NORTHEAST UTILITIES

THE CONNECTICUT LIGHT AND POWER COMPANY
WESTERN MASSACHUSETTS ELECTRIC COMPANY
HOLYOKE WATER POWER COMPANY
NORTHEAST UTILITIES SERVICE COMPANY
NORTHEAST NUCLEAR ENERGY COMPANY

General Offices • Selden Street, Berlin, Connecticut

P.O. BOX 270
HARTFORD, CONNECTICUT 06141-0270
(203) 665-5000

November 20, 1984

Docket No. 50-423
B11368

Director of Nuclear Reactor Regulation
Mr. B. J. Youngblood, Chief
Licensing Branch No. 1
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

References:

(1) B. J. Youngblood to W. G. Counsil, Request for Additional Information for Millstone Nuclear Power Station, Unit No. 3, dated May 31, 1983.

(2) W. G. Counsil to B. J. Youngblood, Millstone Nuclear Power Station, Unit No. 3, N-1 Loop Operation, dated April 9, 1984.

(3) B. J. Youngblood to W. G. Counsil, Millstone Unit No. 3 Safety Evaluation Report, dated August 2, 1984.

Dear Mr. Youngblood:

Millstone Nuclear Power Station, Unit No. 3
Three (N-1) Loop Operation

In Reference (1), Northeast Nuclear Energy Company (NNECO) received NRC Core Performance Branch (CPB) question number 492.3 which requested NNECO to provide additional information concerning our intention to use N-1 loop operation at Millstone Unit No. 3. In Reference (2), NNECO stated its intention to pursue an operating license which permits such operation and also committed to make changes to affected portions of the Millstone 3 FSAR to take into account N-1 loop operation. This was identified as confirmatory item (#15) in Reference (3).

We have recently completed LOCA and non-LOCA accident analyses in support of operation of Millstone Unit No. 3 with one reactor coolant loop out of service and isolated. The enclosed analyses package (Enclosure I) consists of FSAR change pages which include thermal hydraulic design parameters (Chapter 4, Reactor), mass and energy release data (Chapter 6, Engineered Safety Features) and accident analysis (Chapter 15). These FSAR change pages will be incorporated in a future FSAR amendment.

Enclosure II describes the design basis information for the modification to the solid state protection system such that the system can be configured for 3-loop operation. It also provides identification of parameters and conditions associated with 3-loop operation. As agreed in a discussion between your

B001
3/40

Ms. E. L. Doolittle and our Mr. R. G. Joshi during an October 2, 1984 meeting, information contained in Enclsoure II has been prepared to facilitate the NRC's review of our request. Appropriate changes to the FSAR pages (Chapter 7, Instrumentation and Control) will be forwarded to the NRC in a future amendment.

NNECO is hereby requesting for approval to operate Millstone Unit No. 3 (a 4-loop Westinghouse plant) with only three active coolant loops. The purpose of this request is to allow the plant to continue operating with one loop out of service in the event of an equipment failure in that loop. Three loop operation is defined as power operation of Millstone Unit No. 3 with one of the four reactor coolant loops isolated by loop isolation valves in both the hot leg and cold leg of the inactive loop at reduced power levels.

The proposed Technical Specifications for 3-loop operation for Millstone Unit No. 3 are being developed at this time and will be submitted to the NRC by the end of January, 1985. All procedures related to 3-loop operation and surveillance, including revised or new emergency operating procedures are being developed at this time and these procedures will be uniquely identified.

A human factors evaluation will be performed to review the adequacy of the interface between the control room operator and plant processes as found in the control room for 3-loop operation. The schedule for completion of this effort and documenting the results in a report for the NRC review will be accomplished to support fuel load in November, 1985.

If you have any questions regarding this submittal, please feel free to contact our licensing staff directly.

Very truly yours,

NORTHEAST NUCLEAR ENERGY COMPANY
et. al.

BY NORTHEAST NUCLEAR ENERGY COMPANY
Their Agent

W. G. Counsil

W. G. Counsil
Senior Vice President

STATE OF CONNECTICUT)
) ss. Berlin
COUNTY OF HARTFORD)

Then personally appeared before me W. G. Counsil, who being duly sworn, did state that he is a Senior Vice President of Northeast Nuclear Energy Company, an Applicant herein, that he is authorized to execute and file the foregoing information in the name and on behalf of the Applicants herein and that the statements contained in said information are true and correct to the best of his knowledge and belief.

Lorraine J. D'Amico
Notary Public

My Commission Expires March 31, 1988

NORTHEAST UTILITIES

THE CONNECTICUT LIGHT AND POWER COMPANY
WESTERN MASSACHUSETTS ELECTRIC COMPANY
HOLYOKE WATER POWER COMPANY
NORTHEAST UTILITIES SERVICE COMPANY
NORTHEAST NUCLEAR ENERGY COMPANY

General Offices • Selden Street, Berlin, Connecticut

P.O. BOX 270
HARTFORD, CONNECTICUT 06141-0270
(203) 665-5000

November 20, 1984

Docket No. 50-423
B11368

Director of Nuclear Reactor Regulation
Mr. B. J. Youngblood, Chief
Licensing Branch No. 1
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

References: (1) B. J. Youngblood to W. G. Counsil, Request for Additional Information for Millstone Nuclear Power Station, Unit No. 3, dated May 31, 1983.

(2) W. G. Counsil to B. J. Youngblood, Millstone Nuclear Power Station, Unit No. 3, N-1 Loop Operation, dated April 9, 1984.

(3) B. J. Youngblood to W. G. Counsil, Millstone Unit No. 3 Safety Evaluation Report, dated August 2, 1984.

Dear Mr. Youngblood:

Millstone Nuclear Power Station, Unit No. 3
Three (N-1) Loop Operation

In Reference (1), Northeast Nuclear Energy Company (NNECO) received NRC Core Performance Branch (CPB) question number 492.3 which requested NNECO to provide additional information concerning our intention to use N-1 loop operation at Millstone Unit No. 3. In Reference (2), NNECO stated its intention to pursue an operating license which permits such operation and also committed to make changes to affected portions of the Millstone 3 FSAR to take into account N-1 loop operation. This was identified as confirmatory item (#15) in Reference (3).

We have recently completed LOCA and non-LOCA accident analyses in support of operation of Millstone Unit No. 3 with one reactor coolant loop out of service and isolated. The enclosed analyses package (Enclosure I) consists of FSAR change pages which include thermal hydraulic design parameters (Chapter 4, Reactor), mass and energy release data (Chapter 6, Engineered Safety Features) and accident analysis (Chapter 15). These FSAR change pages will be incorporated in a future FSAR amendment.

Enclosure II describes the design basis information for the modification to the solid state protection system such that the system can be configured for 3-loop operation. It also provides identification of parameters and conditions associated with 3-loop operation. As agreed in a discussion between your

Ms. E. L. Doolittle and our Mr. R. G. Joshi during an October 2, 1984 meeting, information contained in Enclsoure II has been prepared to facilitate the NRC's review of our request. Appropriate changes to the FSAR pages (Chapter 7, Instrumentation and Control) will be forwarded to the NRC in a future amendment.

NNECO is hereby requesting for approval to operate Millstone Unit No. 3 (a 4-loop Westinghouse plant) with only three active coolant loops. The purpose of this request is to allow the plant to continue operating with one loop out of service in the event of an equipment failure in that loop. Three loop operation is defined as power operation of Millstone Unit No. 3 with one of the four reactor coolant loops isolated by loop isolation valves in both the hot leg and cold leg of the inactive loop at reduced power levels.

The proposed Technical Specifications for 3-loop operation for Millstone Unit No. 3 are being developed at this time and will be submitted to the NRC by the end of January, 1985. All procedures related to 3-loop operation and surveillance, including revised or new emergency operating procedures are being developed at this time and these procedures will be uniquely identified.

A human factors evaluation will be performed to review the adequacy of the interface between the control room operator and plant processes as found in the control room for 3-loop operation. The schedule for completion of this effort and documenting the results in a report for the NRC review will be accomplished to support fuel load in November, 1985.

If you have any questions regarding this submittal, please feel free to contact our licensing staff directly.

Very truly yours,

NORTHEAST NUCLEAR ENERGY COMPANY
et. al.

BY NORTHEAST NUCLEAR ENERGY COMPANY
Their Agent

W. G. Counsil

W. G. Counsil
Senior Vice President

STATE OF CONNECTICUT)
) ss. Berlin
COUNTY OF HARTFORD)

Then personally appeared before me W. G. Counsil, who being duly sworn, did state that he is a Senior Vice President of Northeast Nuclear Energy Company, an Applicant herein, that he is authorized to execute and file the foregoing information in the name and on behalf of the Applicants herein and that the statements contained in said information are true and correct to the best of his knowledge and belief.

Lorraine J. DiAmico
Notary Public

My Commission Expires March 31, 1988

Enclosure I

FSAR Changes Related to
3 Loop Operation
of
Millstone Unit No. 3

guide thimble cooling flow, head cooling flow, baffle leakage, and leakage to the vessel outlet nozzle.

4.4.1.4 Hydrodynamic Stability Design Basis

Basis

Modes of operation associated with Condition I and II events shall not lead to hydrodynamic instability.

4.4.1.5 Other Considerations

The above design bases, together with the fuel clad and fuel assembly design bases given in Section 4.2.1, are sufficiently comprehensive so additional limits are not required.

Fuel rod diametral gap characteristics, moderator-coolant flow velocity and distribution, and moderator void are not inherently limiting. Each of these parameters is incorporated into the thermal and hydraulic models used to ensure the above-mentioned design criteria are met. For instance, the fuel rod diametral gap characteristics change with time (Section 4.2.3.3) and the fuel rod integrity is evaluated on that basis. The effect of the moderator flow velocity and distribution (Section 4.4.2.2) and moderator void distribution (Section 4.4.2.4) are included in the core thermal (THINC) evaluation and thus affect the design bases.

Meeting the fuel clad integrity criteria covers possible effects of clad temperature limitations. As noted in Section 4.2.3.3, the fuel rod conditions change with time. A single clad temperature limit for Condition I or Condition II events is not appropriate since, of necessity, it would be overly conservative. A clad temperature limit is applied to the loss-of-coolant accident (Section 15.6.5), control rod ejection accident, and locked rotor accident.

4.4.2 Description

4.4.2.1 Summary Comparison

The design of Millstone 3, as described in this report, has thermal-hydraulic parameters similar to the SNUPPS Units (Docket Nos. STN-50-482, STN-50-483, STN-50-485, and STN-50-486).

Values of pertinent parameters, along with critical heat flux ratios, fuel temperatures, and linear heat generation rates, are presented in Table 4.4-1. The key design parameters for N-1 (three-loop) operation are provided in Table 4.4-1A. In power capability evaluation, there has not been any change in this design criteria. The reactor is still designed to a minimum DNBR ≥ 1.30, as well as no fuel centerline melting during normal operation, operational transients, and faults of moderate frequency.

All DNB analyses performed for this application have included a DNBR multiplier of 0.86. The results of 17 x 17-geometry DNB tests

(WCAP-8298-FB and WCAP-8299-A) are discussed in Section 4.4.2.2.1 and 1.52
indicate that a DNBR multiplier of 0.88 is required for the "R" grid 1.53
DNB correlation. Thus, the multiplier used results in conservative 1.54
DNB evaluations.

Fuel densification has been considered in the DNB and fuel 1.55
temperature evaluations utilizing the methods and models described in 1.56
WCAP-8218-P-A and WCAP-8219-A.

4.4.2.2 Critical Heat Flux Ratio or Departure from Nucleate Boiling Ratio and Mixing Technology 1.59

The minimum DNBRs for the rated power, design overpower, and 2.1
anticipated transient conditions are given in Table 4.4-1. The 2.4
minimum DNBR in the limiting flow channel will be downstream of the
peak heat flux location (hot spot) due to the increased downstream 2.5
enthalpy rise.

