Docket No. 50-336

Attachment 1

Millstone Nuclear Power Station, Unit No. 2

Proposed Revisions to

Technical Specifications

October, 1982

LIMITING CONDITION FOR OPERATION

b) The CEA group(s) with the inoperable position indicator is fully inserted and subsequently maintained fully inserted, while maintaining the withdrawl sequence and THERMAL POWER Level required by Specification 3.1.3.6 and when this CEA group reaches its fully inserted position, the "Full In" limit of the CEA with the inoperable position indicator is actuated and verifies this CEA to be fully inserted. Subsequent operation shall be within the limits of Specification 3.1.3.6.

4. If the failure of the position indicator channel(s) is during STARTUP, the CEA group(s) with the inoperable position indicator channel must be moved to the "Full Out" position and verified to be fully withdrawn via a "Full Out" indicator within 4 hours. The Provisions of Specification 3.0.4 are not applicable.

c. With a maximum of one reed switch position indicator channel per group or one pulse counting position indicator channel per group inoperable and the CEA(s) with the inoperable position indicator channel at either its fully inserted position or fully withdrawn position, operation may continue provided:

1. The position of this CEA is verified immediately and at least once per 12 hours thereafter by its "Full In" or "Full Out" limit (as applicable),

2. The fully inserted CEA group(s) containing the inoperable position indicator channel is subsequently maintained fully inserted and

3. Subsequent operation is within the limits of Specification 3.1.3.6.

d. With more than one pulse counting position indicator channels inoperable, operation in MODES 1 and 2 may continue for up to 24 hours provided all of the reed switch position indicator channels are OPERABLE.

SURVEILLANCE REQUIREMENTS

4.1.3.3 Each position indicator channel shall be determined to be OPERABLE by verifying the pulse counting position indicator channels and the reed switch position indicator channels agree within 6 steps at least once per 12 hours except during time intervals when the Deviation circuit is inoperable, then compare the pulse counting position indicator and reed switch position indicator channels at least once per 4 hours.

SAFETY INJECTION TANKS

LIMITING CONDITION FOR OPERATION

3.5.1 Each reactor coolant system safety injection tank shall be OPERABLE with:

a. The isolation valve open and the power to the valve operator removed,

b. Between 1080 and 1190 cubic feet of borated water,

c. A minimum boron concentration of 1720 PPM, and

d. A nitrogen cover-pressure of between 200 and 250 psig.

APPLICABILITY: MODES 1, 2 and 3.*

ACTION:

a. With one safety injection tank inoperable, except as a result of a closed isolation valve, restore the inoperable tank to OPERABLE status within one hour or be in HOT SHUTDOWN within the next 8 hours.

b. With one safety injection tank inoperable due to the isolation valve being closed, either immediately open the isolation valve or be in HOT STANDBY within one hour and be in HOT SHUTDOWN within the next 8 hours.

SURVEILLANCE REQUIREMENTS

4.5.1 Each safety injection tank shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1. Verifying the water level and nitrogen cover-pressure in the tanks, and

2. Verifying that each safety injection tank isolation valve is open.

* With pressurizer pressure ≥ 1750 psia.

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 Primary CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: MODES 1,2,3 and 4.

ACTION:

Without primary CONTAINMENT INTEGRITY*, restore CONTAINMENT INTEGRITY within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.1 Primary CONTAINMENT INTEGRITY shall be demonstrated:

a. At least once per 31 days by verifying that all penetrations not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges or deactivated automatic valves secured in their positions, except as provided in Table 3.6-2 of Specification 3.6.3.1.

b. At least once per 31 days, by verifying the equipment hatch is closed and sealed.

c. By verifying the containment air lock is OPERABLE per Specification 3.6.1.3.

d. After each closing of a penetration subject to type B testing (except the containment air lock), if opened following a Type A or B test, by leak rate testing the seal with gas at P_a (54 psig) and verifying that when the measured leakage rate for these seals is added to the leakage rate determined pursuant to Specification 4.6.1.2.d for all other Type B and C penetrations, the combined leakage rate is less than or equal to $0.60\ L_a$.

* Operation within the time allowances of the ACTION statements of Specification 3.6.1.3 does not constitute a loss of CONTAINMENT INTEGRITY.

CONTAINMENT AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.1.3 Each containment air lock shall be OPERABLE with:

a. Both doors closed except when the air lock is being used for normal transit entry and exit through the containment, then at least one air lock door shall be closed and

b. An overall air lock leakage rate of less than or equal to 0.05 L_a at P_a (54 psig).

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With one containment air lock door inoperable:

1. Maintain at least the OPERABLE air lock door closed and either restore the inoperable air lock door to OPERABLE status within 24 hours or lock the OPERABLE air lock door closed.

2. Operation may then continue until performance of the next required overall air lock leakage test provided that the OPERABLE air lock door is verified to be locked closed at least once per 31 days.

3. Otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

4. The provisions of Specification 3.0.4 are not applicable if the outer air lock door is inoperable.

b. With the containment air lock inoperable, except as the result of an inoperable air lock door, maintain at least one air lock door closed; restore the inoperable air lock to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

a.* After each opening, except when the air lock is being used for multiple entries, then at least once per 72 hours, by verifying no detectable seal leakage by pressure decay when the volume between the door seals is pressurized to greater than or equal to 25 psig for at least 15 minutes,

b. At least once per 6 months by conducting an overall air lock leakage test at P_a (54 psig) and by verifying that the overall air lock leakage rate is within its limit and

c. At least once per 6 months by verifying that only one door in each air lock can be opened at a time.

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* Exemption to Appendix "J" of 10CFR50.

Docket No. 50-336

Attachment 2

Millstone Nuclear Power Station, Unit No. 2

Large Break Loss-of-Coolant Accident

Analysis Results

October, 1982

The Loss of Coolant Accident (LOCA) has been reanalyzed for Millstone Unit 2 with 15.3% (1300 tubes) steam generator tube plugging and 5.5% primary flow reduction. The following information amends the Safety Analysis Report section on Major Reactor Coolant System Pipe Ruptures. The results are consistent with acceptance criteria provided in Reference [1].

The description of the various aspects of the Westinghouse LOCA analysis methodology is given in Reference [2]. This document describes the major phenomena modeled, the interfaces among the computer codes, and the features of the codes which ensure compliance with the Acceptance Criteria. The SATAN-VI, WREFLOOD, COCO, and LOCTA-IV codes which are used in the LOCA analysis are described in detail in References [3] through [6]; code modifications are specified in References [7] through [13]. These codes are used to assess the core heat transfer geometry and to determine if the core remains amenable to cooling throughout and subsequent to the blowdown, refill, and reflood phases of the LOCA. The SATAN-VI computer code analyzes the thermal-hydraulic transient in the RCS during blowdown, and the WREFLOOD computer code is used to calculate this transient during the refill and reflood phases of the accident. The COCO computer code is used to calculate the Containment pressure transient throughout the LOCA analysis. Similarly, the LOCTA-IV computer code is used to compute the thermal transient of the hottest fuel rod during the entire analysis.

SATAN-VI is used to calculate the RCS pressure, enthalpy, density, and the mass and energy flow rates in the RCS, as well as steam generator energy transfer between the primary and secondary systems as a function of time during the blowdown phase of the LOCA. SATAN-VI also calculates the accumulator water flow rates and internal pressure and the pipe break mass and energy flow rates that are assumed to be vented to the Containment during blowdown. At the end of the blowdown phase, these

data are transferred to the WREFLOOD code. The mass and energy release rates during blowdown are utilized in the COCO code for use in the determination of the Containment pressure response during this first phase of the LOCA. Additional SATAN-VI output data including the core flow rates and enthalpy, the core pressure, and the core power decay transient, are transferred to the LOCTA-IV code.

With initial information from the SATAN-VI code, WREFLOOD uses a system thermal-hydraulic model to determine the core flooding rate (i.e., the rate at which coolant enters the bottom of the core), the coolant pressure and temperature, and the core water level during the refill and reflood phases of the LOCA. WREFLOOD also calculates the mass and energy flow addition to the Containment through the break. Since the mass flow rate to the Containment depends upon the core flooding rate and the local core pressure, which is a function of the Containment backpressure, the WREFLOOD and COCO codes are interactively linked. WREFLOOD is also linked to the LOCTA-IV code in that thermal-hydraulic parameters from WREFLOOD are used by LOCTA-IV in its calculation of the fuel temperature. LOCTA-IV is used throughout the analysis of the LOCA transient to calculate the fuel clad temperature and metal-water reaction of the hottest rod in the core.

The analysis presented here was performed with the 1981 version of the evaluation model which includes the NUREG-0630.[13] Reactor Coolant pumps are assumed to continue to run during blowdown unless otherwise noted.

Results

The analysis of the loss of coolant accident is performed at 102 percent of the licensed core power rating. The peak linear power and total core power used in the analysis are given in Table 2. Since there is margin between the value of peak linear power density used in this analysis and the value of the peak linear power density expected during plant operation, the peak clad temperature calculated in this analysis is greater than the maximum clad temperature expected to exist.

Table 1 presents the occurrence time for various events throughout the accident transient.

Table 2 presents selected input values and results from the hot fuel rod thermal transient calculation. For these results, the hot spot is defined as the location of maximum peak clad temperatures. That location is specified in Table 2 for the worst break case analyzed. The location is indicated in feet which presents elevation above the bottom of the active fuel stack.

