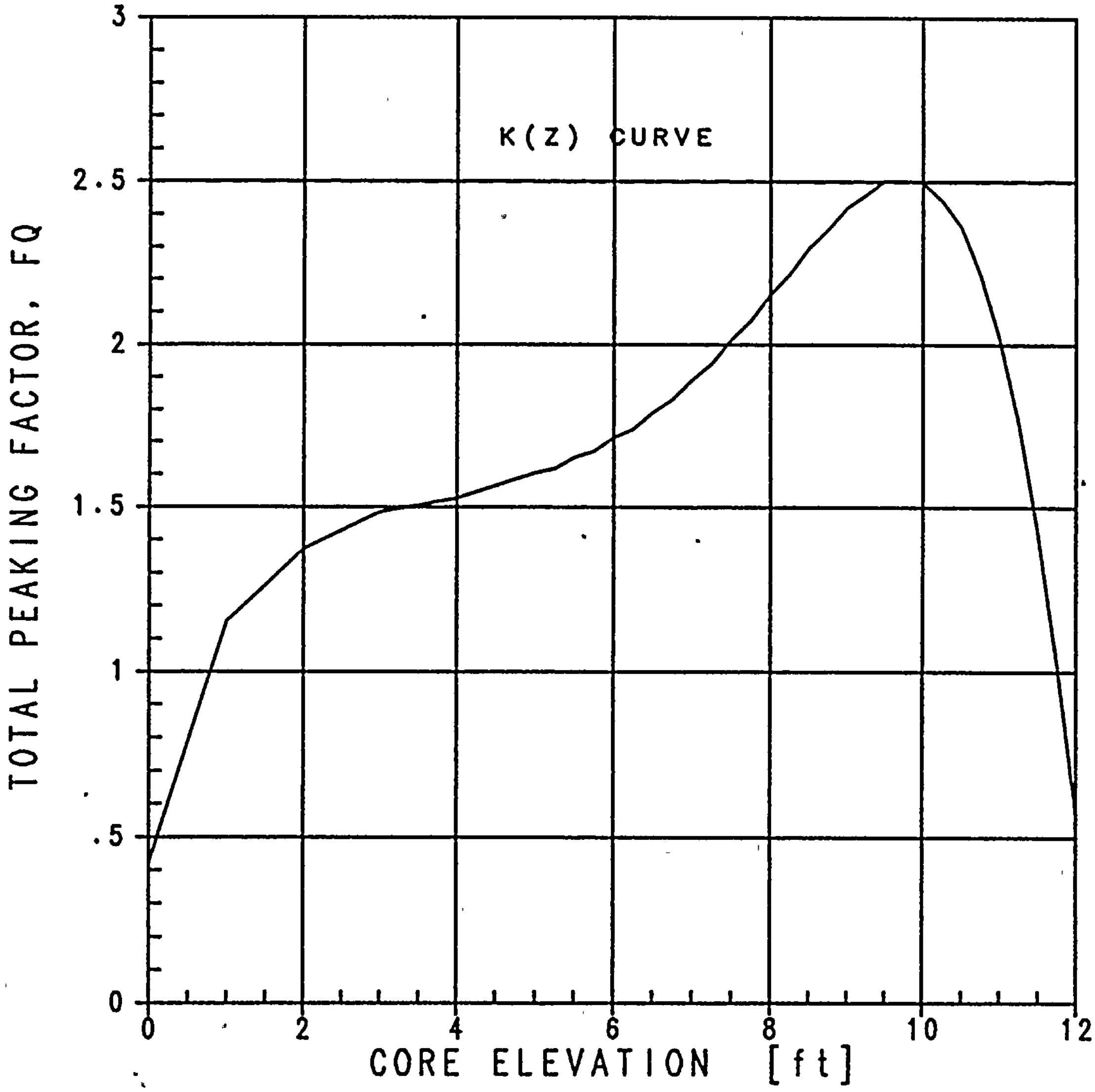

Figure 1: Small Break Hot Rod Power Shape

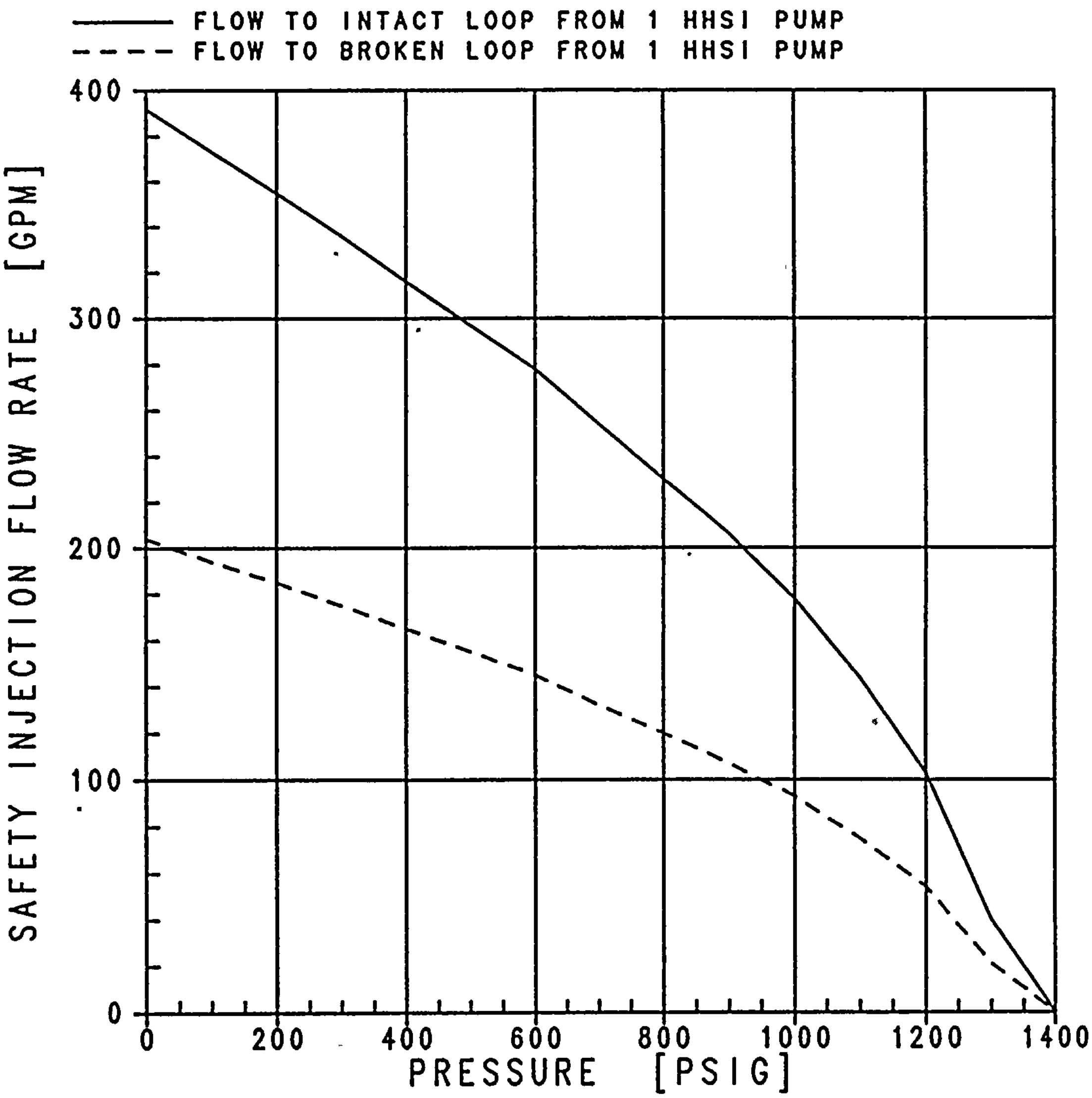

Figure 2: Small Break Pumped Safety Injection Flow Rate - 1 HHSI Pump

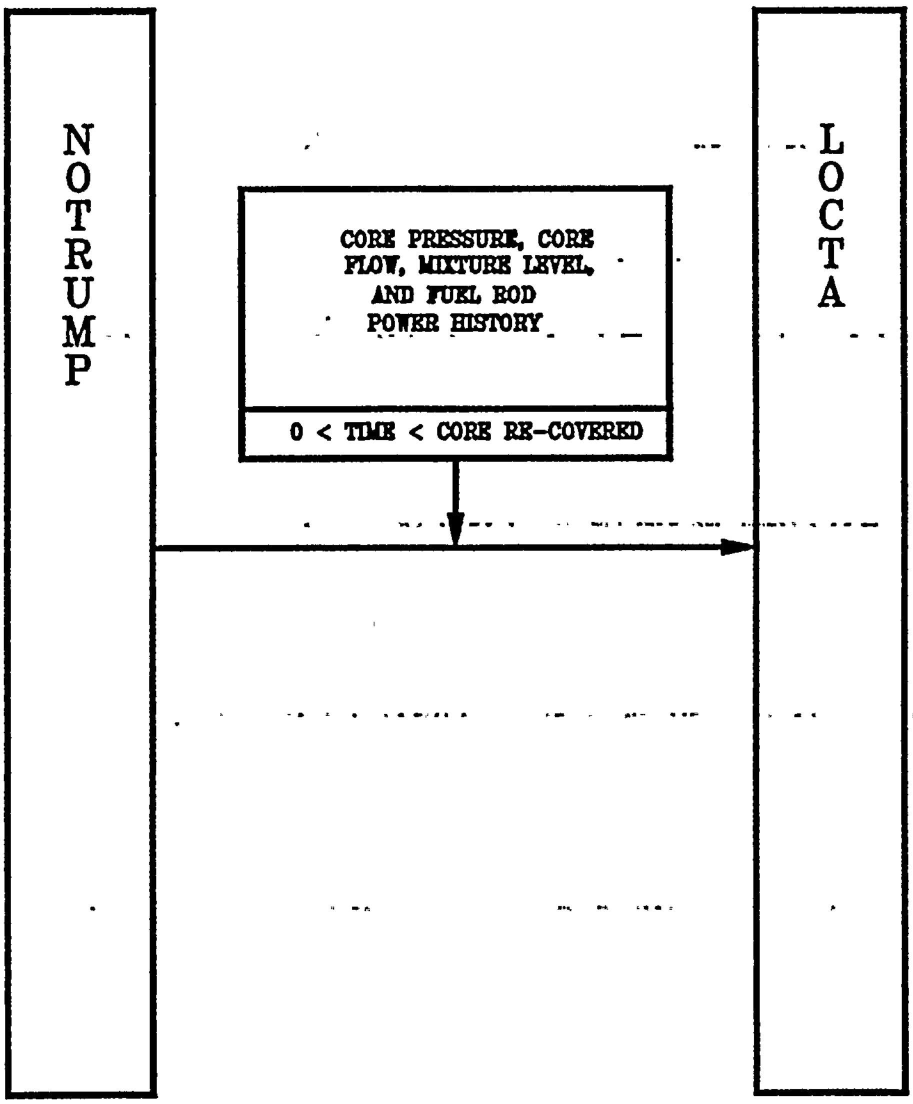

Figure 3: Code Interface Description for the Small Break LOCA Model