DNBRs are calculated by using the correlation and definitions 2.6
described in Sections 4.4.2.2.1 and 4.4.2.2.2. The THINC-IV computer 2.7
code (discussed in Section 4.4.4.5.1) is used to determine the flow
distribution in the core and the local conditions in the hot channel 2.8
for use in the DNB correlation. The use of hot channel factors is 2.9
discussed in Section 4.4.4.3.1 (nuclear hot channel factors) and in
Section 4.4.2.2.4 (engineering hot channel factors). 2.10

4.4.2.2.1 Departure from Nuclear Boiling Technology 2.12

The W-3 correlation and several modifications have been used in 2.13
Westinghouse CHF calculations. The W-3 was originally developed from 2.15
single tube data (Tong 1972), but was subsequently modified to apply
to the 0.442-inch O.D. rod "R"-grid (WCAP-7695-P-A and WCAP-7958-A), 2.16
and "L"-grid (WCAP-7988 and WCAP-8030-A), as well as the 0.374-inch 2.18
O.D. (WCAP-8298-P-A and WCAP-8299-A; WCAP-8296-P-A and WCAP-8297),
rod bundle data. These modifications to the W-3 correlation have 2.19
been demonstrated to be adequate for reactor rod bundle design.

A description of the 17 x 17-fuel assembly test program and a summary 2.20
of the results are described in detail in WCAP-8298-P-A and 2.21
WCAP-8299-A. A correlation factor was developed to adopt the W-3 2.22
correlation to 17 x 17-assemblies with top split mixing vane grids 2.23
referred to as R-grid. This correlation factor, termed the "modified 2.24
spacer factor," was developed as a multiplier on the W-3 correlation.

Figure 4.4-1 shows the 17 x 17-data obtained in this test program. 2.25
The predicted heat flux includes a 0.88 multiplier which is part of 2.26
the 17 x 17-modified spacer factor. However, as noted previously 2.27
(Section 4.4.2.1), a multiplier of 0.860 has been conservatively
applied for all DNB analyses. 2.28

The test results indicated that a reactor core using this geometry 2.29
may operate with a minimum DNBR of 1.28 and satisfy the design 2.30
criterion. However, as stated in 4.4.1.1, a minimum DNBR of 1.30 is 2.31
conservatively used in this application.

TABLE 4.4-1A

THERMAL-HYDRAULIC DESIGN PARAMETERS FOR
ONE OF FOUR COOLANT LOOPS OUT OF SERVICE

Total core heat output, MWt	2,560
Total core heat output, 10^6 Btu/hr	8,735
Heat generated in fuel, %	97.4
Nominal system pressure, psia	2,250

Coolant Flow

Effective thermal flow rate for heat transfer, 10^6 lbm/hr	105.5
Effective flow area for heat transfer, ft^2	51.1
Average velocity along fuel rods, ft/sec	12.9
Average mass velocity, 10^6 lbm/hr-ft^2	2.07

12

Coolant Temperature

Design nominal inlet, °F	550.6
Average rise in core, °F	61.4
Average in core, °F	582.7

Heat Transfer

Active heat transfer surface area, ft^2	59,700
Average heat flux, Btu/hr-ft^2	142,400
Minimum DNBR at nominal conditions	>2.02
Minimum DNBR for design and anticipated transients	$\geq$1.30

Determination of Dryout Time 1.10

During the blowdown following a steam line rupture, a point may be 1.12
reached when all the initial fluid inventory of the affected steam 1.13
generator, including that added from outside sources, will be
depleted. At that time, the blowdown rate out of the break will be 1.14
limited by the rate at which water is added to the steam generator 1.15
from the auxiliary feedwater system. This point in the transient is 1.16
termed "dryout." The time of dryout can be determined as described 1.17
in WCAP-8860.

Additional mass and energy flow from the affected steam generator to 1.18
containment results from:

1. Liquid flashing in the unisolated portion of the main 1.20
feedwater pipe

2. Pumped main feedwater 1.21

3. Auxiliary feedwater flow before isolation 1.22

The main feedwater flow to the affected steam generator is 1.24
conservatively assumed to be at runout flow for the various power 1.25
levels depicted in Table 6.2-59 from the time of the break until the
main feedwater isolation valve (FWIV) receives a signal to close. 1.26
The unaffected steam generator FWIVs are then assumed to close 1.27
instantaneously while the affected steam generator FWIV is assumed to 1.28
have a longer closing time.

All main feedwater is diverted to the affected steam generator for 1.29
the duration of the valve closure time. During the FWIV closure at 1.30
the affected steam generator, feedwater flow is conservatively
increased to 4 times the value in Table 6.2-59. No flow reduction is 1.32
assumed to occur during the closure sequence. The main feedwater 1.33
system runout flow at various power levels, feedwater temperature,
and FWIV closure times are listed in Table 6.2-59. The time of main 1.35
feedwater isolation is dependent upon the time that the isolation
signal setpoint is reached, which varies with the break condition. 1.36

The auxiliary feedwater flow to the affected steam generator is 1.37
42.4 lbm/sec at 120°F. The flow is limited by passive flow control 1.38
devices (cavitating venturis) installed in the line to each steam
generator (Section 10.4.9). 1.39

The initial inventory of the affected steam generator is increased to 1.40
include the liquid in the unisolated portion of its main feedwater
pipe.

6.2.1.5 Minimum Containment Pressure Analysis for Performance 1.43
Capability Studies of Emergency Core Cooling System 1.44

The containment backpressure used for the limiting case break for the 1.46
emergency core cooling system analysis (Section 15.6.4) is depicted 1.48
on Figure 6.2-59 for the N-loop analysis, and on Figure 6.2-59A for

the N-1 loop analysis. The containment backpressure is calculated using the methods and assumptions described in Appendix A of WCAP-8339 (1974). This section describes the input parameters including the containment initial conditions, net containment volume, passive heat sink materials, thicknesses, surface areas, and starting time and performance parameters of containment cooling systems used in the analysis. 12

6.2.1.5.1 Mass and Energy Release Data

The mass and energy releases to the containment during the blowdown and reflood portions of the limiting break transient are presented in Tables 6.2-72, ~~and~~ 6.2-73 and 6.2-74 for the N-loop analysis, and in Tables 6.2-~~74~~ 75 and 6.2-~~75~~ 76 for the N-1 loop analysis. 12

The mathematical models which calculate the mass and energy releases to the containment are described in Section 15.6.5 and conform to 10CFR Part 50, Appendix K, ECCS Evaluation Models. A break spectrum analysis is performed (references in Section 15.6.5) that considers various break sizes, break locations, and Moody discharge coefficients for the double-ended cold leg guillotines which do not affect the mass and energy released to the containment. This effect is considered for each case analyzed. During refill, the mass and energy released to the containment is assumed to be zero, which minimizes the containment pressure. During reflood, the effect of steam water mixing between the safety injection water and the steam flowing through the reactor coolant system intact loops reduces the available energy released to the containment vapor spaces and therefore tends to minimize containment pressure.

6.2.1.5.2 Initial Containment Internal Conditions

The following initial values were used in the analysis:

1. A containment pressure of 9.5 psia
2. A containment temperature of 90°F
3. A refueling water storage tank temperature of 40.0°F
4. An outside temperature of -20.0°F

These containment initial conditions are representatively low values anticipated during normal full power operation.

6.2.1.5.3 Containment Volume

The volume used in the analysis is 2.35 x 10^6 cubic feet, the maximum estimated volume. This value was determined by calculating the containment gross volume and subtracting the volumes of all of the containment internal structures and equipment.

The gross volume was maximized by assuming the containment liner is erected at the maximum radial tolerance. The internal volume subtracted from the gross volume was minimized by reducing the nominal values for internal concrete and some equipment by 5 percent. 2.40 2.41 2.42

6.2.1.5.4 Active Heat Sinks 27.47

The quench spray system operates to remove heat from the containment. 27.48
Table 6.2-60 gives the pertinent data for this system. 27.50

The heat removal capacity of this system is maximized by using the 27.51
minimum RWST water temperature, the minimum delay time for the system 27.52
to become effective, and maximum system flow rates.

6.2.1.5.5 Steam Water Mixing 27.54

Water spillage rates from the broken loop accumulator are determined 27.55
as part of the core reflooding calculation and are included in the 27.56
containment code (COCO) calculational model.

6.2.1.5.6 Passive Heat Sinks 6.2-77 ~~6.2-76~~ 27.59

The passive heat sinks used in the analysis, with their 27.60
thermophysical properties, are given in Table 6.2-1. The passive 28.2
heat sinks and thermophysical properties were derived in compliance
with Branch Technical Position CSB 6-1, Minimum Containment Pressure 28.3
Model for PWR ECCS Performance Evaluation.

6.2.1.5.7 Heat Transfer to Passive Heat Sinks 6.2-60 28.5

The condensing heat transfer coefficients used for heat transfer to 28.6
the steel containment structures are given on Figure 6.2-12 for the 28.7
limiting break. ~~The containment pressure transient for the limiting~~ 28.9
~~break is shown on Figure 6.2-14.~~

6.2.1.5.8 Other Parameters 28.12

No other parameters, including the operation of the containment mini- 28.13
purge system, have a substantial effect on the minimum containment 28.14
pressure analysis.

6.2.1.6 Testing and Inspection 28.17

Preoperational and periodic tests are performed on the containment 28.19
structure and supporting systems. These are discussed in the 28.20
sections as referenced.

Test	Section	28.25
Containment shell leakage	6.2.6	28.27
Containment valve and penetration leakage	6.2.6	28.28
Containment spray system	6.2.2.4	28.30
Containment atmosphere recirculation	9.4.7.3.4	28.31
ESF sump test	6.2.2.4	28.32
High head safety injection	6.3.4	28.33
Low head safety injection	6.3.4	28.34
Residual heat removal (RHR)	5.4.7.4	28.35

MASS AND ENERGY

TABLE 3.6.2-72

RELEASE DUE TO BROKEN LOOP INJECTION SPILL TO CONTAINMENT ~~(CD=0.6) (MAX SI)~~

DURING BLOWDOWN ~~(CD=0.6, MAX SI)~~

FOR THE LIMITING BREAK

TIME (sec)	MASS (lb/sec)	ENERGY (Btu/sec)	ENTHALPY (Btu/lb)
0.000	3087.145	184055.606	59.620
1.010	2858.045	170396.659	59.620
2.010	2675.961	159540.769	59.620
3.010	2525.853	150591.358	59.620
4.010	2399.673	143068.529	59.620
5.010	2290.872	136581.774	59.620
6.010	2195.482	130894.624	59.620
7.010	2110.787	125845.125	59.620
8.010	2034.766	121312.753	59.620
9.010	1966.066	117216.843	59.620
10.010	1903.785	113503.645	59.620
11.010	1847.136	110126.266	59.620
12.010	1795.268	107033.861	59.620
13.010	1747.501	104186.002	59.620
14.010	1703.190	101544.188	59.620
15.010	1661.934	99084.506	59.620
16.010	1623.636	96801.181	59.620
17.010	1588.062	94680.249	59.620
18.010	1554.880	92701.961	59.620
19.010	1523.809	90849.468	59.620
20.010	1494.690	89113.403	59.620
21.010	1467.827	87511.825	59.620
22.010	1443.082	86036.534	59.620
23.010	1419.715	84643.408	59.620
24.010	1397.439	83315.306	59.620
25.010	1376.467	82064.951	59.620
26.010	1356.808	80892.883	59.620
27.010	1338.211	79784.156	59.620
28.010	1488.012	73875.512	49.647
29.010	1474.120	72904.721	49.456
30.010	1460.817	71974.923	49.270
31.010	1448.020	71081.103	49.088

TABLE ~~Ø~~ 6.2-73

DECLG BLOWDOWN MASS AND ENERGY RELEASES FOR THE LIMITING BREAK ~~(CD=0.6) MAX SG~~)

TIME (sec)	MASS FLOW (lbm/sec)	ENERGY FLOW (million Btu/sec)
ø		
0.0	0.0	0.0
0.05	61,359.	33.73
0.20	64,895.	35.70
1.0	60,881.	33.64
2.0	52,596.	29.47
3.0	40,650.	23.10
4.0	33,483.	19.52
5.0	31,749.	19.02
6.0	29,026.	17.59
7.0	26,837	16.53
8.0	24,925.	15.77
9.0	21,593.	14.31
10.0	18,600.	12.74
12.0	15,344.	10.54
14.00	13,413.	9.27
16.00	9,949.	7.38
18.00	6,833.	5.68
20.00	4,533.	4.09
22.00	2,411.	2.09
24.00	6,219.	2.70
26.00	4,063.	1.47
28.00	3,045.	1.02
31.68	69.	0.04

TABLE 6.2-79

DECLG REFLOOD MASS AND ENERGY RELEASES FOR THE LIMITING BREAK ~~OF~~ ~~(CD=0.6)~~ ~~(MAX SI)~~

TIME (sec)	MASS FLOW (lbm/sec)	ENERGY FLOW (Btu/sec)
44.576	0	0
49.957	34.150	44,409.
59.593	853.10	193,768.
74.388	1054.57	221,162.
94.288	1085.56	217,750.
117.488	1103.40	211,503.
143.788	1119.22	204,667.
209.188	1152.72	189,998.
361.488	1208.44	168,941.

