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SUBJECT: Informs that SBLOCA analysis redone w/larger peaking factors to support conversion to eighteen month fuel cycle scheduled to begin at Spring 1996 startup.Encl rept summarizing SBLOCA evaluation will be used to update Section 15.6.4.1 of UFSAR.

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ROBERT C. MECREDY
Vice President
Nuclear Operations

June 19, 1995

U.S. Nuclear Regulatory Commission
Document Control Desk
Attn: Allen R. Johnson
Project Directorate I-1
Washington, D.C. 20555

Subject: Small Break Loss of Coolant Accident Analysis
R. E. Ginna Nuclear Power Plant
Docket No. 50-244

Dear Mr. Johnson:

The small break loss of coolant accident (SBLOCA) analysis has been redone with larger peaking factors to support conversion to an eighteen month fuel cycle scheduled to begin at the Spring 1996 Startup.

The attached report summarizes the SBLOCA evaluation and will be used to update section 15.6.4.1 of the Ginna UFSAR.

It is requested that this evaluation become the analysis of record with the startup of Cycle 26, currently scheduled for June 5, 1996.

Very truly yours,

Thomas A. Marlow for
Robert C. Mecredy

Attachment
RWE\383

xc: Mr. Allen R. Johnson (Mail Stop 14B2)
Project Directorate I-1
Washington, D.C. 20555

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US NRC Ginna Senior Resident Inspector

15.6.4.1 LOSS OF REACTOR COOLANT FROM SMALL RUPTURED PIPES OR FROM CRACKS IN LARGE PIPES, WHICH ACTUATES EMERGENCY CORE COOLING SYSTEM

15.6.4.1.1 Identification of Causes and Accident Description

A loss-of-coolant accident is defined as a rupture of the reactor coolant system piping or of any line connected to the system. Ruptures of small cross section will cause expulsion of the coolant at a rate that can be accommodated by the charging pumps, which would maintain an operational water level in the pressurizer permitting the operator to execute an orderly shutdown. Since no clad failure would occur, the coolant that would be released to the containment contains only the fission products existing in it.

The maximum break size for which the normal makeup system can maintain the pressurizer level is obtained by comparing the calculated flow from the reactor coolant system through the postulated break against the charging pump makeup flow at normal reactor coolant system pressure -- i.e., 2,280 psia. A makeup flow rate from one charging pump is typically adequate to sustain pressurizer level at 2,280 psia for a break through a 0.125 inch diameter hole. This break results in a loss of approximately 15 gallons per minute.

Should a larger break occur, depressurization of the reactor coolant system causes fluid to flow to the reactor coolant system from the pressurizer, resulting in a pressure and level decrease in the pressurizer. Reactor trip occurs when the reactor trip low pressurizer pressure setpoint is reached. The safety injection system is actuated when the appropriate safety injection low pressurizer pressure setpoint is reached. The consequences of the accident are limited in two ways:

1. Reactor trip and borated water injection complement void formation in causing rapid reduction of nuclear power to a residual level that corresponds to the delayed fission and fission product decay.

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

Before the break occurs, the plant is in an equilibrium condition; i.e., the heat generated in the core is being removed via the secondary system. Reactor trip and Reactor Coolant Pump (RCP) trip is assumed

coincident with Loss of Offsite Power (LOOP). During blowdown, heat from decay, hot internals, and the vessel continues to be transferred to the reactor coolant system. Heat transfer between the reactor coolant system and the secondary system may be in either direction depending on the relative temperatures. In the case of continued heat addition to the secondary system, pressure increases and steam dump may occur. Makeup to the secondary side is automatically provided by the auxiliary feedwater pumps. The safety injection signal stops normal feedwater flow by closing the main feedwater line isolation valves, and initiates emergency feedwater flow by starting auxiliary feedwater pumps. The secondary flow aids in the reduction of reactor coolant system pressure. When the reactor coolant system depressurizes to 715 psia, the accumulators begin to inject water into the reactor coolant loops. The reactor coolant pumps are assumed to be tripped at the initialization of the accident, and the effects of pump coastdown are included in the blowdown analysis.

15.6.4.1.2 Analysis of Effects and Consequences

15.6.4.1.2.1 Method of Analysis

For loss-of-coolant (LOCA) accidents less than 1.0 ft^2, LOCA analyses were performed with the Westinghouse 1985 small break LOCA ECCS NOTRUMP Evaluation Model[1,2,3]. This evaluation model consists of the NOTRUMP code[1,2], which predicts the transient depressurization of the reactor coolant system and the mass and enthalpy of the flow through the break, and the LOCTA code[3], which predicts the cladding heatup response during the LOCA.

15.6.4.1.2.2 Small Break LOCA Analysis Using NOTRUMP

The NOTRUMP Small Break LOCA Evaluation Model[1,2,3] was developed by Westinghouse to provide a better estimate calculation of small break LOCA behavior in a Westinghouse designed NSSS while complying with the requirements of 10 CFR 50.46 and Appendix K[4] of 10 CFR 50. In order to provide a better estimate calculation of the Reactor Coolant System's (RCS) response to a small break LOCA, the NOTRUMP Small Break LOCA EM employed better estimate models and modeling assumptions, whenever possible and wherever allowed by Appendix K[4].

The NOTRUMP computer code is a state-of-the-art one-dimensional general network code consisting of a number of advanced features. Among these features are the 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, and regime-dependent heat transfer correlations.

In NOTRUMP, the reactor coolant system is nodalized into volumes interconnected by flowpaths. The broken loop is modelled explicitly, with the intact loops lumped into a second loop for plants with more than 2 loops. The transient behavior of the system is determined from the governing conservation equations of mass, energy, and momentum applied throughout the system. The use of NOTRUMP in the analysis involves, among other things, the representation of the reactor core as heated control volumes with an associated bubble rise model to permit a transient mixture height calculation. The multinode capability of the program enables an explicit and detailed spatial representation of various system components. In particular, it enables a proper calculation of the behavior of the loop seal during a loss-of-coolant accident.

As detailed in Reference 10, the loop seal clearing process has a significant effect on the RCS response for small break LOCAs. In three and four loop plant models, the intact loops are lumped together into a single loop. As a result, it is necessary to restrict loop seal clearing to the broken loop only for three and four loop plants to prevent unrealistic loop seal clearing behavior. For R. E. Ginna, which is a two loop plant, there is no lumped intact loop, but it was conservative to leave the restriction on the intact loop seal in place. For breaks above the critical break size, approximately 6 inches in diameter and larger, in which loop seal clearing occurs in more than one loop, the restriction on the intact loop seal is removed after the broken loop seal clears. A detailed description of the NOTRUMP code is provided in References 1 and 2.

The small break LOCA analysis assumed nominal accumulator water volume (1140 ft^3) with a minimum cover gas pressure of 715 psia. Minimum ECCS availability is assumed for the analysis, and pumped ECCS water is conservatively assumed to be at the maximum RWST temperature. Power is assumed to be at 102% of maximum licensed core power. Those inputs important to small break LOCA were selected in accordance with sensitivities calculated for small break LOCA using NOTRUMP[(10)].

Included in the R. E. Ginna small break LOCA NOTRUMP analysis was a T_{AVE} operating temperature window from 559°F to 573.5°F. High T_{AVE} conditions were used to determine the limiting break size, then the sensitivity to T_{AVE} was determined by running the limiting high T_{AVE} break at low T_{AVE} conditions.

High T_{AVE} was determined to be the limiting conditon.

The most limiting single active failure for small break LOCA has been determined to be the failure of a diesel generator to start for the limiting power availability condition, Loss-Of-Offsite-Power (LOOP). This scenario results in the loss of one High Head Safety Injection pump (HHSI), one low head safety injection pump (the Residual Heat Removal (RHR) pump), and one motor driven auxiliary feedwater (MDAFW) pump. However, since the R. E. Ginna ECCS configuration contains a "swing pump" (a HHSI pump that automatically aligns to the affected cold leg whenever there is loss of the normal HHSI pump to that affected loop) there is always one HHSI pump connected to a cold leg. This analysis assumes two HHSI pumps, each connected to a cold leg.

The R. E. Ginna emergency core cooling system is designed such that the RHR pumps inject directly to the Reactor Vessel in the upper plenum region during the cold leg injection phase. The effect of flow from the RHR pumps is generally not important in the small break LOCA analyses since the shutoff head is lower than the RCS pressure during the typical time portion of the transient considered here. Therefore, modeling upper plenum injection of the RHR flows is not usually important for small break LOCA, and is usually not modelled in NOTRUMP. In the case of R. E. Ginna, the NOTRUMP model did not include any upper plenum injection RHR flow.

Westinghouse LOCA methodology conservatively assumes that one of the injection lines is spilling during a LOCA. The safety injection flows used in LOCA analysis are calculated assuming that the branch line with least resistance (Loop B) is spilling rather than injecting to the reactor coolant system. For small breaks that are equal to or larger than the injection line diameter, one injection line is assumed to spill to containment backpressure (assumed to be 0 psig). In contrast, for small breaks that are smaller than the injection line diameter, one injection line is assumed to spill to RCS backpressure.

The HHSI pumps for R. E. Ginna inject through lines which have an 8.75 inch inner diameter where they join the cold leg. For breaks smaller than this diameter, the SI flows would be assumed to spill to RCS pressure if the break is postulated to be in the SI line or the cold leg. For breaks larger than this diameter, the SI flow is assumed to spill to containment pressure, conservatively assumed to be 0 psig. Since the break spectrum analyzed was between 3 and 6 inches equivalent diameter, the spilling branch lines for all HHSI pumps would normally be assumed to spill to RCS backpressure. However, because the HHSI pumps are not headered the spilling assumption has no effect on delivered flow.

The pumped safety injection characteristics as a function of RCS pressure used in the analysis are provided in Table 15.6.4.1-1. This characteristic assumes five percent pump degradation. Credit for safety injection flow delivery to the RCS was delayed 30 seconds after the safety injection signal was generated at the low-low pressurizer pressure setpoint. Additional important assumptions used in the analysis are contained in Table 15.6.4.1-2 and Table 15.6.4.1-3.

Peak cladding temperature calculations are performed with the LOCTA-IV[3] code using the NOTRUMP calculated core pressure, fuel rod power history, uncovered core steam flow and mixture heights as boundary conditions. While NOTRUMP models an average core rod, LOCTA-IV models the hot rod and the average hot assembly rod. The typical small break LOCA power shape represents a distribution with power concentrated in the upper regions of the core. Such a distribution is limiting for a small break LOCA because it minimizes coolant level swell, while maximizing vapor superheating and fuel rod heat generation at the uncovered elevations. The small break LOCA analysis assumes the core continues to operate at full rated power until the control rods are completely inserted. The fuel rod initial conditions, which include temperature, rod internal pressure and fission gas composition are calculated in accordance with the Westinghouse Improved Fuel Performance Models[5].

