Docket No. 50-336

Attachment 1

Millstone Nuclear Power Station, Unit No. 2

Reload Safety Analysis

April, 1983

## TABLE OF CONTENTS

# LIST OF TABLES

## LIST OF FIGURES

# 1.0 INTRODUCTION AND SUMMARY

## 1.1 OBJECTIVES

This report presents an evaluation for Millstone Nuclear Power Station Unit 2, Cycle 6, which demonstrates that the core reload will not adversely affect the safety of the plant. This evaluation was accomplished utilizing the methodology described in Reference 1.

Based upon the above referenced methodology, only those incidents analyzed and reported in the Basic Safety Report [2] (BSR) which could potentially be affected by fuel reload and tube plugging have been reviewed for the Cycle 6 design described herein. The results of new analyses are included and the justification for the applicability of previous results for the remaining incidents is provided.

## 1.2 GENERAL DESCRIPTION

The Millstone II reactor core is comprised of 217 fuel assemblies arranged in the configuration shown in Figure 1. Each fuel assembly has a skeletal structure consisting of five (5) zircaloy guide thimble tubes, nine (9) Inconel grids, a stainless steel bottom nozzle, and a stainless steel top nozzle. One hundred seventy-six fuel rods are arranged in the grids to form a 14x14 array. The fuel rods consist of slightly enriched uranium dioxide ceramic pellets contained in Zircaloy-4 tubing which is plugged and seal welded at the ends to encapsulate the fuel.

Nominal core design parameters utilized for Cycle 6 are as follows:

| | |
|---|---|
| Core Power (Mwt) | 2700 |
| System Pressure (psia) | 2250 |
| Reactor Coolant Flow (GPM) | 350,000* |
| Core Inlet Temperature ($^{o}F$) | 549 |
| Average Linear Power Density (kw/ft) (based on best estimate hot, densified core average stack height of 136.4 inches) | 6.065 |

*Minimum guaranteed safety analysis flow-assumes plugging level of 2500 steam generator tubes

The core loading pattern for Cycle 6 is shown in Figure 1. Twenty-four (24) interior feed assemblies containing 2.7 w/o U235 and forty-eight (48) peripheral feed assemblies containing 3.2 w/o U235 are replacing seventy-two (72) Combustion Engineering (CE) batch E assemblies. The batch B assembly in Cycle 5 is replaced by a batch B assembly that was discharged at the end of Cycle 1.

A summary of the Cycle 6 fuel inventory is given in Table 1.

## 1.3 CONCLUSIONS

From the evaluation presented in this report, it is concluded that the Cycle 6 design does not result in the previously acceptable safety limits for any incident to be exceeded. This conclusion is based on the following:

1. Cycle 5 expected burnup of 11,500 $^{+0}_{-1000}$ MWD/MTU.

2. There is adherance to plant operating limitations as given in the Technical Specifications.

# 2.0 MECHANICAL DESIGN

## 2.1 GENERAL DISCUSSION

The mechanical design of the Cycle 6 fuel assemblies is essentially identical to that of the Cycle 5 assemblies. The Westinghouse fuel assemblies are designed to be fully compatible with all resident Millstone 2 fuel assemblies and core components (e.g. adequate clearances for insertion of CEA's, plugging devices, etc.). Table 1 summarizes pertinent design parameters of the various fuel regions.

## 3.0 THERMAL AND HYDRAULIC DESIGN

A description of the thermal and hydraulic design of the Westinghouse Millstone II reload fuel assembly to be utilized in Cycle 6 is given in Chapter 3 of the BSR.

As discussed in the BSR, the Westinghouse fuel assemblies have been designed and shown through testing to be hydraulically compatible with all resident Millstone II fuel assemblies.

The DNB analyses for Cycle 6 were performed for a minimum reactor coolant flow rate of 350,000 gpm and a radial peaking factor, $F_r$, of 1.565. This is a decrease in the flow rate and peaking factor assumed in the Cycle 4 and initial Cycle 5 analyses of 370,000 gpm and 1.63. As indicated by the power and flow sensitivities reported in the Cycle 4 Reload Safety Evaluation Report (Reference 6) a flow reduction can be offset by a power (or $F_r$) reduction in a 2:1 ratio to maintain a constant DNBR. Thus the reduction in flow has been more than offset by the reduction in radial peaking factor and this has been confirmed in the Cycle 6 analyses. A reduction in flow from 370,000 gpm to 362,600 gpm and a conservative reduction in $F_r$ from 1.63 to 1.597 was previously implemented during Cycle 5 operation. The Cycle 6 analysis takes a partial credit of 3.0% of the net conservatism which exists between convoluting and summing the uncertainties of various measured plant parameters in power space. This partial credit was applied in previous cycles and is discussed in more detail in the Cycle 4 Reload Safety Evaluation Report (Reference 6).

## 4.0 NUCLEAR DESIGN

The Westinghouse nuclear design procedures, computer programs, and calculation models utilized in the Millstone II, Cycle 6 reload design are presented in the BSR. Similar to the Cycle 4 and Cycle 5 evaluations[6,7], Cycle 6 accident simulations take credit for the variable high power trip by terminating accidents 5% above the variable high power trip. Also $P_L$ values (see BSR Section 6.0) are computed only if the maximum allowed power density of 21 kw/ft is exceeded.

The Cycle 6 core loading results in a maximum linear heat rate of less than 15.6 kw/ft at all fuel heights at rated power. Table 2 provides a summary of changes in the Cycle 6 kinetics characteristics compared with the current limit based on the reference safety analysis.[2][6][7] It can be seen from the table that all of the Cycle 6 values fall within current limits. Table 3 provides the control rod worths and requirements at the most limiting condition during the cycle. The required shutdown margin is based on accident analyses presented in Section 5.0.

# 5.0 ACCIDENT ANALYSIS

## 5.1 INTRODUCTION AND SUMMARY

The power capability of Millstone II is evaluated considering the consequences of those incidents examined in the BSR[2], using the previously accepted design basis specified in Section 1.2. It is concluded that the core reload will not adversely affect the ability to safely operate at 100% of rated power during Cycle 6. For the overpower transient, the fuel centerline temperature limit of 4700°F can be accommodated with margin in the Cycle 6 core. The burnup dependent densification model[3] was used for fuel temperature evaluations. The LOCA limit at rated power can be met by maintaining peak linear heat rates at or below 15.6 kw/ft.

## 5.2 ACCIDENT EVALUATION

The effects of the reload on the design basis and postulated incidents analyzed in the BSR[2] and updated in the Cycle 4 and 5 reload safety analyses[6,7] were examined. In most cases, it was found that the effects were accommodated within the conservatism of the initial assumptions used in the previous safety analysis. For those incidents which were reanalyzed, it was determined that the applicable design bases are not exceeded, and, therefore, the conclusions presented previously are still valid. For the reanalysis, parameters were selected to bound a plugging level of 2500 steam generator tubes.

A core reload can typically affect accident analysis input parameters in the following areas: core kinetic characteristics, shutdown margin, CEA worths, and core peaking factors. Cycle 6 parameters in each of these areas were examined as discussed below to ascertain whether new accident analyses were required.

A safety evaluation was performed to assess the impact of the increased level of steam generator tube plugging on the non-LOCA accident analyses. The approach used was to identify the important parameters for each transient, determine which of these parameters were affected

by the higher steam generator tube plugging levels and determine the impact on the safety analysis. In addition, a study was made of each currently applicable accident analysis to identify margins to safety limits which could be used to offset flow reduction penalties. Based on this evaluation, operation at a ~5% reduction in thermal design flow and a maximum of ~15% effective steam generator tube plugging level will not result in violation of the safety limits.