TABLE 6.2-74 75

BROKEN LOOP INJECTION SPILL DURING BLOWDOWN

(DECLG [Active Loop Break] C_D = 0.4)
(N-1 loop operation)

Time (sec)	Mass (lbm/sec)	Energy (Btu/sec)	Enthalpy (Btu/lbm)
0.000	3048.576	181756.118	59.620
1.010	2775.613	165482.036	59.620
2.010	2568.818	153152.907	59.620
3.010	2403.708	143309.101	59.620
4.010	2267.928	135213.878	59.620
5.010	2153.510	128392.292	59.620
6.010	2055.089	122524.400	59.620
7.010	1969.152	117400.826	59.620
8.010	1892.950	112857.702	59.620
9.010	1824.701	108788.694	59.620
10.010	1763.089	105115.356	59.620
11.010	1707.124	101778.752	59.620
12.010	1656.062	98734.427	59.620
13.010	1609.241	95942.952	59.620
14.010	1566.126	93372.426	59.620
15.010	1526.205	90992.350	59.620
16.010	1488.961	88771.876	59.620
17.010	1454.129	86695.143	59.620
18.010	1421.611	84756.472	59.620
19.010	1391.227	82944.957	59.620
20.010	1362.808	81250.616	59.620
21.010	1336.138	79660.556	59.620
22.010	1311.025	78163.326	59.620
23.010	1287.296	76748.563	59.620
24.010	1264.954	75416.560	59.620
25.010	1244.180	74178.002	59.620
26.010	1224.596	73010.401	59.620
27.010	1205.820	71890.991	59.620
28.010	1188.017	70829.569	59.620
29.010	1186.953	69952.315	58.934
30.010	1171.073	69005.455	58.925
31.010	1155.971	68104.972	58.916
32.010	1141.605	67248.412	58.977
33.010	1127.937	66433.430	58.898
34.010	1114.800	65650.107	58.890
35.010	1102.232	64900.729	58.881

12

TABLE 6.2-15 76

REFLOOD MASS AND ENERGY RELEASES
(DECLG Active Loop Break C_D = 0.4)
(N-1 loop operation)

Time (sec)	m Total (lbm/sec)	mh Total (Btu/sec)
50.751	0	0
51.326	0.053	69.
51.926	0.053	69.
52.126	0.928	1205.
57.313	30.39	40262.
68.882	342.63	120971.
89.132	386.56	123465.
114.532	403.24	122106.
143.532	414.32	119453.
175.632	423.49	116251.
249.532	440.32	109099.
343.532	461.75	101733.

6.2-~~25~~ 77

PASSIVE ~~STRUCTURAL~~ HEAT SINK DATA

Wall	Material	Thickness (in)	Area (Ft^2)
1	Paint	0.004	
	Carbon [redacted] Steel	0.11	191922.6
2	Stainless Steel	0.45	8540.00
3	Concrete	25.92	1732.0
4	Concrete	18.0	15348.0
5	Paint	0.004	
	Concrete	16.32	133277.0
6	Paint	0.004	
	Concrete	25.32	17926.0
7	Paint	0.004	
	Concrete	36.0	6563.0
8	Paint	0.004	
	Concrete	21.0	2007.0
9	Paint	0.004	
	Concrete	24.0	
	Carbon Steel	0.25	
	Concrete	120.0	12269.0
10	Paint	0.004	
	Carbon Steel	0.514	
	Concrete	54.0	24675.0
11	Paint	0.004	
	Carbon Steel	0.514	
	Concrete	54.0	38493.0

TABLE ~~6~~ ~~(Page 2 of 2)~~ 6.2-77 (CONT.)

PASSIVE ~~STRUCTURAL~~ HEAT SINK DATA

Wall	Material	Thickness (in)	Area (Ft^2)
12	Paint	0.004	
	Carbon Steel	0.554	
	Concrete	30.72	34100.0
13	Stainless Steel	1.29	1722.0
14	Paint	0.004	
	Carbon Steel	0.710	552.0
15	Stainless Steel	0.24	13200.0
16	Stainless Steel	0.658	2063.0
17	Paint	0.004	
	Carbon Steel	0.99	1282.0
18	Paint	0.004	
	Carbon Steel	0.383	239261.0
19	Paint	0.004	
	Stainless Steel	0.728	4342.0
20	Paint	0.004	
	Carbon Steel	1.158	21803.0

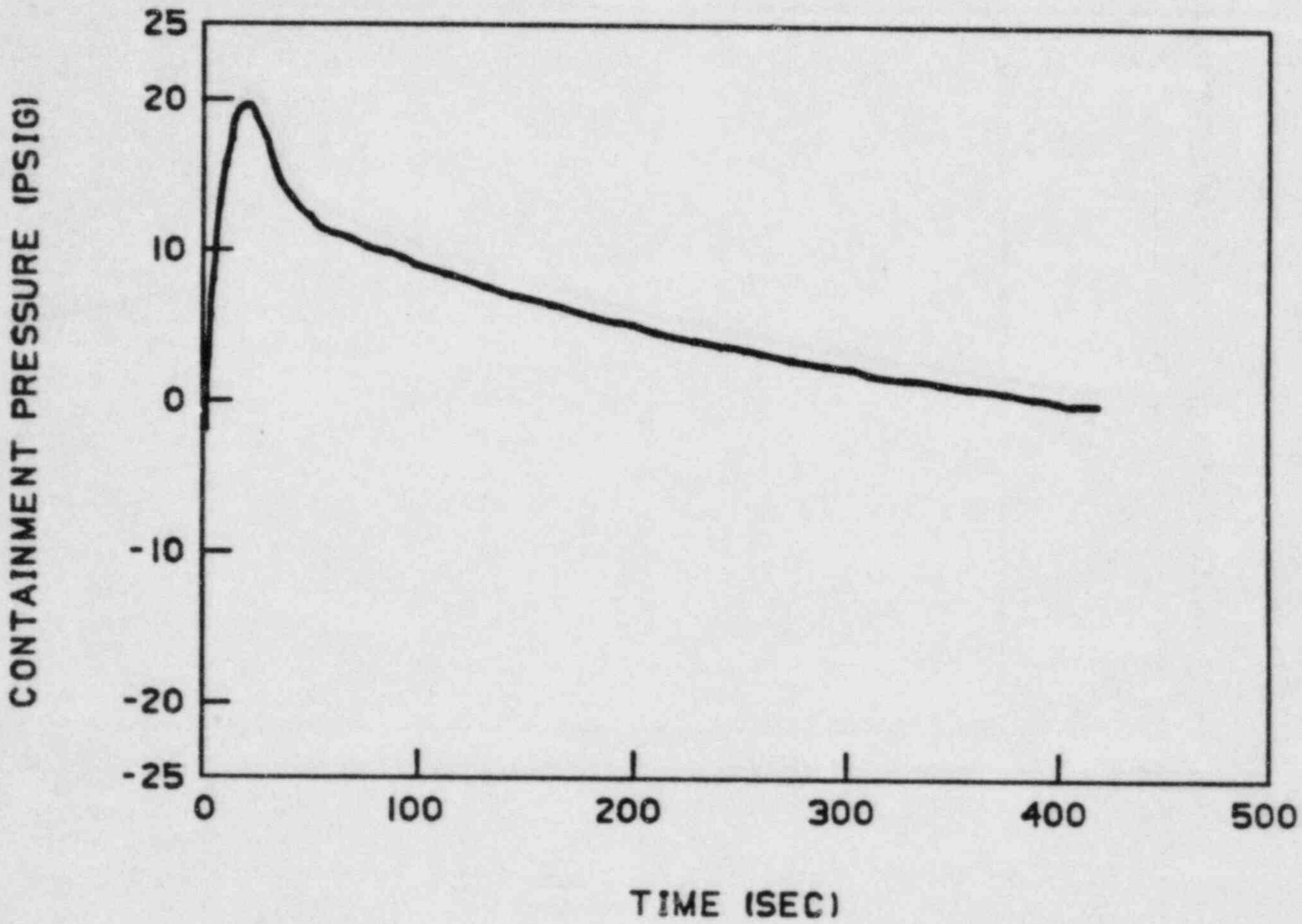
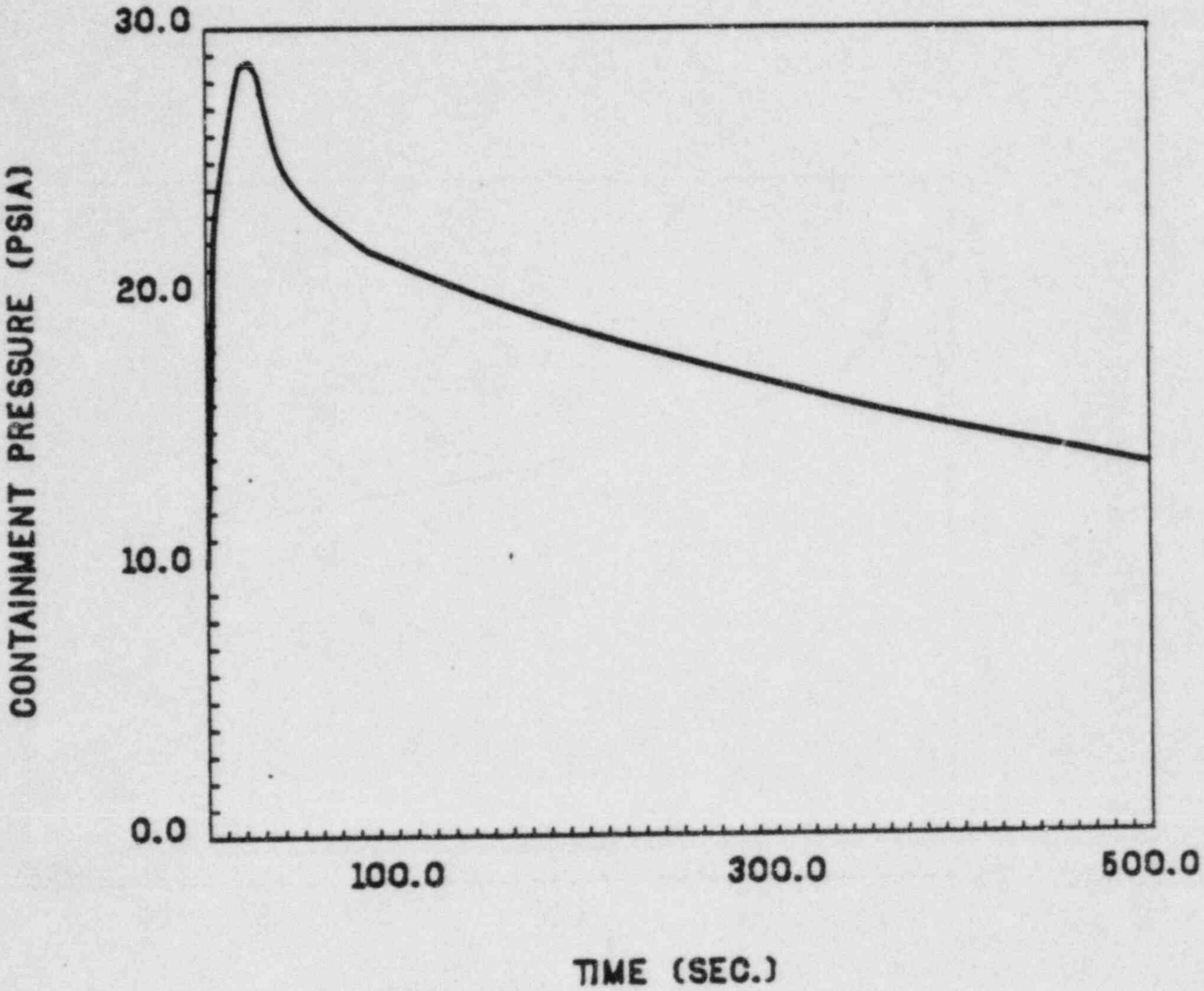


