Docket No. 50-336

Attachment

Millstone Nuclear Power Station, Unit No. 2

Cycle 7 Final Reload Safety Analysis

June, 1985

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 7, 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 have been reviewed for the Cycle 7 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) grids (eleven assemblies have zircaloy grids, two hundred six assemblies have 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 7 are as follows:

Core Power (Mwt)	2700
System Pressure (psia)	2250
Reactor Coolant Flow (GPM)	350,000*
Core Inlet Temperature (°F)	549
Average Linear Power Density (kw/ft) (based on best estimate hot, densified core average stack height of 136.4 inches)	6.067

*Minimum guaranteed safety analysis flow

The core loading pattern for Cycle 7 is shown in Figure 1. The feed fuel for the Millstone II, Cycle 7 core will consist of twenty-four (24) zoned-enrichment interior feed assemblies, each containing sixty (60) fuel rods at 2.62 w/o U235 and one-hundred sixteen (116) fuel rods at 2.91 w/o U235, and forty-eight (48) zoned-enrichment peripheral assemblies, each containing sixty (60) fuel rods at 2.91 w/o U235 and one-hundred sixteen (116) fuel rods at 3.29 w/o U235. The zoned-enrichment assembly configuration is shown in Figure 2. The feed fuel will replace twenty (20) Combustion Engineering (CE) Batch A assemblies, one (1) CE Batch B assembly, and fifty-one (51) Westinghouse Batch F assemblies. An additional five (5) Westinghouse Batch F assemblies will be discharged from the end of Cycle 6, and will be replaced by five (5) Westinghouse Batch F assemblies which were removed from the core at the end of Cycle 5. Due to fuel defects in Cycle 6 and subsequent symmetry considerations, fourteen (14) Westinghouse Batch G assemblies, seven (7) Westinghouse Batch F assemblies (these Batch F and G assemblies were removed from the core at the end of Cycle 5), and four (4) CE Batch A assemblies (discharged at the end of Cycle 1) are needed as well. As as result of fuel reconstitution, the fuel rods from seven (7) Westinghouse reload assemblies to be used in Cycle 7 have been placed in Combustion Engineering (CE) skeletons. Also, twenty-one (21) fuel rods have been replaced with stainless steel rods in Cycle 7. The twenty-one stainless steel rods are distributed among eleven (11) fuel assemblies, with the number of stainless steel rods in each of these assemblies ranging from one to five.

A summary of the Cycle 7 fuel inventory is given in Table 1.

2.0 MECHANICAL DESIGN

2.1 GENERAL DISCUSSION

The mechanical design of the Cycle 7 fuel assemblies is essentially identical to that of the Cycle 6 assemblies[7]. 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 Westingnouse Millstone II reload fuel assembly to be utilized in Cycle 7 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 stainless steel rods in the reconstituted fuel assemblies were treated as heated rods in the THINC DNB analysis. This is conservative since it results in higher subchannel enthalpy predictions.

No significant variations in thermal margins result from the Cycle 7 reload. The Cycle 7 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[5].

4.0 NUCLEAR DESIGN

The Westinghouse nuclear design procedures, computer programs, and calculation models utilized in the Millstone II, Cycle 7 reload design are presented in the BSR. Similar to the Cycle 6 evaluation[7], Cycle 7 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 7 core loading results in a maximum linear heat rate of less than 15.6 kw/ft at all fuel heights at rated power. The safety analysis has specifically included the 21 stainless steel rods. Table 2 provides a summary of changes in the Cycle 7 kinetics characteristics compared with the current limit based on the reference safety analysis.[2,5,6,7] It can be seen from the table that all of the Cycle 7 values fall within current limits with a small exception in the delayed neutron fraction noted in Table 2. Table 3 provides the contol 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 7. For the overpower transient, the fuel centerline temperature limit of 4700°F can be accommodated with margin in the Cycle 7 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.

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, 6, and preliminary Cycle 7 reload safety analyses[5,6,7,8] 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.