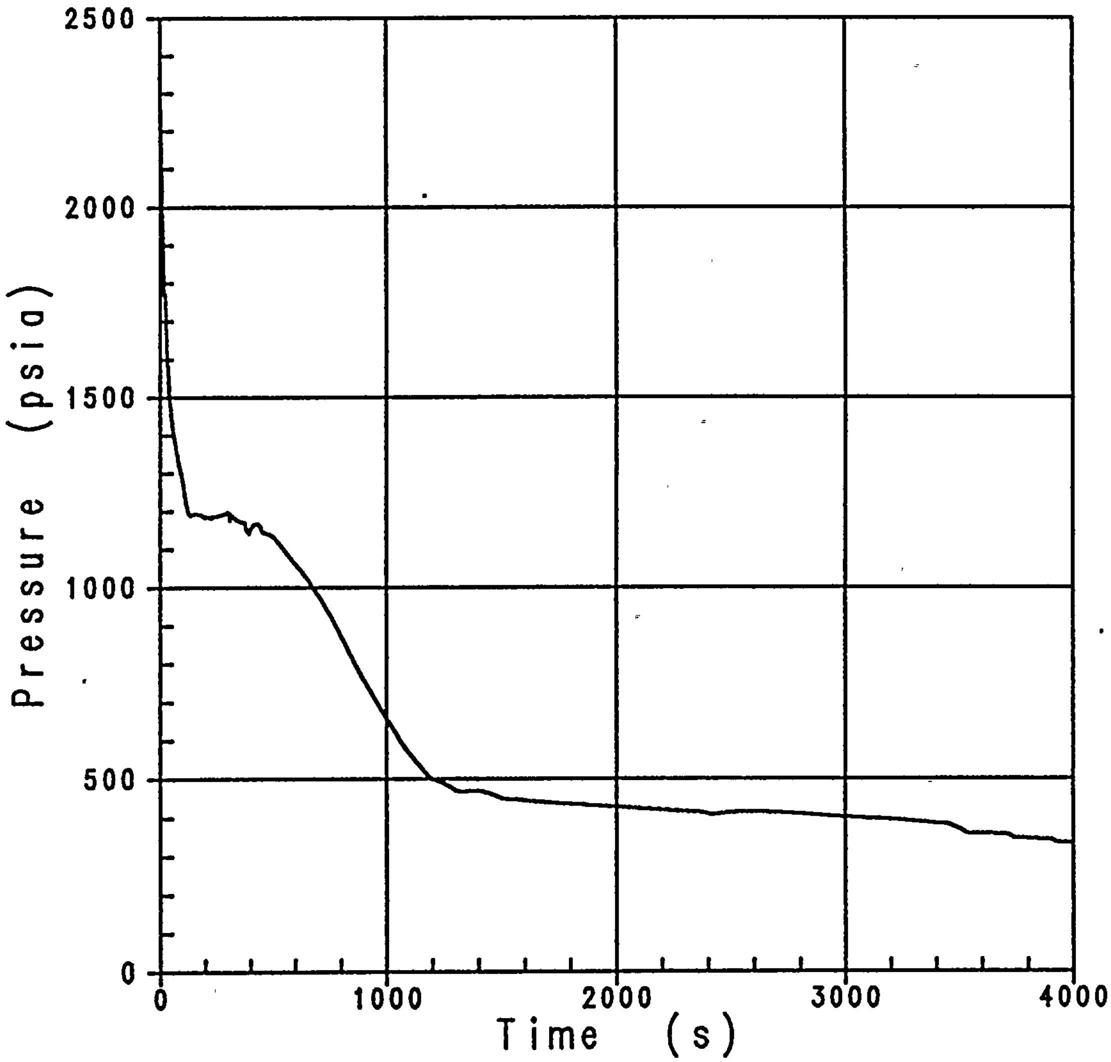

Figure 4: RCS Depressurization Transient, Limiting 3-Inch Break, High T_{AVG}

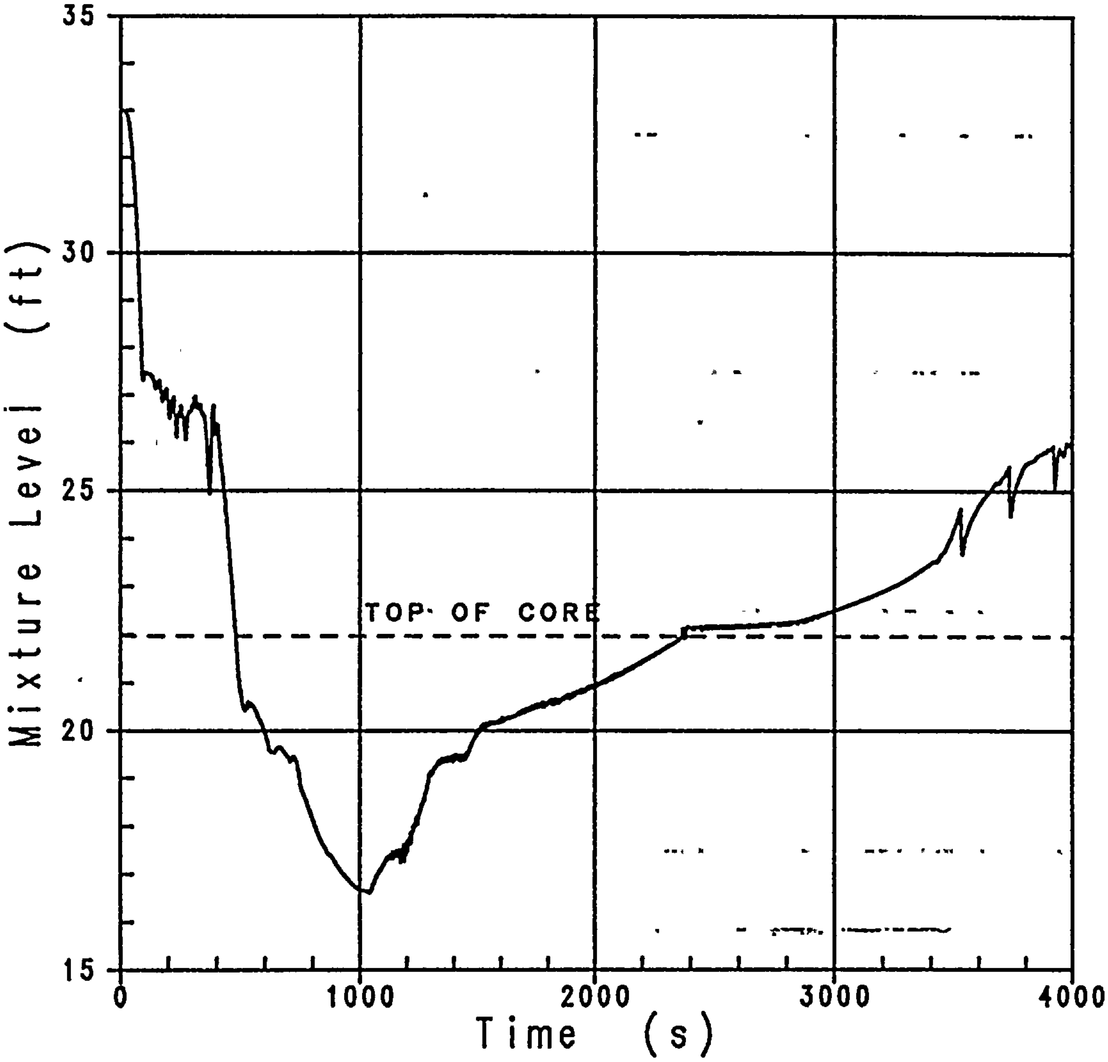

Figure 5: Core Mixture Level, 3-Inch Break, High T_{AVG}

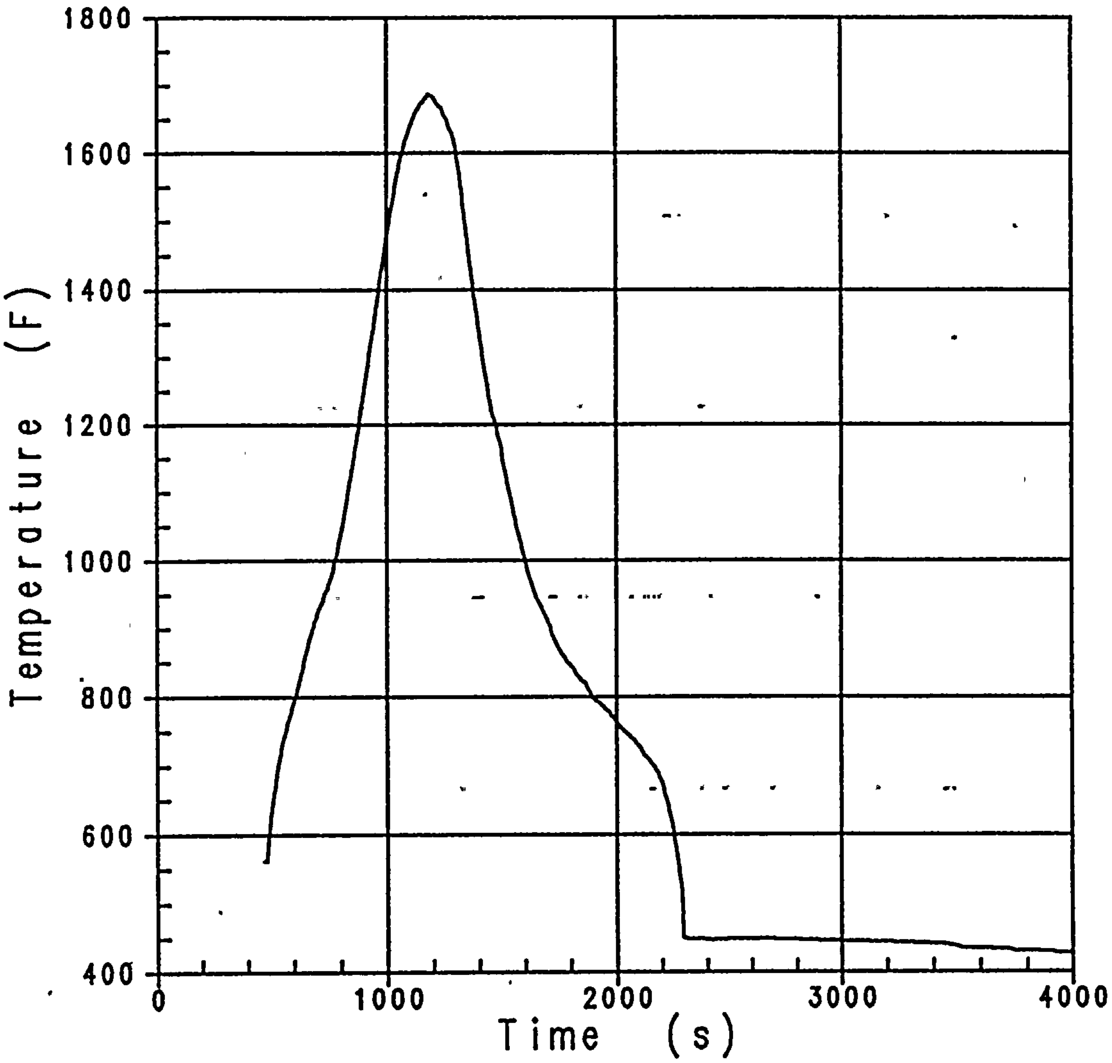

Figure 6: Peak Cladding Temperature - Hot Rod, 3-Inch Break, High T_{AVG}