Figure 6.2-59

Containment Pressure

LIMITING BREAK (N-loop operation)

Delete existing Fig. in the FSAR and Replace with
~~Note: Replace with this~~ figure this figure



59 a

FIGURE 6.2-(6.2.1.5)b

CONTAINMENT PRESSURE

DECLG (ACTIVE LOOP BREAK CD=0.4)

(N-1 LOOP OPERATION)

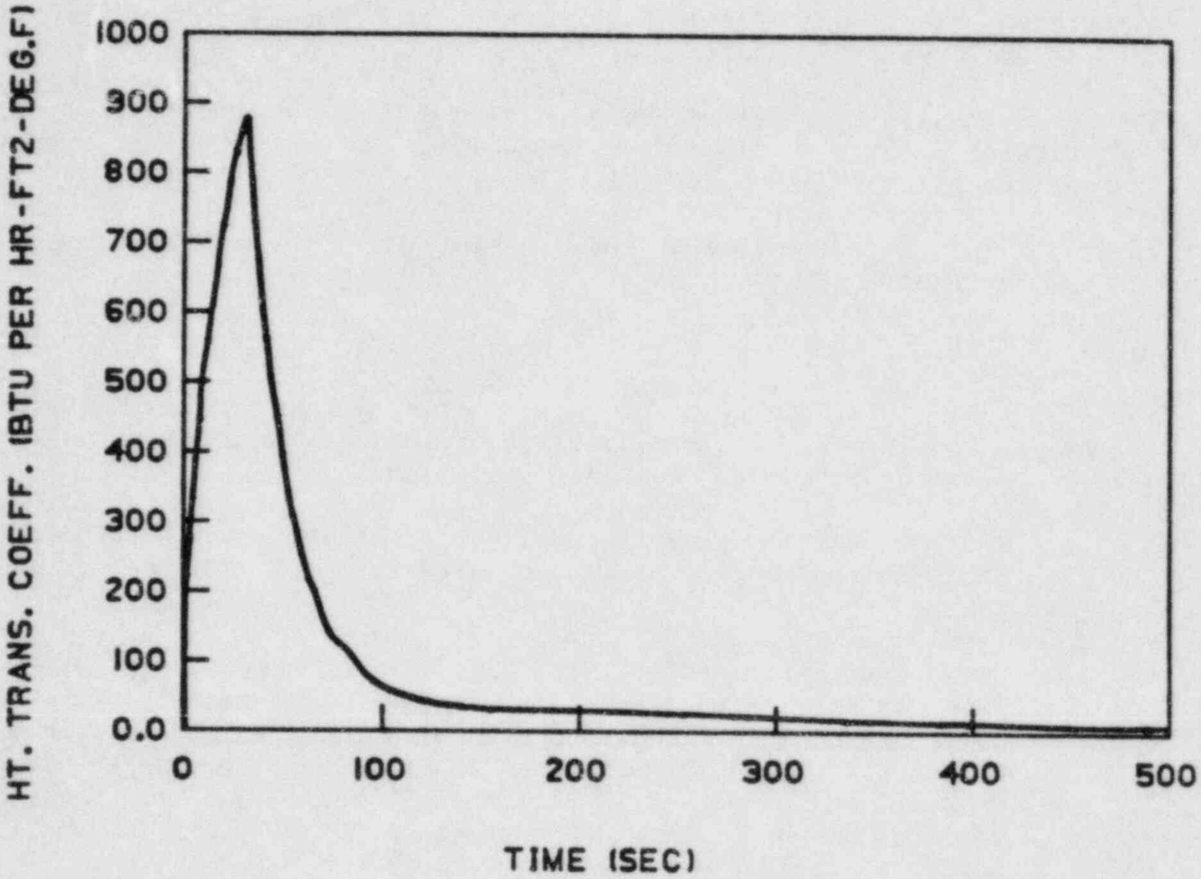


Figure 6.2-60

Condensing Wall Heat Transfer Coefficients

LIMITING BREAK

(N loop operation)

will automatically maintain prescribed conditions in the plant even 1.12
under a conservative set of reactivity parameters with respect to both
system stability and transient performance. 1.13

For each mode of plant operation, a group of optimum controller 1.14
setpoints is determined. In areas where the resultant setpoints are 1.15
different, compromises based on the optimum overall performance are made
and verified. A consistent set of control system parameters is derived 1.17
satisfying plant operational requirements throughout the core life and 1.18
for various levels of power operation.

The study will comprise an analysis of the following control systems: 1.19
rod cluster control assembly, steam dump, steam generator level, 1.20
pressurizer pressure and pressurizer level.

15.0.3 Plant Characteristics and Initial Conditions Assumed in the Accident Analyses 1.24

15.0.3.1 Design Plant Conditions 1.27

Table 15.0-1 lists the principal power rating values which are assumed 1.28
in analyses performed in this report. Four ratings are given: two for 1.31 12-
N-loop operation and two for N-1 loop operation.

1. The guaranteed NSSS thermal power output. This power output 1.34
includes the thermal power generated by the reactor coolant
pumps.

2. The engineered safety features design rating. The NSSS 1.36
supplied engineered safety features are designed for thermal
power higher than the guaranteed value in order not to preclude 1.37
realization of future potential power capability. This higher 1.38
thermal power value is designated as the engineered safety
features design rating. This power output includes the thermal 1.39
power generated by the reactor coolant pumps.

Where initial power operating conditions are assumed in accident 1.41
analyses, the guaranteed NSSS thermal power output, plus allowance for 1.42
errors in steady state power determination, is assumed. Where 1.43
demonstration of adequacy of the containment and engineered safety
features are concerned, the engineered safety features design rating, 1.44
plus allowance for error, is assumed. The thermal power values used for 1.45
each transient analyzed are given in Table 15.0-2. In all cases where 1.46
the 3,579 megawatt thermal (MWt) rating is used in an analysis, the
resulting transients and consequences are conservative compared to using 1.47
the 3,425 MWt rating.

The values of other pertinent plant parameters utilized in the accident 1.48
analyses are given in Table 15.0-3.

15.0.3.2 Initial Conditions

For accident evaluation, the initial conditions are obtained by adding the maximum steady state errors to rated values. The following steady state errors are considered:

1.	Core power	± 2 percent allowance for calorimetric error
2.	Average reactor coolant system temperature	± 6.5°F (6.1) allowance for controller deadband and measurement error and steam generator fouling penalty
3.	Pressurizer pressure	± 30 psi allowance for steady state fluctuations and measurement penalty

Initial values for core power, average reactor coolant system temperature, and pressurizer pressure are selected to minimize the initial departure from nucleate boiling ratio (DNBR) unless otherwise stated in the sections describing specific accidents. Table 15.0-2 summarizes the initial conditions and computer codes used in the accident analyses.

15.0.3.3 Power Distribution

The transient response of the reactor system is dependent on the initial power distribution. The nuclear design of the reactor core minimizes adverse power distribution through the placement of control rods and operating instructions. Power distribution may be characterized by the radial factor ($F\Delta H$) and the total peaking factor (F_q). The peaking factor limits are given in the Technical Specifications.

For transients which may be DNB limited, the radial peaking factor is of importance. The radial peaking factor increases with decreasing power level due to rod insertion. This increase in $F\Delta H$ is included in the core limits illustrated on Figure 15.0-1. All transients that may be DNB limited are assumed to begin with a $F\Delta H$ consistent with the initial power level defined in the Technical Specifications.

The axial power shape used in the DNB calculation is the 1.55 chopped cosine as discussed in Section 4.4. The radial and axial power distributions described above are input to the THINC Code as described in Section 4.4.

For transients which may be overpower limited, the total peaking factor (F_q) is of importance. All transients that may be overpower limited are assumed to begin with plant conditions including power distributions which are consistent with reactor operation as defined in the Technical Specifications.

For overpower transients which are slow with respect to the fuel rod thermal time constant, for example the chemical and volume control

system malfunction that results in a decrease in the boron concentration
in the reactor coolant incident which lasts many minutes, and the 2.36
excessive increase in secondary steam flow incident which may reach
equilibrium without causing a reactor trip, the fuel rod thermal 2.37
evaluations are performed as discussed in Section 4.4. For overpower 2.39
transients which are fast with respect to the fuel rod thermal time
constant, e.g., the uncontrolled rod cluster control assembly bank 2.40
withdrawal from subcritical or low power startup and rod cluster control
assembly ejection incidents which result in a large power rise over a 2.41
few seconds, a detailed fuel heat transfer calculation must be 2.42
performed. Although the fuel rod thermal time constant is a function of 2.43
system conditions, fuel burnup and rod power, a typical value at 2.44
beginning-of-life for high power rods is approximately 5 seconds.

15.0.4 Reactivity Coefficients Assumed in the Accident Analyses 2.46

The transient response of the reactor system is dependent on reactivity 2.47
feedback effects, in particular the moderator temperature coefficient 2.49
and the Doppler power coefficient. These reactivity coefficients and 2.50
their values are discussed in detail in Chapter 4. (See Figure 15.0-6).

In the analysis of certain events, conservatism requires the use of 2.51
large reactivity coefficient values whereas in the analysis of other 2.52
events, conservatism requires the use of small reactivity coefficient
values. Some analyses such as loss of reactor coolant from cracks or 2.53
ruptures in the reactor coolant system do not depend on reactivity 2.54
feedback effects. The values used are given in Table 15.0-2. Reference 2.56
is made in that table to Figure 15.0-2, which shows the upper and lower
bound Doppler power coefficients as a function of power, used in the 2.57
transient analysis. The justification for use of conservatively large 2.58
versus small reactivity coefficient values are treated on an event-by- 2.59
event basis. In some cases, conservative combinations of parameters are 2.60
used to bound the effects of core life. For example, in a load increase 3.1
transient, it is conservative to use a small Doppler defect and a small
moderator coefficient. 3.2

15.0.5 Rod Cluster Control Assembly Insertion Characteristics 3.4

The negative reactivity insertion following a reactor trip is a function 3.5
of the acceleration of the rod cluster control assemblies and the 3.7
variation in rod worth as a function of rod position. With respect to 3.8
accident analyses, the critical parameter is the time of insertion up to
the dashpot entry or approximately 85 percent of the rod cluster travel. 3.9
The rod cluster control assembly position versus time assumed in 3.10
accident analyses is shown on Figure 15.0-3. The rod cluster control 3.11
assembly insertion time to dashpot entry is taken as 2.3 seconds, unless 2.2
otherwise noted in the discussion. The use of such a long insertion 3.13
time provides the most conservative results for all accidents and is
intended to be applicable to all types of rod cluster control assemblies 3.14
which may be used throughout plant life. Drop time testing requirements 3.15
are dependent on the type of rod cluster control assemblies actually
used in the plant and are specified in the plant Technical 3.16
Specifications.