Table 3 presents a summary of the various containment systems parameters and structural parameters which were used as input to the COCO computer code[5] used in this analysis.

Figures 1 through 14 present the parameters of principal interest from the large break ECCS analysis. The following items are noted:

Figure 1: Hot spot clad temperature.
Figure 2: Coolant pressure in the reactor core.
Figure 3: Water level in the core and downcomer during reflood.
Figure 4: Containment pressure transient
Figure 5: Core flow during blowdown
Figure 6: Fuel rod heat transfer coefficients.
Figure 7: Hot spot fluid temperature.
Figure 8: Mass released to Containment during blowdown.
Figure 9: Energy released to containment during blowdown.
Figure 10: Fluid quality in the hot assembly
Figure 11: Mass velocity
Figure 12: Safety injection tank water flow rate into RCS during blowdown (per tank).
Figure 13: Pumped safety injection water flow rate during reflood.
Figure 14: Core reflooding rate.

Conclusions - Thermal Analysis

For breaks up to and including the double ended severance of a reactor coolant pipe, the Emergency Core Cooling System will meet the Acceptance Criteria as presented in 10CFR50.46.[1] That is:

1. The calculated peak clad temperature does not exceed 2200°F based on a peak core linear power of 15.6 kw/ft.

2. The amount of fuel element cladding that reacts chemically with water or steam does not exceed 1 percent of the total amount of Zircalloy in the reactor.

3. The clad temperature transient is terminated at a time when the core geometry is still amenable to cooling. The cladding oxidation limits of 17% are not exceeded during or after quenching.

4. The core temperature is reduced and decay heat is removed for an extended period of time, as required by the long-lived radioactivity remaining in the core.

REFERENCES

1. "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Cooled Nuclear Power Reactors," 10CFR50.46 and Appendix K of 10CFR50. Federal Register, Volume 39, Number 3, January 4, 1974.

2. Bordelon, F. M., Massie, H. W. and Zordan T. A., "Westinghouse ECCS Evaluation Model - Summary," WCAP-8339, July 1974.

3. Bordelon, F. M., et al., "SATAN-VI Program: Comprehensive Space-Time Dependent Analysis of Loss of Coolant," WCAP-8302 (Proprietary) and WCAP-8306 (Non-Proprietary), June 1974.

4. Kelly, R. D., et al., "Calculational Model for Core Reflooding After a Loss of Coolant Accident (WREFLOOD Code)," WCAP-8170 (Proprietary) and WCAP-8171 (Non-Proprietary), June 1974.

5. Bordelon, F. M. and Murphy, E. T., "Containment Pressure Analysis Code (COCO)," WCAP-8327 (Proprietary) and WCAP-8326 (Non-Proprietary), June 1974.

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

7. Ferguson, K. L., and Kemper, R. M., ECCS Evaluation Model for Westinghouse Fuel Reloads of Combustion Engineering NSSS, WCAP-9528 (Proprietary) and WCAP-9529 (Non-Proprietary), June 1979.

8. Ferguson, K. L., and Kemper, R. M., Addendum to ECCS Evaluation Model for Westinghouse Fuel Reloads of Combustion Engineering NSSS, October 1979.

9. Bordelon, F. M., et al., "Westinghouse ECCS Evaluation Model - Supplementary Information," WCAP-8471 (Proprietary) and WCAP-8472 (Non-Proprietary), April 1975.

10. "Westinghouse ECCS Evaluation Model - October 1975 Version," WCAP-8622 (Proprietary) and WCAP-8623 (Non-Proprietary), November 1975.

11. Letter NS-CE-924, dated January 23, 1976, C. Eicheldinger (Westinghouse) to D. B. Vassallo (NRC).

12. Eicheldinger, C., "Westinghouse ECCS Evaluation Model,, February 1978 Version," WCAP-9220-P-A (Proprietary Version), WCAP-9221-A (Non-Proprietary Version), February 1978.

13. "Westinghouse ECCS Evaluation Model - 1981 Version," WCAP-9220-P-A Revision I (Proprietary Version), WCAP-9221-A Revision I (Non-Proprietary Version), February 1982.

TABLE 1

LARGE BREAK
TIME SEQUENCE OF EVENTS

	C_D=0.6 DECLG (Sec)
START	0.0
S. I. Signal*	0.68
S. I. Tank Injection	15.4
End of Blowdown	21.3
Bottom of Core Recovery	34.0
S. I. Tank Empty	63.8
End of Bypass	21.3

*from containment pressure sensor

TABLE 2

LARGE BREAK

Results	C_D=0.6 DECLG
Peak Clad Temp. °F	2055
Peak Clad Location, Ft.	7.0
Local Zr/H_2O Rxn(max) %	4.5
Local Zr/H_2O Location, Ft.	7.0
Total Zr/H_2O Rxn, %	<0.3
Hot Rod Burst Time, sec	28.6
Hot Rod Burst Location, Ft.	5.7

Calculation Assumptions	
NSSS Power, Mwt, 102% of	2700
Peak Core Linear Power, kw/ft	15.6
S.I. Tank Actuation Pressure, psia	215
S.I. Tank Water Volume, ft^3 per tank	1080

TABLE 3

Millstone Unit 2
Containment Physical Parameters

Net Free Volume	1.938×10^6 ft^3
Containment Initial Conditions:	
Humidity	99 %
Containment Temperature	60°F
Enclosure Building Temperature	60°F
Ground Temperature	40°F
Initial Pressure	14.7 psia
Initial Time for:	
Spray Flow	26 seconds
Fans (3)	0.0 seconds
Additional Fan	14.0 seconds
Containment Spray Water:	
Temperature	50°F
Flow Rate (Total, 2 pumps)	3300 gpm

Fan Cooling Capacity (Per Fan)

Vapor Temperature (°F)	Capacity (BTU/Sec)
60	0.0
145	3360.0
165	5280.0
300	28800.0
350	32400.0

Containment Heat Absorbing Surfaces

1. Surface Areas and Thicknesses
   a. Shell and dome - 71,870 Ft2
      (1) Paint - 0.003 In. (one side exposed to containment atmosphere)
      (2) Carbon steel - 0.25 In.
      (3) Concrete - 3.0 Ft. (one side exposed to enclosure building atmosphere)

   b. Unlined Concrete - 62,800 Ft2
      (1) Concrete - 2.0 Ft. (one side exposed to containment atmosphere, one side insulated)

   c. Galvanized Steel - 120,000 Ft2
      (1) Zinc - 0.0036 In. (one side exposed to containment atmosphere)
      (2) Carbon steel - 0.20 In. (one side insulated)

Millstone Unit 2
Containment Physical Parameters

d. Painted Thin Steel - 56,350 Ft^2
   (1) Paint - 0.003 In. (one side exposed to containment atmosphere)
   (2) Carbon steel - 0.2 In. (one side insulated)

e. Painted Steel - 32,600 Ft^2
   (1) Paint - 0.003 In. (one side exposed to containment atmopshere)
   (2) Carbon steel - 0.26 In. (one side insulated)

f. Painted Steel - 22,425 Ft^2
   (1) Paint - 0.003 In. (one side exposed to containment atmosphere)
   (2) Carbon steel - 0.86 In. (one side insulated)

g. Painted Thick Steel - 4,230 Ft^2
   (1) Paint - 0.003 In. (one side exposed to containment atmosphere)
   (2) Carbon steel - 2.94 In. (one side insulated)

h. Containment Penetration Area - 3,000 Ft^2
   (1) Paint - 0.003 In. (one side exposed to containment atmosphere)
   (2) Carbon steel - 0.75 In.
   (3) Concrete - 3.75 Ft. (one side exposed to enclosure building atmosphere)

i. Stainless Steel Line Concrete - 8,340 Ft^2
   (1) Stainless steel - 0.25 In. (one side exposed to containment atmosphere)
   (2) Concrete - 2.0 Ft. (one side insulated)

j. Base Slab - 11,130 Ft^2
   (1) Concrete - 8.0 Ft. (one side exposed to containment sump, one side exposed to ground)

k. Neutron Shield - 1400 Ft^2
   (1) Stainless steel - 0.024 Ft. (both sides exposed to containment atmosphere)

l. CEDM Cable Support Structure - 1380 ft^2
   (1) Paint - 0.006 In.
   (2) Stainless Steel - 0.1094 ft. (both sides exposed to containment atmosphere)

Millstone Unit 2
Containment Physical Parameters

2. Thermal Properties

Material	Conductivity (BTU/hr-ft-°F)	Heat Capacity (BTU/ft^3-°F)
a. Concrete	2.0	36
b. Carbon Steel	35.0	55
c. Stainless Steel	10.0	62
d. Paint	1.5	32
e. Zinc	70.0	45