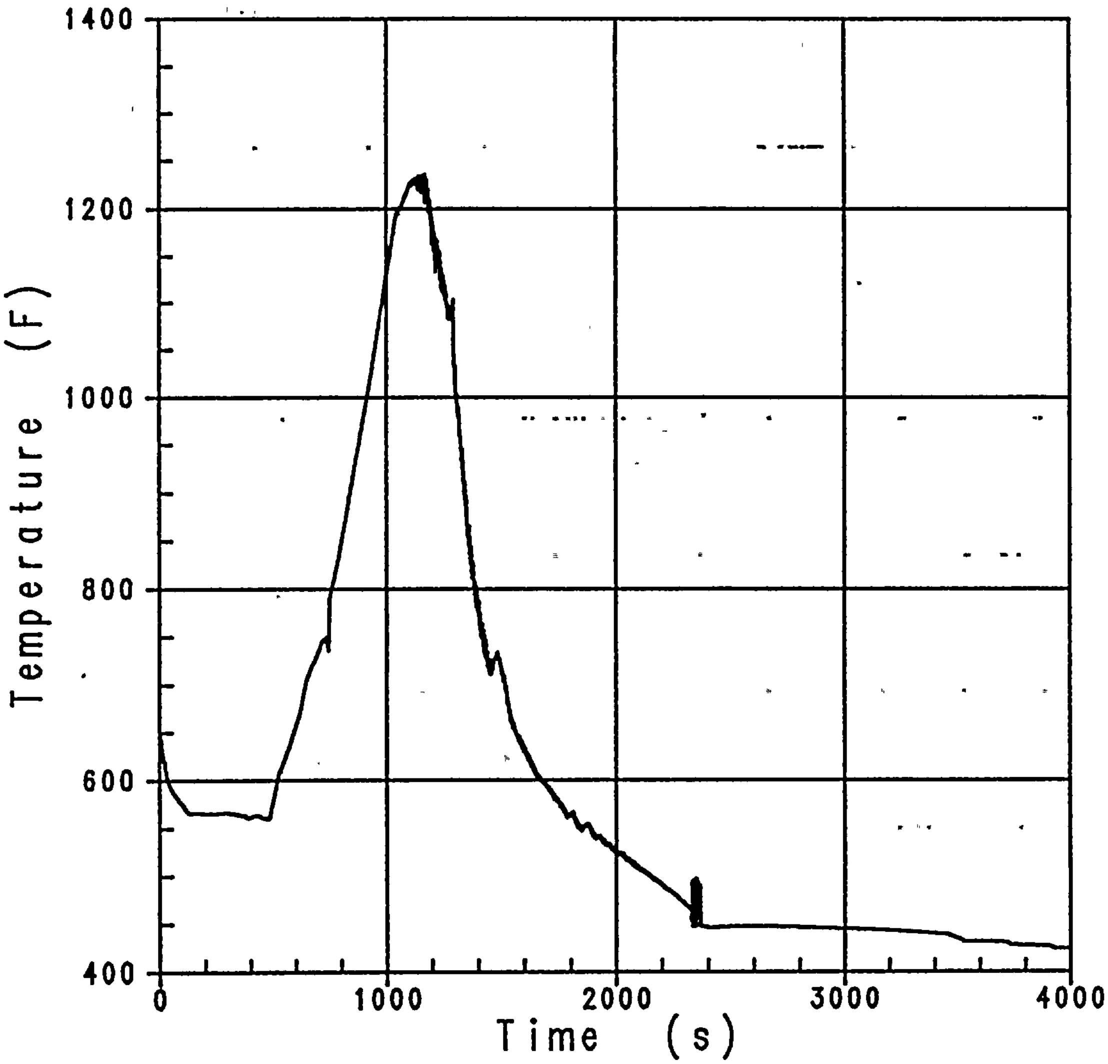

Figure 7: Top Core Node Vapor Temperature, 3-Inch Break, High T_{AVG}

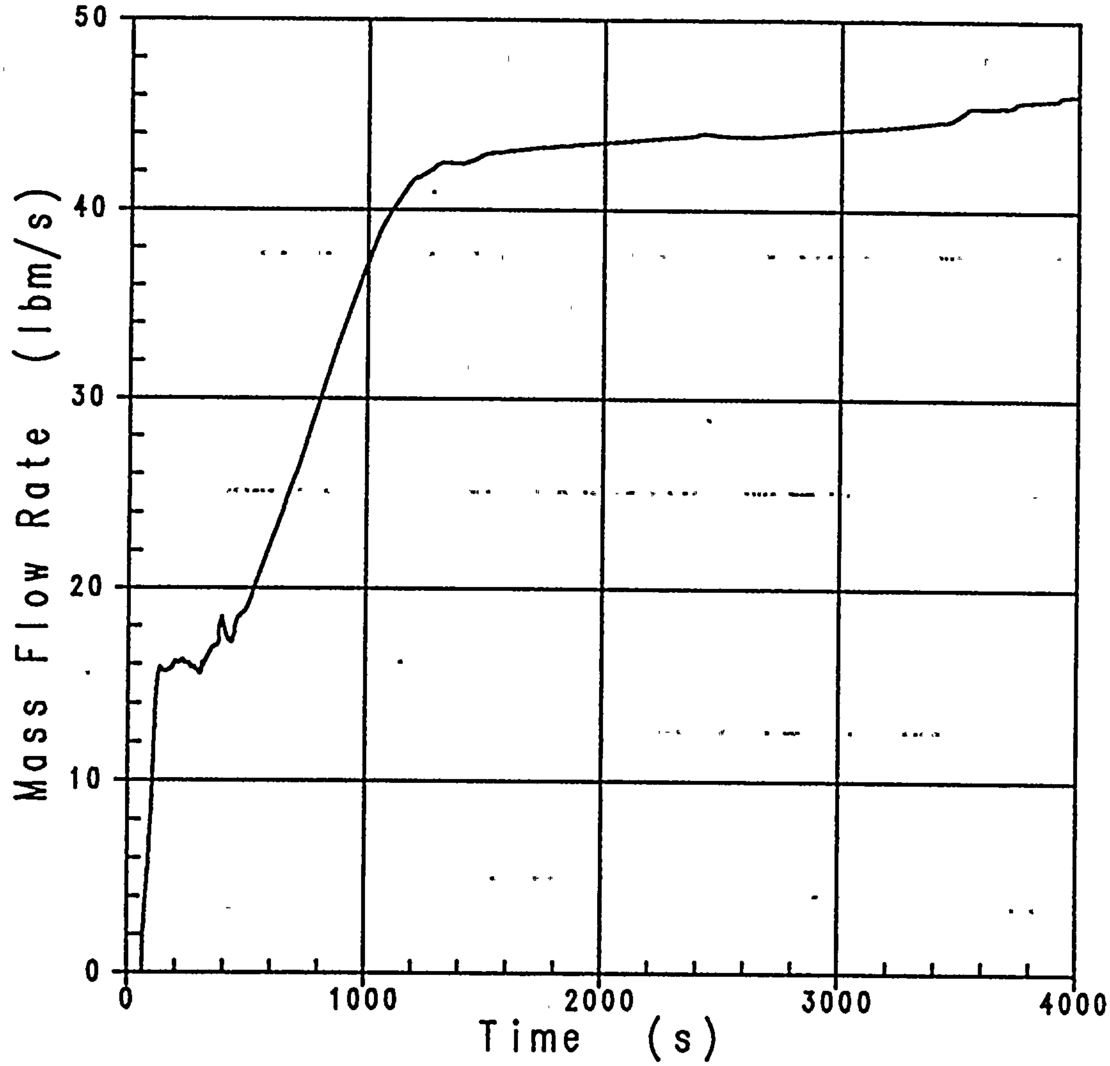

Figure 8: ECCS Pumped Safety Injection - Intact Loop, 3-Inch Break, High T_{AVG}

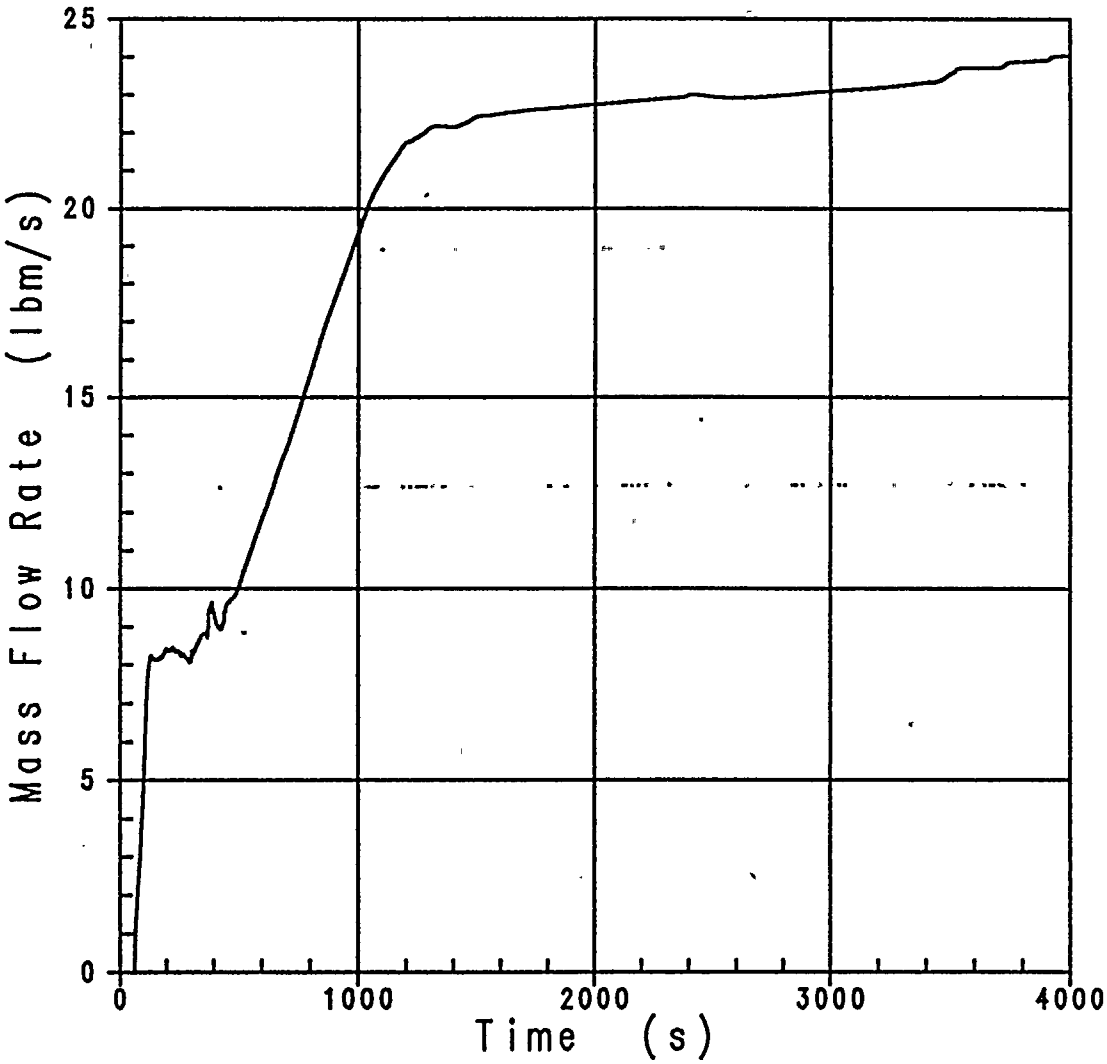