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 7 parameters in each of these areas were examined as discussed below to ascertain whether new accident analyses were required.

5.2.1 KINETICS PARAMETERS

A comparison of Cycle 7 kinetics parameters with the current limits, established by the BSR and Cycles 4, 5, and 6 reload safety analyses, is presented in Table 2. The maximum delayed neutron fraction and the least negative doppler power coefficient above 30% power exceed the current limit slightly. These parameters were evaluated, [7] and the previous analysis was determined to be adequate. Therefore, no reanalysis is required.

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 7 shutdown margin requirements are the same as Cycle 6.

5.2.3 CEA WORTHS

Changes in CEA worths may affect shutdown margin. Table 3 shows that the Cycle 7 shutdown margin requirements are satisfied.

5.2.4 CORE PEAKING FACTORS

All core peaking factors for Cycle 7 were within the reference cycle limits.

5.3 INCIDENTS REANALYZED

5.3.1 COMPLETE LOSS OF REACTOR COOLANT FLOW

The loss of flow accident was reanalyzed for Cycle 7 assuming a $+0.4 \times 10^{-4}$ $\Delta\rho/°F$ moderator temperature coefficient as specified in the Millstone II Technical Specifications and Table 2, and utilizing cycle specific design parameters.

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 3, 4 and 5.

The results show that the reactor coolant pump speed sensing system provides sufficient protection against clad and fuel damage. The calculated minimum DNBR is 1.30.

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. Letter, Counsil to Miller, Millstone Nuclear Power Station Unit No. 2 - Supplement to the Reload Safety Analyses, November 17, 1983.

8. Letter, Counsil to Miller, Millstone Nuclear Power Station, Unit No. 2 Cycle 7 Refueling - Preliminary Reload Safety Analysis - Proposed Revisions to Technical Specifications, February 6, 1985.

TABLE 1

Millstone Unit 2 Cycle 7
Core Loading

Region	Type	Number of Assemblies	Initial Enrichment w/o U235	%Theoretical Density	BOC** Burnup Average (MWD/MTU)
A	CE	4	1.93	95.0	15960
F1	W	4	2.70	94.5	25200
F2	W	5	3.30	94.9	22200
F2	W*	3	3.30	94.9	21560
G1	W	19	2.72	95.0	23470
G2	W	32	3.19	94.7	19290
G2	W*	4	3.19	94.7	9970
H1	W	30	2.73	95.2	13790
H2	W	44	3.22	94.8	9560
J1	W	24	2.62/2.91	95.2/95.1	0
J2	W	48	2.91/3.29	95.1/95.2	0

* Westinghouse fuel reassembled using CE skeletons.

** EOL Cycle 6 burnup assumed: 11,500 MWD/MTU.

TABLE 2

MILLSTONE UNIT 2 KINETICS CHARACTERISTICS

	Current Limit	Cycle 7
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	-1.2 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 7

Control Rod Worth (%Δρ)	Cycle 6		Cycle 7	
	BOC	EOC	BOC	EOC
All Rods Inserted	7.49	8.37	7.83	8.60
All Rods Inserted Less Worst Stuck Rod	6.51	6.67	6.64	6.95
(1) Less 10 Percent	5.86	6.00	5.98	6.26
Control Rod Requirements (%Δρ)				
Reactivity Defects (Combined Doppler, T_{avg}, Void and Redistribution Effects)	1.86	2.64	1.67	2.58
Rod Insertion Allowance	0.52	0.38	0.87	0.41
(2) Total Requirements	2.38	3.02	2.54	2.99
Shutdown Margin ((1) - (2)) (%Δρ)	3.48	2.98	3.44	3.27
Required Shutdown Margin (%Δρ)	2.90	2.90	2.90	2.90