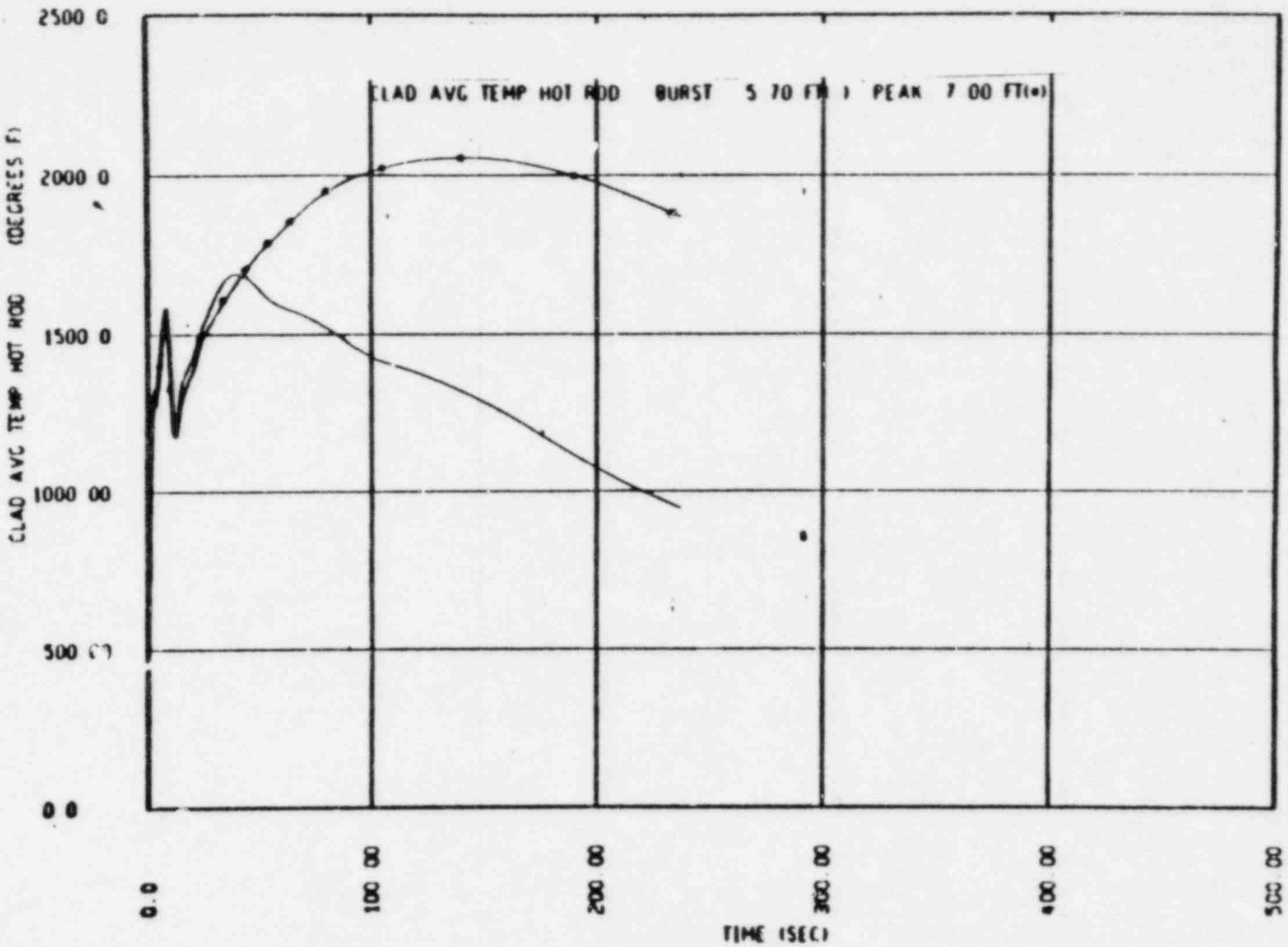

Figure 1 - Hot Spot Clad Temperature, 0.6 DECLG

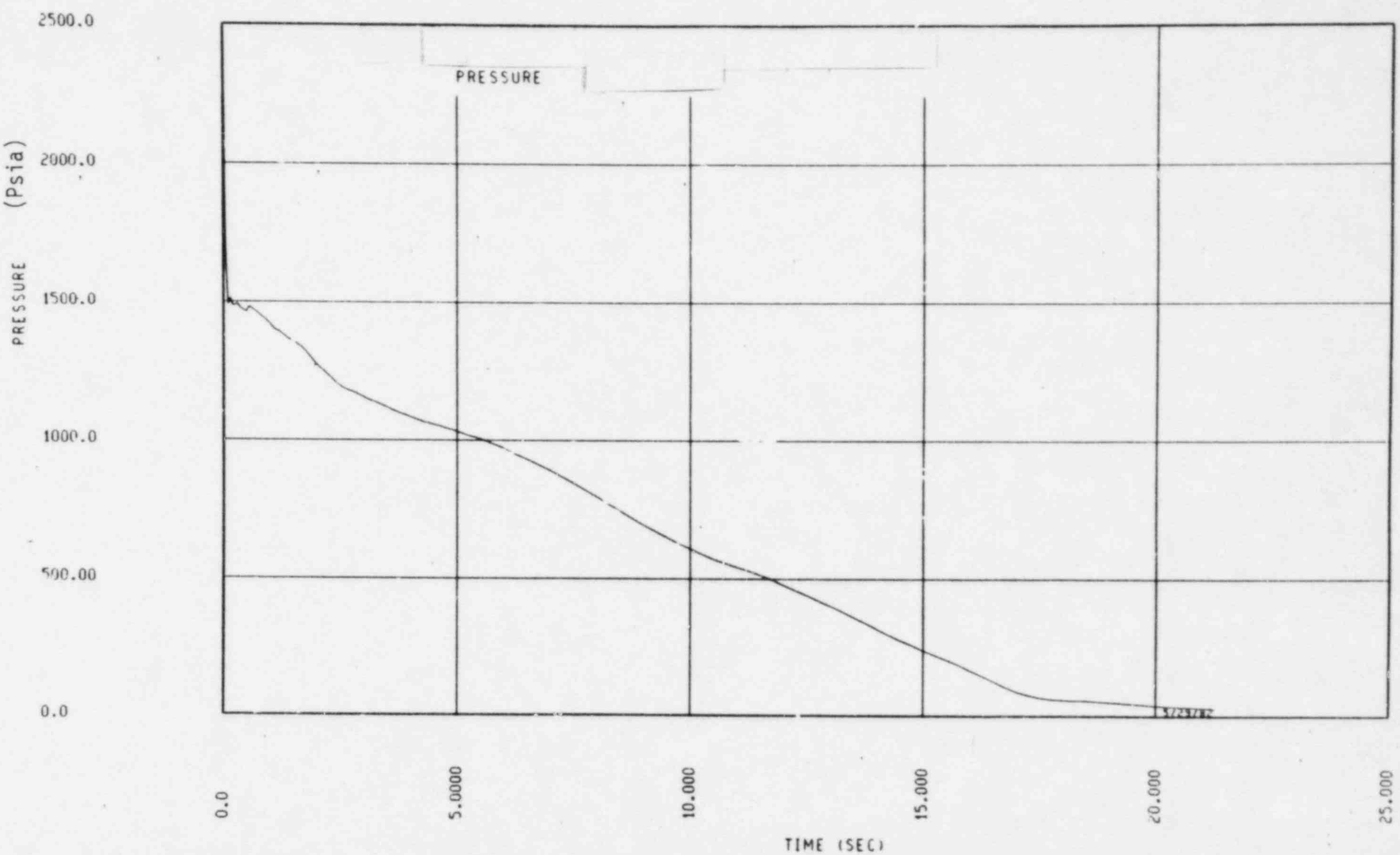

Figure 2 - Reactor Coolant Pressure, 0.6 DECLG

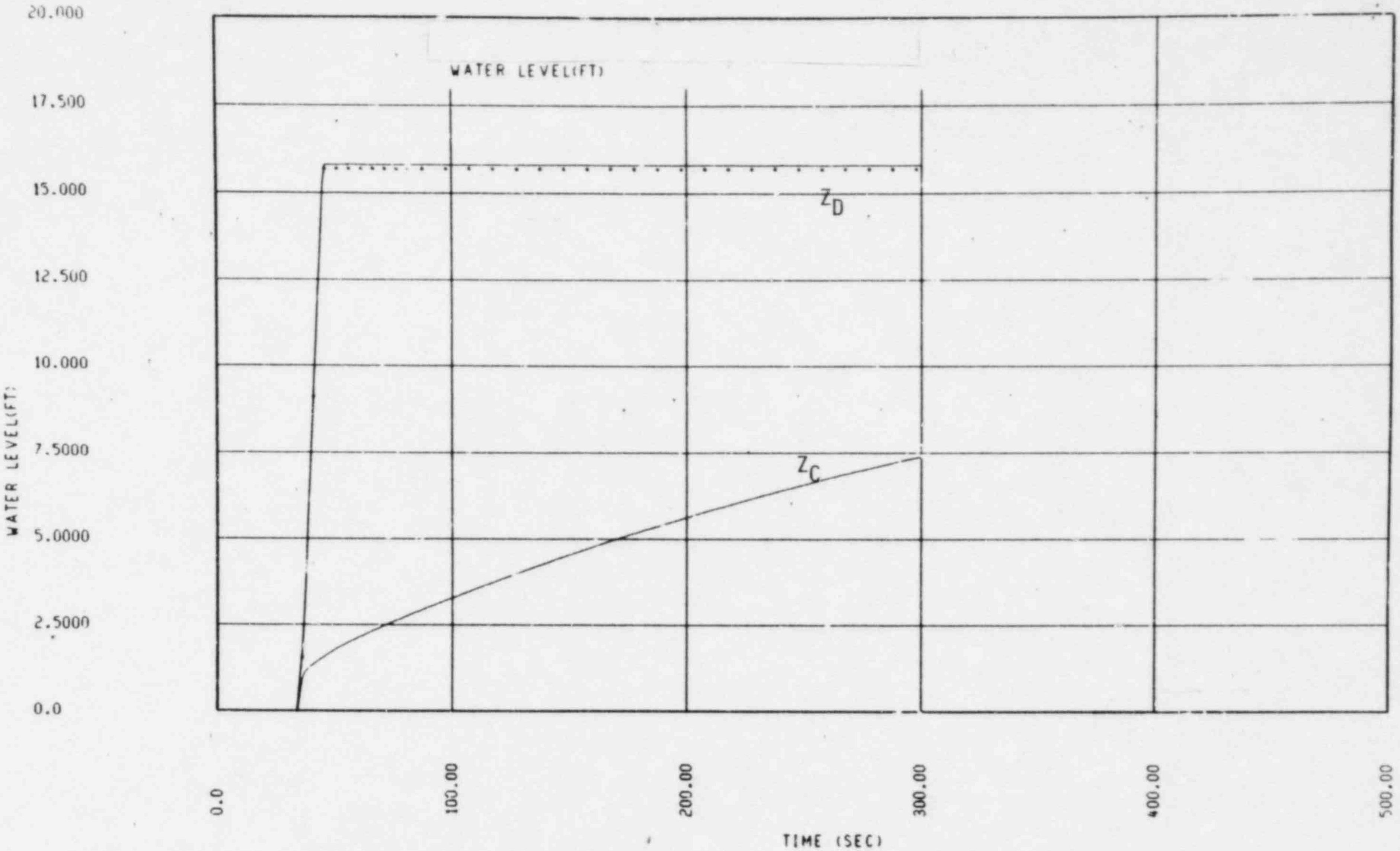

Figure 3 - Water Level in the Core and Downcomer During Reflood, 0.6 DECLG

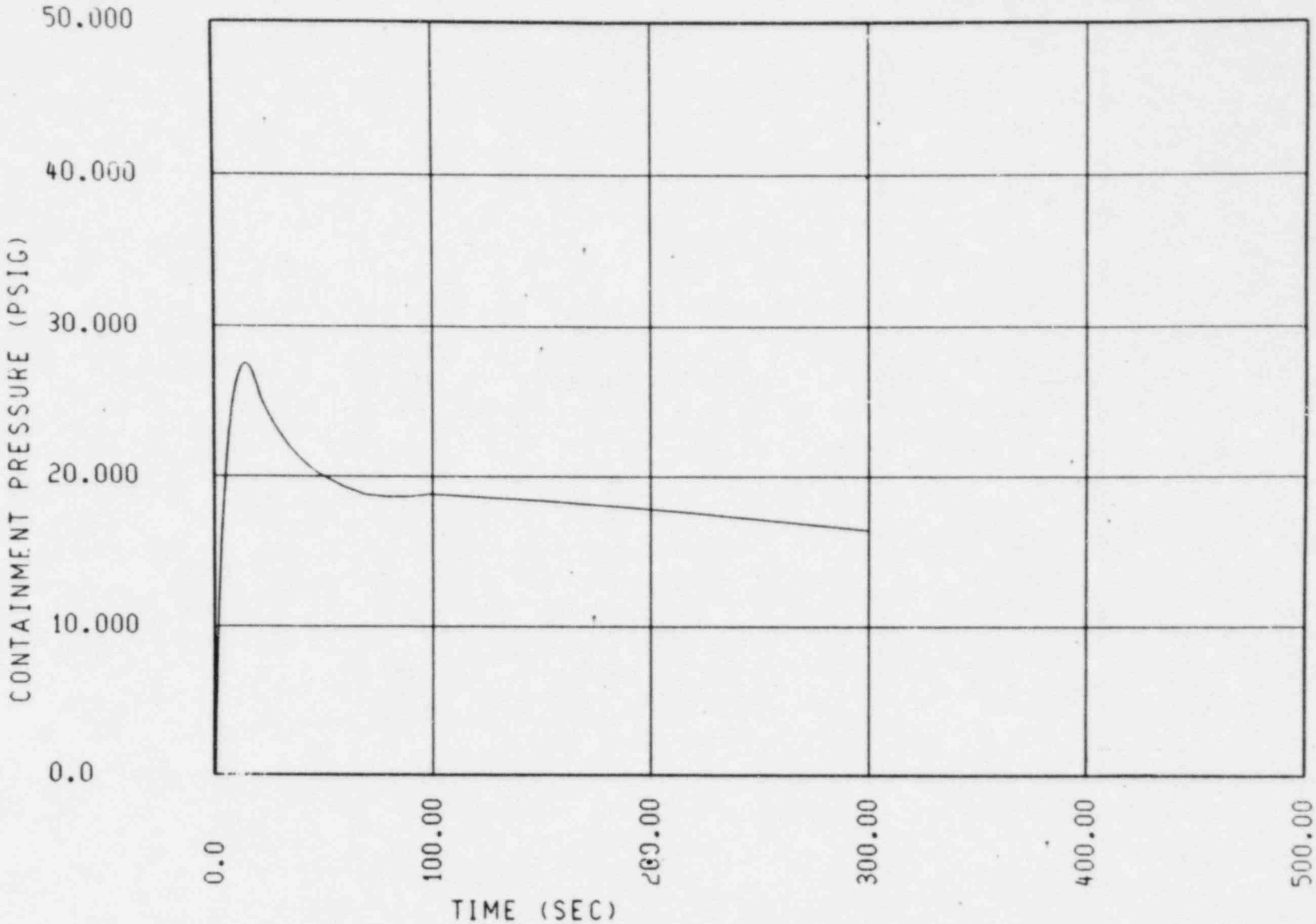

Figure 4 - Containment Pressure, 0.6 DECLG

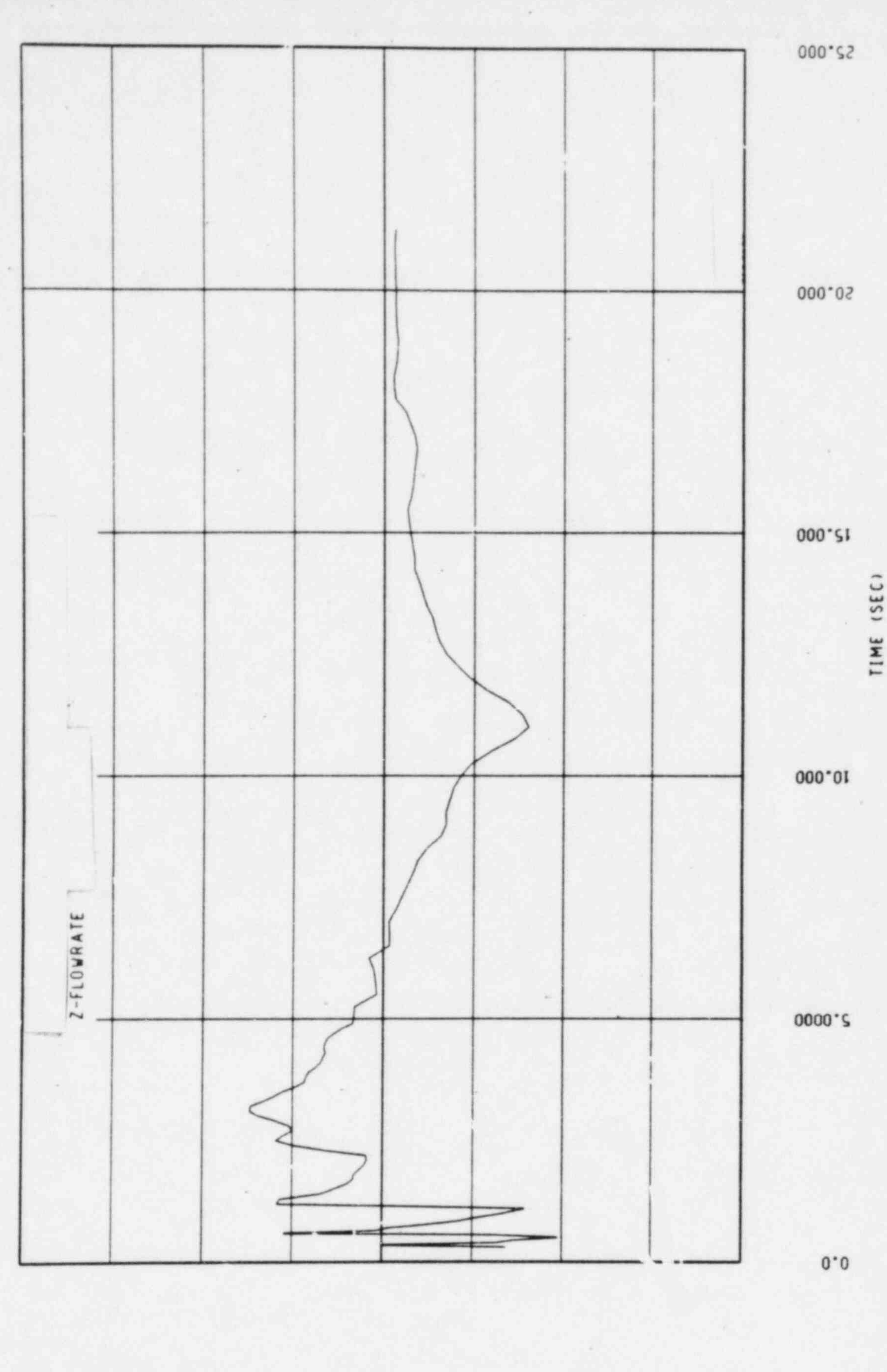

Figure 5 - Core Flow During Blowdown 0.6 DECLG

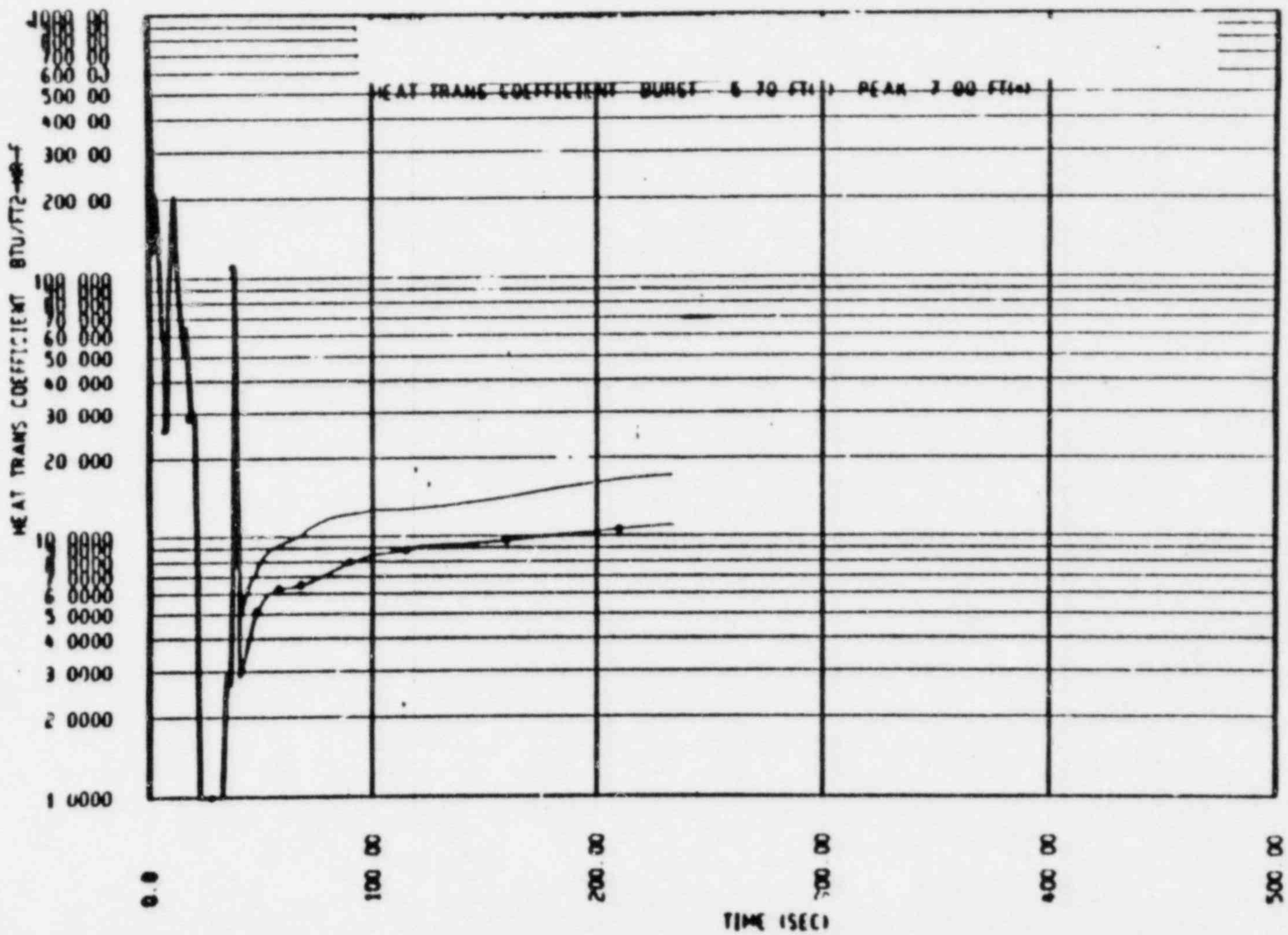

Figure 6 - Fuel Rod Heat Transfer Coefficients, 0.6 DECLG

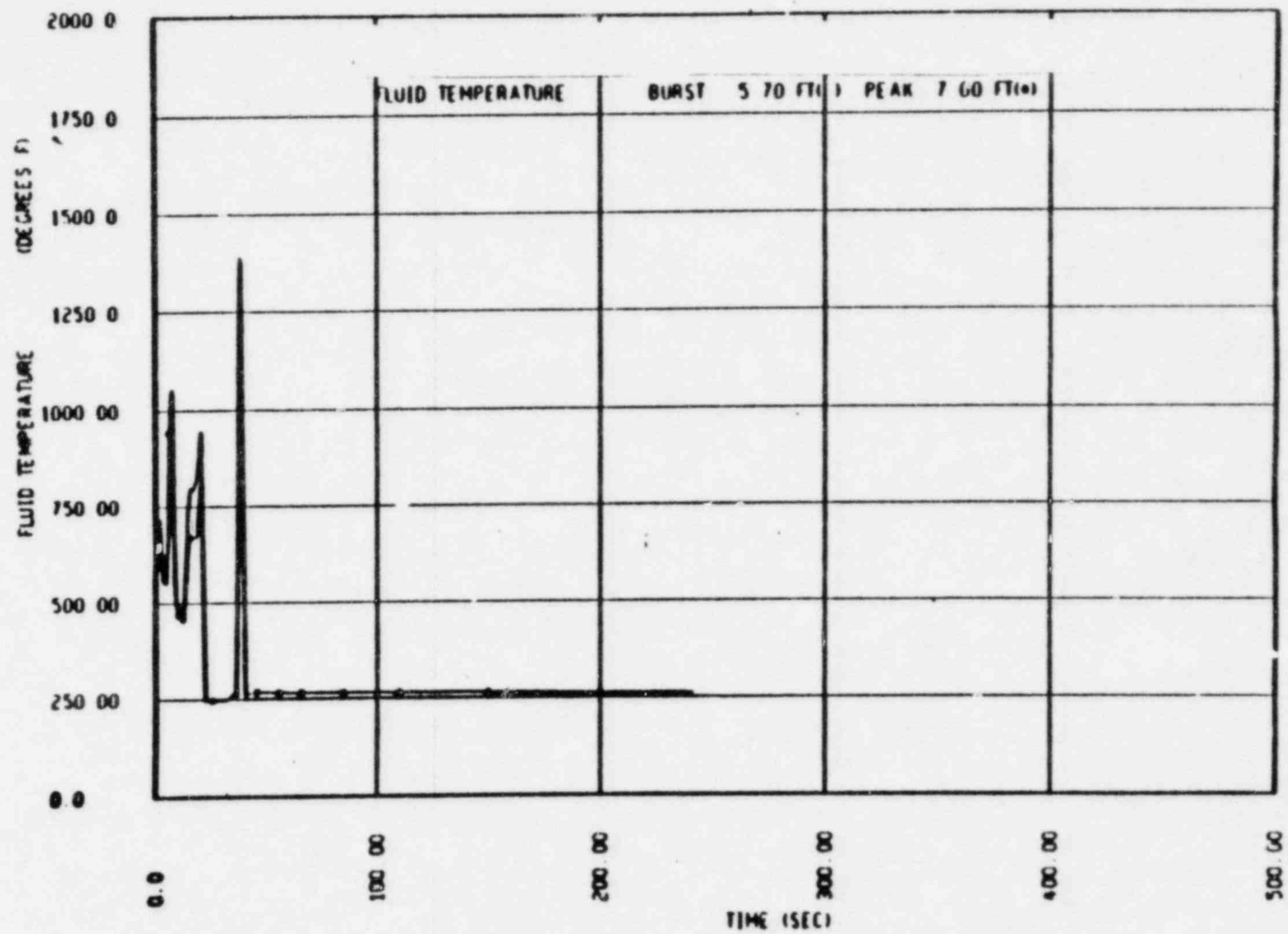

Figure 7 - Hot Spot Fluid Temperature, 0.6 DECLG

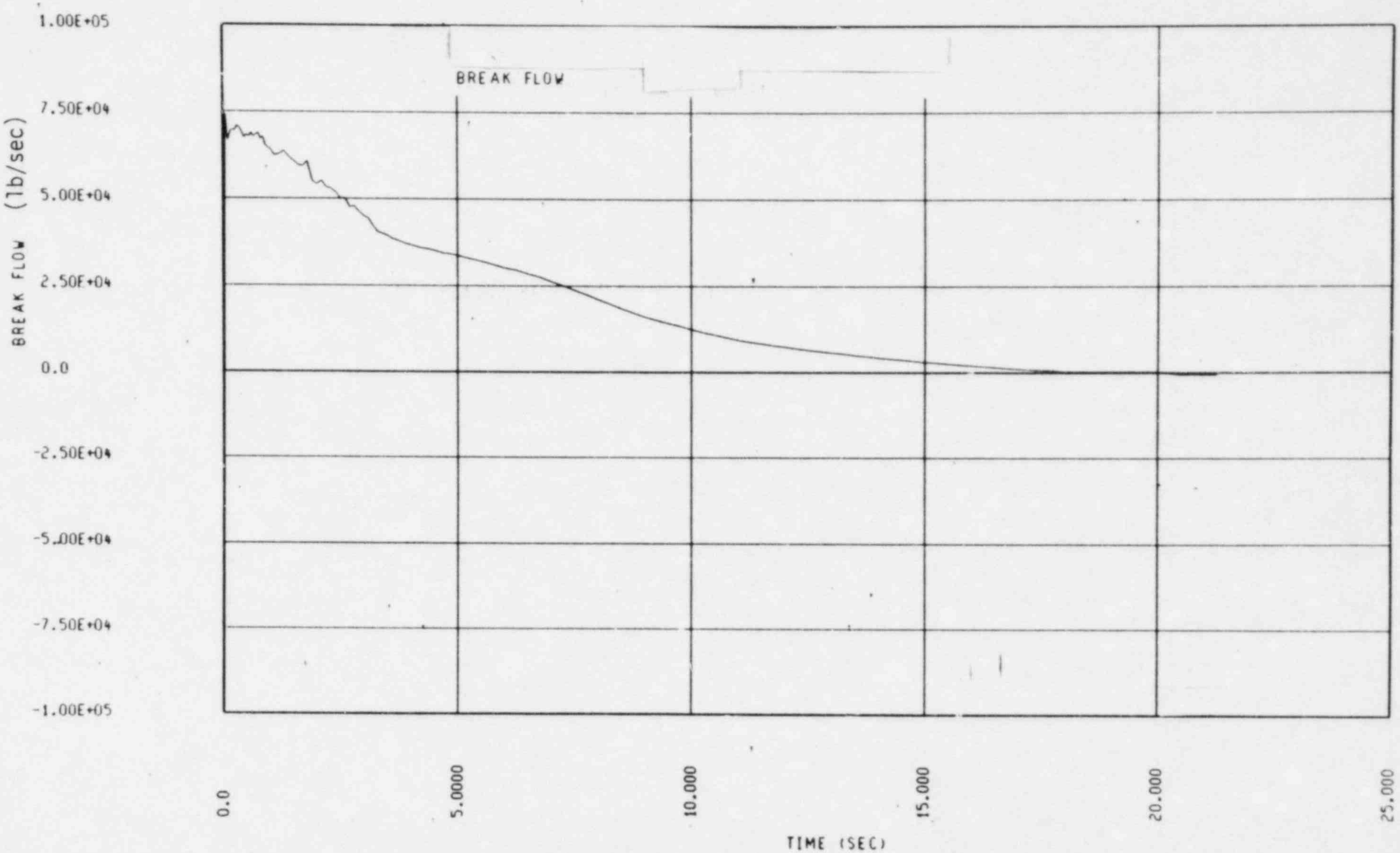

Figure 8 - Mass Flow Rate into Containment During Blowdown 0.6 DECLG

Figure 9 - Energy Flow Rate into Containment During Blowdown, 0.6 DECLG

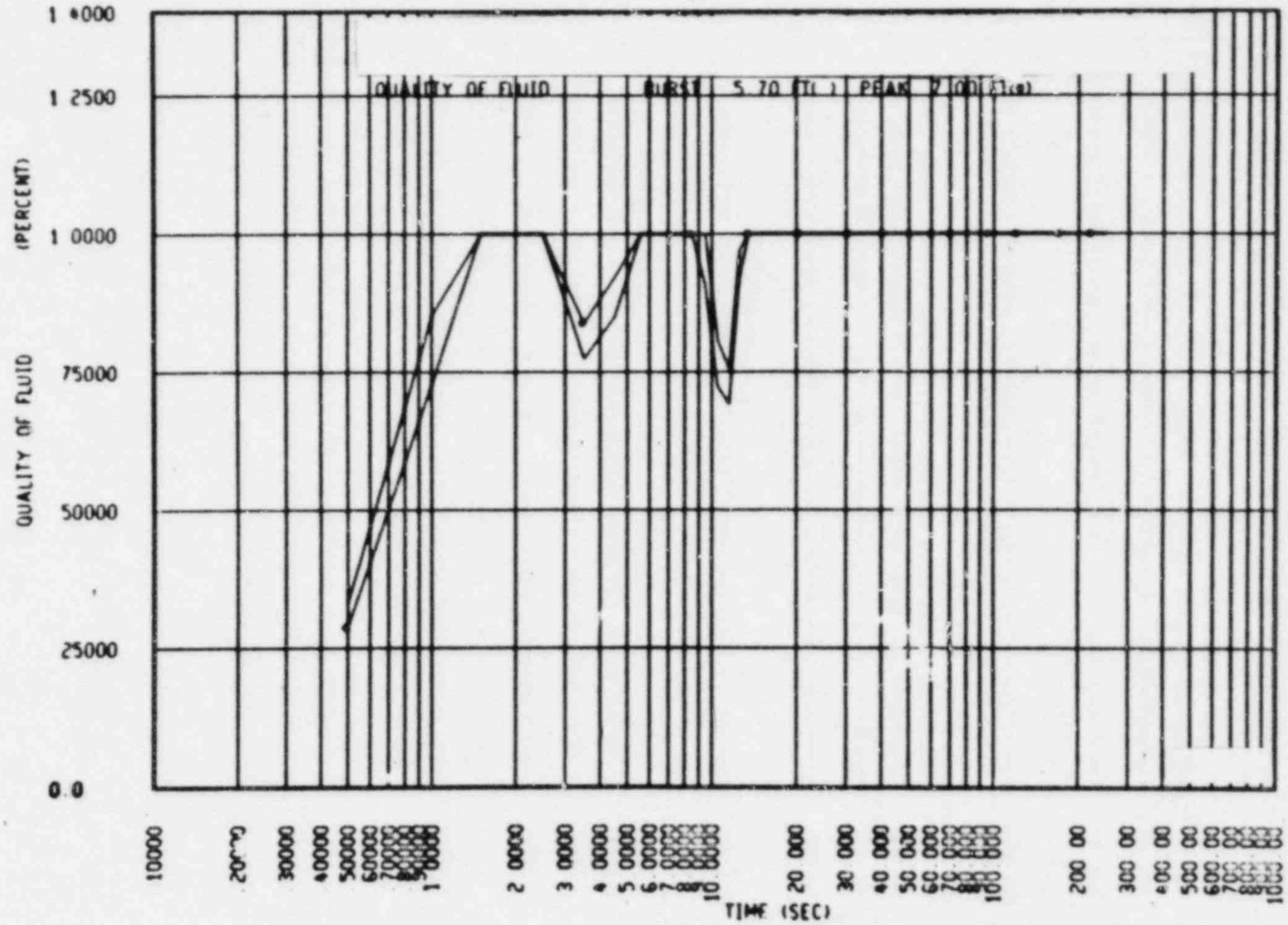

Figure 10 - Fluid Quality in the Hot Assembly, 0.6 DECLG

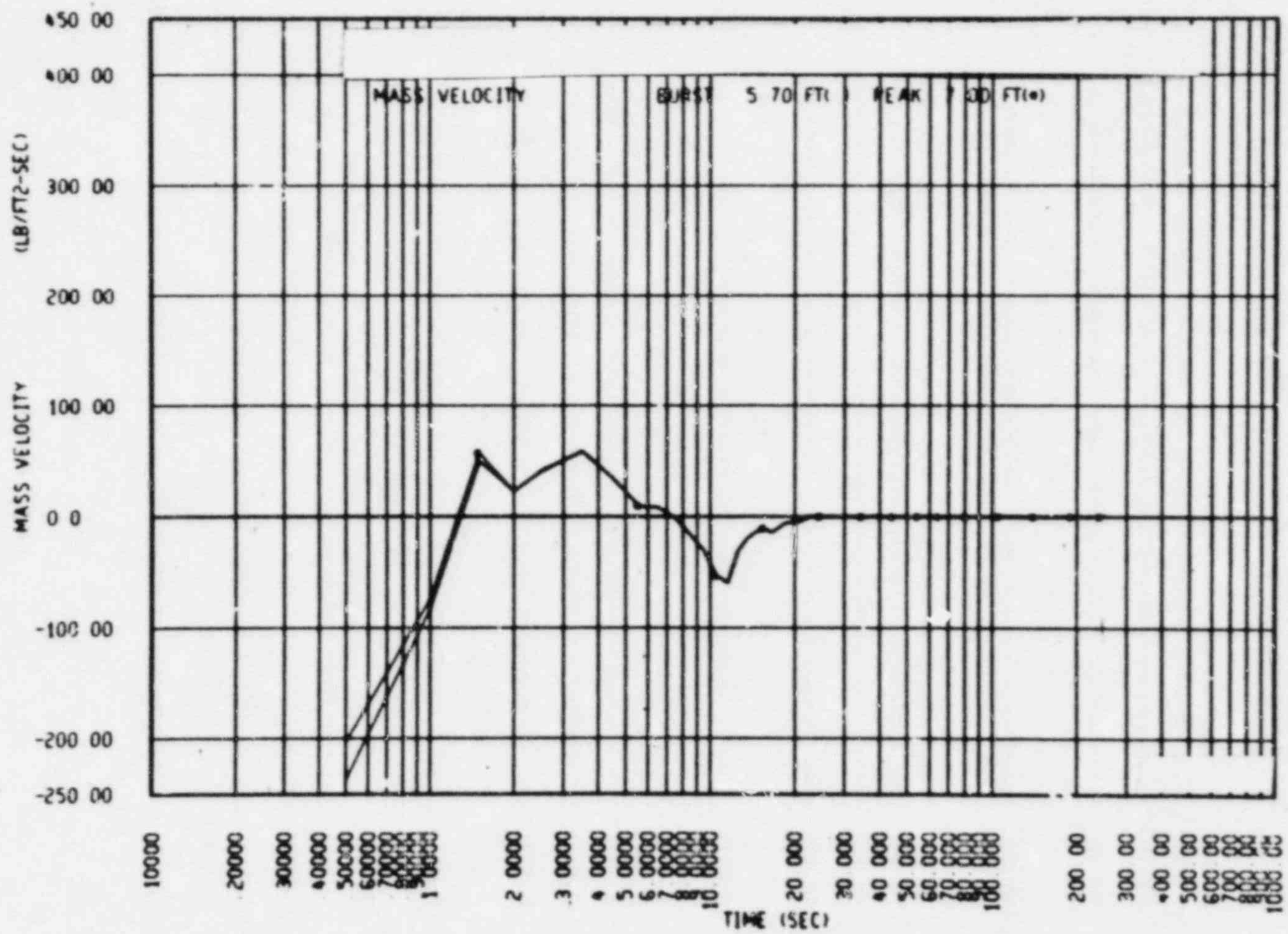

Figure 11 - Mass Velocity, 0.6 DECLG

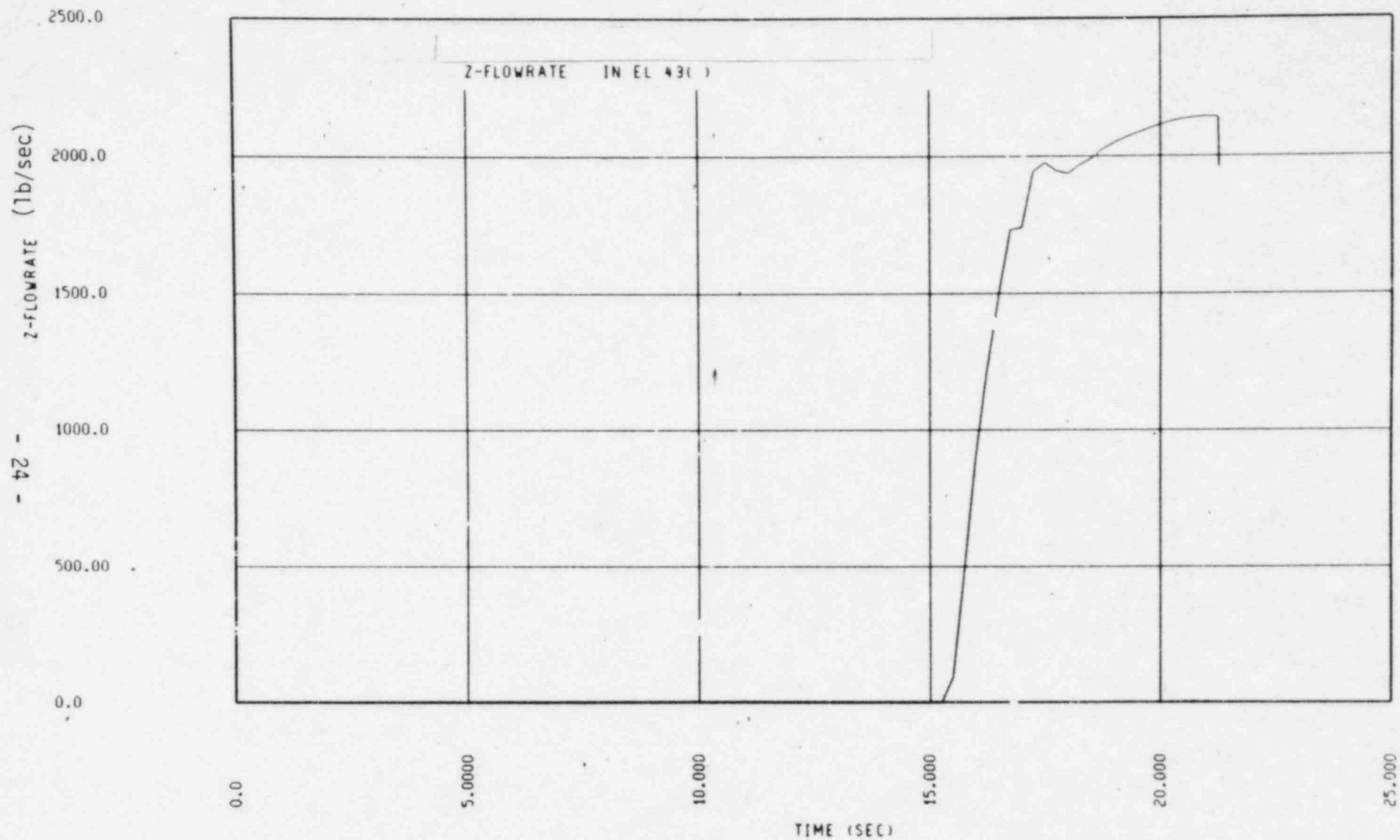

Figure 12 - Safety Injection Tank Flow Rate, 0.6 DECLG

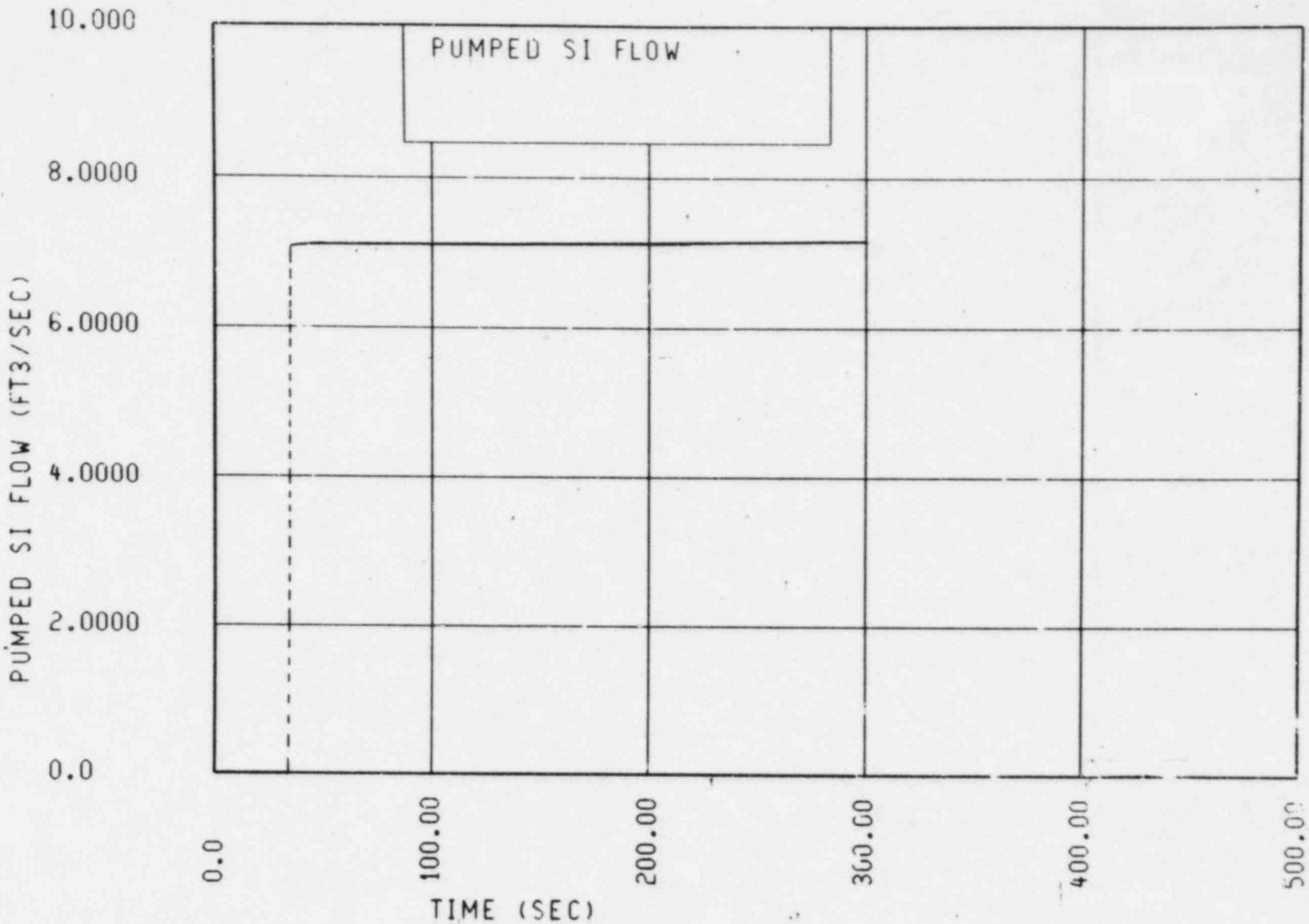

Figure 13 - Pumped SI Flow During Reflood, 0.6 DECLG

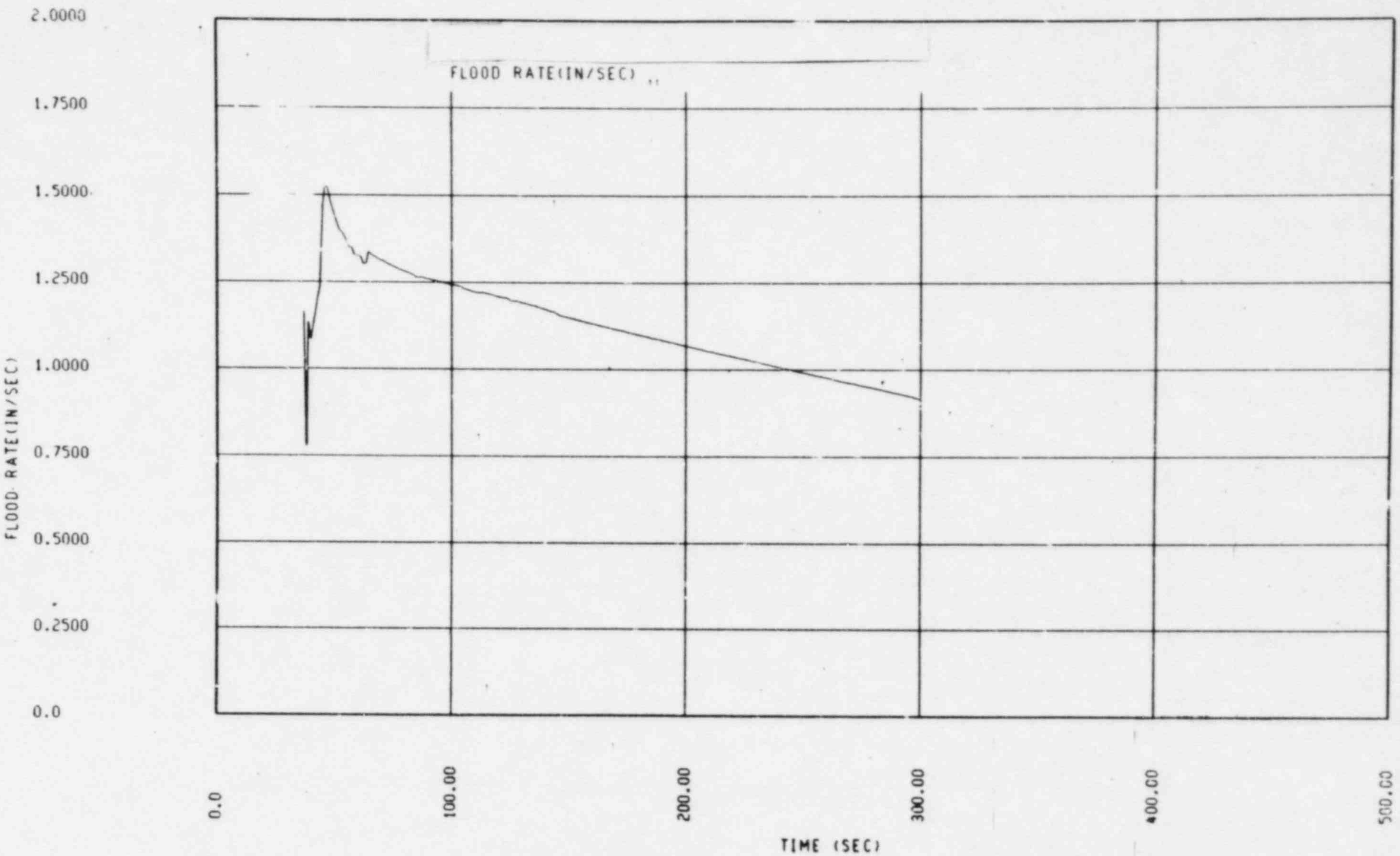

Figure 14 - Core Reflooding Rate

Docket No. 50-336

Attachment 3