Figure 9: ECCS Pumped Safety Injection - Broken Loop, 3-Inch Break, High T_{AVG}

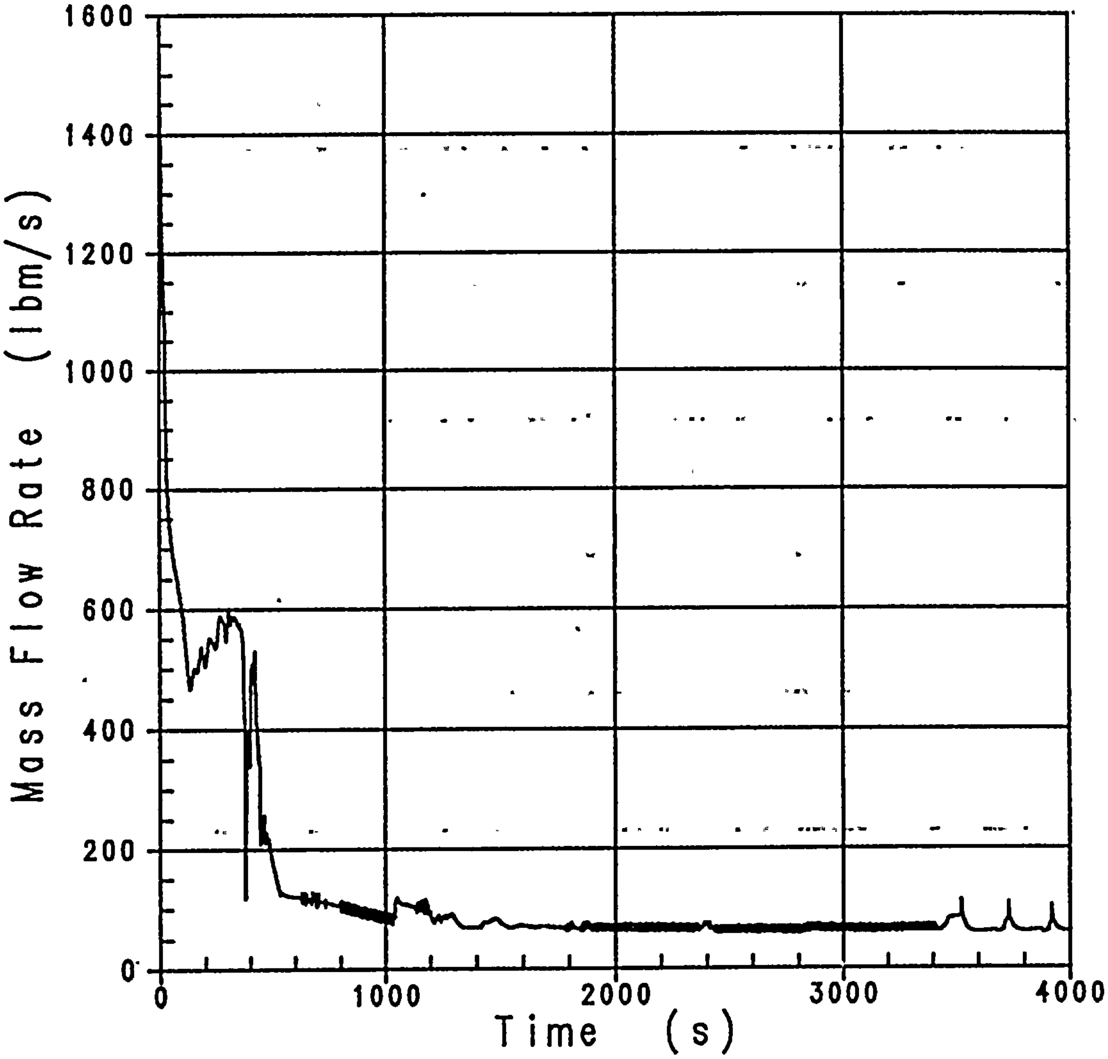

Figure 10: Cold Leg Break Mass Flow, 3-Inch Break, High T_{AVG}

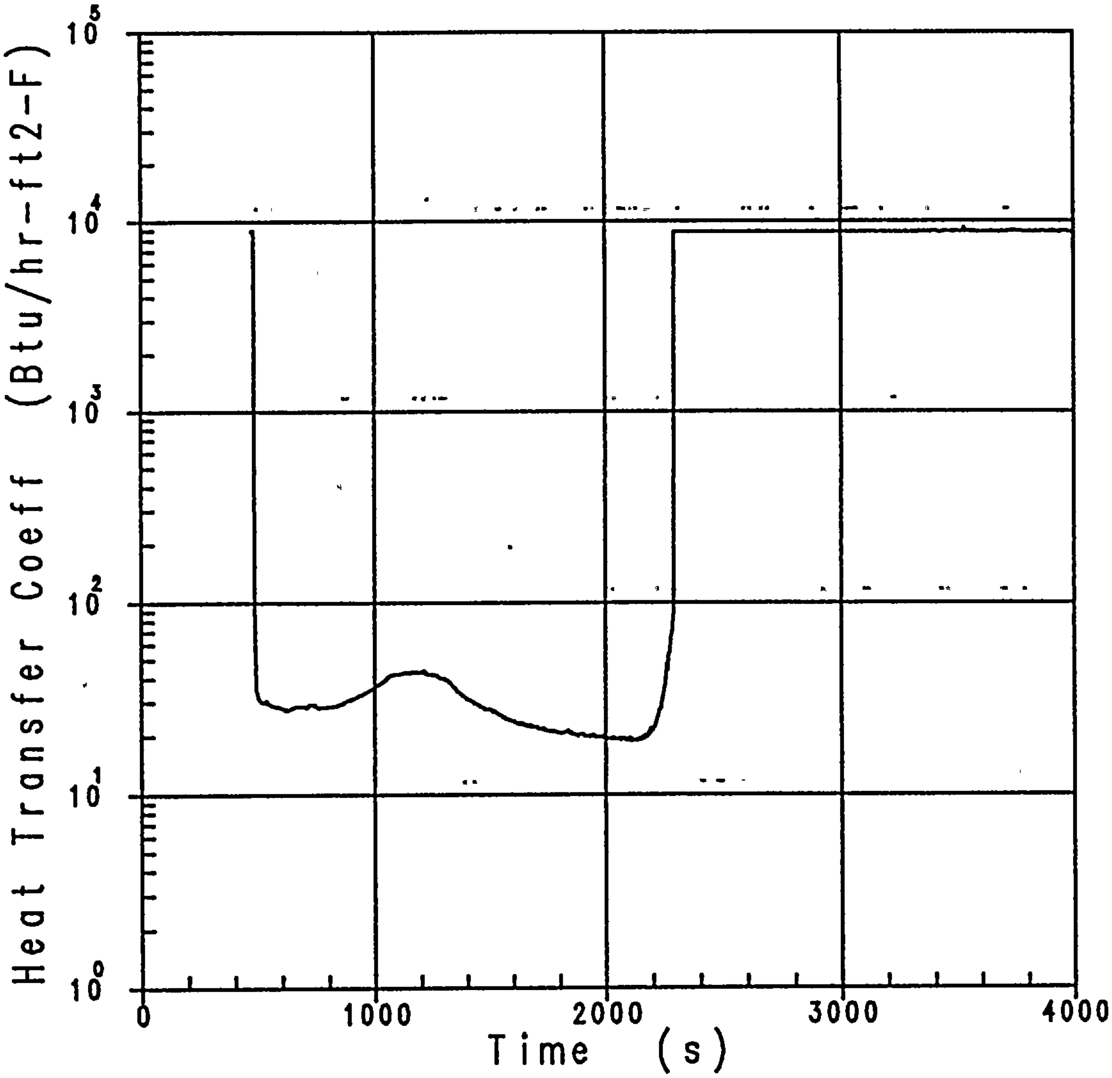

Figure 11: Hot Rod Surface Heat Transfer Coefficient - Hot Spot, 3-Inch Break, High T_{AVG}