### 5.2.1 KINETICS PARAMETERS

A comparison of Cycle 6 kinetics parameters with the current limits, established by the BSR and Cycle 4 and 5 reload safety analyses, is presented in Table 2. None of the parameters in Table 2 exceeded the limiting range of values established by the previous safety analyses. However, the total trip reactivity curve as a function of position calculated for Cycle 6 was more limiting than that calculated for Cycle 5. The Cycle 6 curve was therefore used in all accident reanalysis.

### 5.2.2 SHUTDOWN MARGIN

Changes in minimum shutdown margin requirements may impact the safety analyses, particularly the steamline break and boron dilution accident analyses. Cycle 6 shutdown margin requirements are the same as Cycle 5.

### 5.2.3 CEA WORTHS

Changes in CEA worths may affect shutdown margin. Table 3 shows that the Cycle 6 shutdown margin requirements are satisfied.

### 5.2.4 CORE PEAKING FACTORS

Peaking factor evaluations were performed for rod out of position, and steam line break accidents to ensure that the DNB design limits are not exceeded. These evaluations were performed utilizing the existing transient statepoint information from the reference cycle (BSR) and

peaking factors determined for the reload core design. In each case, it was found that the peaking factor for Cycle 6 yielded results that were within the DNB design limits.

CEA peaking factors for Cycle 6 were within the reference cycle limits.

## 5.3 INCIDENTS REANALYZED

### 5.3.1 CEA WITHDRAWAL AT POWER

The CEA withdrawal at power accident was reanalyzed for Cycle 6 to assess the impact of increased steam generator tube plugging and the corresponding reduction in thermal design flow.

Table 4 gives the key parameters assumed in the analysis. Figure 2 shows the minimum DNBR as a function of reactivity insertion rate from initial full power operation for both minimum and maximum reactivity feedback cases.

The results show that the thermal margin low pressure trip provides core protection over the full range of reactivity insertion rates. The minimum DNBR remains above 1.30.

### 5.3.2 COMPLETE LOSS OF REACTOR COOLANT FLOW

The loss of flow transient is sensitive to the reduced thermal design flow because of the higher loop resistances which lead to a more rapid flow coastdown and the lower initial flows which reduce the margin to the 1.30 DNBR limit.

The loss of flow accident was reanalyzed for Cycle 6 to determine the effect of steam generator tube plugging on the minimum DNBR reached during the incident.

This analysis reflects the change in the trip reactivity curve, the reduction in normal operation radial peaking factor Fr, and appropriate fuel temperature curves for Cycle 6.

Table 5 gives the time sequence of events for this accident. The reactor coolant flow, nuclear power, heat flux, and DNB transients are shown in Figures 3, 4 and 5.

The results show that the reactor coolant pump speed sensing system provides sufficient protection against clad and fuel damage. The DNBR does not decrease below 1.30 during the transient.

5.3.3 STEAMLINE RUPTURE

The steamline rupture accident was reanalyzed for Cycle 6 to incorporate a more bounding trip reactivity curve and kinetics coefficients.

The transient which was reanalyzed is the most limiting case and assumes the steamline rupture of a pipe inside the containment at the outlet of the steam generator. The plant initially is at no load conditions with offsite power available.

The reanalysis was performed with the assumption that auxiliary feedwater flow is initiated automatically during this transient. It was assumed that 2800 gpm of auxiliary feedwater, 35% more than the maximum runout flow, is delivered to the affected steam generator three minutes after the beginning of the transient. This is conservative with respect to the expected time of auxiliary feedwater initiation since automatic actuation of the auxiliary feedwater system would occur on a low steam generator water level trip signal. The three minute time delay for auxiliary feedwater delivery is also conservatively small.

The assumption was also made that the minimum capability for injection of boric acid solution (1720 ppm) corresponds to the most restrictive single failure in the safety injection system. This corresponds to the flow delivered by one high pressure safety injection pump and one low pressure safety injection pump delivering full flow to the cold leg header.

The steam generator pressure, RCS temperature, pressurizer pressure, reactivity, and core heat flux transients for this case are shown in Figures 6-10. Table 6 lists the time sequence of events.

As shown in Figure 9 , the core returns to critical after CEA insertion (assuming the most reactive CEA is stuck in the withdrawn position). This is due to the high cooldown rate, in the presence of a negative moderator temperature coefficient, resulting from the steam discharge and feedwater addition. However, the addition of boron from the high pressure safety injection pump brings the core subcritical again. The peak heat flux attained during this transient is small, approximately 7 percent. By this time the faulted steam generator is essentially depressurized and the primary system cooldown is due mainly to the boiling of feedwater.

At 180 seconds, auxiliary feedwater flow is delivered to the faulted steam generator, which further cools down the RCS. However, the positive reactivity insertion due to this additional cooldown is offset by the addition of boron via safety injection, and the core remains subcritical.

Results show that the DNB margin design basis will not be violated. That is, DNB will not occur on at least 95 percent of the limiting fuel rods at a 95 percent confidence level.

## 6.0 REFERENCES


1. Bordelon, F. M., et. al., "Westinghouse Reload Safety Evaluation Methodology", WCAP-9273, March, 1978.

2. Millstone Unit 2, "Millstone Unit 2 Basic Safety Report", Docket No. 50-336, March, 1980.

3. Miller, J. V. (Ed), "Improved Analytical Model used in Westinghosue Fuel Rod Design Computations", WCAP-8785, October, 1976.

4. Hellman, J. M. (Ed.), "Fuel Densification Experiemental Results and Model for Reactor Operation", WCAP-8219-A, March 1975.

5. Letter, Counsil to Reed, Millstone Nuclear Power Station, Unit No. 2, Proposed License Amendment, Power Uprating, February 12, 1979

6. Letter, Counsil to Clark, Millstone Nuclear Power Station Unit No. 2, Cycle 4 Refueling - Reload Safety Analysis, June 3, 1980

7. Letter, Counsil to Clark, Millstone Nuclear Power Station Unit No. 2, Cycle 5 Refueling - Reload Safety Analysis, November 17, 1981.

TABLE 1

Millstone Unit 2 Cycle 6
Core Loading

| Region | Type | Number of Assemblies | Initial Enrichment w/o U235 | %Theoretical Density | BOC** Burnup Average (MWD/MTU) |
|---|---|---|---|---|---|
| B+ | CE | 1 | 2.336 | 95 | 17450 |
| F1 | W | 24 | 2.697 | 94.54 | 24020 |
| F2 | W | 48 | 3.297 | 94.87 | 22560 |
| G1 | W | 24 | 2.720 | 95.00 | 13940 |
| G2 | W | 48 | 3.191 | 94.70 | 9370 |
| H1 | W | 24 | 2.70 | 95* | 0 |
| H2 | W | 48 | 3.20 | 95* | 0 |

* The Region H1 and H2 densities are nominal. Average densities of 94.5 theoretical were used for Region H1 and H2 nuclear design evaluations.