15.6.4.1.3 <u>Results</u>

15.6.4.1.3.1 Reactor Coolant System Pipe Breaks

To identify the limiting break size, a spectrum of breaks ranging from 3 to 6 inches in diameter were analyzed. A single parameter sensitivity to T_{AVE} with temperatures ranging from 559° to 573.5°F was performed to identify the limiting RCS operating condition for a T_{AVE} window. The thermohydraulic transient results for the breaks in the spectrum analyzed are summarized in Table 15.6.4.1-4, while the rod heat up results for those transients are summarized in Table 15.6.4.1-5. This section presents the results of the most limiting break size in terms of the highest peak cladding temperature. The worst break (small break) is the 4-inch equivalent diameter high T_{AVE} break. The depressurization transient for this break is shown in Figure 15.6.4.1-1. The vessel mixture level, showing the extent of core uncovery, is shown in Figure 15.6.4.1-2.

The peak cladding temperature of 1308.°F occurred at 310. seconds. The cladding temperature transient is shown in Figure 15.6.4.1-3 for the break with the highest cladding temperature. The core steam flow

rate is shown in Figure 15.6.4.1-4. The core heat transfer coefficients for the core uncovery time are shown in Figure 15.6.4.1-5. The hot spot fluid temperature for this worst break is shown in Figure 15.6.4.1-6.

Figure 15.6.4.1-7 presents the core axial power shape utilized to perform the small break analysis. This power shape was chosen because it provides a limiting distribution of power versus core height while maximizing local power in the upper regions of the reactor core.

15.6.4.1.3.2 Additional Break Sizes

Additional break sizes were analyzed to identify the limiting break size. Figures 15.6.4.1-8, 15.6.4.1-9, and 15.6.4.1-10 present the reactor coolant system pressure transient for the 3-inch, 6-inch, and 4-inch low T_{AVE} breaks respectively, and Figures 15.6.4.1-11, 15.6.4.1-12, and 15.6.4.1-13 present the core mixture level plots for those same three breaks. The peak cladding temperatures for the 3-inch, 6-inch, and 4-inch low T_{AVE} breaks were lower than the 4-inch high T_{AVE} break. The peak cladding temperature transients for the 3-inch, 6-inch, and 4-inch low T_{AVE} breaks are given in Figures 15.6.4.1-14, 15.6.4.1-15 and 15.6.4.1-16.

15.6.4.1.4 <u>Conclusions</u>

Analyses presented in this section show that the high head portion of the emergency core cooling system, together with accumulators, provides sufficient core flooding to keep the calculated peak cladding temperatures below the required limits of 10 CFR 50.46[4]. Hence, adequate protection is afforded by the emergency core cooling system in the event of a small break, loss-of-coolant accident.

15.6.4.7 REFERENCES

1. Meyer, P. E., et. al., "NOTRUMP - A Nodal Transient Small Break and General Network Code," WCAP-10079-P-A (Proprietary) and WCAP-10080-NP-A (Non-Proprietary), August 1985.

2. 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.

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

4. 10 CFR 50.46 and 10 CFR 50 Appendix K, "Acceptance Criteria for Emergency Core Cooling Systems for Water Cooled Nuclear Power Reactors" Federal Register, Volume 39, Number 3, January 1974, as amended in Federal Register, Volume 53, September 1988.

5. Weiner, R. A., et al., "Improved Fuel Performance Models for Westinghouse Fuel Rod Design and Safety Evaluations," WCAP-10581-P-A (Proprietary) and WCAP-11873-A (Non-Proprietary), August 1988.

6. Burnett, T. W. T., et al., "LOFTRAN Code Description," WCAP-7907-P-A (Proprietary), and WCAP-7907-A (Non-Proprietary), April 1984.

7. Hargrove, H. G., "FACTRAN, a FORTRAN-IV Code for Thermal Transients in a UO_2 Fuel Rod," WCAP-7908-A, December 1989.

8. Hochreiter, L. E., Chelemer, H., and Chu, P. T., "THINC-IV An Improved Program for Thermal-Hydraulic Analysis of Rod Bundle Cores," WCAP-7956, June 1973.

9. USNRC Regulatory Guide 1.24, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Pressurized Water Reactor Radioactive Gas Storage Tank Failure," March 3, 1973.

10. 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-A (Non-Proprietary), October 1986.

TABLE 15.6.4.1-1

TOTAL DELIVERED SMALL BREAK LOCA SAFETY INJECTION FLOW

RCS Pressure (psia)	HHSI Delivered Flow per Loop (lb/sec)
14.7	53.277
114.7	51.047
214.7	48.815
314.7	46.582
414.7	44.598
514.7	42.362
614.7	40.125
714.7	37.887
814.7	35.147
914.7	31.849
1014.7	27.909
1114.7	23.174
1214.7	17.407
1314.7	8.630
1389.7	0.000

Table 15.6.4.1-2

SMALL BREAK LOCA KEY ASSUMPTIONS

Core Power (MWt)		102% of 1520
Total Peaking Factor, F_Q		2.5
Hot Channel Enthalpy Rise Factor, $F_{\Delta H}$		1.75
Maximum Assembly Average Power, $\bar{P}_{HA}$		1.55
Axial Offset (%)		+25
Fuel Assembly Array		14x14 OFA
Accumulator Water Volume (ft^3)		1140.
Min Accumulator Gas Pressure (psia), (w/ uncertainties)		715.
Thermal Design Flow per Loop (gpm)		85100.
Vessel Inlet Temperature (°F)	T_{AVE}=573.5°F T_{AVE}=559°F	542.44 527.42
Vessel Outlet Temperature (°F)	T_{AVE}=573.5°F T_{AVE}=559°F	604.56 590.58
RCS Pressure (psia), (with uncertainties)		2280.
Steam Generator Tube Plugging Level (%), maximum		15.
Reactor Trip Setpoint (psia) (Safety Analysis)		1730.
SI Signal Setpoint (psia) (Safety Analysis)		1715.
Reactor Trip Delay Time (sec)		5.0[1]
Safety Injection Flow (lbs/sec)		Table 15.6.4.1-1
Main Steam Safety Valve Data		Table 15.6.4.1-3
Safety Injection Delay Time (sec)		30.
Feedwater Trip Processing Delay Time After SI Signal (sec)		5.0
Time for Main Feedwater Flow Coastdown After SI Signal (sec)		10.0
Auxiliary Feedwater Pump Start Delay Time (sec)		50.

[1] This value assumes a 2.0 second Reactor Trip Signal Processing Delay Time, a 1.8 second rod drop time which corresponds to the Technical Specification value, and an additional 1.2 second delay for added conservatism.

Table 15.6.4.1-3

SMALL BREAK LOCA MAIN STEAM SAFETY VALVE ASSUMPTIONS

MAIN STEAM SAFETY VALVE DATA	VALUE	UNITS
Number of Safety Valves per Steam Generator	4	-
Set Pressure for Valve 1	1085.	psig
Set Pressure for Valve 2	1140.	psig
Set Pressure for Valve 3	1140.	psig
Set Pressure for Valve 4	1140.	psig
Percent Uncertainty for Valves 1-4	1.0	%
Percent Accumulation for Valves 1-4	3.0	%
Rated Flow for Valve 1	797700.	lbm/hr
Rated Flow for Valve 2	837600.	lbm/hr
Rated Flow for Valve 3	837600.	lbm/hr
Rated Flow for Valve 4	837600.	lbm/hr

TABLE 15.6.4.1-4

SMALL BREAK LOCA TIME SEQUENCE OF EVENTS

Event	Equivalent Break Diameter			
	4 Inch High T_{AVE}	3 Inch* High T_{AVE}	6 Inch High T_{AVE}	4 Inch Low T_{AVE}
Break Occurs (sec)	0.0	0.0	0.0	0.0
Reactor Trip Signal (sec)	7.45	10.99	5.95	6.62
Safety Injection Signal (sec)	7.72	11.34	6.15	6.77
Safety Injection Begins (sec)	39.22	42.84	37.65	38.27
Loop Seal Venting Begins (sec)	156.48	254.30	42.87	160.19
Top Of Core Uncovered (sec)	200.2	N/A	107.2	206.8
Peak Cladding Temperature Occurs (sec)	309.9	N/A	190.	244.
Accumulator Injection Begins (sec)	339.1	623.	148.9	336.2
Top Of Core Covered (sec)	430.4	N/A	219.4	264.
Hot Rod Burst Occurs (sec)	N/A	N/A	N/A	N/A

*The top of the core did not uncover in the 3 inch high T_{AVE} case, so there is no calculated PCT.

TABLE 15.6.4.1-5

SMALL BREAK LOCA FUEL CLADDING RESULTS

Results	Equivalent Break Diameter			
	4 Inch High T_{AVE}	3 Inch* High T_{AVE}	6 Inch High T_{AVE}	4 Inch Low T_{AVE}
Peak Clad Temperature (°F)	1308.	N/A	956.	912.
Peak Clad Temperature Location (ft)	11.0	N/A	10.75	10.5
Peak Clad Temperature Time (sec)	310.	N/A	190.	244.
Local Zr/H_2O Reaction (Maximum %)	0.074	N/A	0.003	0.002
Local Zr/H_2O Reaction Location (ft)	11.0	N/A	10.75	10.75
Total Zr/H_2O Reaction (%)	< 1.0	N/A	< 1.0	< 1.0
Hot Rod Burst Time (sec)	N/A	N/A	N/A	N/A
Hot Rod Burst Location (ft)	N/A	N/A	N/A	N/A

* The three inch high T_{AVE} break did not uncover the core, therefore the PCT is reported as N/A