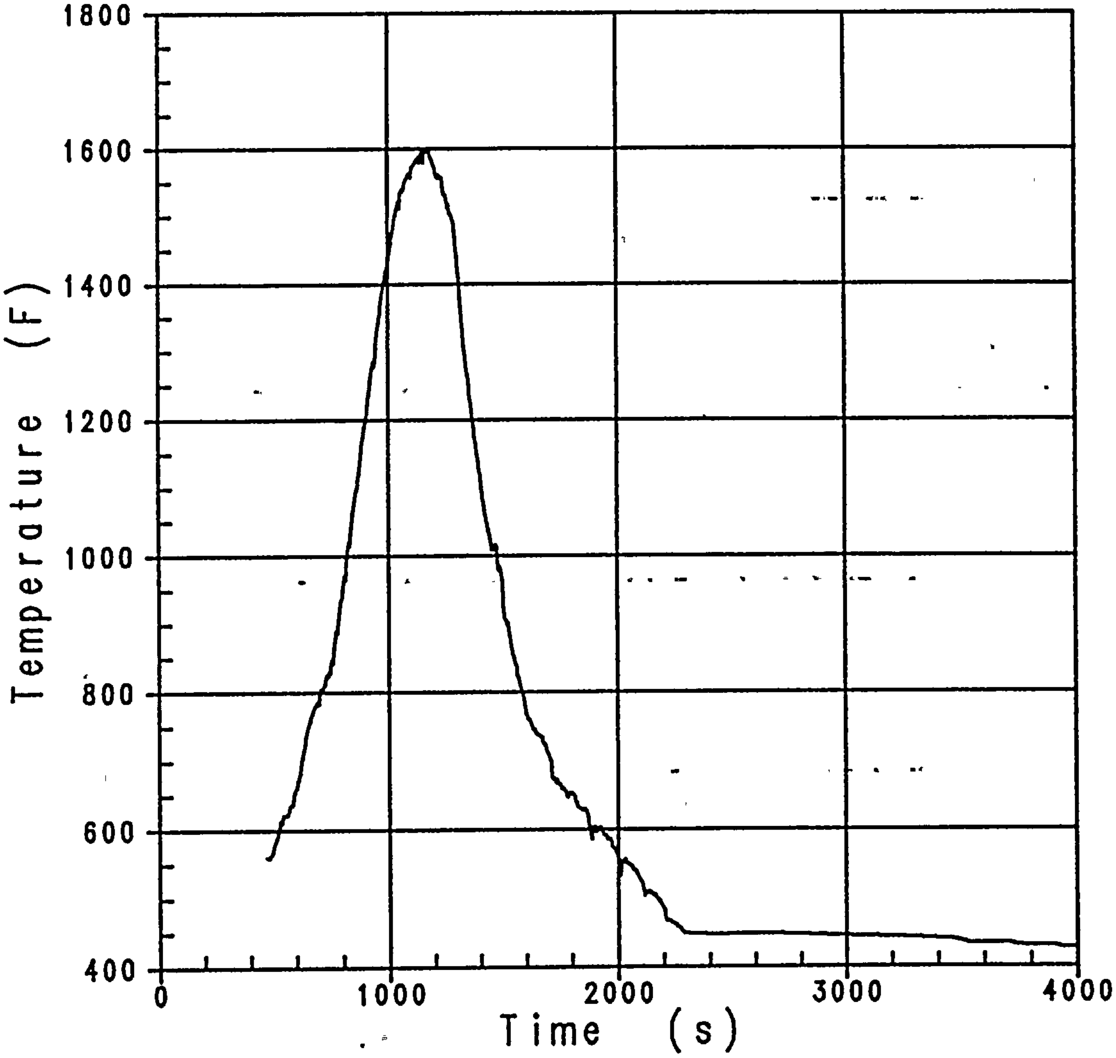

Figure 12: Fluid Temperature - Hot Spot, 3-Inch Break, High T_{AVG}

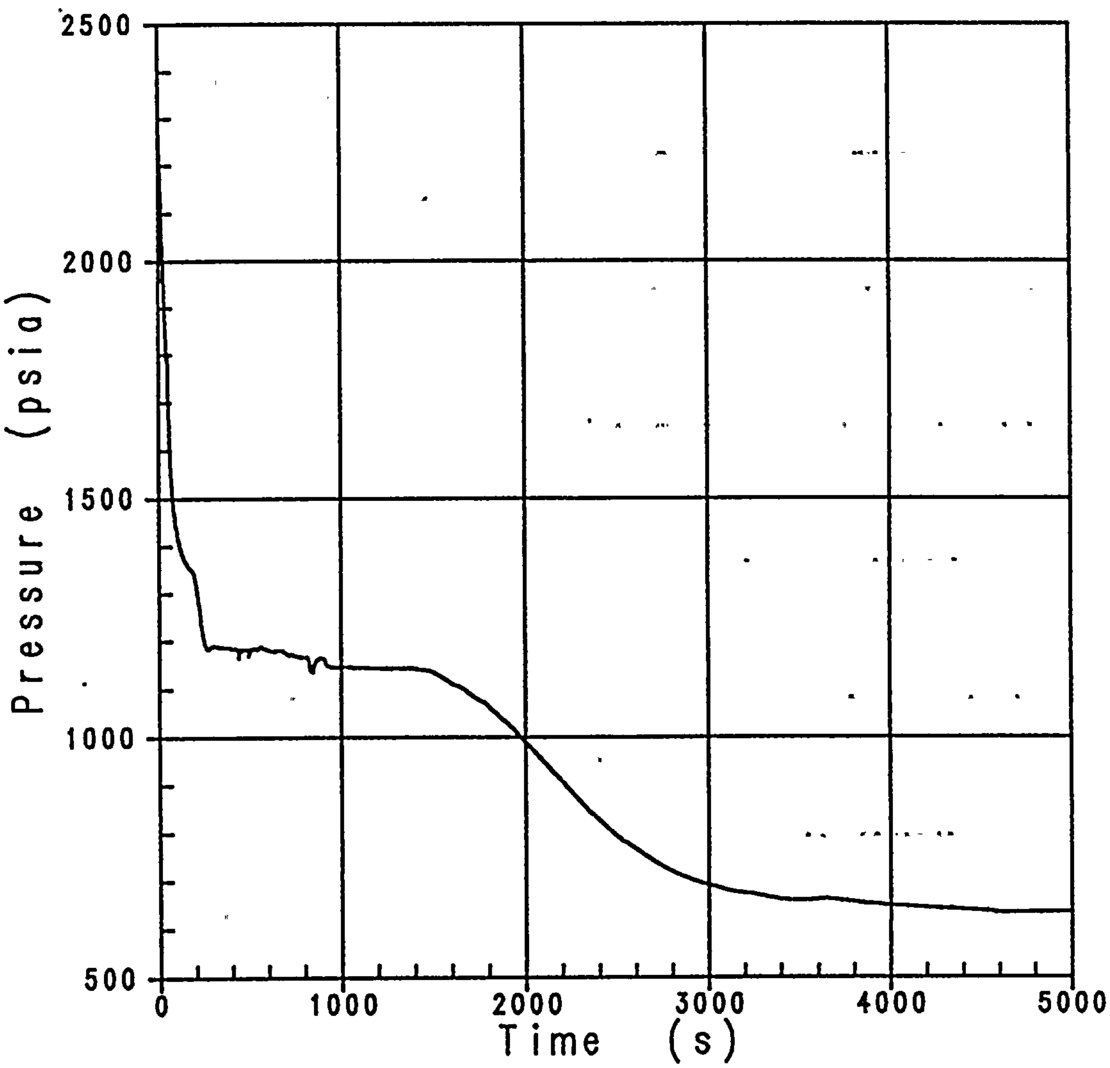

Figure 13: RCS Depressurization Transient, 2-Inch Break, High T_{AVG}

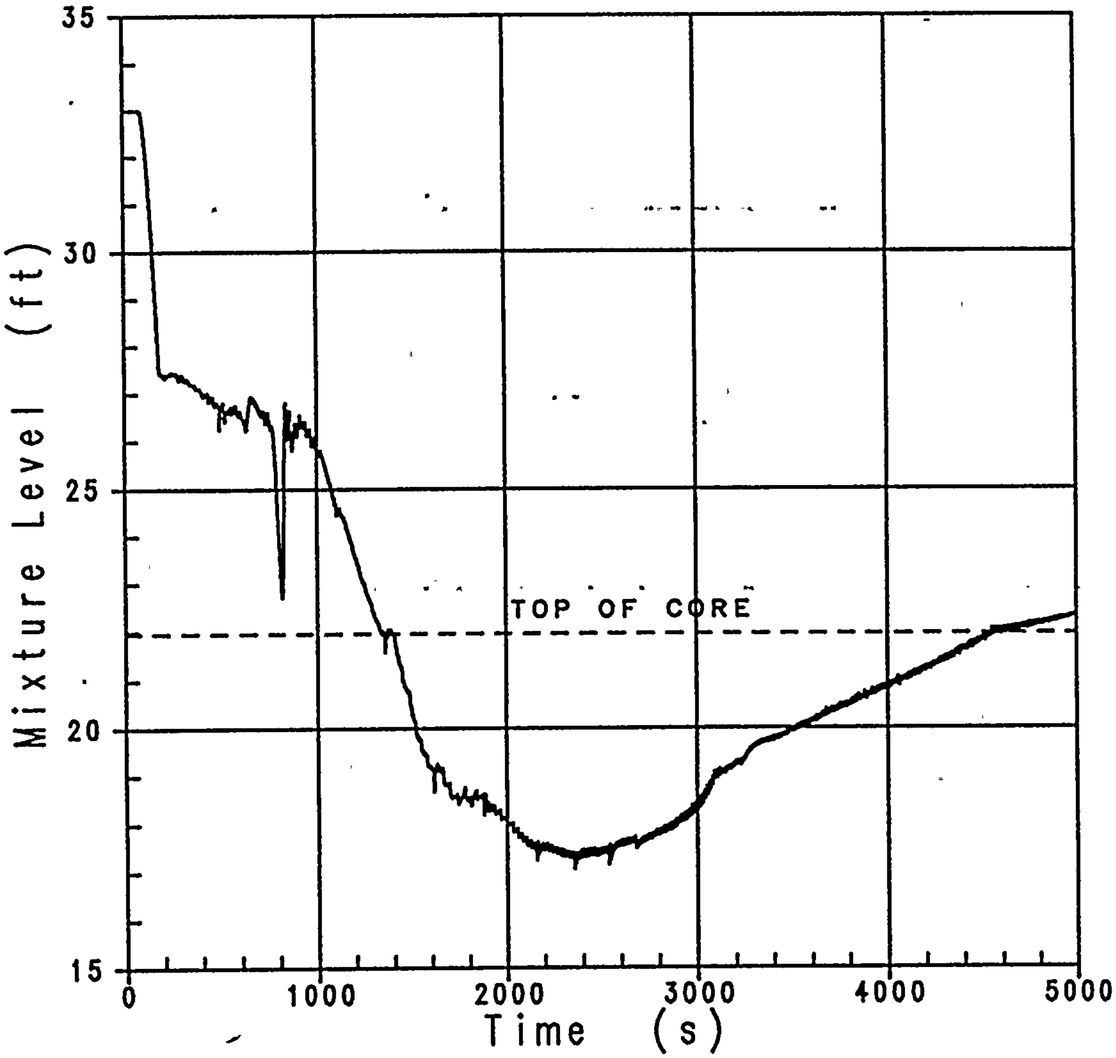