** EOL Cycle 5 burnup assumed: 11,500 MWD/MTU.

TABLE 2

MILLSTONE UNIT 2 KINETICS CHARACTERISTICS

| | Current Limit | Cycle 6 |
|---|---|---|
| Most Positive Moderator Temperature Coefficient ($\Delta\rho$/°F) x $10^{-4}$ | +0.5 from 0 to 70% Power<br>+0.4 from 70 to 100% Power | +0.5 from 0 to 70% Power<br>+0.4 from 70 to 100% Power |
| Most Negative Moderator Temperature Coefficient ($\Delta\rho$/°F) x $10^{-4}$, ARI | -3.8 | -3.8 |
| Doppler Temperature Coefficient ($\Delta\rho$/°F x $10^{-5}$) | -1.2 to -1.92 | -1.2 to -1.92 |
| Delayed Neutron Lifetime $\beta_{eff}$ (%) | .479 to .634 | .479 to .634 |
| Prompt Neutron Fraction ($\mu$sec) | <32.2 | <32.2 |
| Maximum Differential Rod Worth of two CEA groups moving together at HZP (pcm/in) | 36.6 | 36.6 |

TABLE 3

SHUTDOWN REQUIREMENTS AND MARGINS

MILLSTONE UNIT 2 - CYCLE 6

| Control Rod Worth (%Δρ) | Cycle 5 | | Cycle 6 | |
|---|---|---|---|---|
| | BOC | EOC | BOC | EOC |
| All Rods Inserted | 7.84 | 8.67 | 7.94 | 8.65 |
| All Rods Inserted Less Worst Stuck Rod | 6.47 | 6.59 | 6.49 | 6.70 |
| (1) Less 10 Percent | 5.82 | 5.93 | 5.84 | 6.03 |
| **Control Rod Requirements** | | | | |
| Reactivity Defects (Combined Doppler, $T_{avg}$, Void and Redistribution Effects) | 1.94 | 2.64 | 1.93 | 2.69 |
| Rod Insertion Allowance | 0.36 | 0.36 | 0.41 | 0.41 |
| (2) Total Requirements | 2.30 | 3.00 | 2.34 | 3.10 |
| Shutdown Margin ((1) - (2)) (%Δρ) | 3.52 | 2.93 | 3.50 | 2.93 |
| Required Shutdown Margin (%Δρ) | 2.90 | 2.90 | 2.90 | 2.90 |

TABLE 4

PARAMETERS USED IN THE CEA WITHDRAWAL ANALYSIS

| Parameter | Units | Cycle 6 |
|---|---|---|
| Initial Core Power Level | MWt | 102 percent of 2700 |
| Core Inlet Coolant Temperature | °F | 551 |
| Reactor Coolant System Pressure | psia | 2200 |
| Moderator Temperature Coefficent | $10^{-4}\Delta\rho/°F$ | -2.5 and + 0.5 |
| Doppler Coefficient Multiplier | - | 1.15 and 0.85 |
| CEA Worth at Trip | $10^{-2}\Delta\rho$ | -2.9 |
| Reactivity Insertion Rate | $x10^{-4}\Delta\rho/sec$ | 0 to 2.44 |
| Holding Coil Delay Time | sec | 0.5 |
| CEA Time to 90 Percent Insertion (Including Holding Coil Delay) | sec | 2.75 |

TABLE 5

SEQUENCE OF EVENTS - LOSS OF COOLANT FLOW

Four pumps in operation, all pumps coasting down

| Event | Time (sec) |
|---|---|
| Loss of power to all pumps | 0.0 |
| Reactor coolant pump low speed setpoint reached | .91 |
| CEA's begin to drop | 1.56 |
| Minimum DNBR occurs | 3.7 |

TABLE 6

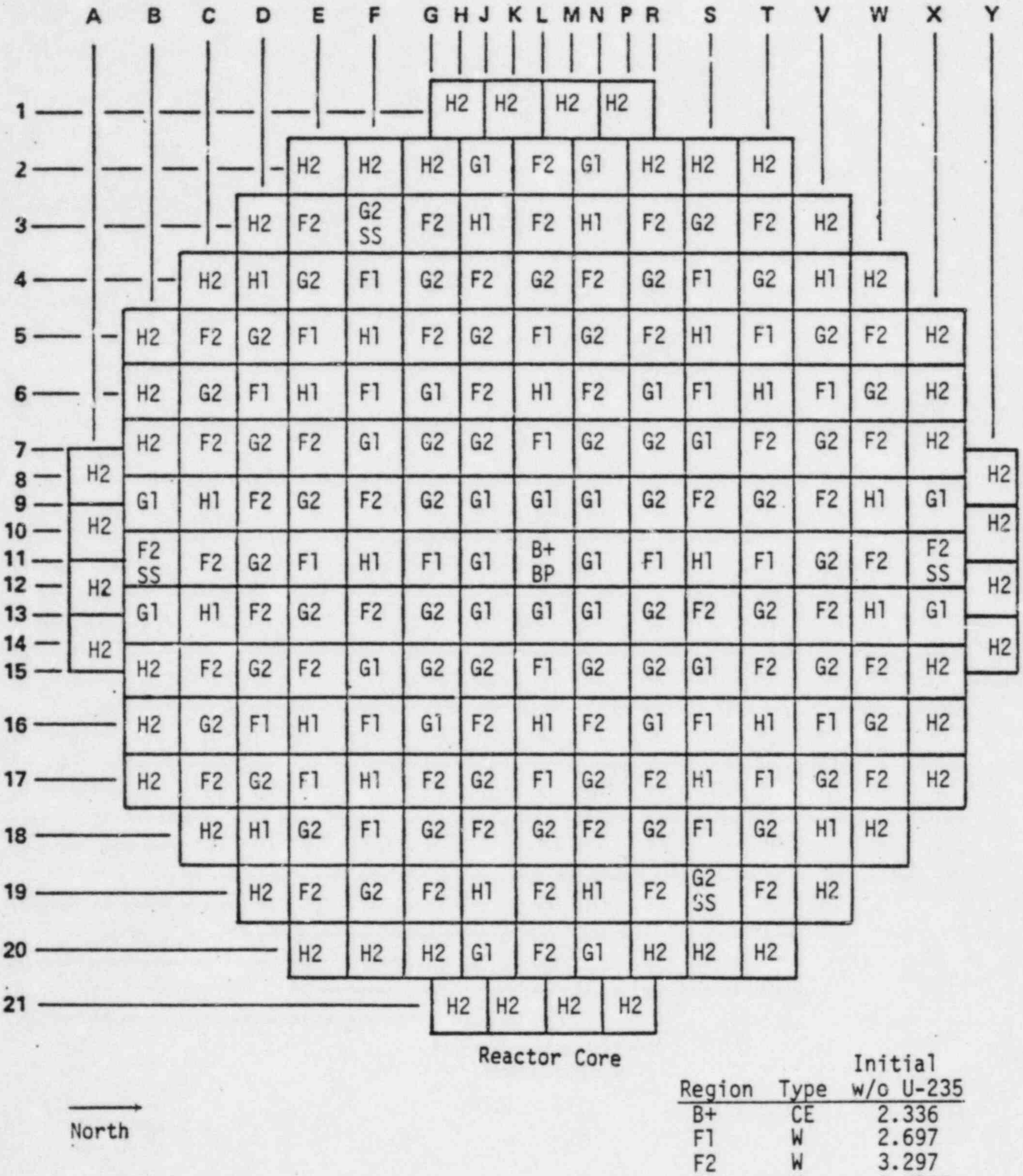
SEQUENCE OF EVENTS, STEAMLINE RUPTURE AT NO LOAD

| Time (Sec) | Event | Setpoint or Value |
|---|---|---|
| 0.0 | Steamline Rupture occurs | -- |
| 3.5 | Low Steam Generator Pressure trip signal occurs; main steam isolation begins | 478 psia |
| 4.4 | CEA's begin to drop into the core | -- |
| 10.4 | Main steamline isolation valves closed | -- |
| 11.5 | SIAS initiated on low RCS pressure | 1563 psia |
| 19.5 | Pressurizer empties | -- |
| 59.1 | Peak Reactivity | 0.275% $\Delta\rho$ |
| 139.1 | Peak Heat Flux | 7.1 percent power |
| 180.0 | Auxiliary feedwater initiated | -- |