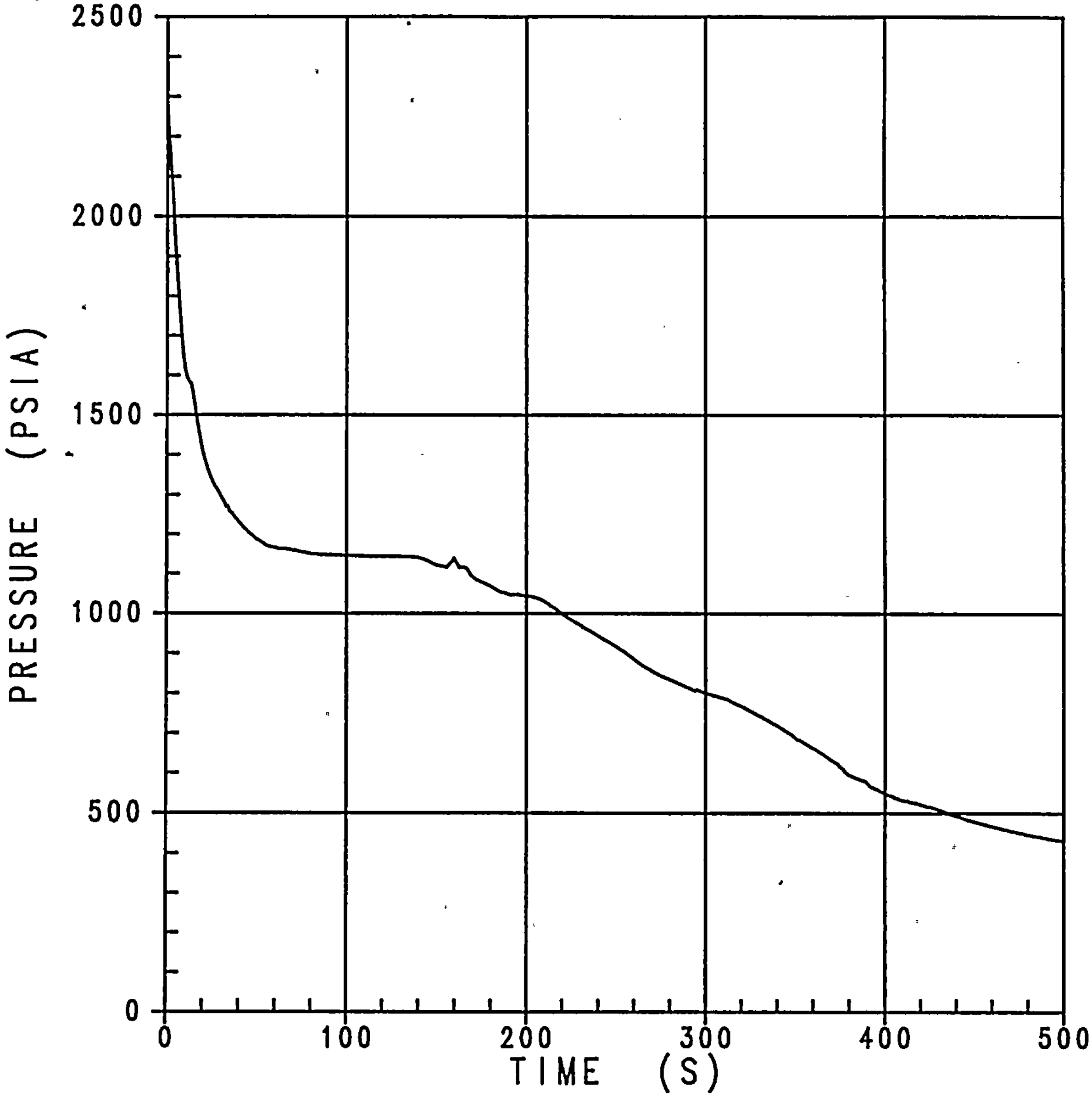

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Figure 15.6.4.1-1
4 Inch High T_{AVE} Break
Reactor Coolant System Pressure

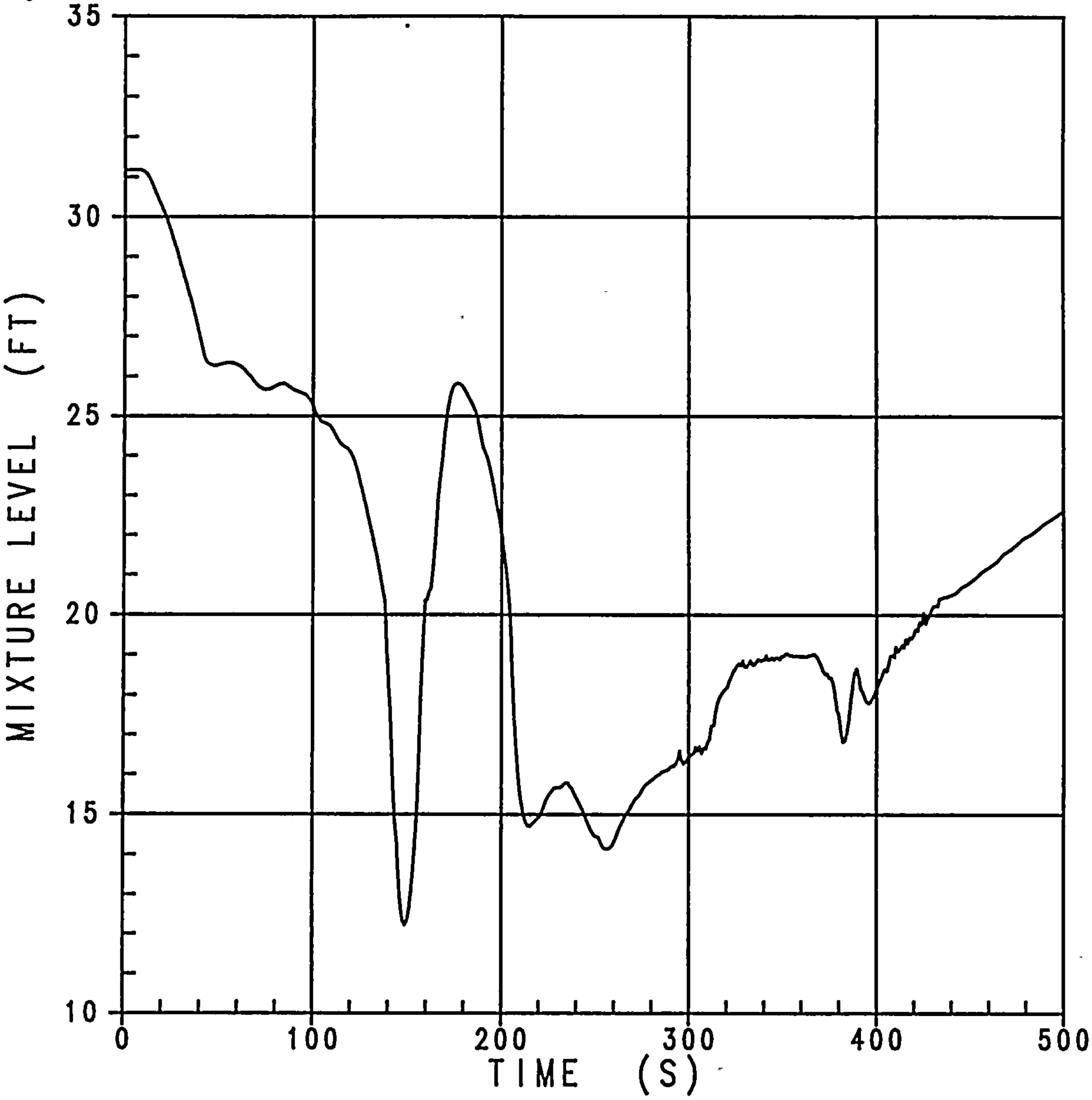

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Figure 15.6.4.1-2
4 Inch High T_{AVE} Break
Core Mixture Level

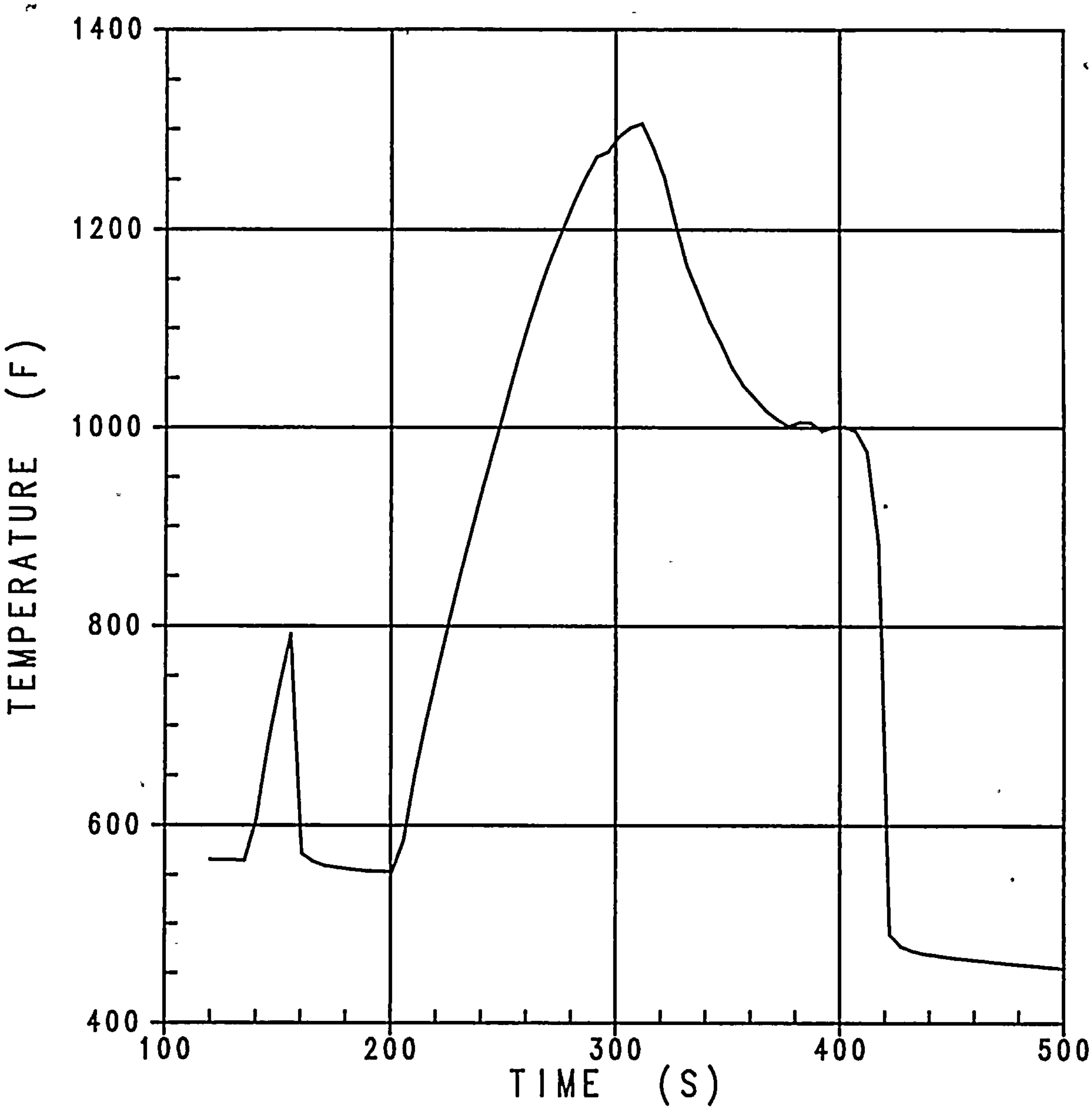

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Figure 15.6.4.1-3
4 Inch High T_{AVE} Break
Peak Cladding Temperature (11.0 Feet)