Figure 14: Core Mixture Level, 2-Inch Break, High T_{AVG}

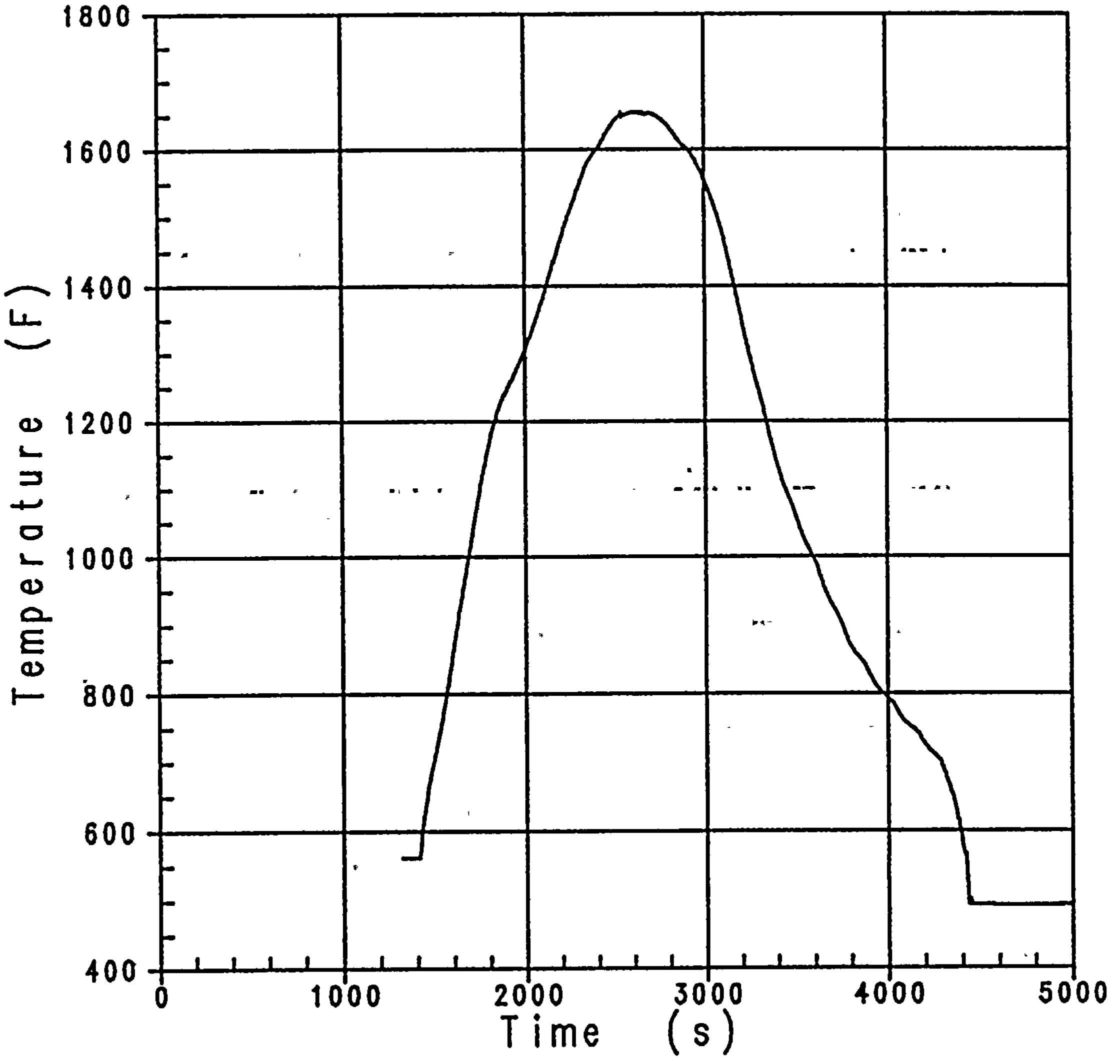

Figure 15: Peak Cladding Temperature - Hot Rod, 2-Inch Break, High T_{AVG}

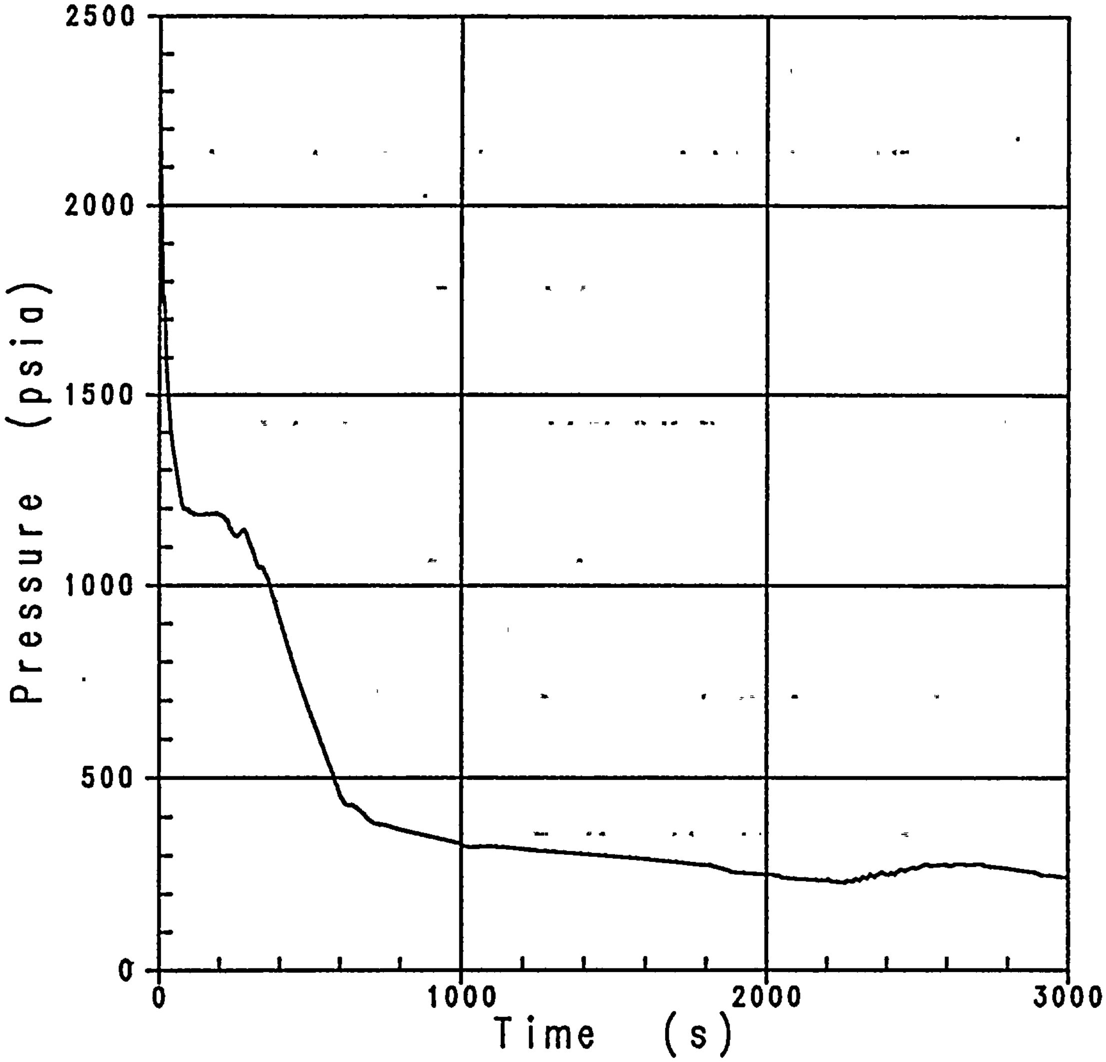

Figure 16: RCS Depressurization Transient, 4-Inch Break, High T_{AVG}