TABLE 4

SEQUENCE OF EVENTS - LOSS OF COOLANT FLOW

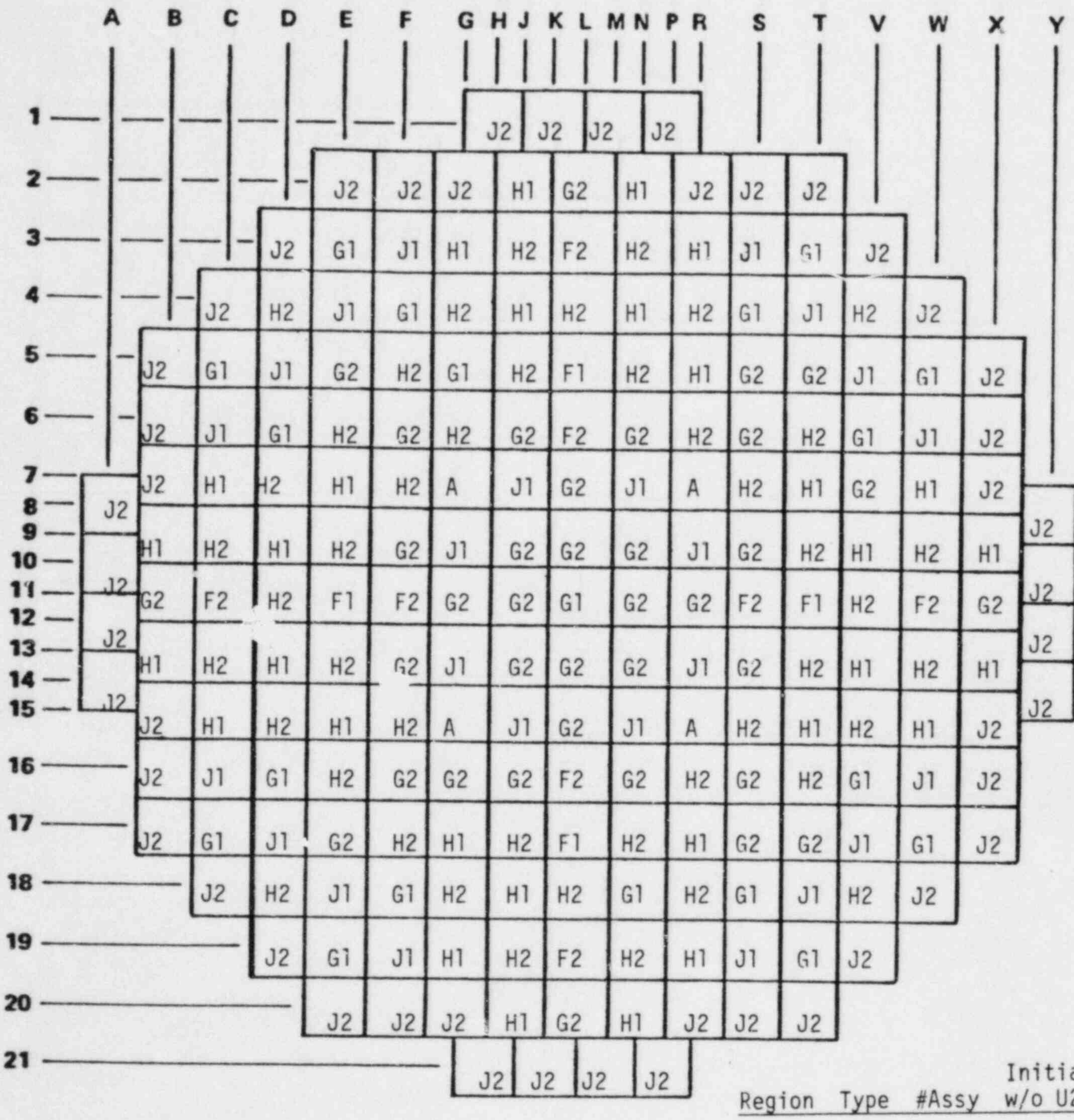
Four pumps in operation, all pumps coasting down