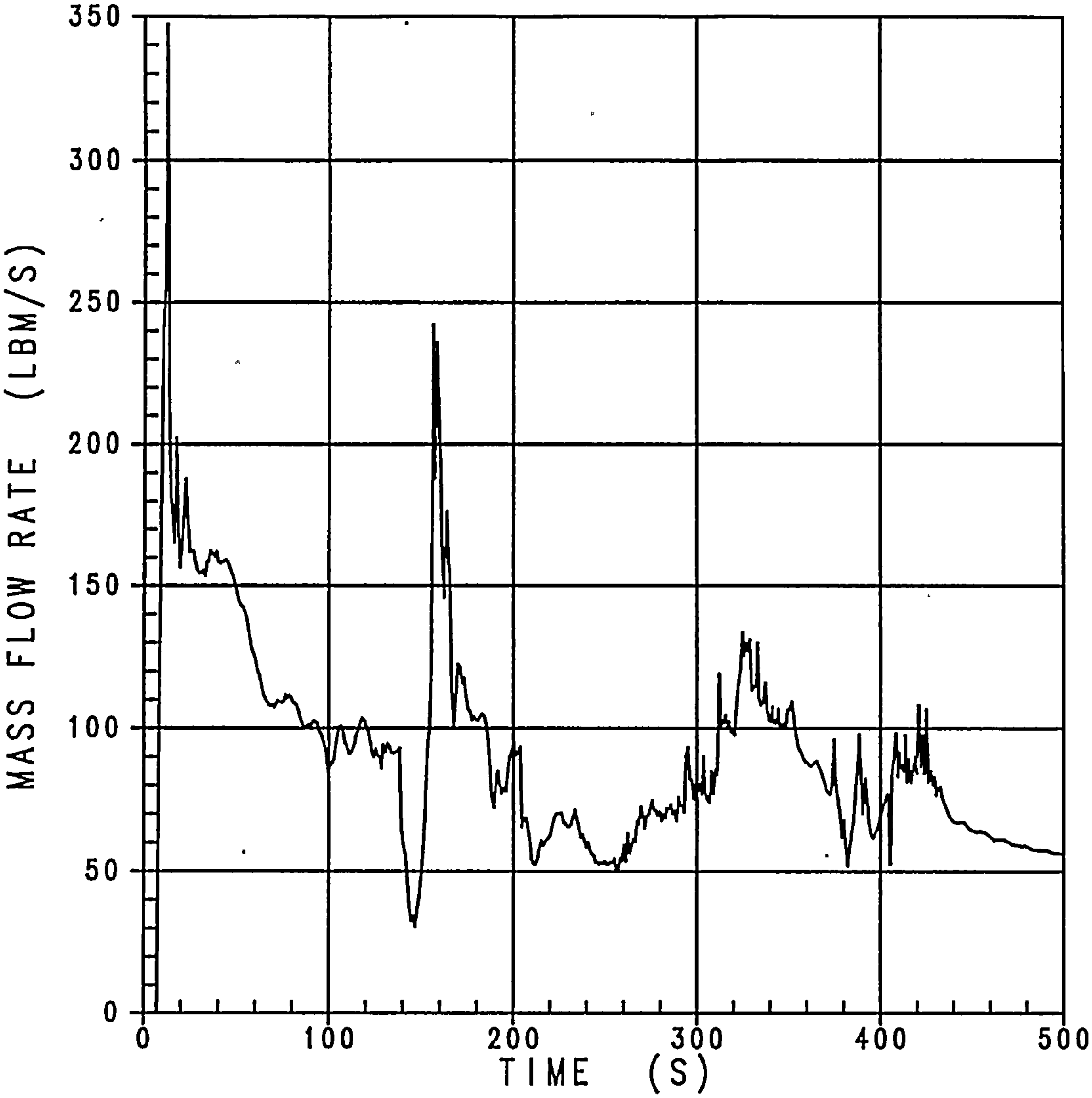

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Figure 15.6.4.1-4
4 Inch High T_{AVE} Break
Core Exit Vapor Mass Flow Rate

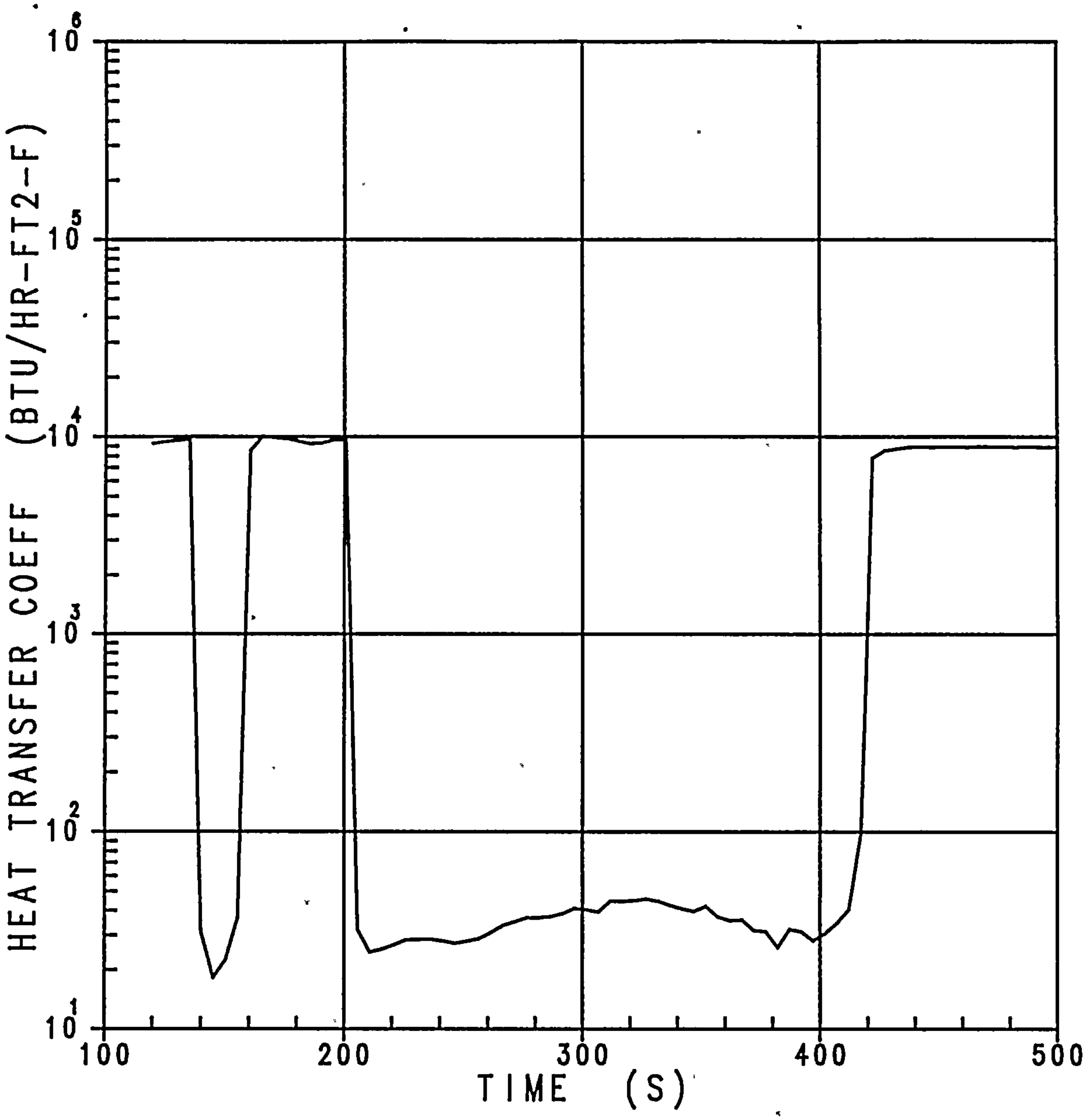

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Figure 15.6.4.1-5
4 Inch High T_{AVE} Break
Clad Hot Spot Heat Transfer Coefficient
(11.0 Feet)