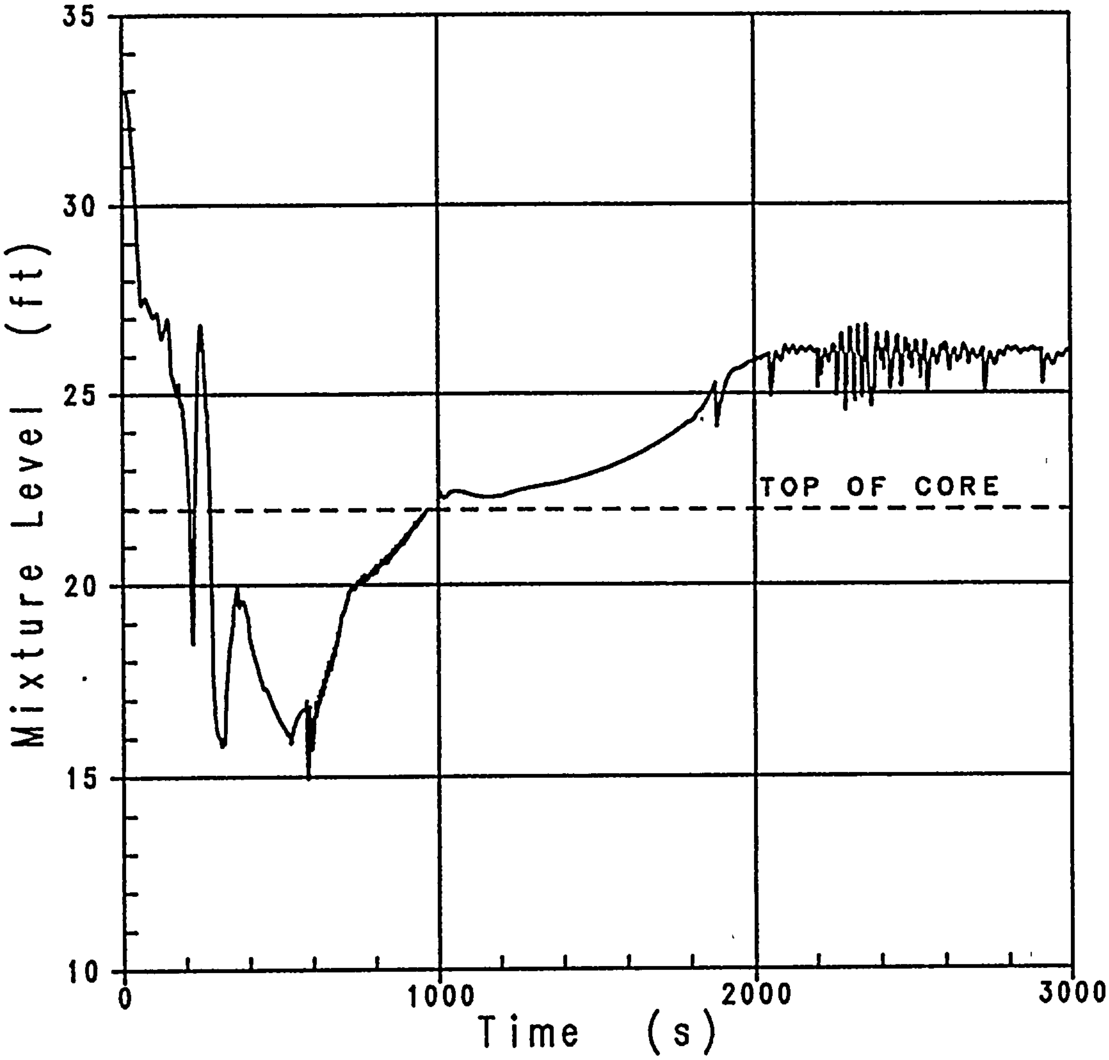

Figure 17: Core Mixture Level, 4-Inch Break, High T_{AVG}

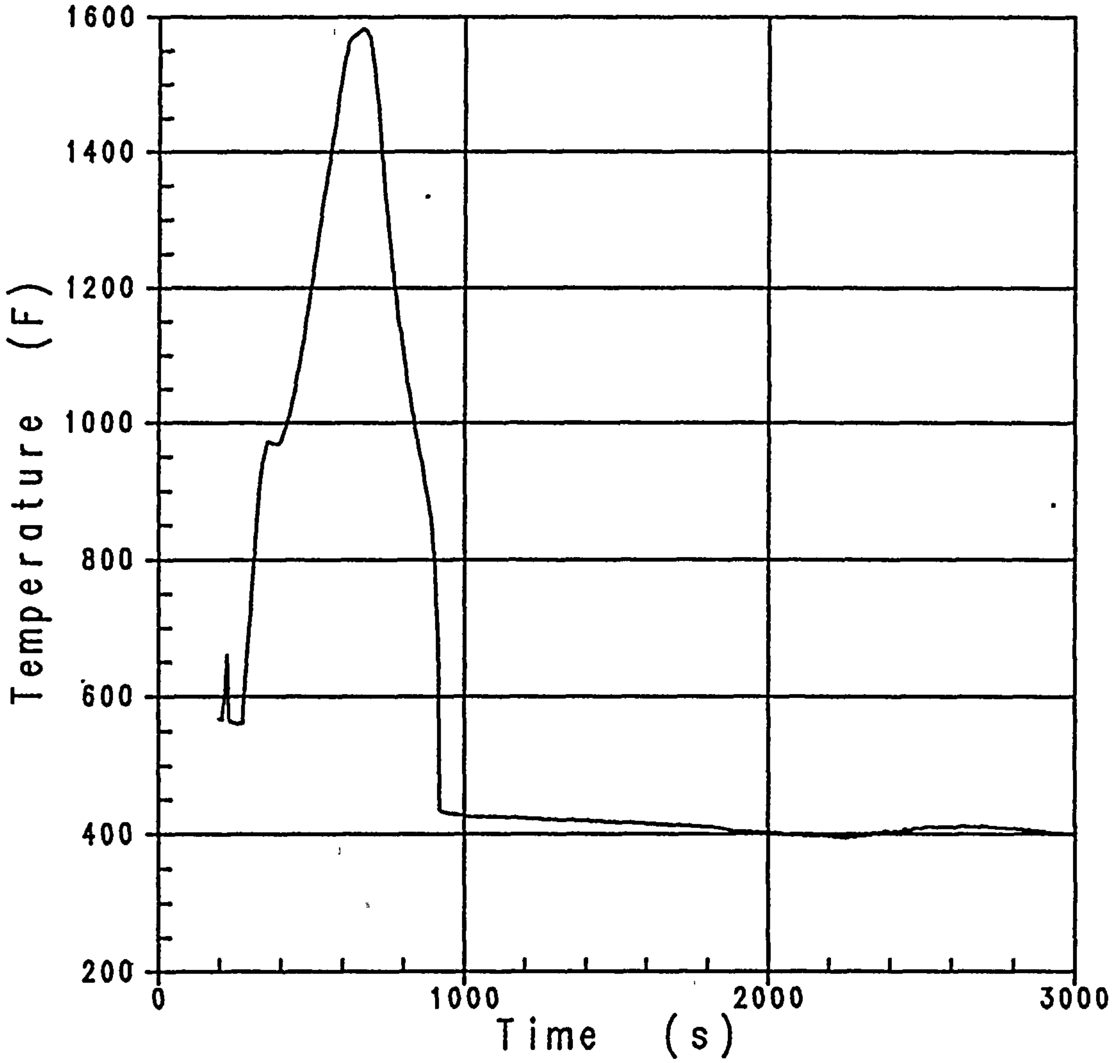

Figure 18: Peak Cladding Temperature - Hot Rod, 4-Inch Break, High T_{AVG}

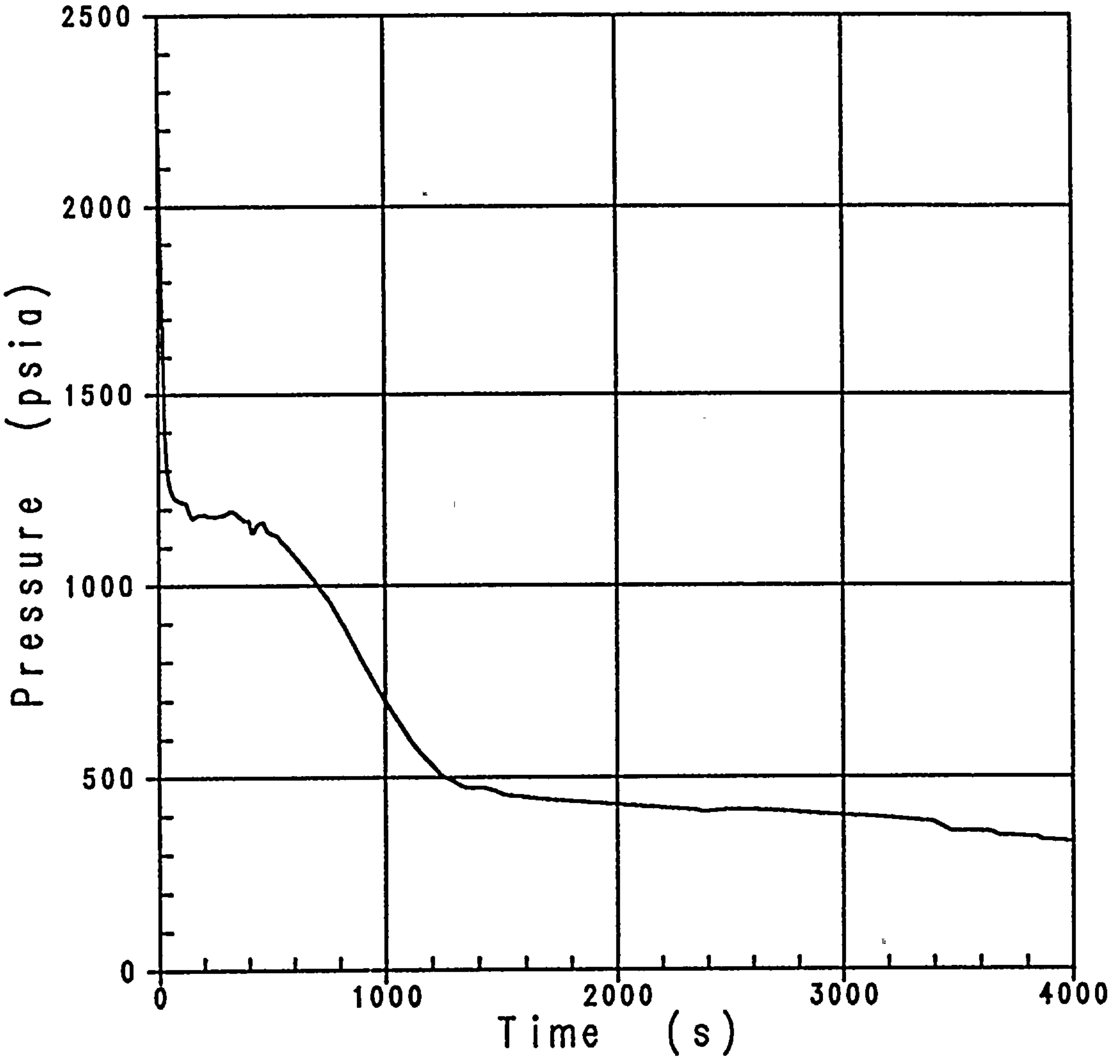