FIGURE 1

CORE LOADING PATTERN

MILLSTONE UNIT 2 - CYCLE 6




| Region | Type | Initial w/o U-235 |
|---|---|---|
| B+ | CE | 2.336 |
| F1 | W | 2.697 |
| F2 | W | 3.297 |
| G1 | W | 2.7195 |
| G2 | W | 3.1909 |
| H1 | W | 2.70 |
| H2 | W | 3.20 |

SS - Secondary Source
BP - Burnable Poison

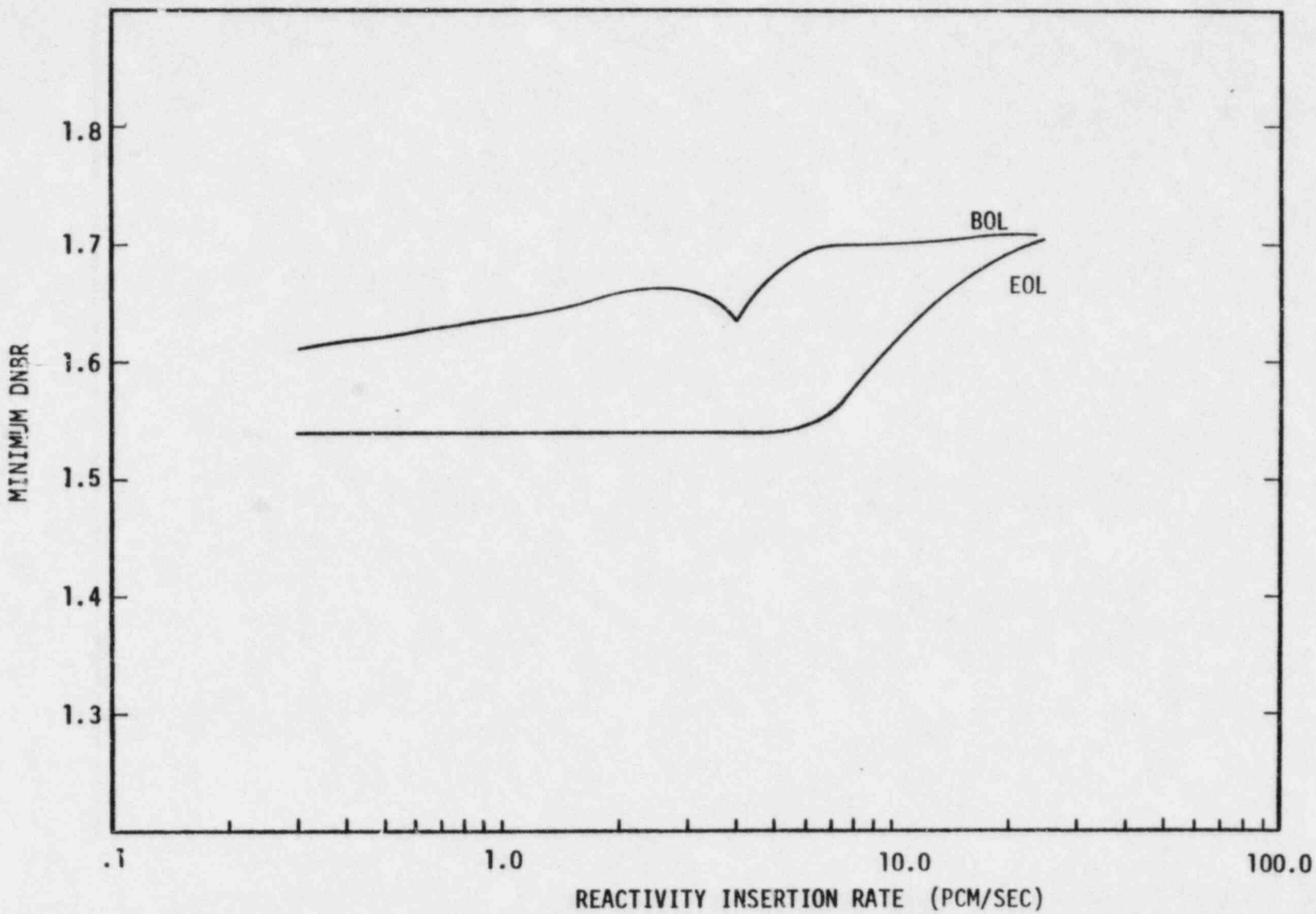



FIGURE 2 MILLSTONE 2 - SAFETY ANALYSIS
CEA WITHDRAWAL AT FULL POWER
REACTIVITY INSERTION RATE VS. DNBR

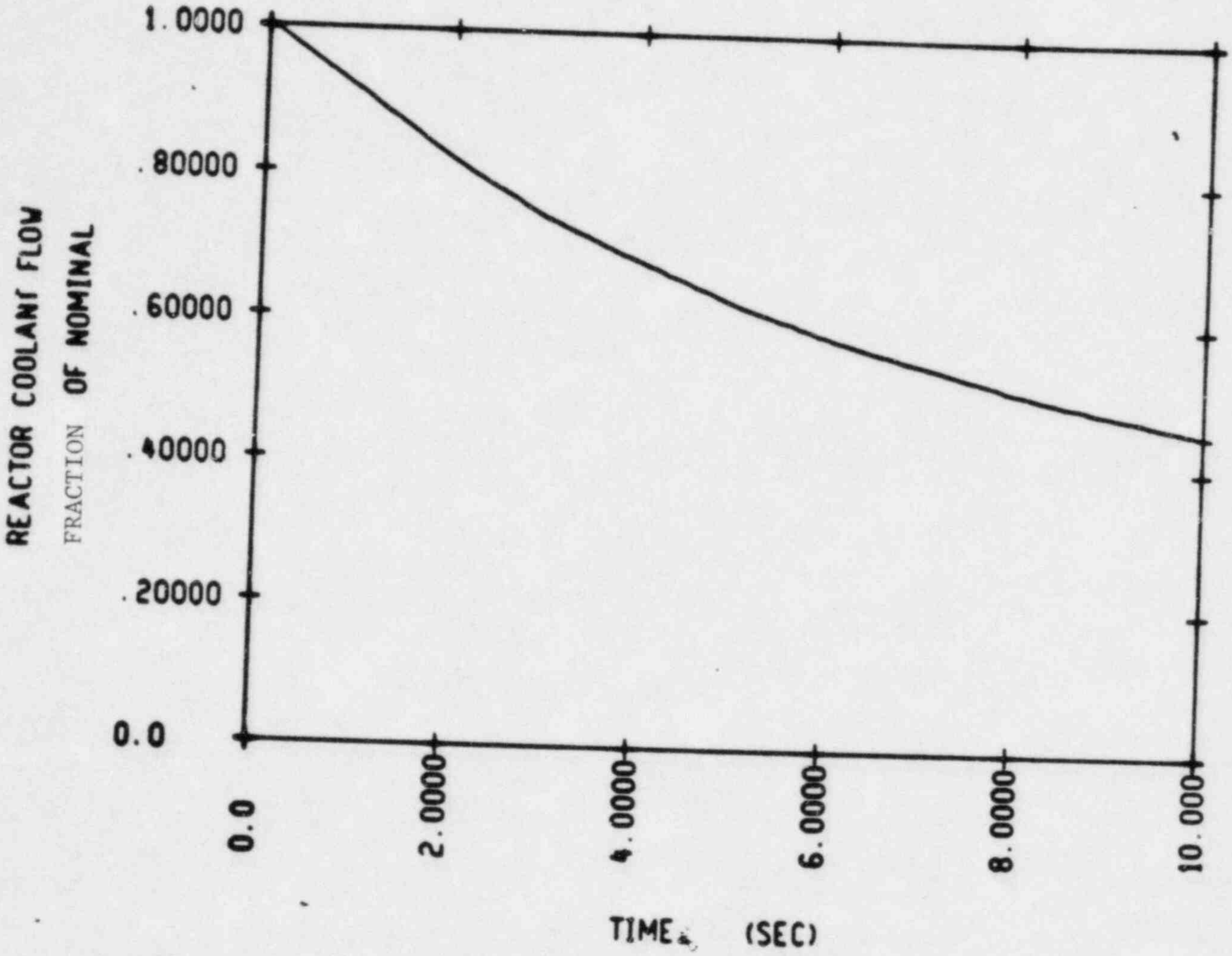



FIGURE 3: MILLSTONE 2
LOSS OF FLOW
REACTOR COOLANT FLOW VERSUS TIME

