Docket No. 50-336

Millstone Nuclear Power Station, Unit No. 2

Supplement to the

Reload Safety Analyses

for

Cycle 6 Operation

November, 1983

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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 redesign will not adversely affect the safety of the plant. This reload redesign is a result of the following:

1. Fuel rod failure (see LER83-19 filed with NRC July, 1983)
2. Removal of two damaged fuel assemblies
3. Removal of the thermal shield.

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

The core loading pattern for Cycle 6 is shown in Figure 1. Twenty-four (24) interior feed assemblies containing 2.7 w/o U-235 and forty-eight (48) peripheral feed assemblies containing 3.2 w/o U-235 are replacing seventy-two (72) Combustion Engineering (CE) batch E assemblies. Due to fuel defects in Cycle 5 and subsequent symmetry considerations, sixteen (16) interior feed assemblies containing 2.70 w/o, U-235, twenty (20) CE assemblies from Batch A and one (1) CE assembly from Batch B (these CE assemblies were discharged at the end of Cycle 1) are needed as well.

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 burnup of 11,500 MWD/MTU.

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

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

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. Based on observations reported in LER's numbers RO 50-336/83-25, 83-25/01-T, 83-26 and 83-26/01-T mechanical design changes were incorporated. These include modifications to the top nozzle holddown springs and flower in reload region H and future regions. These issues are discussed in detail in Reference 8. 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

There are no changes in the thermal and hydraulic analysis as a result of the Cycle 6 reload redesign. The results stated in Reference 7 remain the same.

4.0 NUCLEAR DESIGN

The Westinghouse nuclear design procedures, computer programs, and calculation models utilized in the Millstone II, Cycle 6 reload redesign are presented in the BSR. Similar to the Cycle 4 and Cycle 5 evaluations [5,6], 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.[2,5,6] It can be seen from the table that all of the Cycle 6 values fall within current limits with a small exception in the Doppler Temperature Coefficient and the delayed neutron fraction noted in Table 2. 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 [3,4] 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.

Only the loss of flow accident which changed from that stated in Reference 7 is updated herein.

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, 5 and 6 reload safety analyses [5,6,7] were examined. This is an update to include those incidents that were affected by the reload redesign. 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 the loss of flow incident which was reanalyzed, it was determined that the applicable design bases are not exceeded, and, therefore, the conclusions presented previously are still valid.

The effect of the removal of the thermal shield on Non-LOCA transients has been assessed. The primary result of this modification is to enlarge the downcomer region flow area and volume. The enlarged downcomer flow area and volume produces less resistance to flow in the downcomer area which increases core flow velocities. Increased flow results in a slight penalty for the system transient of cooldown events and a steamline break, however this slight

penalty would be more than offset by the resulting benefit with respect to DNB. Increased flow would also tend to cause a more rapid pump coastdown impacting transients such as the loss of flow/locked rotor transients, however this effect would be more than offset by the DNB benefit resulting from higher flow. The impact of this modification on the safety analysis assumptions and results presented in the BSR and subsequent reanalysis is negligible; therefore the conclusions of the previous safety analysis remain valid and support Cycle 6 operation without the thermal shield.

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. The parameters in Table 2 which exceeded the limiting range of values established by the previous safety analyses are the least negative doppler temperature coefficient, and the doppler temperature coefficient above 30% power, and the maximum delayed neutron fraction.

The least negative doppler temperature coefficient is -1.11 pcm/^{o}F compared to a previous analysis value of -1.18 pcm/^{o}F. The slight difference has a negligible impact on plant transient analysis. Therefore, no reanalysis is required.

The least negative doppler temperature coefficient as a function of T_{eff} is used in the CEA withdrawal from subcritical analysis. Previous analytical results demonstrate that the peak heat flux and coolant temperatures remain well below nominal full power values; therefore, there is a larger margin to DNB. Therefore, no reanalysis is necessary.

The least negative doppler-only power coefficient is slightly outside the bounds of the current limit above 30% power by a minimal amount (an integral difference of less than 1 pcm). This results in a negligible effect on transient analysis; therefore, no additional analysis is required due to the slight changes in this , rameter.

The maximum BOL beta effective is < 1% higher than the previous analysis value. Since the change is minimal there is no impact on the transient analysis therefore no reanalysis is required.

The maximum beta effective primarily impacts the CEA Withdrawal Transients. For CEA withdrawal from subcritical transient, substantial margin to the design limits on core average heat flux and average core water temperature exists. Therefore, previous safety conclusions remain valid.