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	3.4

FIGURE 1

CORE LOADING PATTERN

MILLSTONE UNIT 2 - CYCLE 7



Building North →

Region	Type	#Assy	Initial w/o U235
A	CE	4	1.93
F1	W	4	2.70
F2	W	8	3.30
G1	W	19	2.72
G2	W	36	3.19
H1	W	30	2.73
H2	W	44	3.22
J1	W	24	2.62/2.91
J2	W	48	2.91/3.29

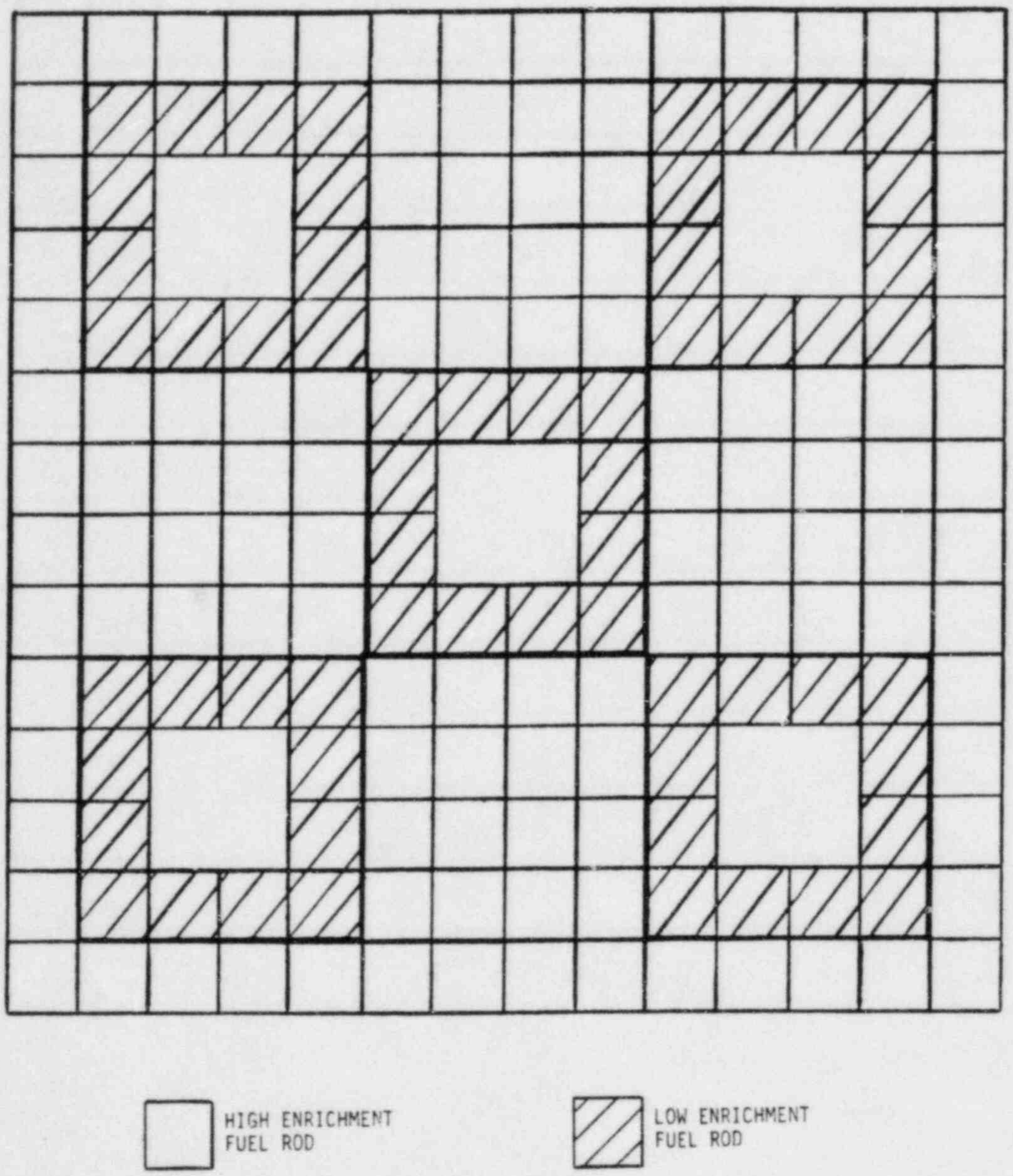


FIGURE 2