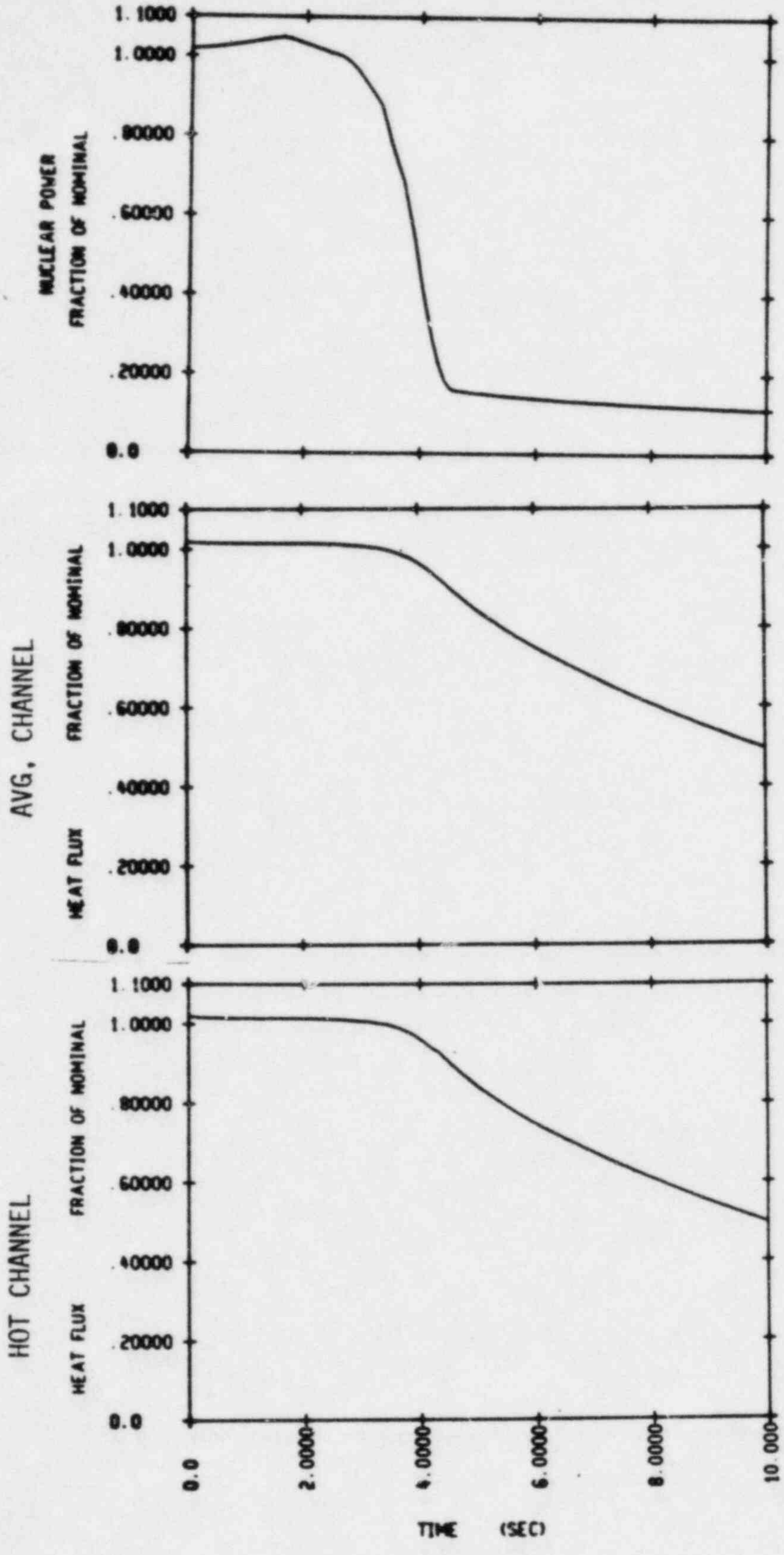



FIGURE 4: MILLSTONE 2
LOSS OF FLOW
NUCLEAR POWER AND HEAT FLUX VERSUS TIME

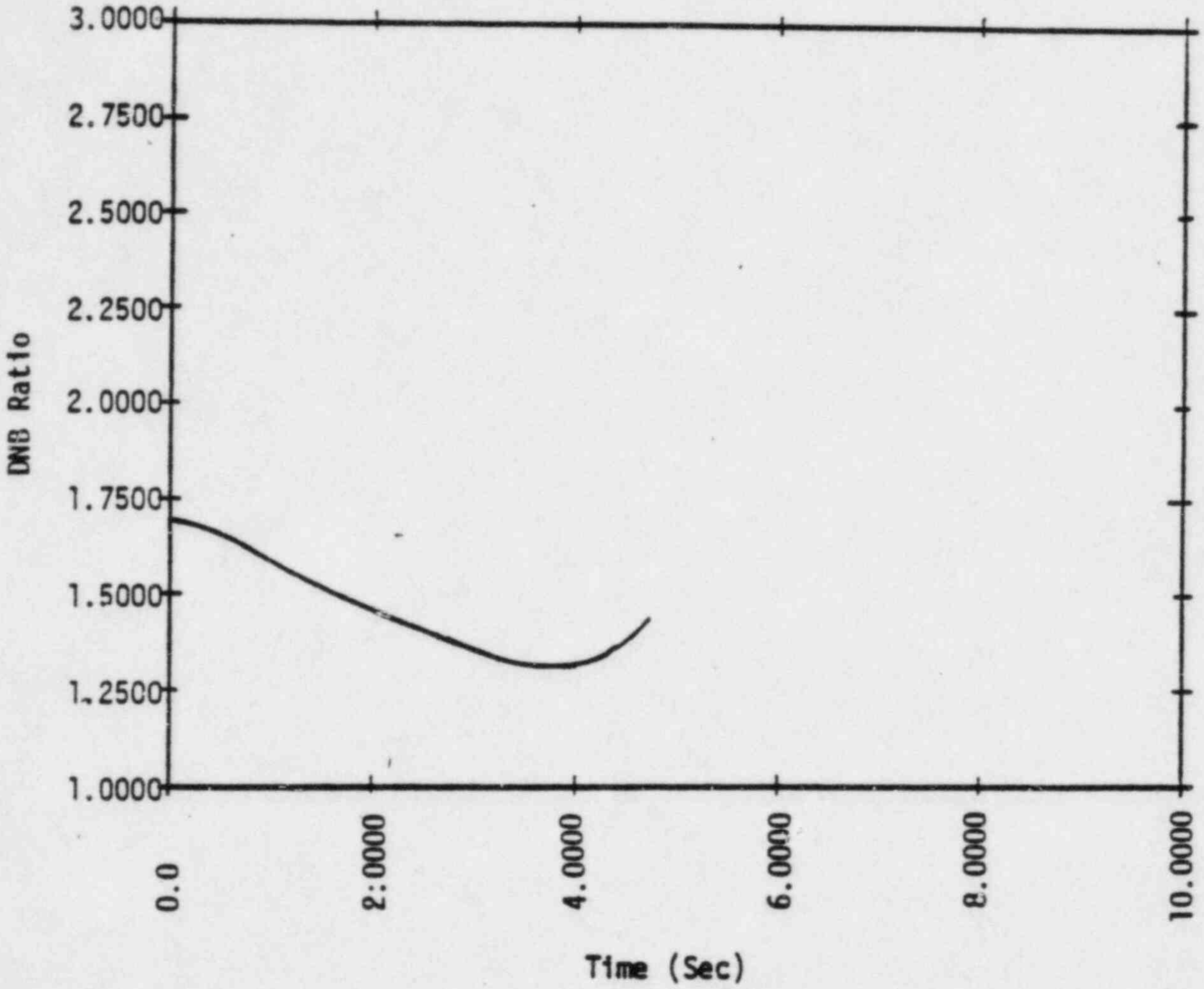



FIGURE 5: MILLSTONE 2
LOSS OF FLOW
DNBR VERSUS TIME

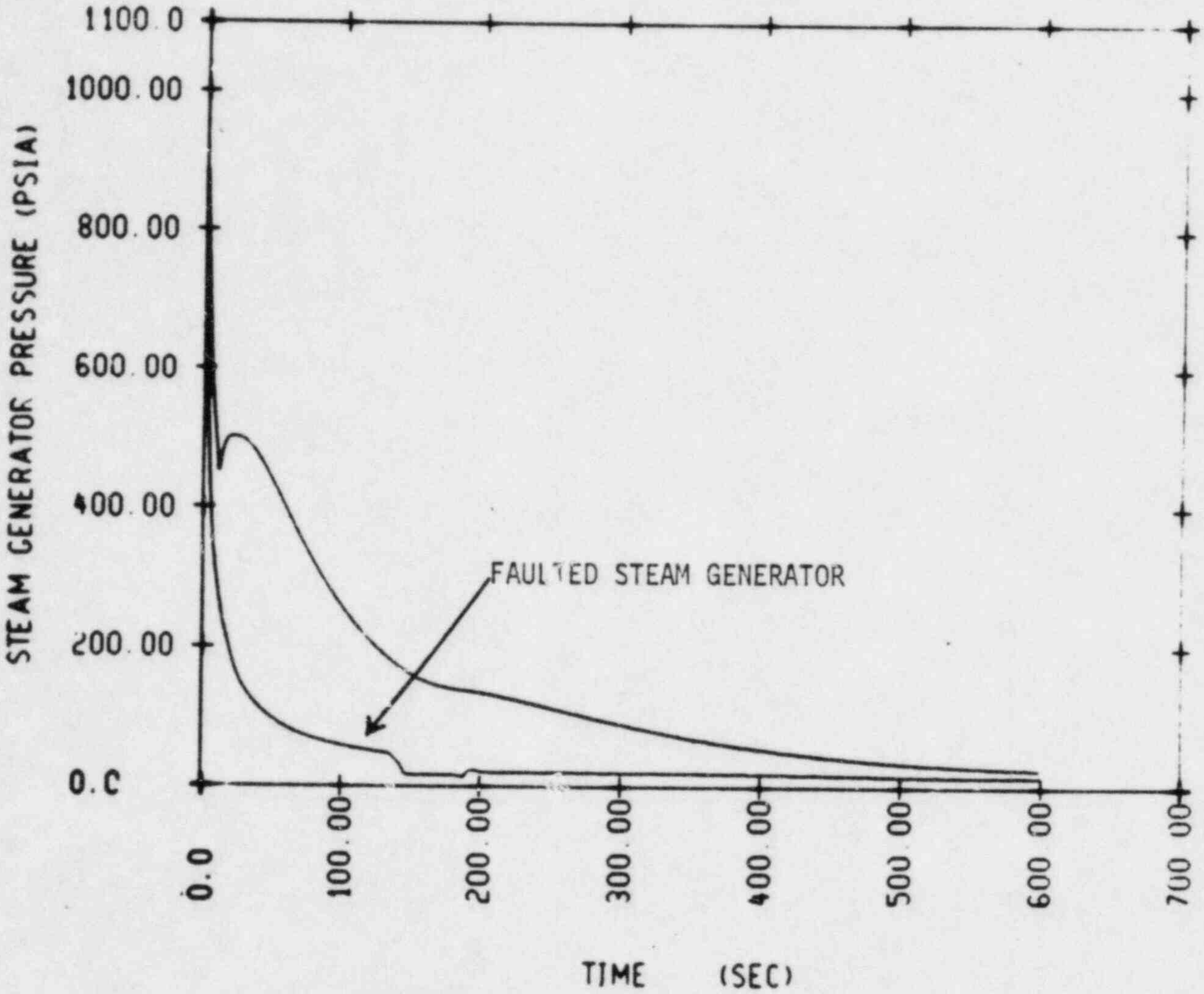