For CEA withdrawal at power the TM/LP trip provides functions to prevent the violation of the limit DNBR. The <1% increase in maximum beta effective would have a negligible impact on the transient. Thus the TM/LP trip function is adequate and all previous conclusions remain valid.

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.

5.3 INCIDENTS REANALYZED

5.3.1 COMPLETE LOSS OF REACTOR COOLANT FLOW

The loss of flow accident was reanalyzed for Cycle 6 as a result of the change in the trip reactivity curve. Table 4 gives the time sequence of events for this accident. The reactor coolant flow, nuclear power, heat flux, and DNB transients are shown in Figures 2, 3 and 4.

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. The calculated minimum DNB is 1.378.

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 Westinghouse Fuel Rod Design Computations", WCAP-8785, October, 1976.

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

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

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

7. Jacobs, G. V., Iorii, J. A., "Reload Safety Evaluation Millstone Nuclear Power Station Unit 2, Cycle 6", February 1983.

8. Letter from Counsil, W. G. (Northeast Utilities) to Miller, J. R. (NRC), Subject: "Millstone Nuclear Power Station Unit Number 2 Faulted Fuel Assemblies", November 4, 1983 Docket No. 50-336.

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)
A	CE	20	1.93	95.0	15900
B+	CE	1	2.34	95.0	17450
F1	W	19	2.70	94.5	23570
F2	W	37	3.30	94.9	22750
G1	W	20	2.72	95.0	13940
G2	W	32	3.19	94.7	9240
H1	W	40	2.73	95.1	0
H2	W	48	3.22	94.8	0

** EOL Cycle 5 burnup: 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 +0.4 from 70 to 100% Power	+0.5 from 0 to 70% Power +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	-1.1 to -1.92
Delayed Neutron Fraction β_{eff} (%)	.479 to .634	.479 to .640
Prompt Neutron Lifetime (μsec)	<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	Cycle 6 Redesign
	BOC	EOC	EOC	EOC
All Rods Inserted	7.84	8.67	8.65	8.37
All Rods Inserted Less Worst Stuck Rod	6.47	6.59	6.70	6.67
(1) Less 10 Percent	5.82	5.93	6.03	6.00
Control Rod Requirements				
Reactivity Defects (Combined Doppler, T_{avg}, Void and Redistribution Effects)	1.94	2.64	2.69	2.64
Rod Insertion Allowance	0.36	0.36	0.41	0.38
(2) Total Requirements	2.30	3.00	3.10	3.02
Shutdown Margin ((1) - (2)) (%Δρ)	3.52	2.93	2.93	2.98
Required Shutdown Margin (%Δρ)	2.90	2.90	2.90	2.90