Millstone Nuclear Power Station, Unit No. 2

Small Break Loss-of-Coolant Accident

Reevaluation

October, 1982

Small Break Loss-of-Coolant Accident Analysis Reevaluation

Northeast Nuclear Energy Company (NNECO) provided the NRC Staff with the results of the small break LOCA analysis conducted for Cycle 3 operation in Reference (1). The results of this analysis have been demonstrated to be applicable and appropriate for cycles 4 and 5 pending the completion of small break LOCA model revisions by our fuel vendor. Small break LOCA model revisions are due to be submitted, in fulfillment of Item II.K.3.30 of NUREG-0737, in March, 1983 (Reference (2)).

During the Cycle 4/5 refueling outage, unanticipated steam generator tube corrosion in the form of pitting necessitated the plugging of approximately seven hundred (700) tubes. NNECO evaluated the resulting effects of reduced primary coolant flow and heat transfer area on the docketed small break LOCA analysis results and concluded in Reference (3) that continued plant operation was bounded by the assumptions utilized in the Reference (1) analysis.

Anticipating the potential for additional steam generator tube plugging, NNECO has performed evaluations to demonstrate the acceptability of the current small break LOCA analysis and results for up to an additional 9.2% reduction in heat transfer area (approximately 3150 plugged tubes, total) and a reactor coolant flow rate of 350,000 gpm, for Cycle 6 operation. This evaluation addresses the impact of these changes on the small break ECCS performance for the limiting small break, the 0.1 ft^2cold leg break. The reduction in heat transfer area is addressed first followed by a discussion of the effect of the reduction in primary coolant flow rate.

Reduced Steam Generator Heat Transfer Area

In performing the evaluation, a total steam generator heat transfer area of 145,000 square feet, which corresponds to an additional reduction of 9.2% in heat transfer area, was chosen for conservatism. Using this conservatively reduced heat transfer area, the analysis determined the maximum increase in peak clad surface temperatures to be 13.2°F for the limiting 0.1 square foot cold leg break at the pump discharge. This insensitivity is not surprising since for that portion of the transient during which steam generator heat transfer is important, the core power is characterized by fission product decay heat generation wherein the power levels are less than 5% since plant trip will have occurred.