Figure 15.0-4 shows the fraction of total negative reactivity insertion versus normalized rod position for a core where the axial distribution is skewed to the lower region of the core. An axial distribution which is skewed to the lower region of the core can arise from an unbalanced xenon distribution. This curve is used to compute the negative reactivity insertion versus time following a reactor trip which is input to all point kinetics core models used in transient analyses. The bottom skewed power distribution itself is not an input into the point kinetics core model.

There is inherent conservatism in the use of Figure 15.0-4 in that it is based on skewed flux distribution which would exist relatively infrequently. For cases other than those associated with unbalanced xenon distributions, significant negative reactivity would have been inserted due to the more favorable axial distribution existing prior to trip.

The normalized rod cluster control assembly negative reactivity insertion versus time is shown on Figure 15.0-5. The curve shown on this figure was obtained from Figures 15.0-3 and 15.0-4. A total negative reactivity insertion following a trip of 4 percent is assumed in the transient analyses except where specifically noted otherwise. This assumption is conservative with respect to the calculated trip reactivity worth available as shown in Table 4.3-3. For Figures 15.0-3 and 15.0-4, the rod cluster control assembly drop time is normalized to 2.2 1.3 seconds, unless otherwise noted for a particular event, in order to provide a bounding analysis for all rod cluster control assemblies to be used in the Millstone 3 cores, as previously stated.

The normalized rod cluster control assembly negative reactivity insertion versus time curve for an axial power distribution skewed to the bottom (Figure 15.0-5) is used in those transient analyses for which a point kinetics core model is used. Where special analyses require use of three dimensional or axial one dimensional core models, the negative reactivity insertion resulting from the reactor trip is calculated directly by the reactor kinetics code and is not separable from the other reactivity feedback effects. In this case, the rod cluster control assembly position versus time of Figure 15.0-3 is used as code input.

15.0.6 Trip Points and Time Delays to Trip Assumed in Accident Analyses

A reactor trip signal acts to open two trip breakers connected in series feeding power to the control rod drive mechanisms. The loss of power to the mechanism coils causes the mechanisms to release the rod cluster control assemblies which then fall by gravity into the core. There are various instrumentation delays associated with each trip function, including delays in signal actuation, in opening the trip breakers, and in the release of the rods by the mechanisms. The total delay to trip is defined as the time delay from the time that trip conditions are reached to the time the rods are free and begin to fall. Limiting trip setpoints assumed in accident analyses and the time delay assumed for each trip function are given in Table 15.0-4.

Reference is made in Table 15.0-4 to the overtemperature and overpower trip shown on Figures 15.0-1 and 15.0-1A. These figures present the allowable reactor coolant loop average temperature and ΔT for the flow and power distribution, as described in Section 4.4, as a function of primary coolant pressure. The boundaries of operation defined by the overpower ΔT trip and the overtemperature ΔT trip are represented as "protection lines" on this diagram. The protection lines are drawn to include all adverse instrumentation and set point errors so that, under nominal conditions, trip would occur well within the area bounded by these lines. The utility of this diagram is that the limit imposed by any given DNBR can be represented as a line. The DNB lines represent the locus of conditions for which DNBR equals the limit value (1.30). All points below and to the left of a DNB line for a given pressure have a DNBR greater than the limit value. The diagram shows that DNB is prevented for all cases if the area enclosed with the maximum protection lines is not traversed by the applicable DNBR line at any point.

The area of permissible operation (power, pressure, and temperature) is bounded by the combination of reactor trips: high neutron flux (fixed set point); high pressure (fixed set point); low pressure (fixed set point); and overpower and overtemperature ΔT (variable set points).

The limit value, which was used as the DNBR limit for all accidents, is conservative compared to the actual design DNBR value required (1.31 for the thimble cell and 1.33 for the typical cell) to meet the DNB design basis as discussed in Section 4.4.

The difference between the limiting trip point assumed for the analysis and the nominal trip point represents an allowance for instrumentation channel error and setpoint error. Nominal trip setpoints are specified in the plant Technical Specifications. During plant startup tests, it will be demonstrated that actual instrument time delays are equal to or less than the assumed values. Additionally, protection system channels are calibrated and instrument response times determined periodically in accordance with the plant Technical Specifications.

15.0.7 Instrumentation Drift and Calorimetric Errors - Power Range Neutron Flux

The instrumentation drift and calorimetric errors used in establishing the power range high neutron flux setpoint are presented in Table 15.0-5.

The calorimetric error is the error assumed in the determination of core thermal power as obtained from secondary plant measurements. The total ion chamber current (sum of the top and bottom sections) is calibrated (set equal) to this measured power on a periodic basis.

The secondary power is obtained from measurement of feedwater flow, feedwater inlet temperature to the steam generators and steam pressure. High accuracy instrumentation is provided for these measurements with accuracy tolerances much tighter than those which would be required to control feedwater flow.

15.0.8 Plant Systems and Components Available for Mitigation of Accident Effects 4.51 4.52

The NSSS is designed to afford proper protection against the possible 4.55
effects of natural phenomena, postulated environmental conditions and 4.57
dynamic effects of the postulated accidents. In addition, the design 4.58
incorporates features which minimize the probability and effects of
fires and explosions. Chapter 17 discusses the quality assurance 4.59
program which has been implemented to assure that the NSSS will
satisfactorily perform its assigned safety functions. The incorporation 5.1
of these features in the NSSS, coupled with the reliability of the
design, ensures that the normally operating systems and components 5.2
listed in Table 15.0-6 will be available for mitigation of the events
discussed in Chapter 15. In determining which systems are necessary to 5.4
mitigate the effects of these postulated events, the classification
system of ANSI-N18.2-1973 is utilized. The design of "systems important 5.6
to safety" (including protection systems) is consistent with IEEE
Standard 379-1972 and Regulatory Guide 1.53 in the application of the 5.7
single failure criterion.

In the analysis of the Chapter 15 events, control system action is 5.8
considered only if that action results in more severe accident results. 5.9
No credit is taken for control system operation if that operation 5.10
mitigates the results of an accident. For some accidents, the analysis 5.12
is performed both with and without control system operation to determine
the worst case. The pressurizer heaters are not assumed to be energized 5.14
during any of the Chapter 15 events.

TABLE 15.0-1

NUCLEAR STEAM SUPPLY SYSTEM POWER RATINGS

	N-Loop	N-1 Loop[1]
Guaranteed NSSS thermal power output (MWt)	3,425	2,569
Thermal power generated by the reactor coolant pumps (MWt)	14	9
Guaranteed core thermal power (MWt)[1]	3,411	2,560
Engineered safety features design rating (maximum calculated turbine rating) (MWt)	3,579	3,579

NOTE:

1. LOCA accident analysis currently limits the guaranteed core thermal power to 2,217 MWt. The original accident analyses were performed with an assumed 75 percent power and if this was found unacceptable, as in the case of the LOCA analysis, a reanalysis was done based on a conservative step decrease in power. The limiting power level is therefore subject to change.

TABLE 15.0-2

SUMMARY OF INITIAL CONDITIONS AND COMPUTER CODES USED

Faults	Computer Codes Utilized	Reactivity Coefficients Assumed (a): Moderator Temperature (Δk/°F)	Reactivity Coefficients Assumed (a): Moderator Density (Δk/gm/cc)	Doppler	Initial NSSS Thermal Power Output Assumed (b) (MWt): N-Loop	Initial NSSS Thermal Power Output Assumed (b) (MWt): N-1 Loop
15.1 Increase in heat removal by the secondary system						
Feedwater system malfunctions that result in an increase in feedwater flow	LOFTRAN	-	0.43	lower (a)	0 & 3,425	2,569
Excessive increase in secondary steam flow	LOFTRAN	-	Fig. 15.0-6 and 0.43	lower (a)	3,425	2,569
Inadvertent opening of a steam generator relief or safety valve	LOFTRAN	-	Function of moderator density (Sec. 15.1.4, Fig. 15.1-11)	-2.2 pcm/°F	0 (Subcritical)	
Steam system piping failure	LOFTRAN, THINC	-	Function of moderator density (Sec. ~~tion~~ 15.1.5, Fig. 15.1-11)	Refer to Sec. 15.1.5	0 (Subcritical)	
15.2 Decrease in heat removal by the secondary system						
Loss of external electrical load and/or turbine trip	LOFTRAN	-	Fig. 15.0-6 and 0.43	upper (a)	3,425	2,569
Loss of nonemergency ac power to the station auxiliaries	LOFTRAN	-	NA	NA	3,579	2,755.83
Loss of normal feedwater flow	LOFTRAN	-	Fig. 15.0-6	upper (a)	3,579	2,755.83
Feedwater system pipe break	LOFTRAN, FACTRAN, THINC	-	Fig. 15.0-6	upper (a)	3,579	2,755.83
15.3 Decrease in reactor coolant system flow rate						
Partial and complete loss of forced reactor coolant flow	LOFTRAN, FACTRAN, THINC	-	Fig. 15.0-6	upper (a)	3,425	2,569

12

TABLE 15.0-2 (Cont)

Faults	Computer Codes Utilized	Reactivity Coefficients Assumed (a): Moderator Temperature (Δk/°F)	Reactivity Coefficients Assumed (a): Moderator Density (Δk/gm/cc)	Doppler	Initial NSSS Thermal Power Output Assumed (b) (MWt): N-Loop	Initial NSSS Thermal Power Output Assumed (b) (MWt): N-1 Loop
Reactor coolant pump shaft seizure (locked rotor)	LOFTRAN, FACTRAN	-	Fig. 15.0-6	upper (a)	3,425	2,569
15.4 Reactivity and power distribution anomalies						
Uncontrolled rod cluster control assembly bank withdrawal from a subcritical or lower power startup condition	TWINKLE, FACTRAN, THINC	Refer to Sec. 15.4.1.2	-	Consistent with lower limit shown on Fig. 15.0-2	0	(Subcritical)
Uncontrolled rod cluster control assembly bank withdrawal at power	LOFTRAN	-	Fig. 15.0-6 and 0.43	lower and upper (a)	3,425	2,569
Rod cluster control assembly misalignment	THINC, TURTLE LOFTRAN, LEOPARD	-	Fig. 15.0-6	lower (a)	3,425	-
Startup of an inactive reactor coolant loop at an incorrect temperature	LOFTRAN, FACTRAN, THINC	-	0.43	lower (a)	2,569 2,398	NA
Chemical and volume control system malfunction that results in a decrease in the boron concentration in the reactor coolant	NA	NA	NA	NA	0 & 3,425	
Inadvertent loading and operation of a fuel assembly in an improper position	LEOPARD, TURTLE	-	NA	NA	3,425	
Spectrum of rod cluster control assembly ejection accidents	TWINKLE, FACTRAN, LEOPARD	Refer to Sec. 15.4.8 min., max. feedback	-	Consistent with lower limit shown on Fig. 15.0-2	0 & 3,425	2,569
15.5 Increase in reactor coolant inventory						
Inadvertent operation of the ECCS during power operation	LOFTRAN	-	Fig. 15.0-6	lower (a)	3,425	2,569

12

TABLE 15.0-2 (Cont)

Faults	Computer Codes Utilized	Reactivity Coefficients Assumed (a): Moderator Temperature (Δk/°F)	Reactivity Coefficients Assumed (a): Moderator Density (Δk/gm/cc)	Doppler	Initial NSSS Thermal Power Output Assumed (b) (MWt): N-Loop	Initial NSSS Thermal Power Output Assumed (b) (MWt): N-1 Loop	
15.6 Decrease in reactor coolant inventory							3.1 3.2
Inadvertent opening of a pressurizer safety or relief valve	LOFTRAN	-	Fig. 15.0-6	upper (a)	3,425	2,569	3.4 3.5 3.6
Steam generator tube failure	NA	NA	NA	NA	3,565		3.9
Loss of coolant accidents resulting from the spectrum of postulated piping breaks within the reactor coolant pressure boundary	SATAN-VI, WFLASH, WREFLOOD, COCO, LOCTA-IV	Refer to Sec. 15.6.5, References	-	Refer to Sec. 15.6.6, References	3,411	2,817 2,7[illegible]	3.11 12 3.12 3.13 3.14 3.15

NOTES: 3.19

(a) Refer to Fig. 15.0-2. 3.22

(b) A minimum 2 percent margin is applied to the values shown for analysis purposes. 3.25

TABLE 15.0-3

NOMINAL VALUES OF PERTINENT PLANT PARAMETERS
UTILIZED IN THE ACCIDENT ANALYSIS[1]

	N-Loop	N-1 Loop
Thermal output of NSSS (MWt)	Table 15.0-2	Table 15.0-2
Core inlet temperature (°F)	557.0	550.6
Vessel average temperature (°F)	587.1	579.6
Reactor coolant system pressure (psia)	2,250	2,250
Reactor coolant flow per loop (gpm)	94,600	99,600
Steam flow from NSSS (lb/hr)	15,050,000	10,780,000
Steam pressure at steam generator outlet (psia)	990	935
Maximum steam moisture content (percent)	0.25	0.25
Assumed feedwater temperature at steam generator inlet (°F)	436.2	404.2
Average core heat flux (Btu/hr-ft²)	189,800	142,400

12-

NOTE:

1. Steady state errors discussed in Section 15.0.3 are added to these values to obtain initial conditions for transient analyses.