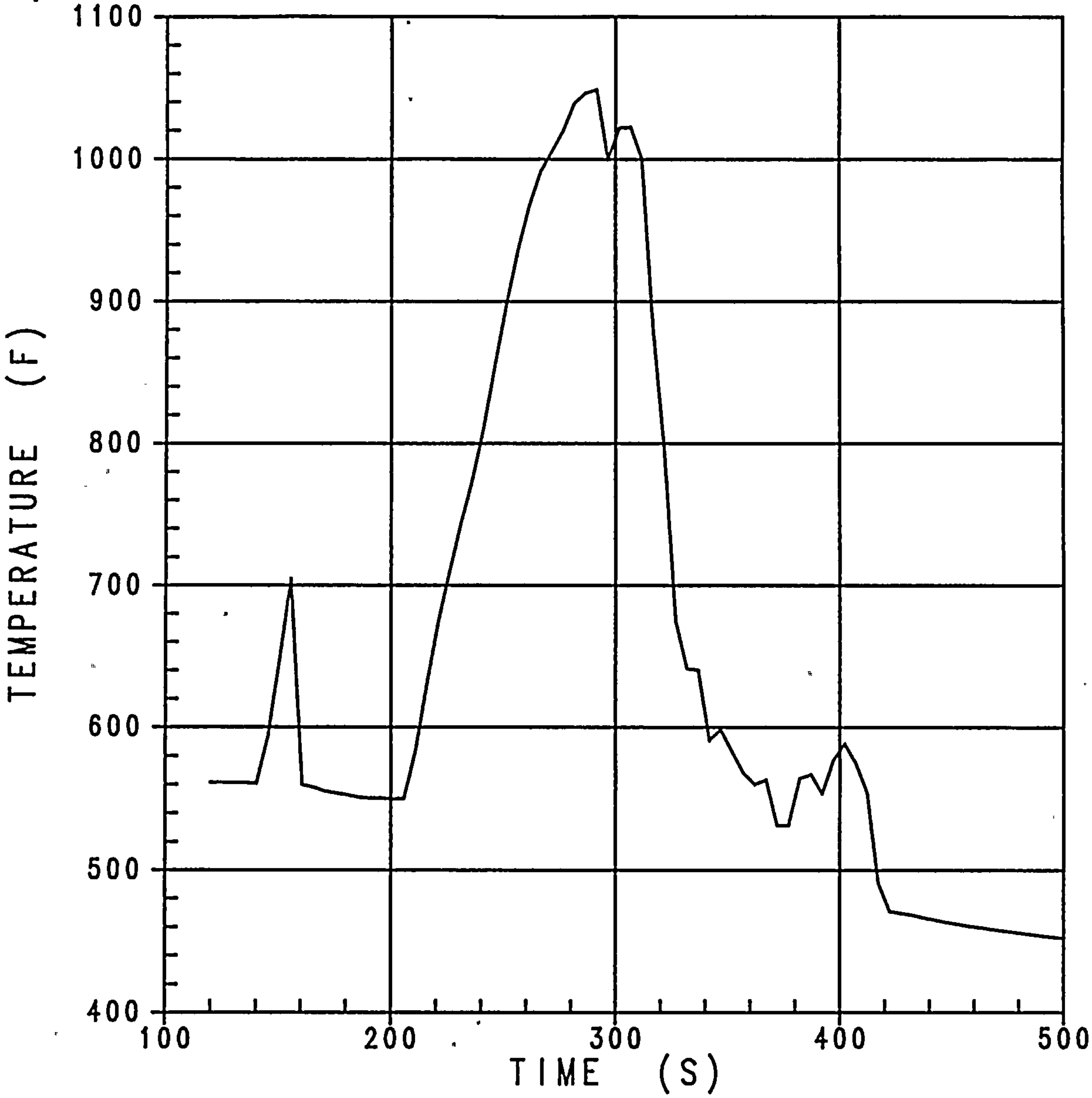

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Figure 15.6.4.1-6
4 Inch High T_{AVE} Break
Hot Spot Fluid Temperature (11.0 Feet)

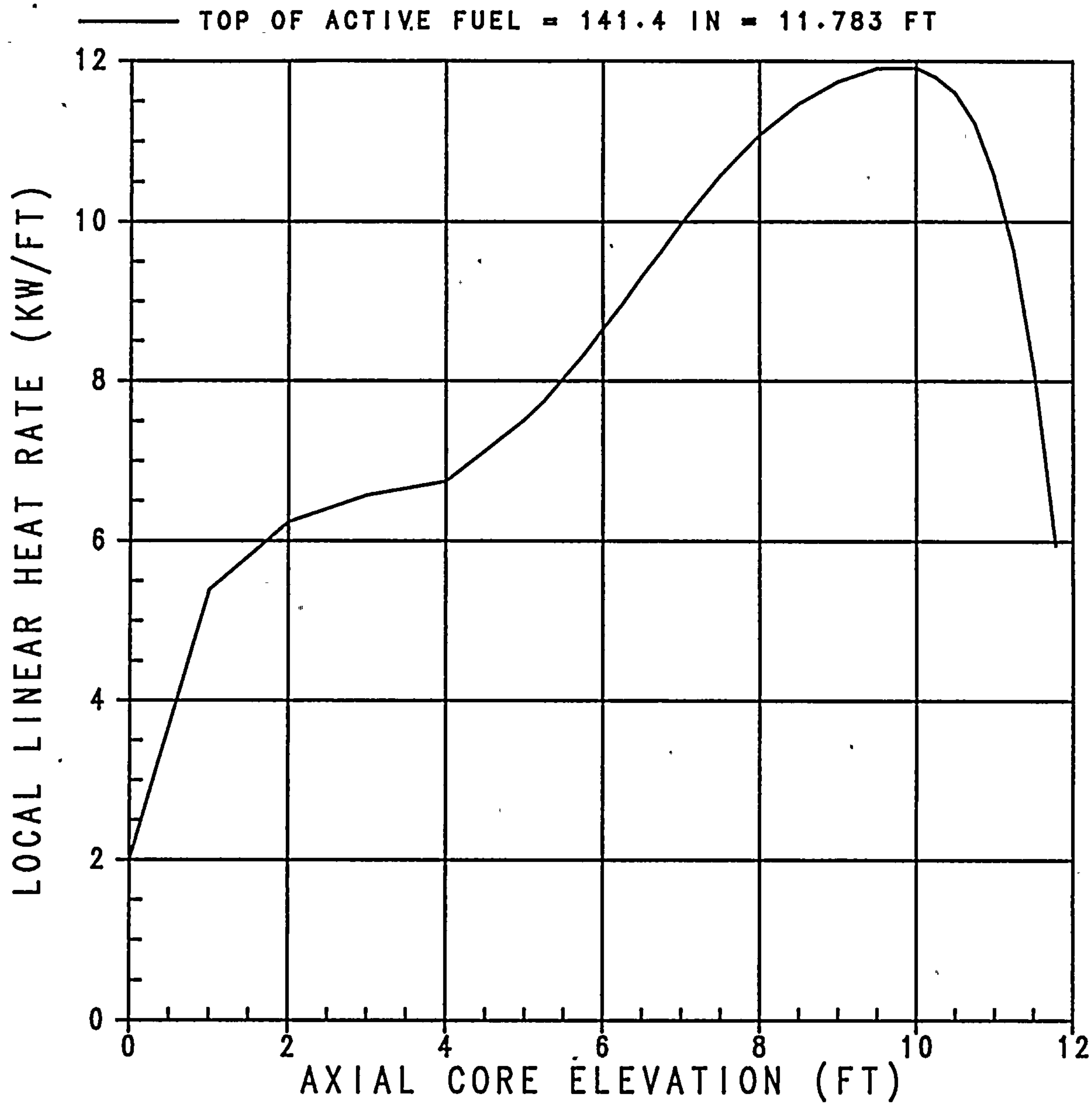

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Figure 15.6.4.1-7
Small Break LOCA
Axial Power Distribution

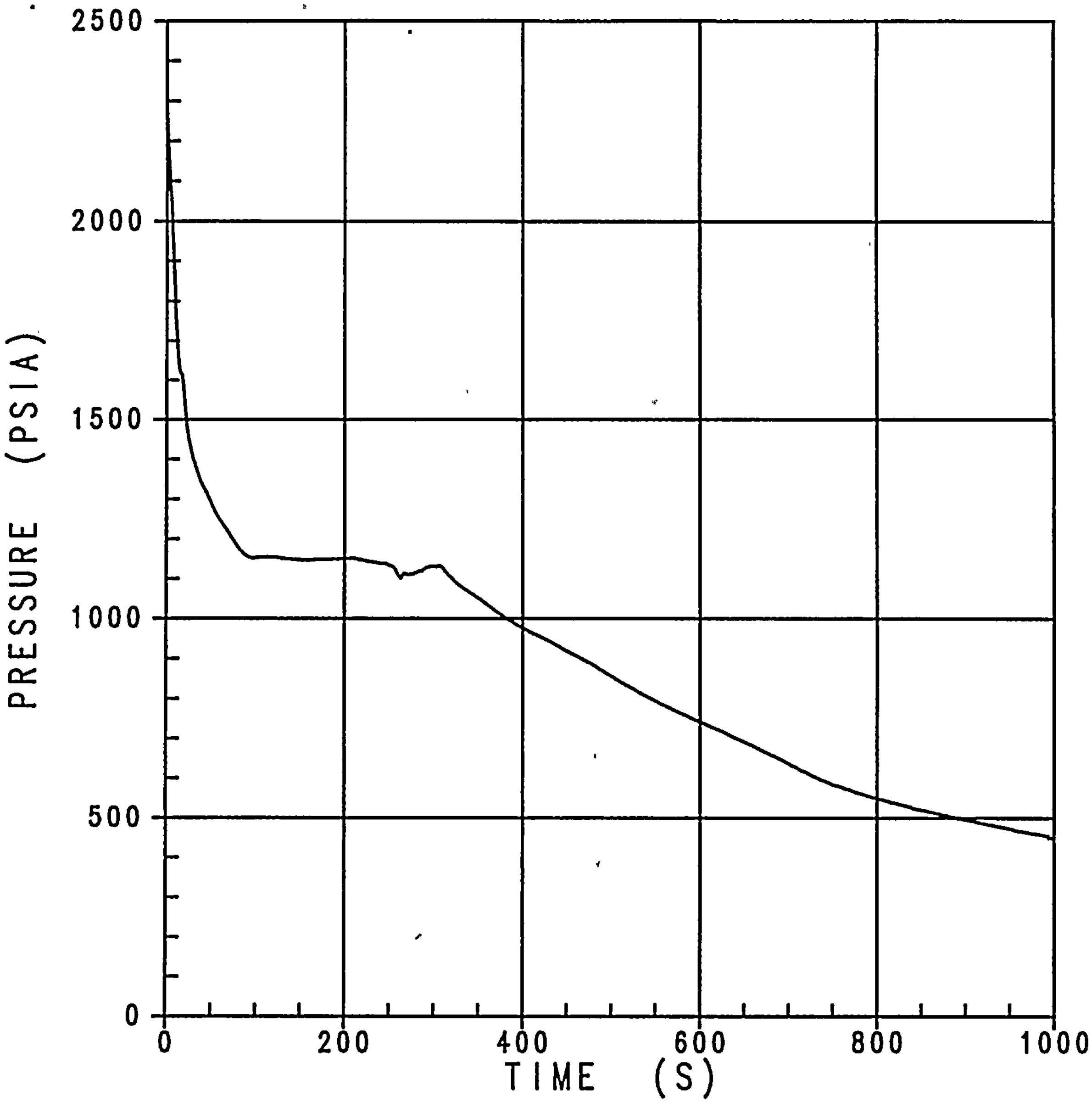

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Figure 15.6.4.1-8
3 Inch High T_{AVE} Break
Reactor Coolant System Pressure