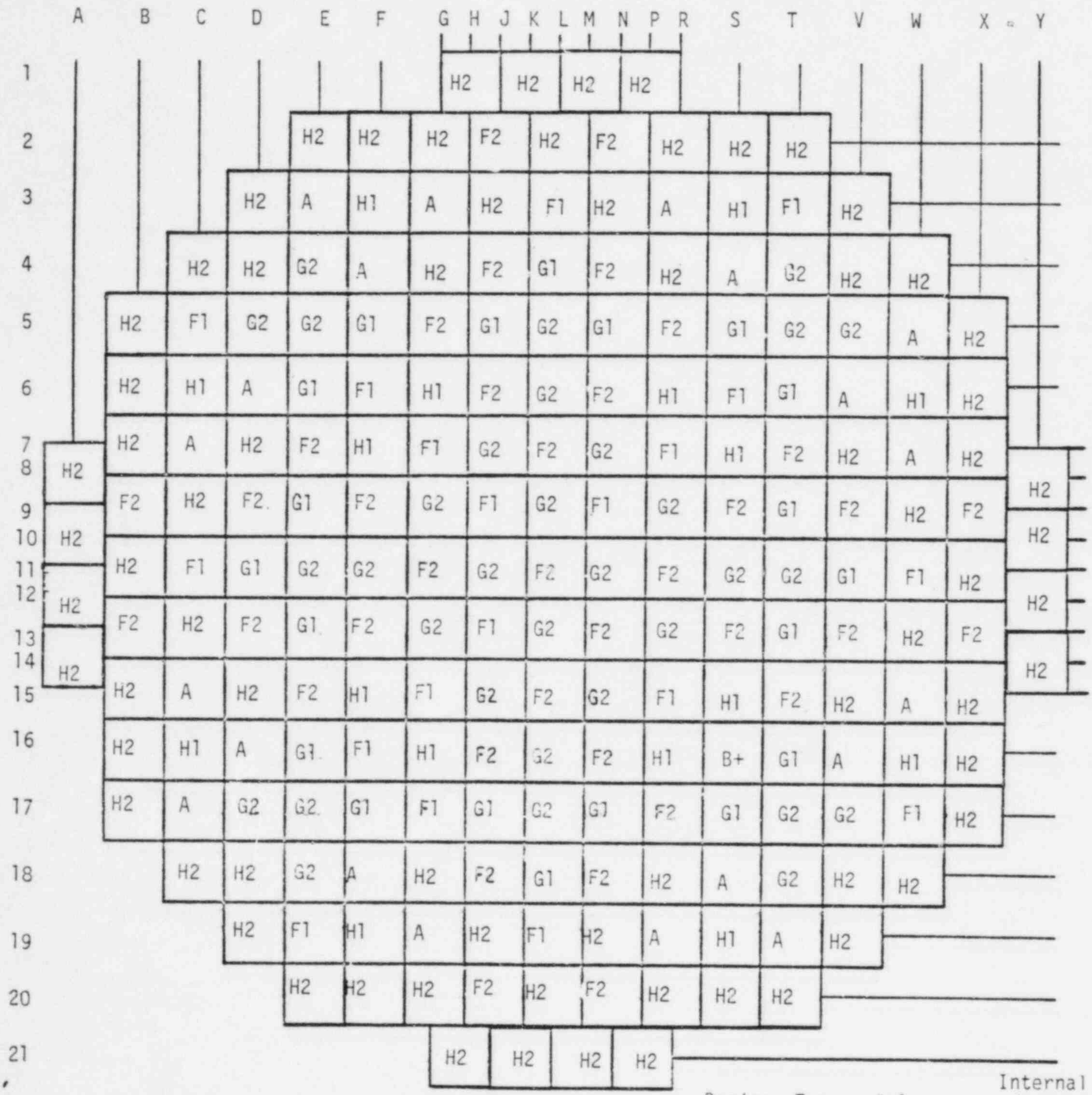
TABLE 4

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	3.5

FIGURE 1: CORE LOADING PATTERN MILLSTONE UNIT 2 - CYCLE 6



Region	Type	# Assy.	Internal w/o U-235
A	CE	20	1.93
B+	CE	1	2.34
F1	W	19	2.70
F2	W	37	3.30
G1	W	20	2.72
G2	W	32	3.19
H1	W	40	2.73
H2	W	48	3.22