Figure 19: RCS Depressurization Transient, 3-Inch Break, Low T_{AVG}

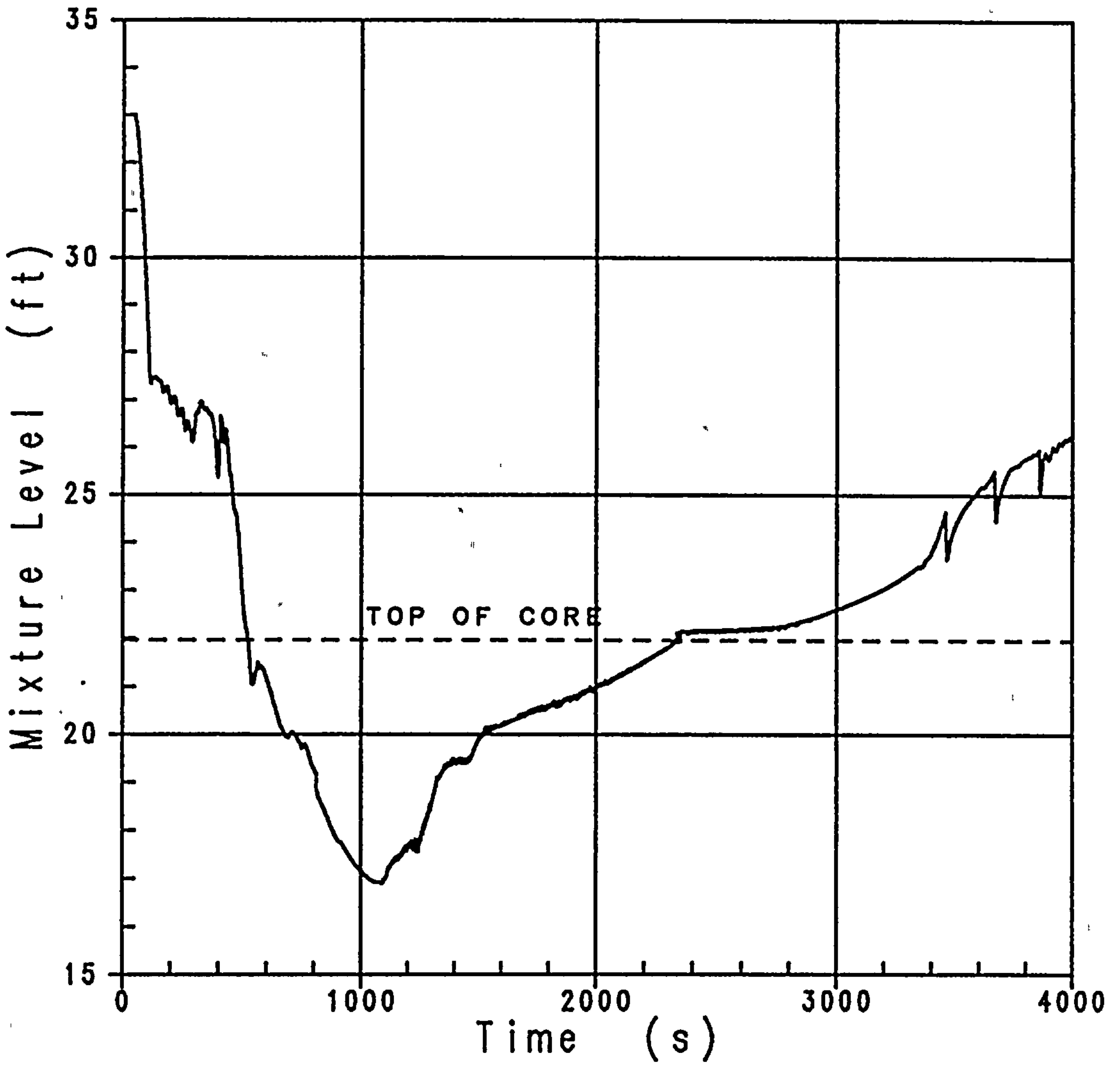

Figure 20: Core Mixture Level, 3-Inch Break, Low T_{AVG}

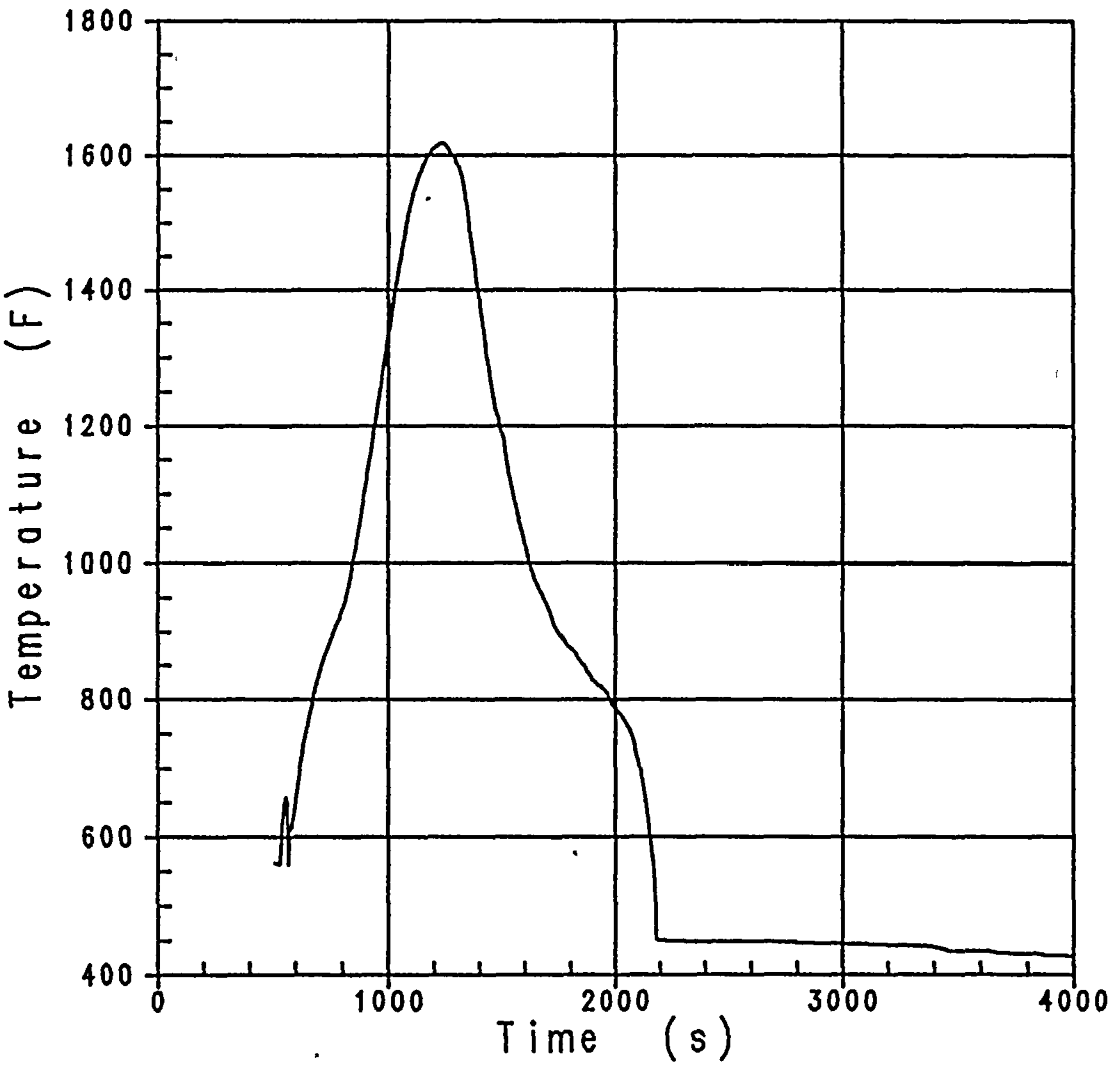

Figure 21: Peak Cladding Temperature - Hot Rod, 3-Inch Break, Low T_{AVG}