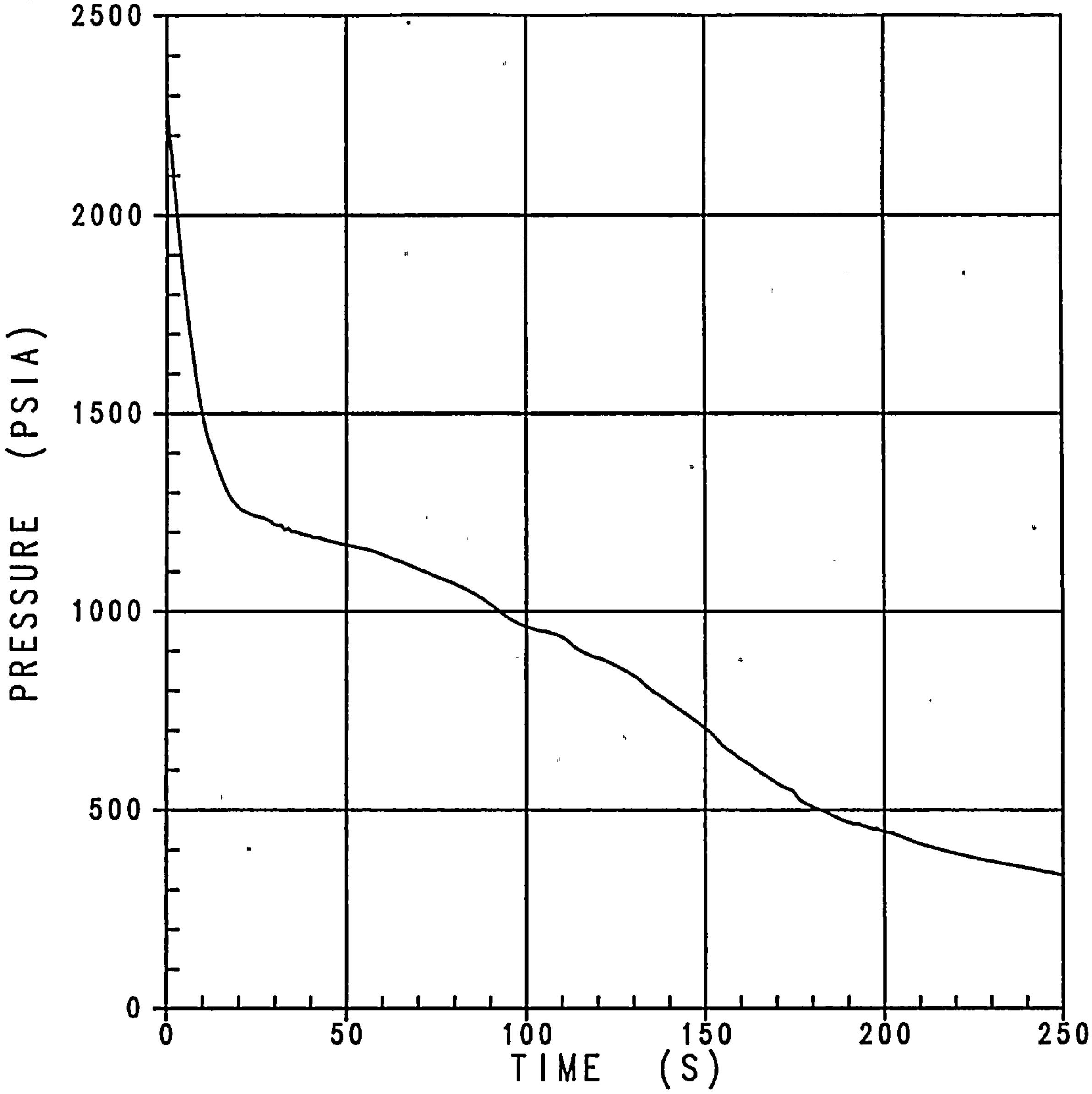

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Figure 15.6.4.1-9
6 Inch High T_{AVE} Break
Reactor Coolant System Pressure

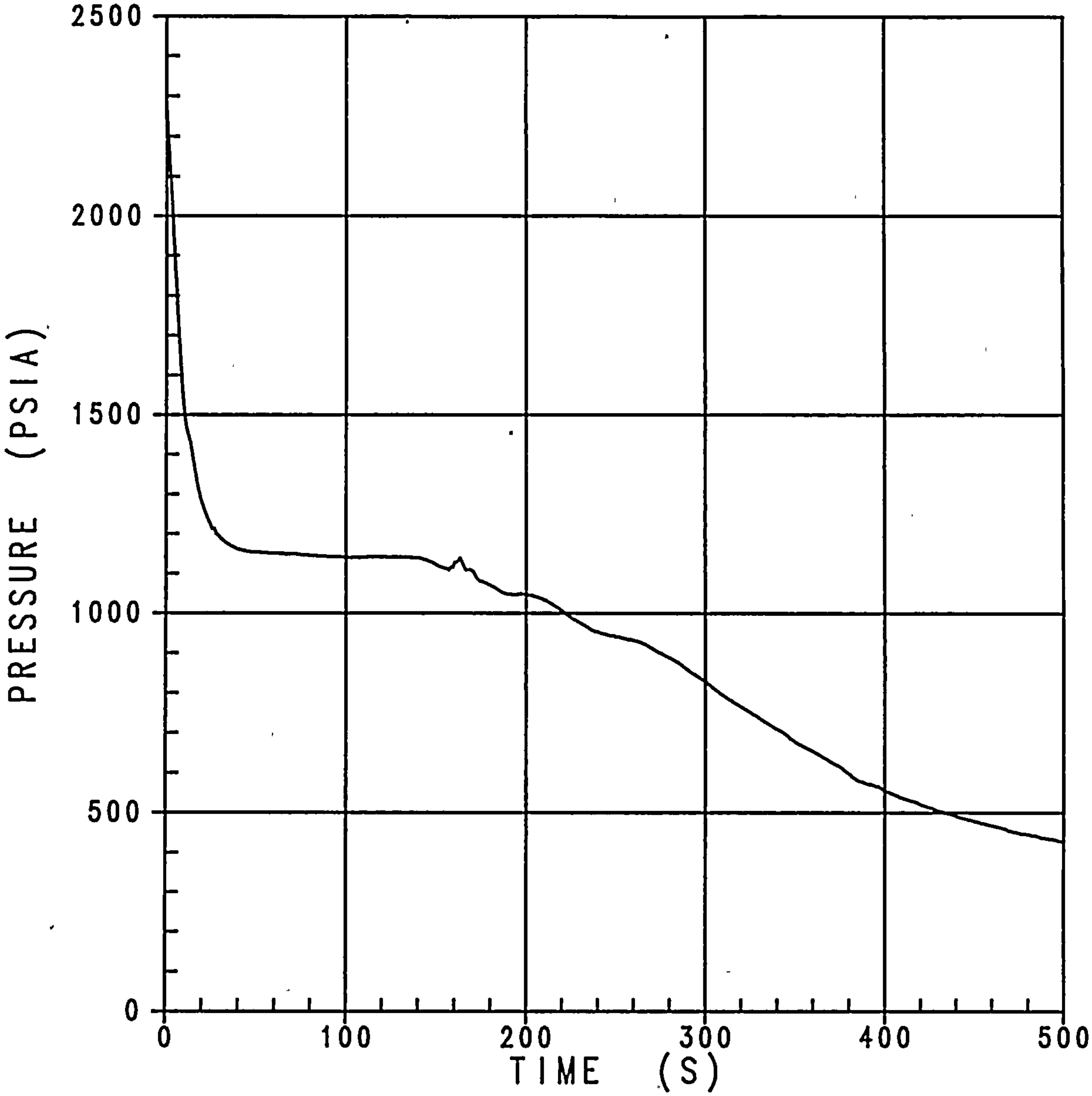

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Figure 15.6.4.1-10
4 Inch Low T_{AVE} Break
Reactor Coolant System Pressure

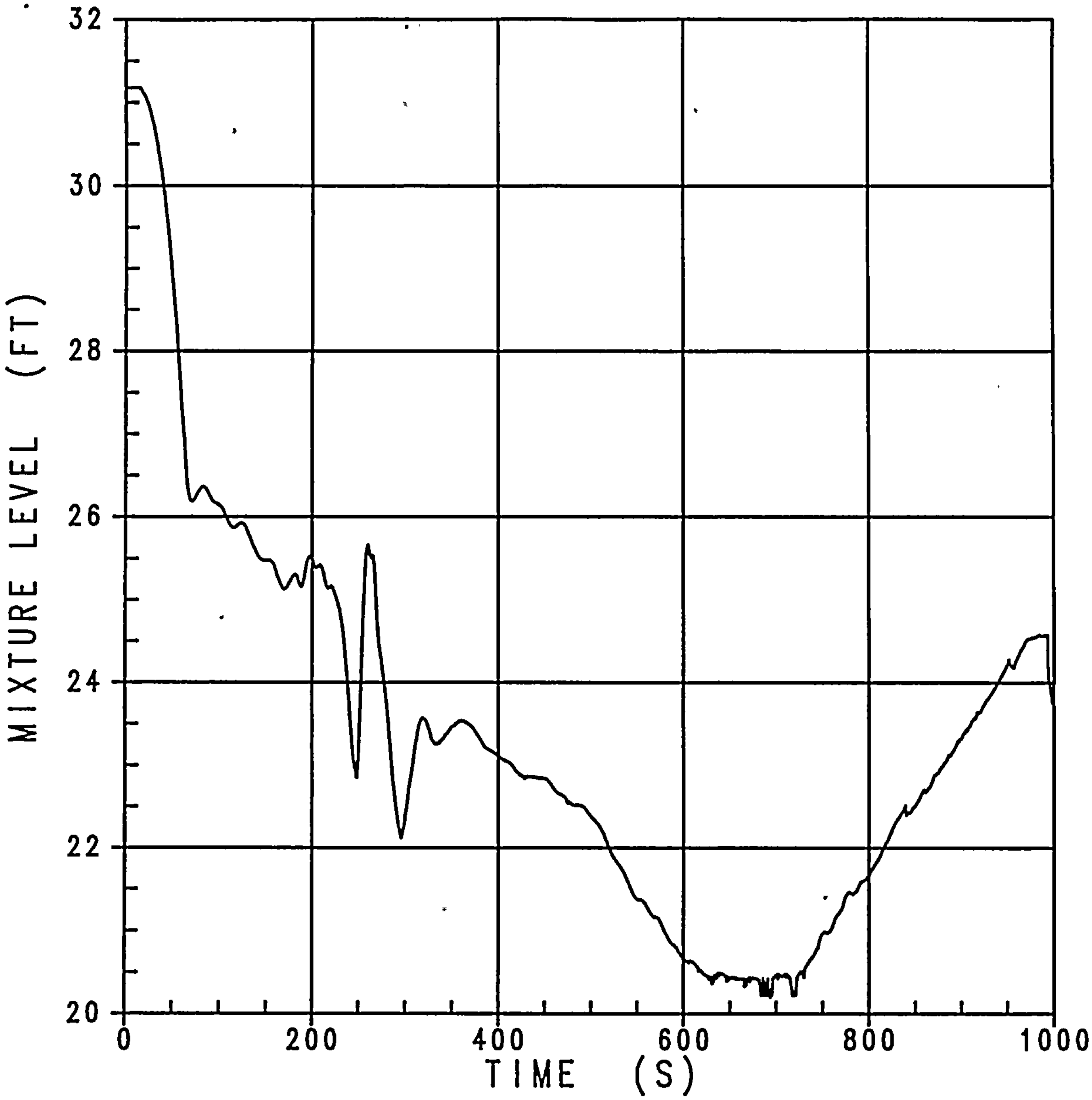

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Figure 15.6.4.1-11
3 Inch High T_{AVE} Break
Core Mixture Level