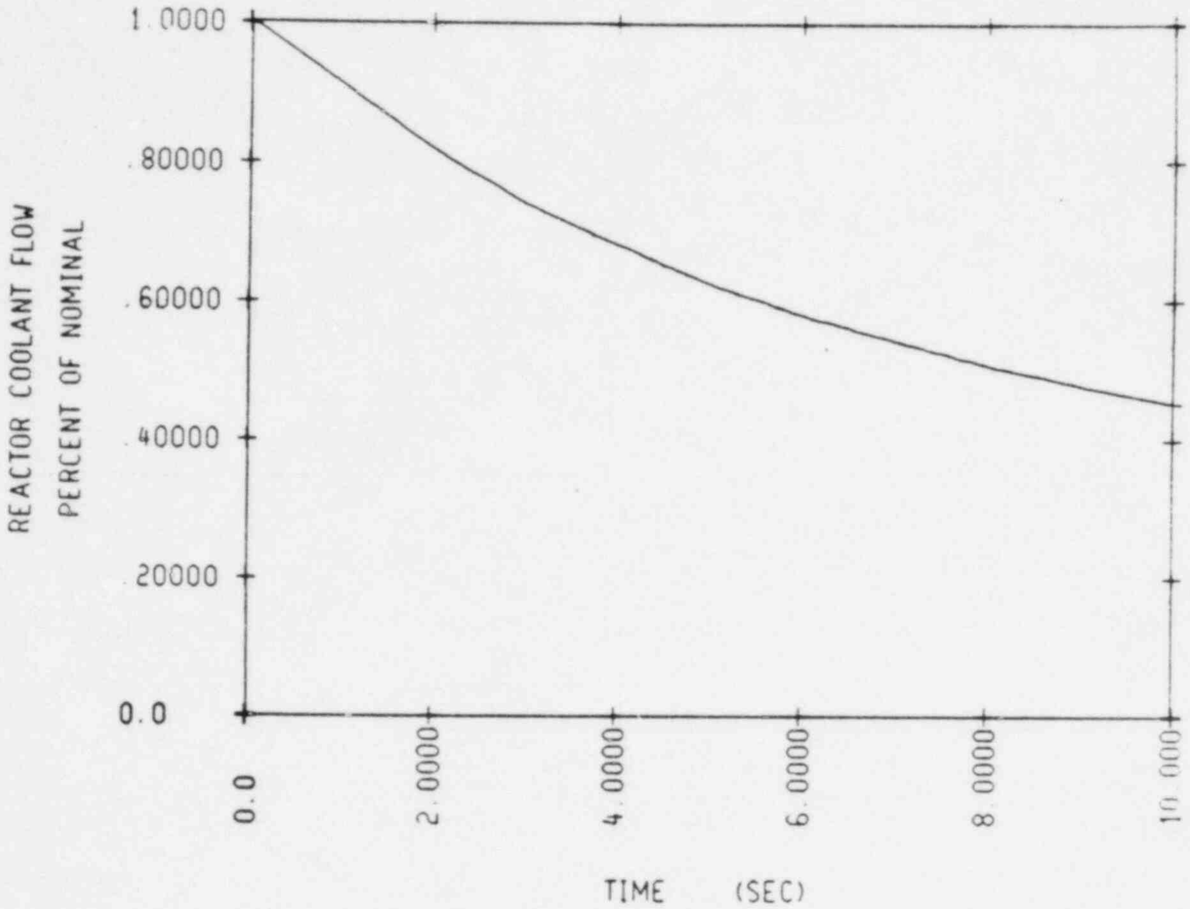


FIGURE 2: Millstone 2
Loss of Flow
Reactor Coolant Flow Vs. Time

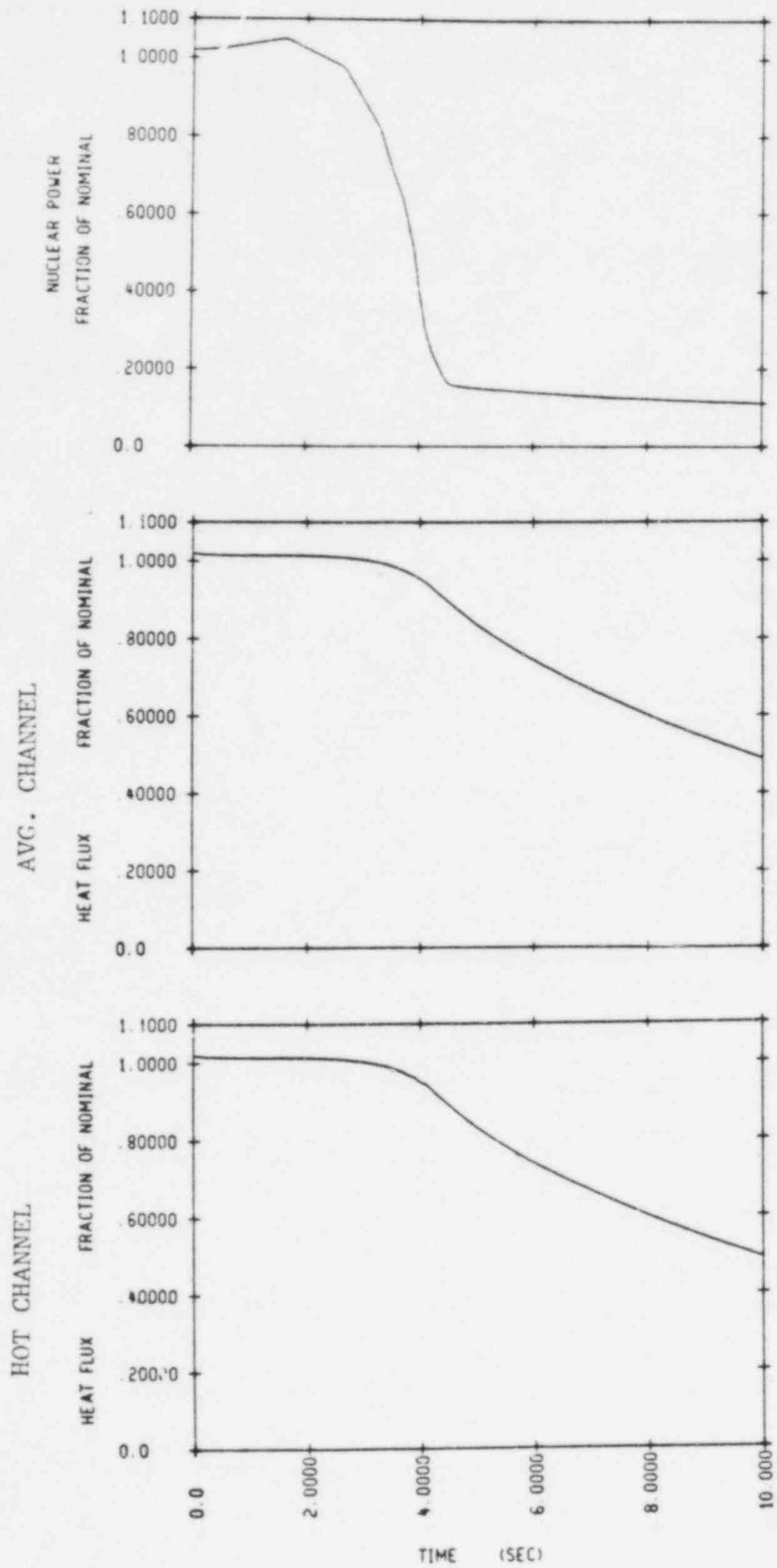