FIGURE 6: MILLSTONE 2
STEAMLINE RUPTURE
STEAM GENERATOR PRESSURE VERSUS TIME

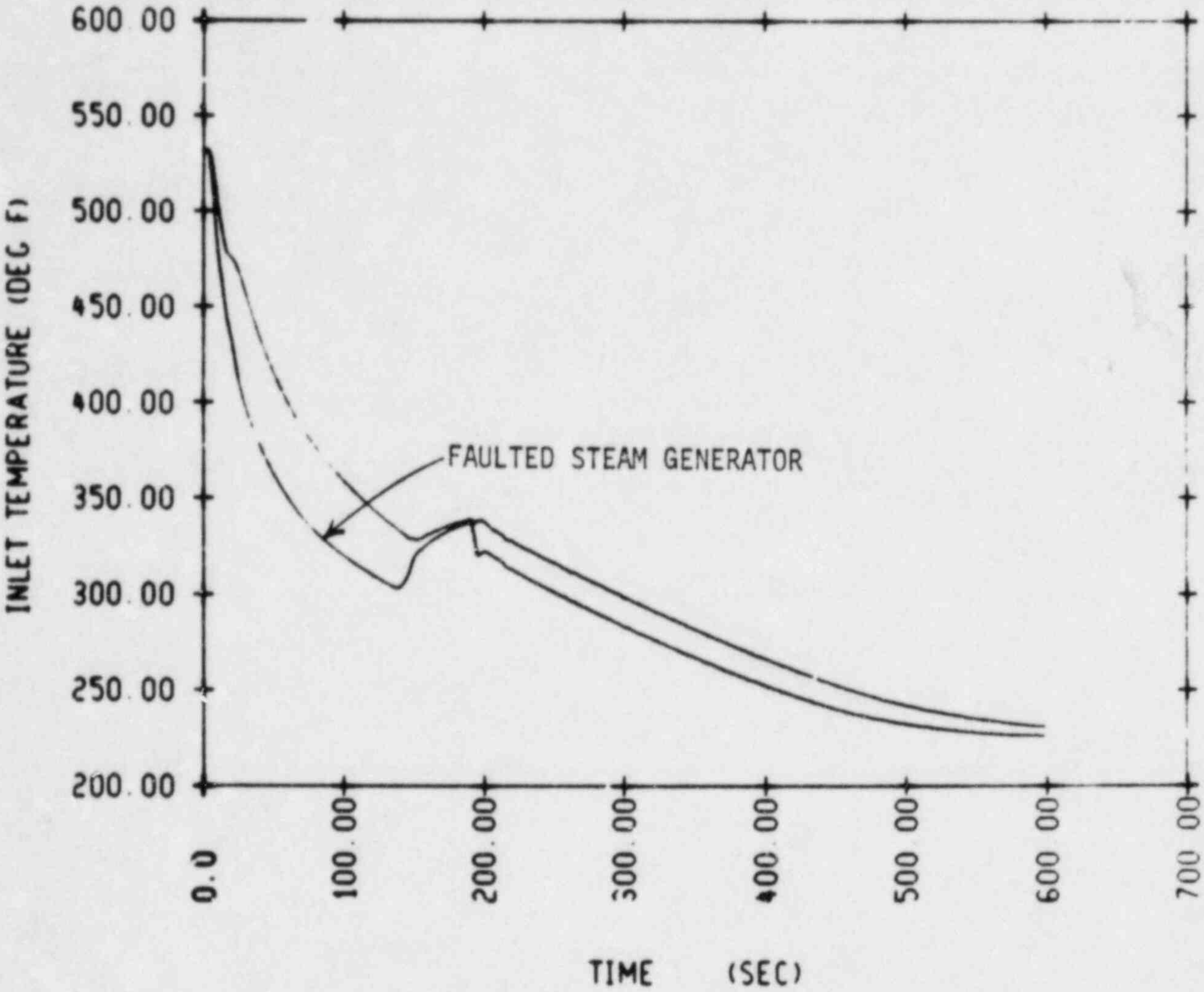



FIGURE 7: MILLSTONE 2
STEAMLINE RUPTURE
INLET TEMPERATURE VERSUS TIME

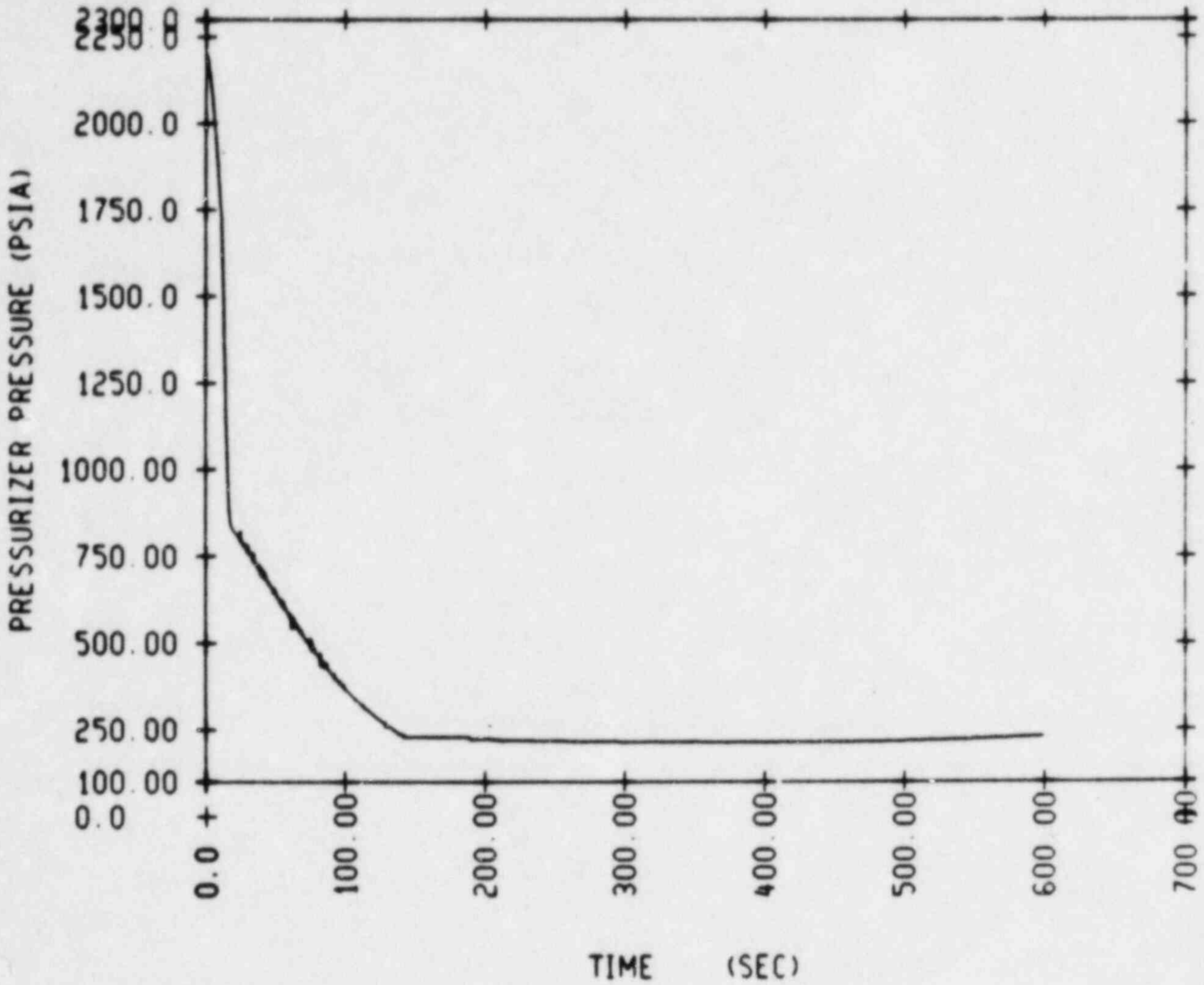



FIGURE 8: MILLSTONE 2
STEAMLINE RUPTURE