TABLE 15.0-8

POTENTIAL OFFSITE DOSES DUE TO ACCIDENTS

Postulated Accident	FSAR Section	Dose (Rem) - 2 Hr Exclusion Area Boundary (524 m) Thyroid	Gamma	Beta	Dose (Rem) - Low Population Zone (3862 m) Thyroid	Gamma	Beta
Main Steam Line Break	15.1.5						
1. N-Loop							
a) Case I: MSLB with 1% Failed Fuel		8.3E+00	1.0E-01 (1)	4.7E-02	6.6E-01	9.6E-03	4.4E-03
b) Case II: MSLB with preaccident iodine spike		2.1E+00	2.7E-03	1.0E-03	1.4E-01	1.2E-04	7.3E-05
2. N-1 Loop							
a) Case I: MSLB with 1% Failed Fuel		1.2E+01	1.5E-01	6.9E-02	7.9E-01	1.4E-02	6.3E-03
b) Case II: MSLB with preaccident iodine spike		2.1E+00	2.8E-03	1.1E-03	1.3E-01	1.8E-04	7.0E-05
Locked Rotor Accident	15.3.3						
1. N-Loop		2.9E+00	1.6E-01	7.9E-02	1.9E+00	3.1E-02	1.4E-02
2. N-1 Loop		5.4E+00	2.9E-01	1.4E-01	3.7E+00	5.0E-02	2.3E-02
Rod Ejection Accident	15.4.8						
1. N-Loop							
a) Primary Side		1.2E+01	2.7E-01	1.3E-01	7.5E-01	3.1E-02	1.2E-02
b) Secondary Side		1.7E-01	2.1E-02	1.7E-02	9.1E-03	1.1E-03	9.3E-04
2. N-1 Loop							
a) Primary Side		1.3E+01	2.8E-01	1.3E-01	7.6E-01	2.7E-02	1.0E-02
b) Secondary Side		2.0E-02	4.4E-03	3.5E-03	1.1E-03	2.3E-04	1.9E-04
Small Line LOCA outside containment	15.6.2	2.1E+01	1.5E-01	3.7E-02	(2)	(2)	(2)
Steam Generator Tube Rupture	15.6.3						
a) Preaccident Iodine Spike		2.1E+00	1.9E-02	9.6E-03	2.4E-01	1.2E-03	5.7E-04
b) Concurrent iodine spike		3.4E-01	1.8E-02	8.9E-03	7.6E-02	1.1E-03	5.3E-04

TABLE 15.0-8 (Cont)

Postulated Accident	FSAR Section	Dose (Rem) - 2 Hr Exclusion Area Boundary (524 m) Thyroid	Gamma	Beta	Dose (Rem) - Low Population Zone (3862 m) Thyroid	Gamma	Beta	
LOCA	15.6.5	2.4E+02	1.7E+01	9.1E+00	1.6E+01	1.6E+00	7.3E-01	2.12
Waste Gas System Failure	15.7.1	0.0E+00	2.2E-01	1.7E-01	(2)	(2)	(2)	2.14 460-11 2.15
Radioactive Liquid Waste System Leak or Failure (Atmospheric Release)	15.7.2	4.3E-01	4.7E-04	2.5E-04	(2)	(2)	(2)	2.17 2.18 2.19 2.20
Fuel Handling Accident	15.7.4	7.6E+00	5.9E-01	6.8E-01	(2)	(2)	(2)	2.23
Spent Fuel Cask Drop	15.7.5	(3)	(3)	(3)	(3)	(3)	(3)	2.25

NOTES: 2.28

1. 1.0E-01 = 1.0×10^{-1} 2.31
2. 2 hours of release or less; a 30-day dose is not applicable. 2.32
3. Not applicable - see Sections 15.7.5.2 and 15.7.5.3. 2.33

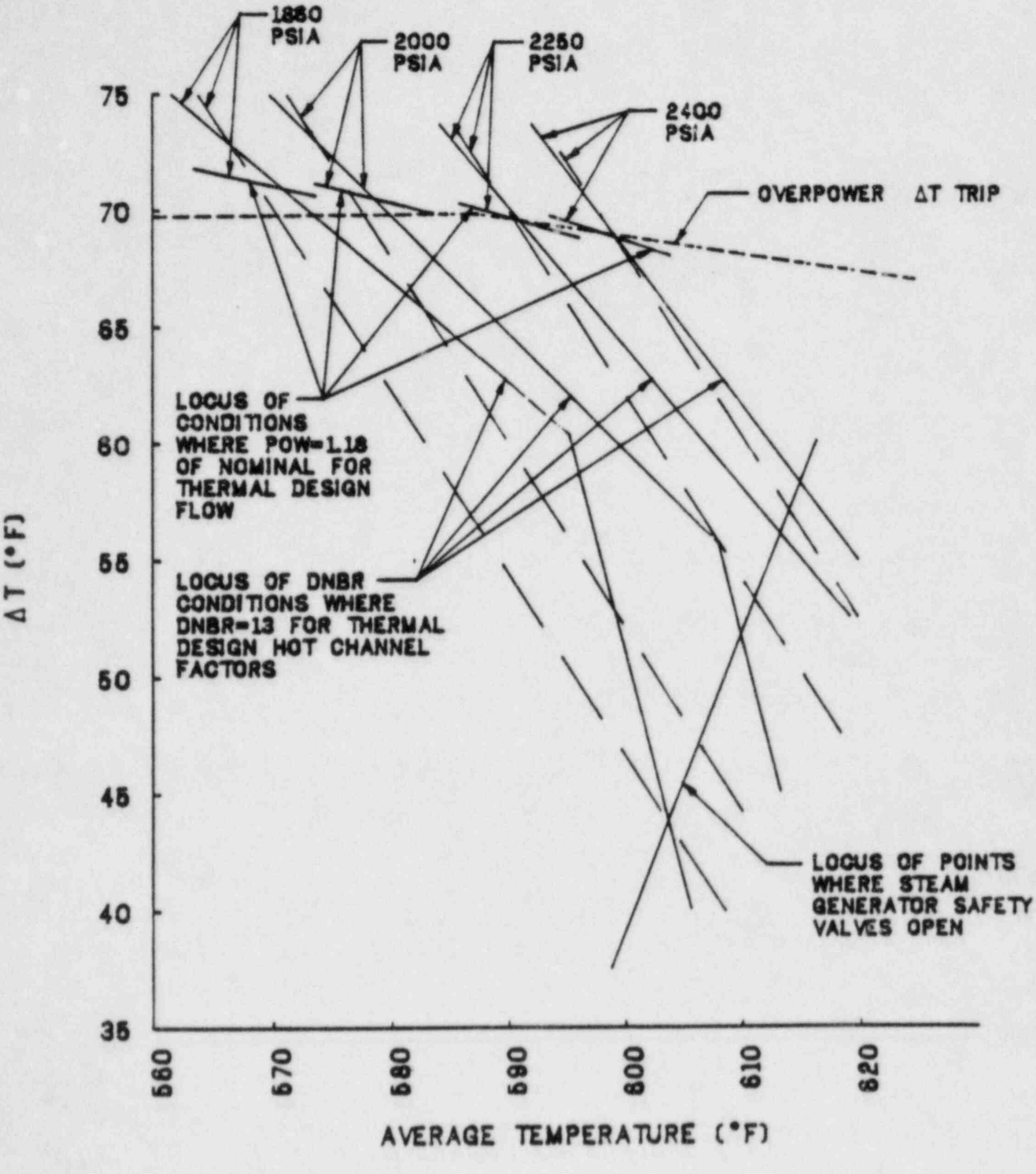


FIGURE 15.0-1a
OVER TEMPERATURE ΔT TRIP
(INCLUDING MAXIMUM INSTRUMENT ERRORS)
AT INDICATED PRESSURES
N-1 LOOP OPERATION

15.1 INCREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

A number of events have been postulated which could result in an increase in heat removal from the reactor coolant system (RCS) by the secondary system. Detailed analyses are presented for several such events which have been identified as limiting cases.

Discussions of the following RCS cooldown events are presented in this section.

1. Feedwater system malfunctions that result in a decrease in feedwater temperature.

2. Feedwater system malfunctions that result in an increase in feedwater flow.

3. Excessive increase in secondary steam flow.

4. Inadvertent opening of a steam generator relief or safety valve.

5. Steam system piping failure.

The above are considered to be American Nuclear Society (ANS) Condition II events, with the exception of steam system piping failures, which are considered to be ANS Condition III (minor) and Condition IV (major) events. Section 15.0.1 contains a discussion of ANS classifications and applicable acceptance criteria.

15.1.1 Feedwater System Malfunctions That Result in a Decrease in Feedwater Temperature

15.1.1.1 Identification of Causes and Accident Description

Reductions in feedwater temperature will cause an increase in core power by decreasing reactor coolant temperature. Such transients are attenuated by the thermal capacity of the secondary plant and of the RCS. The overpower/overtemperature protection (neutron overpower, overtemperature and overpowerΔT trips) prevents any power increase which could lead to a departure from nucleate boiling ratio (DNBR) less than 1.30.

A reduction in feedwater temperature may be caused by the accidental opening of a feedwater bypass valve which diverts flow around a portion of the feedwater heaters. In the event of an accidental opening of the bypass valve, there is a sudden reduction in feedwater inlet temperature to the steam generators. At power, this increased subcooling will create a greater load demand on the RCS.

With the plant at no-load conditions the addition of cold feedwater may cause a decrease in RCS temperature and thus a reactivity insertion due to the effects of the negative moderator coefficient of reactivity. However, the rate of energy change is reduced as load

and feedwater flow decrease, so the no-load transient is less severe 1.45
than the full power case.

The net effect on the RCS due to a reduction in feedwater temperature 1.46
is similar to the effect of increasing secondary steam flow, i.e., 1.47
the reactor will reach a new equilibrium condition at a power level
corresponding to the new steam generator ΔT. 1.48

A decrease in normal feedwater temperature is classified as an ANS 1.49
Condition II event, fault of moderate frequency. See Section 15.0.1 1.50
for a discussion of Condition II events.

The protection available to mitigate the consequences of a decrease 1.51
in feedwater temperature is the same as that for an excessive steam 1.52
flow increase, as discussed in Section 15.0.8 and listed in
Table 15.0-6.