FIGURE 3: Millstone 2
Loss of Flow
Nuclear Power & Heat Flux Vs. Time

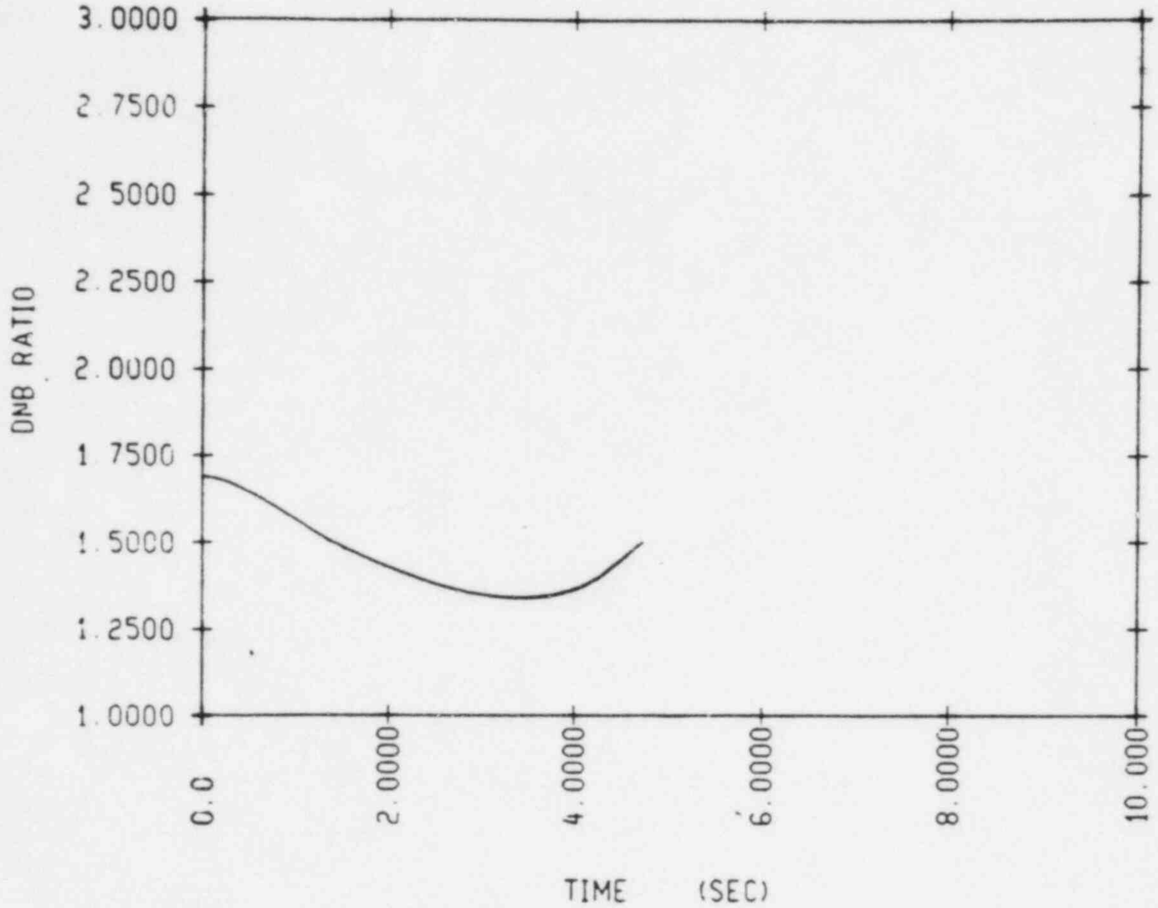


FIGURE 4: Millstone 2
Loss of Flow
DNB Ratio vs. Time