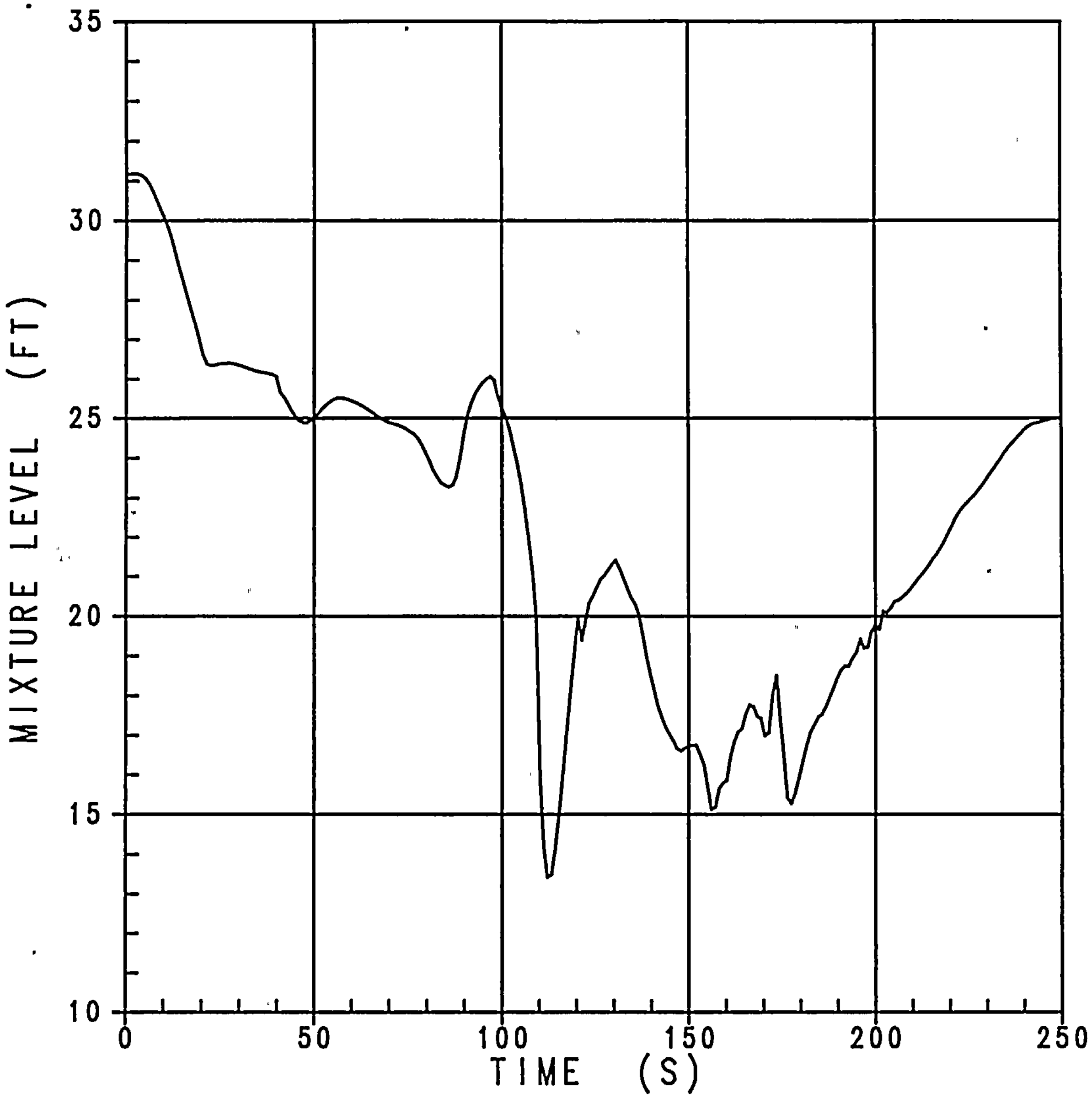

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Figure 15.6.4.1-12
6 Inch High T_{AVE} Break
Core Mixture Level

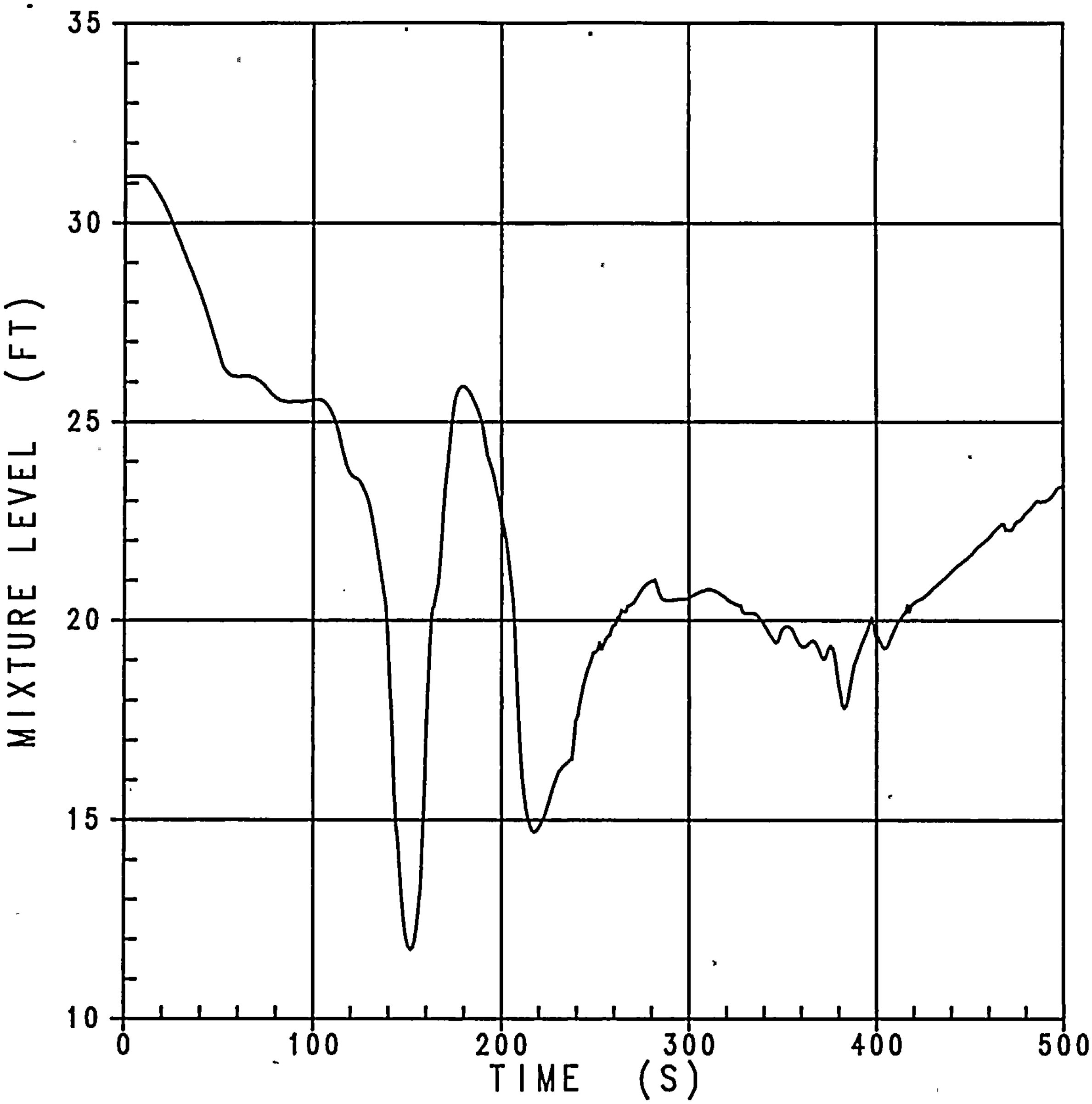

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Figure 15.6.4.1-13
4 Inch Low T_{AVE} Break
Core Mixture Level