Reference 4 presents an analysis applicable to the CE 2700 Mwt class of plants in which complete loss of one steam generator was assumed. The results of the Reference 4 analysis (Case 17) demonstrated that the effect of a loss of one steam generator is insignificant for small breaks requiring steam generator heat transfer and includes break sizes 0.02 square feet and smaller. The 0.02 square foot break is well below the 0.1 square foot limiting small break for Millstone Unit 2.

Since the peak clad temperature for the 0.1 square foot break is 1971°F, in Reference 1, the expected peak clad temperature response due to the reduced steam generator area will be less than 1984.2°F, which is still well below the 10 CFR50.46 limit of 2200°F for acceptability.

Reduced Primary Flow

As noted above, the reduced primary system flow of 350,000 gpm represents a 5.4% reduction relative to the 370,000 gpm flow utilized in the licensing analysis of Reference 1. The effect of the reduced flow will increase the initial fuel stored energy by a small amount. However, this small increase in initial fuel stored energy will have no effect on peak clad temperatures for the limiting 0.1 square foot break since all of the initial fuel stored energy is removed from the fuel rod prior to uncovering the core when the rod heatup begins. That is, uncovering of the core is not initiated until after 500 seconds following the opening of the break. Since reactor trip occurs within the first 15 seconds, the ensuing flow coastdown over the next 500 seconds is more than sufficient to remove the initial fuel stored energy. As illustrated in Reference (1) for the 0.1 square foot break, the clad temperature prior to uncovering of the core is less than 10°F above the coolant temperature. At this time, the fuel pellet temperatures and hence clad temperatures are only a function of the decay heat generation rate which remains unchanged in the evaluations. Thus, increases in initial fuel stored energy resulting from the 5.4% reduction in flow rate, will not affect peak clad surface temperatures since the fuel temperature distribution will always subside to that distribution characterized by decay heat generation due to the large flow coastdown time prior to core uncovery.

SUMMARY

An evaluation has been performed to address the impact of a reduction in steam generator heat transfer area and a reduction in primary system flow rate on the small break LOCA ECCS performed for Millstone Unit 2. The results of the evaluation demonstrated that for the conservatively assumed reduction in area of 9.2%, an insignificant increase in peak clad temperature of only 13.2°F resulted. This insensitivity is consistent with an evaluation applied to the 2700 MWT class plants reported in Reference 4 which demonstrated that a 50% reduction in heat transfer area does not affect the small break LOCA performance.

The 5.4% reduction in flow also does not affect small break LOCA ECCS performance for the 0.1 square foot break due to the lengthy 500-second initial flow coastdown period during which all of the initial fuel stored energy is removed from the fuel rods. Since all of the initial fuel stored energy is removed from the rods prior to uncovering of the core, the small increase in initial fuel stored energy due to the lower flow will have no effect on peak clad temperature for the limiting 0.1 square foot break.

Based on the evaluation contained herein, the effect of reduction in steam generator heat transfer area and primary system flow rate will not impact the Millstone Unit 2 small break LOCA ECCS performance. It can therefore be concluded that operation of Millstone Unit 2 at a power level of 2754 MWT (102% of 2700 MWT) with a reduction in steam generation heat transfer area of 9.2% and a reduction in primary system flow rate of 5.4% is acceptable.

References: (1) W. G. Counsil letter to R. Reid, dated March 22, 1979.

(2) W. G. Counsil letter to R. A. Clark, dated September 2, 1982.

(3) W. G. Counsil R. A. Clark, dated February 4, 1982.

(4) CEN-114, Review of Small Break Transient in Combustion Engineering Nuclear Supply Systems, July 1979.