15.1.1.2 Analysis of Effects and Consequences 1.54

<u>Method of Analysis</u> 1.56

This transient is analyzed by computing conditions at the feedwater 1.58
pump inlet following opening of the heater bypass valve. These 2.1
feedwater conditions are then used to recalculate a heat balance
through the high pressure heaters. This heat balance gives the new 2.2
feedwater conditions at the steam generator inlet.

The following assumptions are made. 2.4

1. Plant initial power level corresponding to guaranteed 2.6
nuclear steam supply system (NSSS) thermal output.

2. Low pressure heater bypass valve opens, resulting in 2.8
condensate flow splitting between the bypass line and the
low pressure heaters; the flow through each path is 2.9
proportional to the pressure drops.

3. Heater drain pumps trip; this increases the effect of the 2.10
cold bypass flow.

Plant characteristics and initial conditions are further discussed in 2.12
Section 15.0.3.

<u>Results</u> 2.15

Opening of a low pressure heater bypass valve and trip of the heater 2.17
drain pumps causes a reduction in feedwater temperature which 2.19
increases the thermal load on the primary system. The calculated 2.20
reduction in feedwater temperature is less than 35°F, resulting in an
increase in heat load on the primary system of less than 10 percent 2.21
of full power. The increased thermal load, due to opening of the low 2.22
pressure heater bypass valve, would result in a transient very 2.23
similar (but of reduced magnitude) to that presented in 2.24
Section 15.1.3 for an excessive increase in secondary steam flow, 2.25

which evaluates the consequences of a 10-percent step load increase.
Therefore, the transient results of this analysis are not presented. 2.26

15.1.1.3 Conclusions 2.28

The decrease in feedwater temperature transient is less severe than 2.29
the increase in feedwater flow event (Section 15.1.2) and the 2.31
increase in secondary steam flow event (Section 15.1.3). Based on 2.32
results presented in Sections 15.1.2 and 15.1.3, the applicable
acceptance criteria for the decrease in feedwater temperature event 2.33
have been met.

15.1.1.4 Radiological Consequences 2.35

There are no radiological consequences associated with a decrease in 2.36
feedwater temperature event, and activity is retained within the fuel 2.37
rods and reactor coolant system.

15.1.2 Feedwater System Malfunctions That Result in an Increase in 2.41
Feedwater Flow 2.42

15.1.2.1 Identification of Causes and Accident Description 2.45

Addition of excessive feedwater will cause an increase in core power 2.46
by decreasing reactor coolant temperature. Such transients are 2.49
attenuated by the thermal capacity of the secondary plant and of the
RCS. The overtemperature ΔT trip and the overpower ΔT trip prevent 2.50
any power increase which could lead to a DNBR less than 1.30. 2.51

An example of excessive feedwater flow would be a full opening of a 2.52
feedwater control valve due to a feedwater control system malfunction 2.53
or an operator error. At power, this excess flow causes a greater 2.54
load demand on the RCS due to increased subcooling in the steam
generator. With the plant at no-load conditions, the addition of an 2.55
excess of feedwater may cause a decrease in RCS temperature and thus 2.56
a reactivity insertion due to the effects of the negative moderator 2.57
coefficient of reactivity.

Continuous addition of excessive feedwater is prevented by the steam 2.58
generator high-high level trip, which closes the feedwater valves. 2.59

An increase in normal feedwater flow is classified as an ANS 2.60
Condition II event, fault of moderate frequency. See Section 15.0.1 3.1
for a discussion of ANS Condition II events.

Plant systems and equipment which are available to mitigate the 3.2
effects of the accident are discussed in Section 15.0.8 and listed in 3.3
Table 15.0-6.

15.1.2.2 Analysis of Effects and Consequences

Method of Analysis

The excessive heat removal due to a feedwater system malfunction transient is analyzed by using the detailed digital computer code LOFTRAN (WCAP-7907). This code simulates a multi-loop system, neutron kinetics, the pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, and steam generator safety valves. The code computes pertinent plant variables including temperatures, pressures, and power level.

The system is analyzed to demonstrate plant behavior in the event that excessive feedwater addition, due to a control system malfunction or operator error which allows a feedwater control valve to open fully, occurs. Two cases are analyzed as follows:

1. accidental opening of one feedwater control valve with the reactor just critical at zero load conditions assuming a conservatively large negative moderator temperature coefficient; and

2. accidental opening of one feedwater control valve with the reactor in automatic control at full power.

An accidental opening of one feedwater control valve with the reactor just critical at zero load conditions, assuming a conservatively large negative moderator temperature coefficient and one loop isolated, is bounded by the N-loop analysis at zero load conditions.

Case 2 is analyzed for operation with four loops in service and for operation with three loops in service.

The reactivity insertion rate following a feedwater system malfunction is calculated with the following assumptions.

1. For the feedwater control valve accident at full power, one feedwater control valve is assumed to malfunction, resulting in a step increase to 140 percent of nominal feedwater flow to one steam generator.

2. For the feedwater control valve accident at zero load condition, a feedwater control valve malfunction occurs which results in an increase in flow to one steam generator from zero to 200 percent of the nominal full load value for one steam generator.

3. For the zero load condition, feedwater temperature is at a conservatively low value of 32°F.

4. No credit is taken for the heat capacity of the RCS and steam generator thick metal in attenuating the resulting plant cooldown.

5. The feedwater flow resulting from a fully open control valve 3.38
is terminated by a steam generator high-high level trip 3.39
signal which closes all feedwater control and isolation 3.40
valves, trips the main feedwater pumps, and trips the
turbine.

Initial operating conditions are assumed at values consistent with 3.42
steady state N and N-1 loop operations.

Plant characteristics and initial conditions are further discussed in 3.43
Section 15.0.3.

Normal reactor control systems and engineered safety systems are not 3.44
required to function. The reactor protection system may function to 3.45
trip the reactor due to overpower or high-high steam generator water
level conditions. No single active failure will prevent operation of 3.47
the reactor protection system. A discussion of anticipated 3.48
transients without trip (ATWT) considerations is presented in
WCAP-8330 (1974).

<u>Results</u> 3.51

In the case of an accidental full opening of one feedwater control 3.53
valve with the reactor at zero power and the above mentioned 3.55
assumptions, the maximum reactivity insertion rate is less than the
maximum reactivity insertion rate analyzed in Section 15.4.1 for an 3.57
uncontrolled rod cluster control assembly bank withdrawal from a
subcritical or low power startup condition, and therefore the results 3.58
of the analysis are not presented here. It should be noted that if 3.59
the incident occurs with the unit just critical at no-load
conditions, the reactor may be tripped by the power range high 4.1
neutron flux trip (low setting) set at approximately 25 percent of
nominal full power.

The full power case (maximum reactivity feedback coefficients, 4.2
without rod control) gives the largest reactivity feedback and 4.3
results in the greatest power increase. Assuming the reactor to be 4.4
in the automatic rod control mode results in a slightly less severe
transient. The rod control system is not required to function for an 4.5
excessive feedwater flow event.

When the steam generator water level in the faulted loop reaches the 4.6
high-high level setpoint, all feedwater control and isolation valves 4.8
and the main feedwater pumps are tripped. This prevents continuous 4.9
addition of the feedwater. In addition, a reactor trip and a turbine 4.10
trip are initiated.

Transient results (Figures 15.1-1 and 15-1-2) show the core heat 4.11
flux, pressurizer pressure, T_{avg} and DNBR, as well as the increase in 4.12
nuclear power and loop ΔT associated with the increased thermal load
on the reactor. The DNBR does not drop below 1.30. Figures 15.1-1A 4.15
through 15.1-3A show the transient results with three loops in
operation.

Since the power level rises during the excessive feedwater flow 4.16
incident, the fuel temperatures will also rise until after reactor 4.17
trip occurs. The core heat flux lags behind the neutron flux 4.18
response due to the fuel rod thermal time constant, hence the peak
value does not exceed 118 percent of its nominal value (i.e., the 4.19
assumed high neutron flux trip setpoint). The peak fuel temperature 4.20
will thus remain well below the fuel melting temperature.

The transient results show that DNB does not occur at any time during 4.21
the excessive feedwater flow incident; thus, the ability of the 4.22
primary coolant to remove heat from the fuel rod is not reduced. The 4.23
fuel cladding temperature therefore does not rise significantly above
its initial value during the transient.

The calculated sequence of events for this accident is shown in 4.24
Table 15.1-1.

15.1.2.3 Conclusions 4.26

The results of the analysis show that the DNBRs encountered for an 4.27
excessive feedwater addition at power are at all times, above the 4.28
limiting value of 1.30; the DNBR design basis as described in 4.30
Section 4.4 is met. Additionally, it has been shown that the 4.31
reactivity insertion rate which occurs at no-load conditions
following excessive feedwater addition is less than the maximum value 4.32
considered in the analysis of the rod withdrawal from a subcritical 4.33
condition analysis. 4.34

15.1.2.4 Radiological Consequences 4.36

There are no radiological consequences associated with this event. 4.37
There is no adverse effect to the core and the RCS, and the activity 4.38
is retained within the fuel rods and reactor coolant system. 4.39

15.1.3 Excessive Increase in Secondary Steam Flow 4.42

15.1.3.1 Identification of Causes and Accident Description 4.43

An excessive increase in secondary system steam flow (excessive load 4.44
increase incident) is defined as a rapid increase in steam flow that 4.46
causes a power mismatch between the reactor core power and the steam
generator load demand. The reactor control system is designed to 4.47
accommodate a 10 percent step load increase or a 5 percent per minute
ramp load increase in the range of 15 to 100 percent of full power. 4.49
Any loading rate in excess of these values may cause a reactor trip 4.50
actuated by the reactor protection system. Steam flow increases 4.51
greater than 10 percent are analyzed in Sections 15.1.4 and 15.1.5.

This accident could result from either an administrative violation 4.52
such as excessive loading by the operator or an equipment malfunction 4.53
in the steam dump control or turbine speed control.

During power operation, steam dump to the condenser is controlled by 4.54
reactor coolant condition signals, i.e., high reactor coolant 4.55

temperature indicates a need for steam dump. A single controller malfunction does not cause steam dump; an interlock is provided which blocks the opening of the valves unless a large turbine load decrease or a turbine trip has occurred.

Protection against an excessive load increase accident is provided by the following reactor protection system signals:

1. overpower ΔT
2. overtemperature ΔT
3. power range high neutron flux; and
4. low pressurizer pressure

An excessive load increase incident is considered to be an ANS Condition II event, fault of moderate frequency. See Section 15.0.1 for a discussion of Condition II events.

15.1.3.2 Analysis of Effects and Consequences

<u>Method of Analysis</u>

This accident is analyzed using the LOFTRAN Code (WCAP-7907). The code simulates the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, steam generator safety valves, and feedwater system. The code computes pertinent plant variables including temperatures, pressures, and power level.

Four cases are analyzed to demonstrate the plant behavior following a 10 percent step load increase from rated load. These cases are as follows.

1. Reactor control in manual with minimum moderator reactivity feedback.
2. Reactor control in manual with maximum moderator reactivity feedback.
3. Reactor control in automatic with minimum moderator reactivity feedback.
4. Reactor control in automatic with maximum moderator reactivity feedback.

These four cases also are analayzed for a 10 percent step load increase from 77 percent power with three reactor coolant pumps in service.

For the minimum moderator feedback cases, the core has the least negative moderator temperature coefficient of reactivity and therefore the least inherent transient capability. For the maximum

moderator feedback cases, the moderator temperature coefficient of
reactivity has its highest absolute value. This results in the 5.31
largest amount of reactivity feedback due to changes in coolant
temperature. For the cases with automatic rod control, no credit 5.32
was taken for ΔT trips on overtemperature or overpower in order to ~~5.33~~
demonstrate the inherent transient capability of the plant. Under 5.34
actual operating conditions, such a trip may occur, after which the
plant would quickly stabilize.

A conservative limit on the turbine throttle valve opening is 5.35
assumed, and all cases are studied without credit being taken for 5.36
pressurizer heaters. Initial operating conditions are assumed at 5.37
extreme values consistent with the steady state full power operation
allowing for calibration and instrument errors. This results in 5.39
minimum margin to core DNB at the start of the accident.