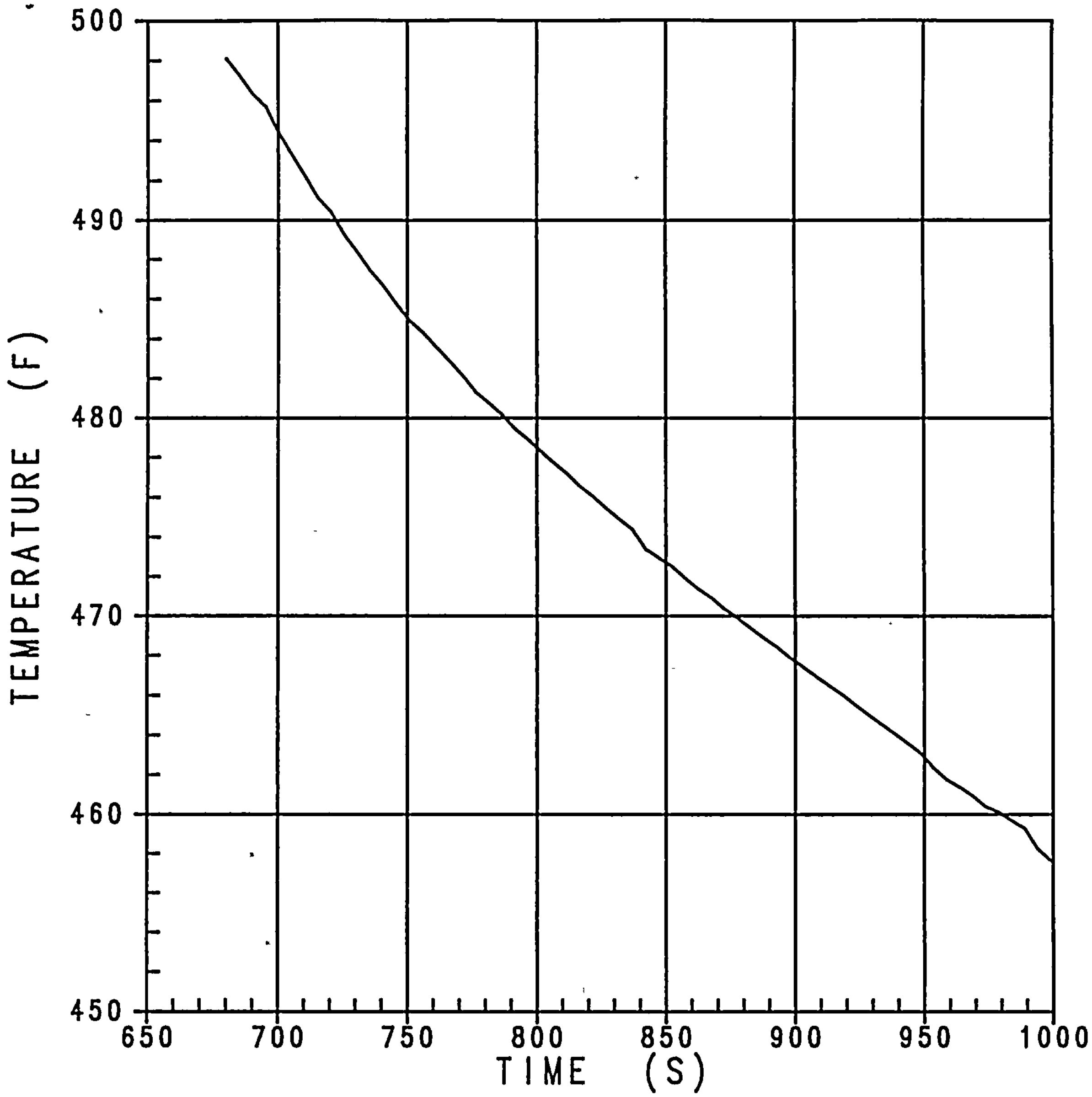

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Figure 15.6.4.1-14
3 Inch High T_{AVE} Break
Peak Cladding Temperature (11.783 Feet)

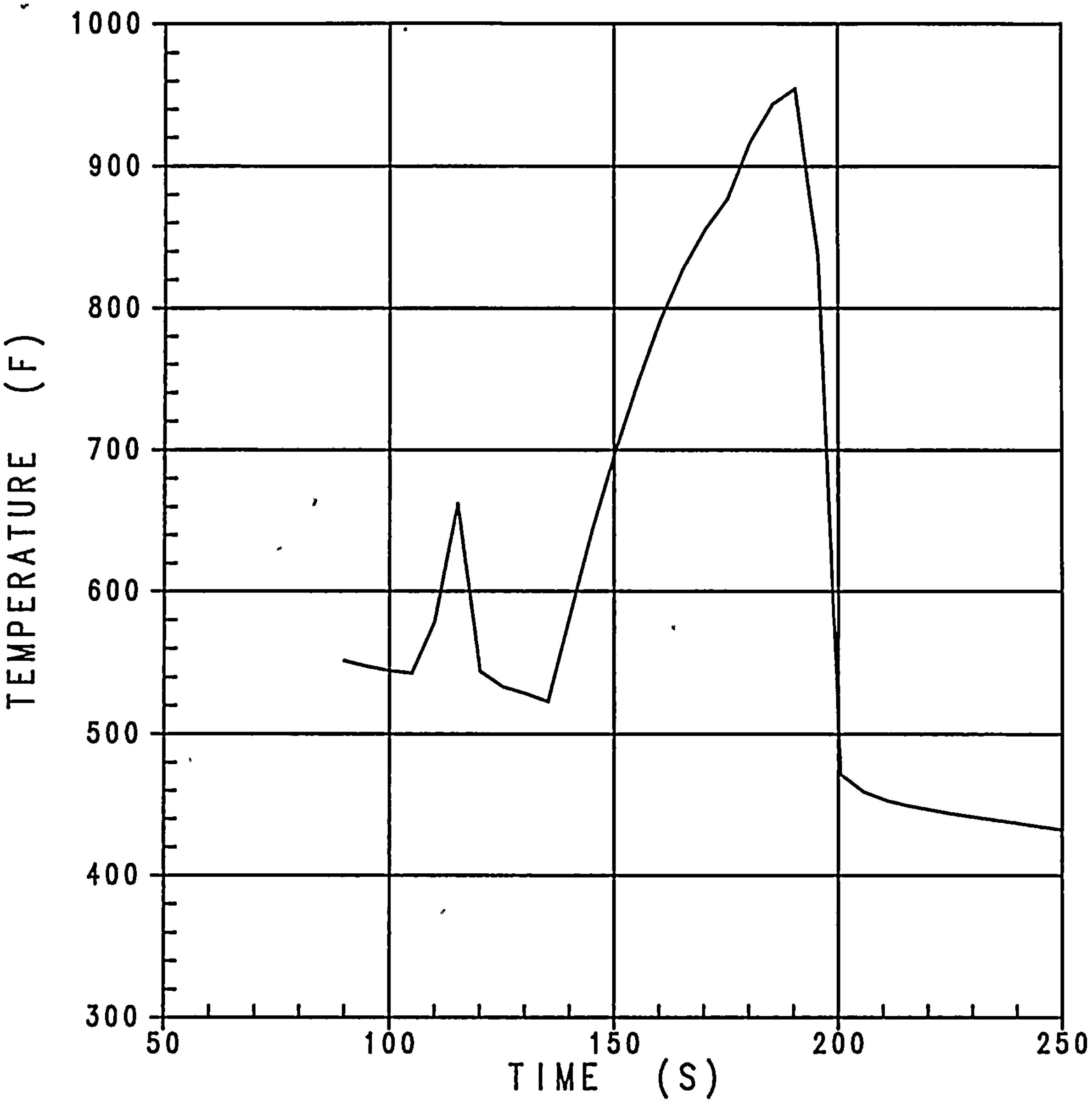

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Figure 15.6.4.1-15
6 Inch High T_{AVE} Break
Peak Cladding Temperature (10.75 Feet)

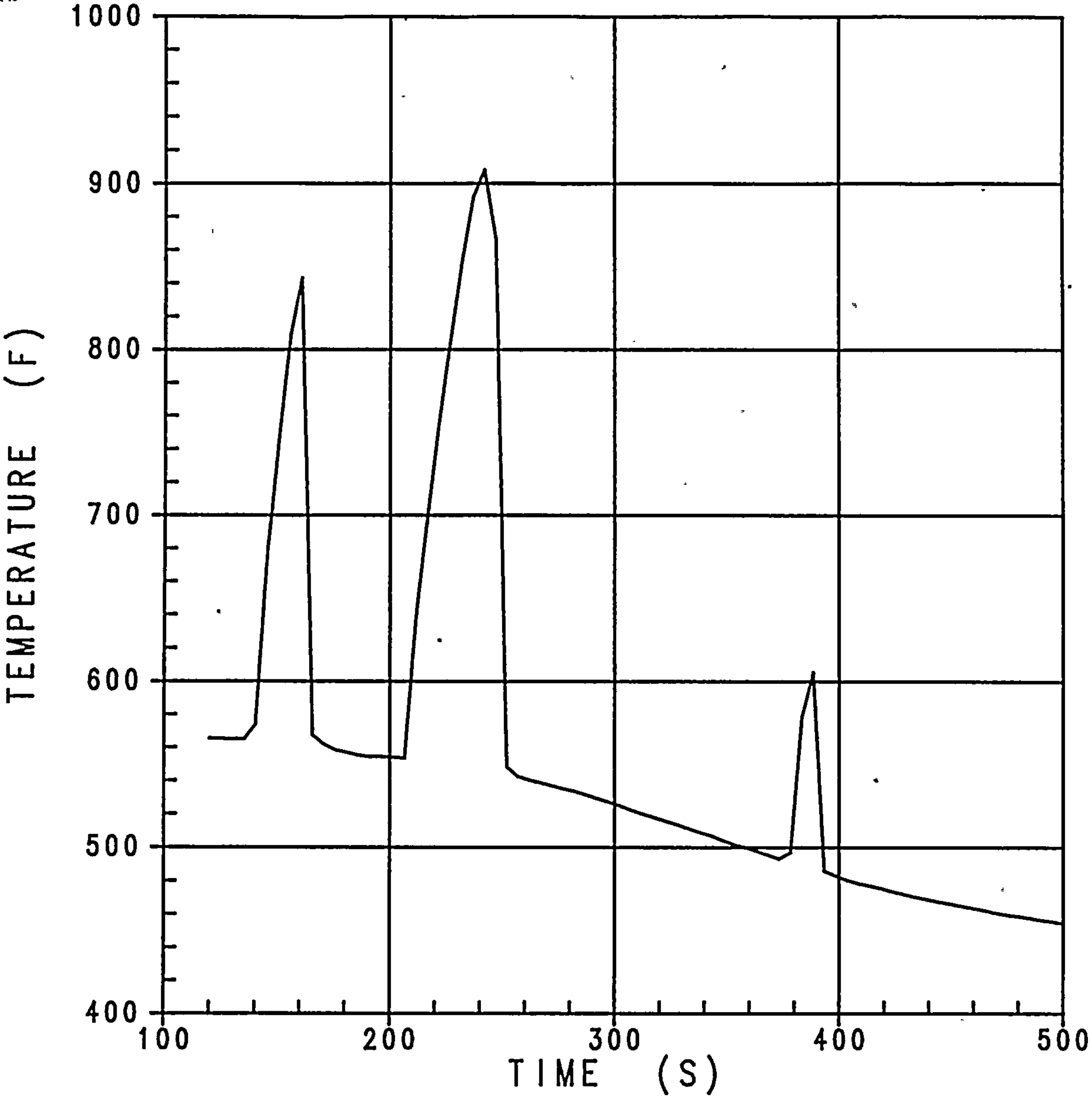

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Figure 15.6.4.1-16
4 Inch Low T_{AVE} Break
Peak Cladding Temperature (10.5 Feet)