ZONED-ENRICHMENT FUEL ASSEMBLY LATTICE

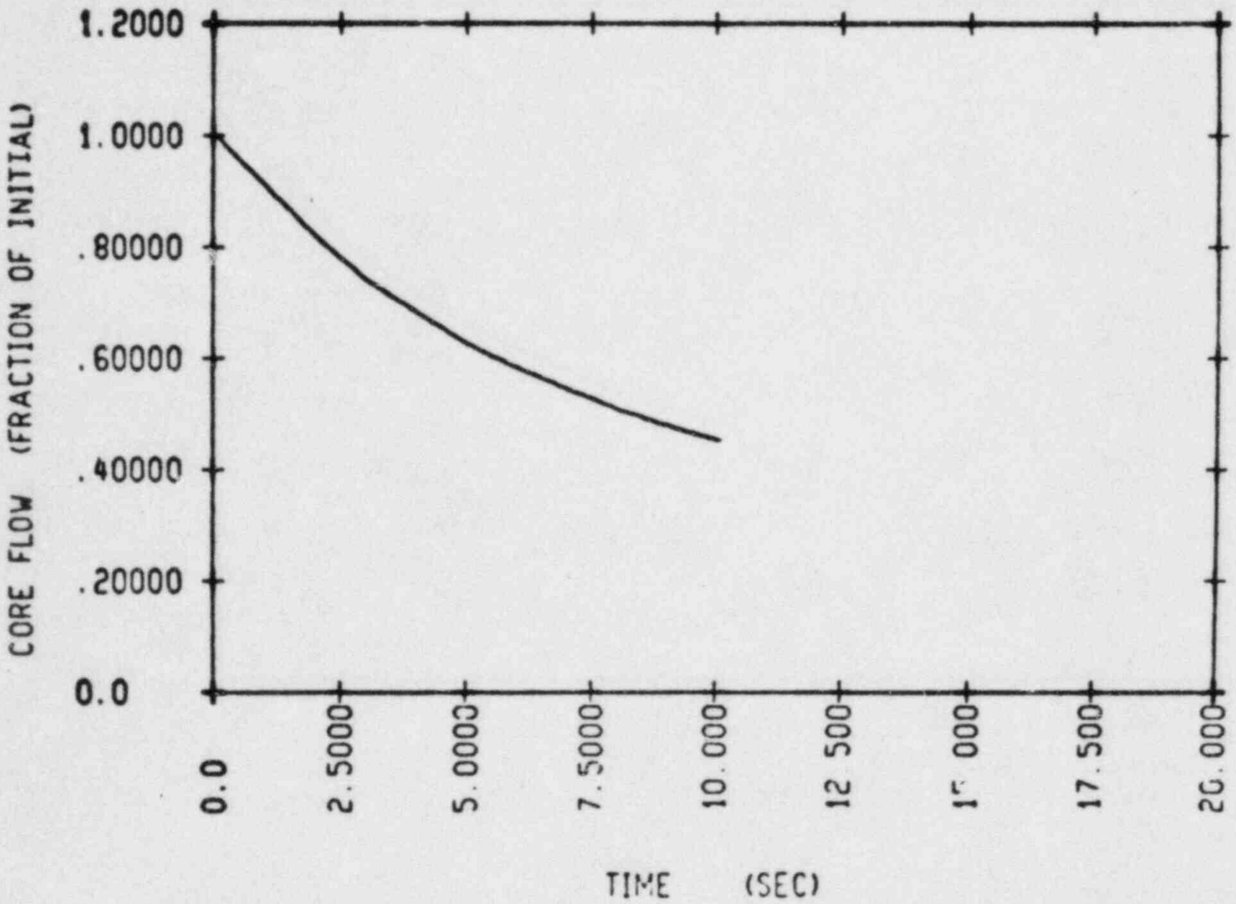


FIGURE 3: Millstone 2
Loss of Flow
Core Flow vs. Time

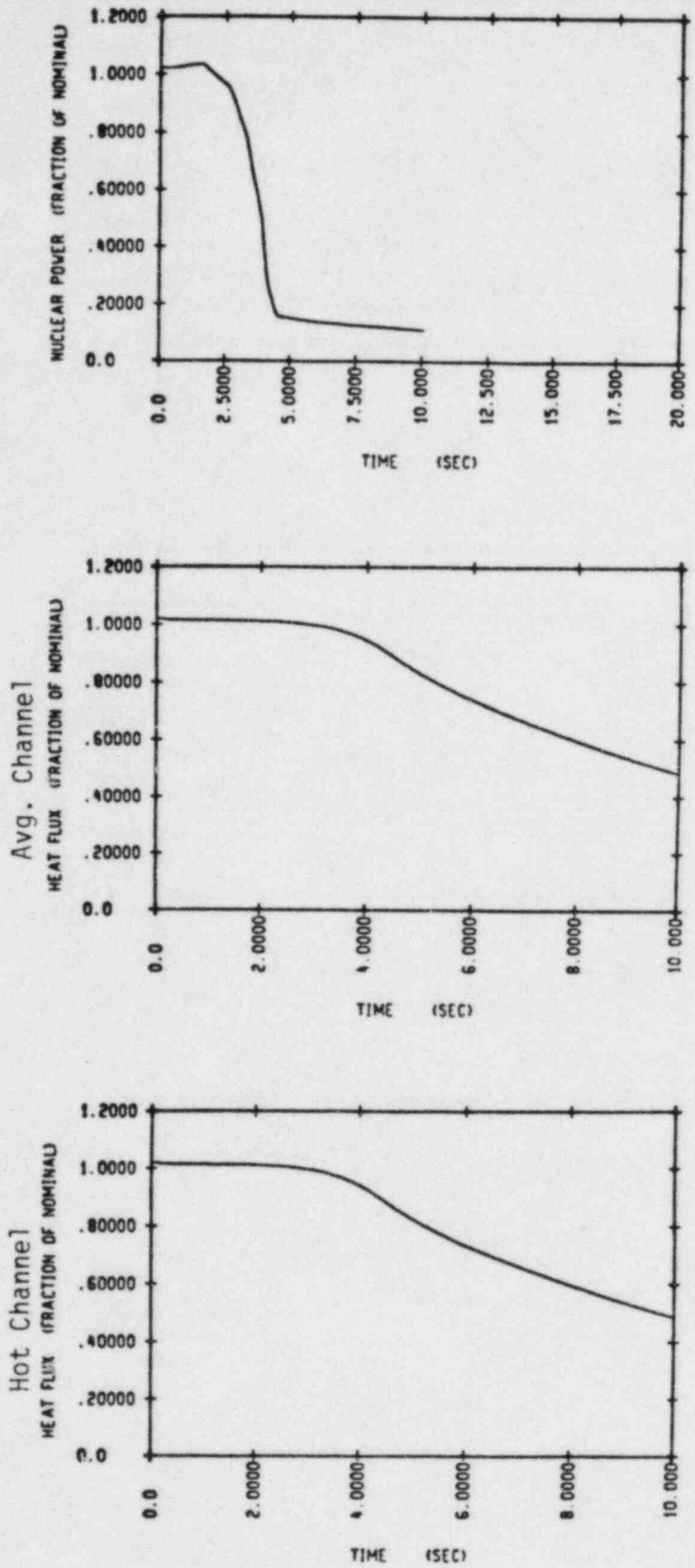