PRESSURIZER PRESSURE VERSUS TIME

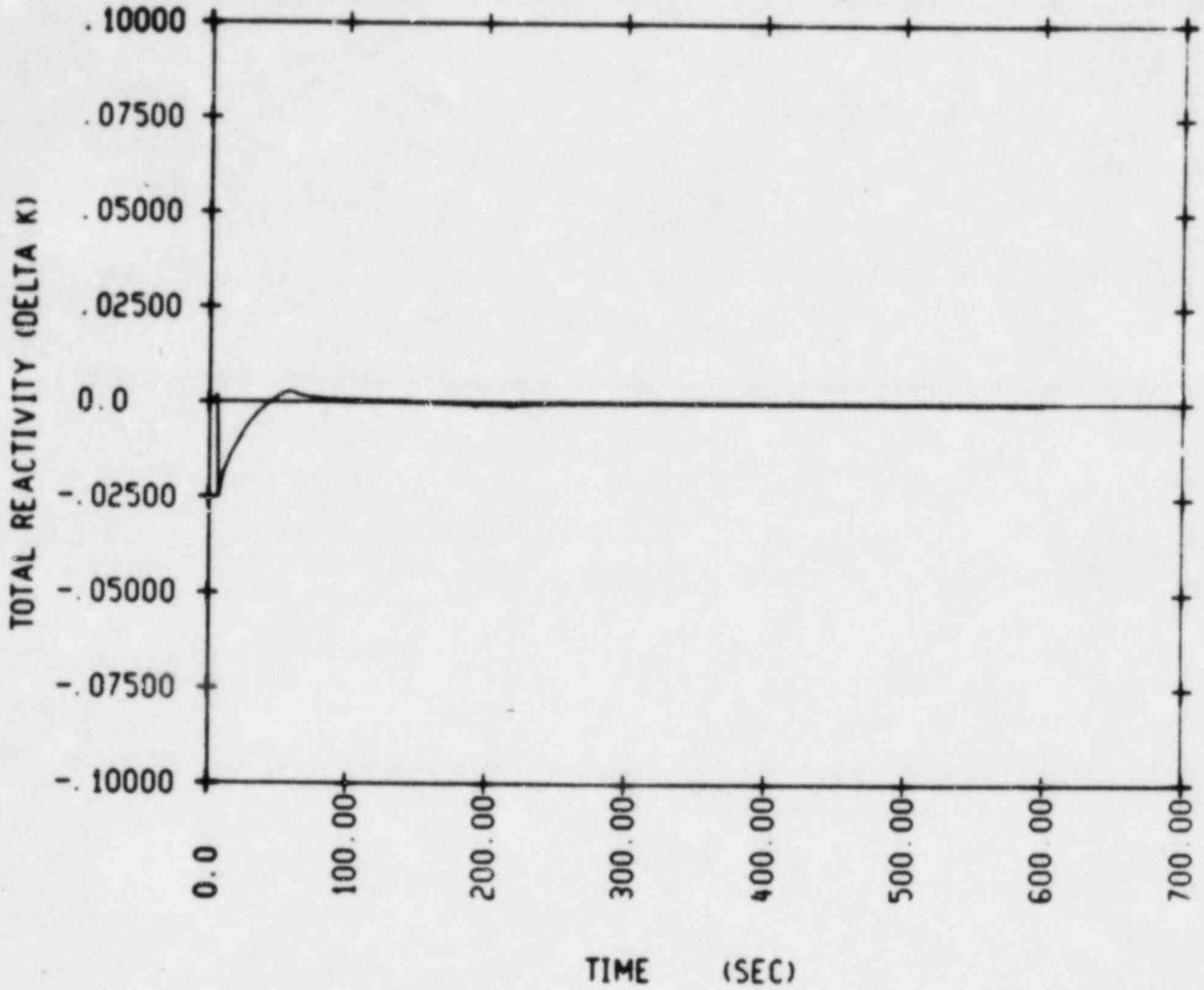



FIGURE 9: MILLSTONE 2
STEAMLINE RUPTURE
TOTAL REACTIVITY VERSUS TIME

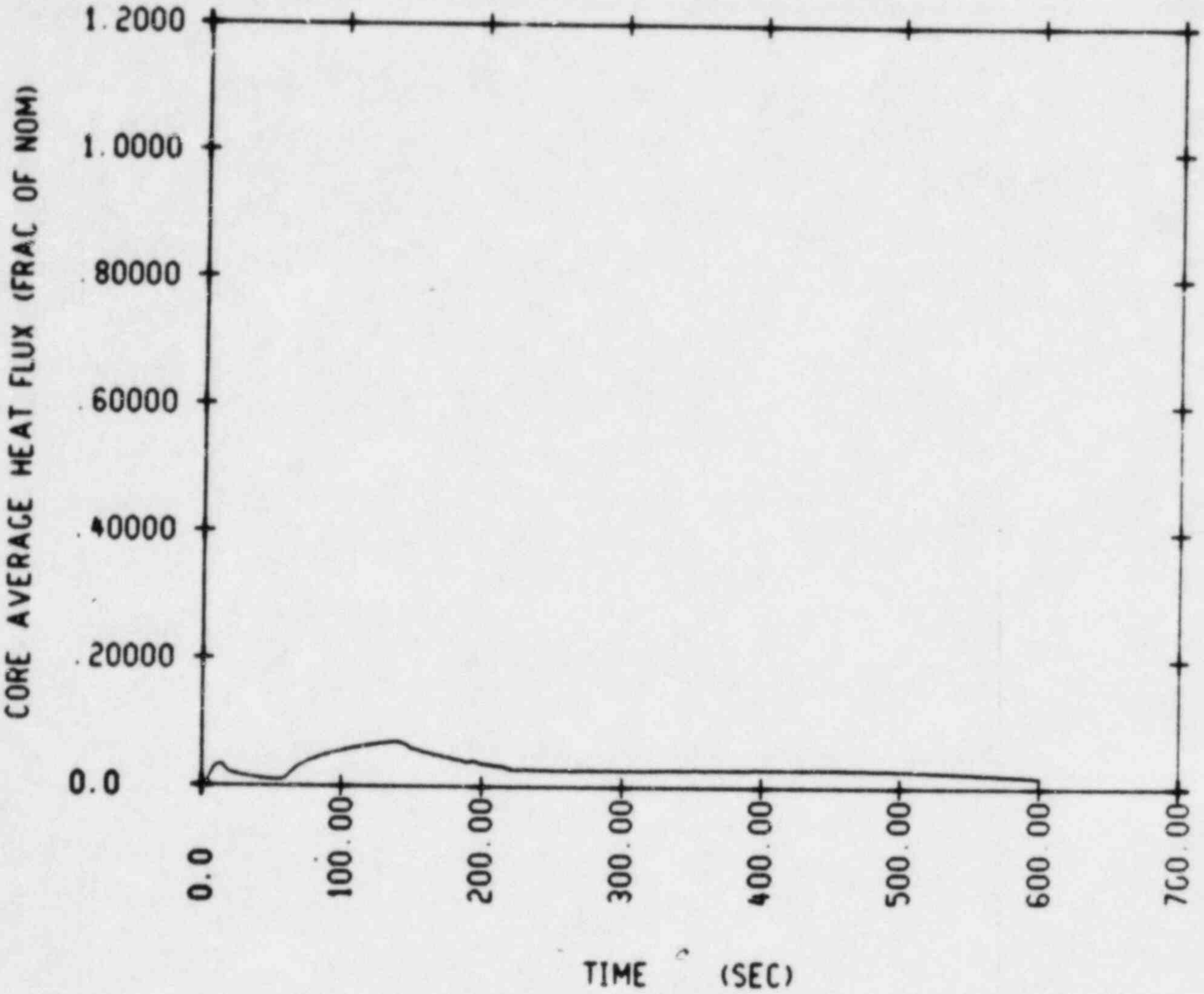



FIGURE 10: MILLSTONE 2
STEAMLINE RUPTURE
CORE AVERAGE HEATFLUX VERSUS TIME