FIGURE 4: Millstone 2
Loss of Flow
Nuclear Power and
Heat Flux vs. Time

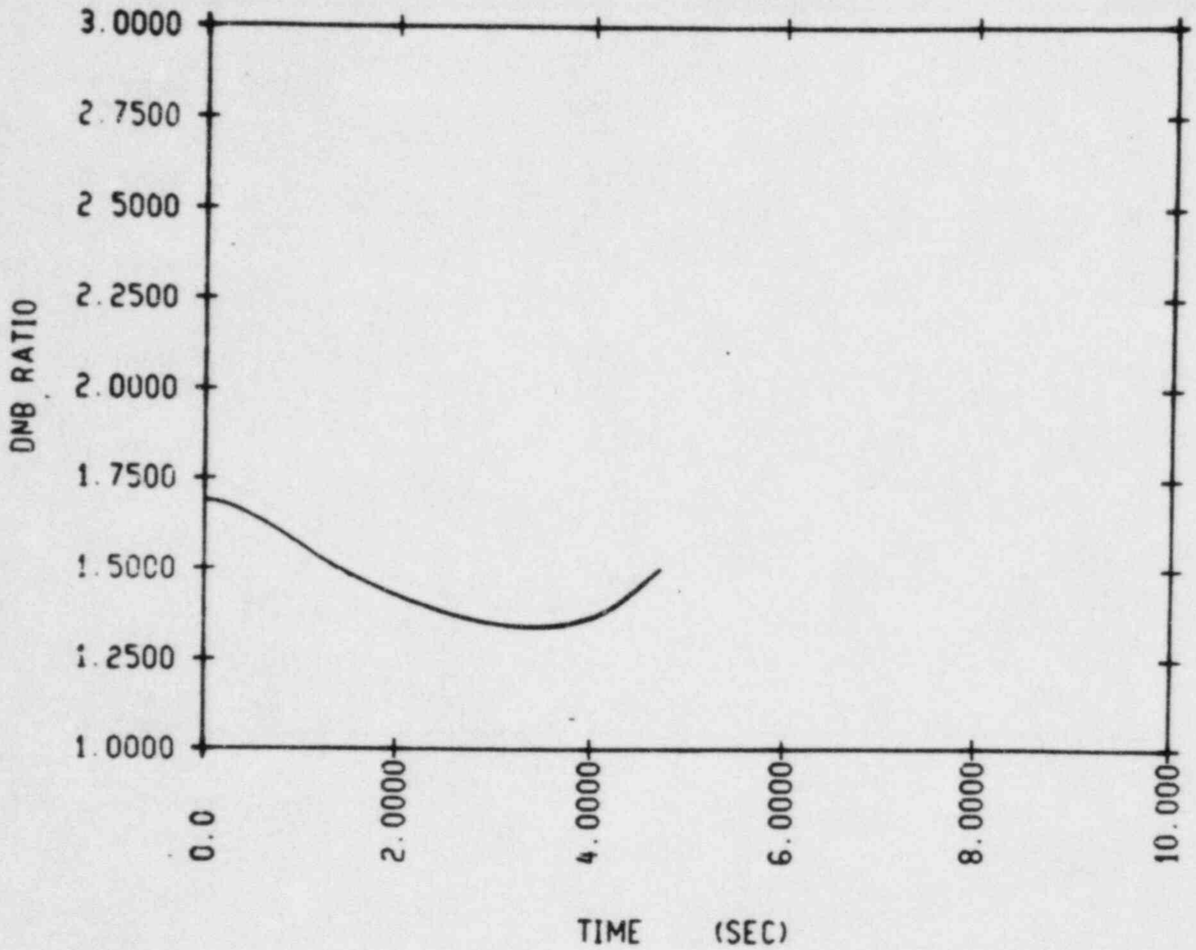


FIGURE 5: Millstone 2
Loss of Flow
DNB Ratio vs. Time