Initial operating conditions are assumed at values consistent with 5.40
steady state N and N-1 loop operation.

Plant characteristics and initial conditions are further discussed in 5.41
Section 15.0.3.

Normal reactor control systems and engineered safety systems are not 5.42
required to function. The reactor protection system is assumed to be 5.43
operable; however, reactor trip is not encountered for any case due
to the error allowances assumed in the setpoints. No single active 5.45
failure will prevent the reactor protection system from performing
its intended function.

The cases which assume automatic rod control are analyzed to ensure 5.46
that the worst case is presented. The automatic function is not 5.47
required.

Results 5.50

Figures 15.1-3 through 15.1-6 illustrate the transient with the 5.52
reactor in the manual control mode. As expected, for the minimum 5.54
moderator feedback case there is a slight power increase, and the
average core temperature shows a large decrease. This results in a 5.56
DNBR which increases above its initial value. For the maximum 5.57
moderator feedback, manually controlled case there is a much larger
increase in reactor power due to the moderator feedback. A reduction 5.60
in DNBR is experienced but DNBR remains above 1.30.

Figures 15.1-7 through 15.1-10 illustrate the transient assuming the 6.1
reactor is in the automatic control mode. Both the minimum and 6.2
maximum moderator feedback cases show that core power increases,
thereby increasing the coolant average temperature and pressurizer 6.3
pressure above their initial value. For both of these cases, the 6.4
minimum DNBR remains above 1.30.

Figures 15.1-3A through 15.1-10A depict the previously described 6.5
cases, but considering three loops in operation.

For all cases, the plant rapidly reaches a stabilized condition at 6.6
the higher power level. Normal plant operating procedures would then 6.7
be followed to reduce power. Note that due to the measurement errors 6.8
assumed in the setpoints, it is possible that reactor trip could
actually occur for the automatic control cases. The plant would then 6.10
reach a stabilized condition following the trip.

The excessive load increase incident is an overpower transient for 6.11
which the fuel temperatures will rise. Reactor trip does not occur 6.12
for any of the cases analyzed, and the plant reaches a new
equilibrium condition at a higher power level corresponding to the 6.13
increase in steam flow.

Since DNB does not occur at any time during the excessive load 6.14
increase transients, the ability of the primary coolant to remove 6.15
heat from the fuel rod is not reduced. Thus, the fuel cladding 6.16
temperature does not rise significantly above its initial value
during the transient.

The calculated sequence of events for the excessive load increase 6.17
incident is shown in Table 15.1-1.

15.1.3.3 Conclusions 6.19

The analysis presented above shows that for a 10 percent step load 6.20
increase, the DNBR remains above 1.30; the design basis for DNBR as 6.21
described in Section 4.4 is met. The plant reaches a stabilized 6.22
condition rapidly following the load increase.

15.1.3.4 Radiological Consequences 6.25

There are no radiological consequences associated with this event and 6.26
activity is retained within the fuel rods and reactor coolant system 6.29
and radionuclide concentrations remain within the limits of the
Technical Specifications.

15.1.4 Inadvertent Opening of a Steam Generator Relief or Safety 6.33
Valve causing a depressurization of the Main Steam System 6.34

15.1.4.1 Identification of Causes and Accident Description 6.37

The most severe core conditions resulting from an accidental 6.38
depressurization of the main steam system are associated with an 6.40
inadvertent opening, with failure to close, of the largest of any
single steam dump, relief, or safety valve. The analyses performed 6.41
assuming a rupture of a main steam line are given in Section 15.1.5.

The steam release as a consequence of this accident results in an 6.42
initial increase in steam flow which decreases during the accident as 6.43
the steam pressure falls. The energy removal from the RCS causes a 6.44
reduction of coolant temperature and pressure. In the presence of a 6.45
negative moderator temperature coefficient, the cooldown results in
an insertion of positive reactivity. 6.46

The analysis is performed to demonstrate that the following criterion 6.47
is satisfied:

Assuming a stuck rod cluster control assembly, with offsite power 6.48
available, and assuming a single failure in the engineered safety 6.49
features system there will be no consequential damage to the core or 440.45
reactor coolant system after reactor trip for a steam release 6.50
equivalent to the spurious opening, with failure to close, of the
largest of any single steam dump, relief, or safety valve. 6.51

Accidental depressurization of the secondary system is classified as 6.52
an ANS Condition II event. See Section 15.0.1 for a discussion of 6.53
Condition II events.

The following systems provide the necessary protection against an 6.54
accidental depressurization of the main steam system due to the 6.55
opening of a steam generator relief or safety valve.

1. Safety injection system actuation from any of the following: 6.57

   a. two out of four pressurizer pressure signals; 6.59

   b. two out of three low steamline pressure signals in a 6.60
   loop.

2. The overpower reactor trips (neutron flux and ΔT) and the ~~7.2~~
   reactor trip occurring in conjunction with receipt of the 7.3
   safety injection signal

3. Redundant isolation of the main feedwater lines 7.4

   Sustained high feedwater flow would cause additional 7.6
   cooldown. Therefore, in addition to the normal control 7.7
   action which will close the main feedwater valves following
   reactor trip, a safety injection signal will rapidly close 7.8
   all feedwater control valves and back up feedwater isolation
   valves, and trip the main feedwater pumps. 7.9

4. Trip of the fast-acting steam line stop valves (designed to 7.11
   close in less than 5 seconds) on:

   a. safety injection system actuation derived from two out 7.13
   of three low steam line pressure signal in any loop
   (above Permissive P-11); 7.14

   b. high negative steam pressure rate indication from two 7.15
   out of three signals in any loop (below Permissive
   P-11).

Plant systems and equipment which are available to mitigate the 7.17
effects of the accident are also discussed in Section 15.0.8 and 7.18
listed in Table 15.0-6.

15.1.4.2 Analysis of Effects and Consequences 7.20

Method of Analysis 7.22

The following analyses of a secondary system steam release are 7.24
performed for this section.

1. A full plant digital computer simulation using the LOFTRAN 7.26
   Code (WCAP-7907) to determine RCS temperature and pressure 7.27
   during cooldown, and the effect of safety injection.

2. Analyses to determine that there is no damage to the core or 7.29 440.45
   the reactor coolant system.

The following conditions are assumed to exist at the time of a 7.31
secondary steam system release.

1. End-of-life shutdown margin at no-load, equilibrium xenon 7.33
   conditions, and with the most reactive rod cluster control 7.34
   assembly stuck in its fully withdrawn position. Operation 7.36
   of rod cluster control assembly banks during core burnup is
   restricted in such a way that addition of positive 7.37
   reactivity in a secondary system steam release accident will
   not lead to a more adverse condition than the case analyzed. 7.38

2. A negative moderator coefficient corresponding to the end- 7.40
   of-life rodded core with the most reactive rod cluster
   control assembly in the fully withdrawn position. The 7.42
   variation of the coefficient with temperature and pressure
   is included. The K_{eff} versus temperature at 1,000 psi ~~7.43~~
   corresponding to the negative moderator temperature
   coefficient used is shown on Figure 15.1-11. 7.44

3. Minimum capability for injection of concentrated boric acid 7.45
   solution corresponding to the most restrictive single
   failure in the safety injection system. This corresponds to 7.47
   the flow delivered by one centrifugal charging/high head
   safety injection pump delivering its full contents to the 7.49
   cold leg header. Unborated water must be swept from the 7.51
   safety injection lines downsteam of the refueling water
   storage tank (RWST) prior to the delivery of concentrated 7.52
   boric acid (1,950 parts per million, ppm) to the reactor
   coolant loops. This effect has been accounted for in the 7.53
   analysis.

4. The case studied is a steam flow of 268 pounds per second at 7.54
   1,200 pounds per square inch absolute (psia) with offsite 7.55
   power available. This is the maximum capacity of any single 7.56
   steam dump, relief, or safety valve. Initial hot shutdown 7.57
   conditions at time zero are assumed since this represents
   the most conservative initial condition.

   Should the reactor be just critical or operating at power at 7.59
   the time of a steam release, the reactor will be tripped by 7.60

the normal overpower protection when power level reaches a
trip point. Following a trip at power, the RCS contains 8.1
more stored energy than at no-load, the average coolant
temperature is higher than at no-load and there is 8.2
appreciable energy stored in the fuel. Thus, the additional 8.3
stored energy is removed via the cooldown caused by the
steam release before the no-load conditions of RCS 8.4
temperature and shutdown margin assumed in the analyses are
reached. After the additional stored energy has been 8.5
removed, the cooldown and reactivity insertions proceed in
the same manner as in the analysis which assumes no-load 8.6
condition at time zero. However, since the initial steam 8.7
generator water inventory is greatest at no-load, the
magnitude and duration of the RCS cooldown are less for 8.8
steam line release occurring at power.

5. In computing the steam flow, the Moody Curve (Moody 1965) 8.10
for fL/D = 0 is used.

6. Perfect moisture separation in the steam generator is 8.11
assumed.

7. Cases are shown for four loops in operation and three loops 8.12
in operation.

Results 8.14

The calculated time sequence of events for this accident is listed in 8.16
Table 15.1-1.

The results presented are a conservative indication of the events 8.18
which would occur assuming a secondary system steam release since it 8.19
is postulated that all of the conditions described above occur
simultaneously.

Figures 15.1-12 and 15.1-13 show the transient results for a steam 8.20
flow of 168 lb/sec at 1,200 psia from one steam generator. 8.21

The assumed steam release is typical of the capacity of any single 8.22
steam dump, relief, or safety valve. Safety injection is initiated 8.23
automatically by low pressurizer pressure. Operation of one 8.24
centrifugal charging/high head safety injection pump is assumed.
Boron solution at 1,950 ppm enters the RCS providing sufficient 8.25
negative reactivity to prevent core damage. The calculated transient 8.26
is quite conservative with respect to cooldown, since no credit is
taken for the energy stored in the system metal other than that of 8.28
the fuel elements or the energy stored in the other steam generator
tubes. Since the transient occurs over a period of about 5 minutes, 8.29
the neglected stored energy is likely to have a significant effect in 8.30
slowing the cooldown.

Figures 15.1-12A and 15.1-13A show the same parameters as 8.31
Figures 15.1-12 and 15.1-13 only for the case with one loop out of 8.32
service. The steam leak is assumed to occur on one of the loops 8.33

which is in service. Safety injection is initiated automatically 8.34
from a low pressurizer pressure SI Signal.

15.1.4.3 Conclusions 8.36

The analysis shows that the criteria stated earlier in this section 8.37
are satisfied. For an accidental depressurization of the main steam 8.39
system the minimum DNBR remains well above the limiting value and no 8.40
system design limits are exceeded. The radiological consequences of 8.41
this event are not limiting.

15.1.4.4 Radiological Consequences 8.43

The inadvertent opening of a single steam dump relief or safety valve 8.44
can result in steam release from the secondary system. Normally, no 8.47
activity release to plant personnel or the public is expected.
However, if steam generator leakage exists coincident with the 8.48
failed fuel conditions, some activity will be released.

The dose, being a function of steam release, will be less than that 8.49
calculated for the steam line break accident (Section 15.1.5). 8.50

15.1.5 Steam System Piping Failure 8.52

15.1.5.1 Identification of Causes and Accident Description 8.53

The steam release arising from a rupture of a main steam line would 8.54
result in an initial increase in steam flow which decreases during 8.56
the accident as the steam pressure falls. The energy removal from 8.57
the RCS causes a reduction of coolant temperature and pressure. In 8.58
the presence of a negative moderator temperature coefficient, the
cooldown results in an insertion of positive reactivity. If the most 8.60
reactive rod cluster control assembly (RCCA) is assumed stuck in its 9.1
fully withdrawn position after reactor trip, there is an increased
possibility that the core will become critical and return to power. 9.2
A return to power following a steam line rupture is a potential 9.3
problem mainly because of the high power peaking factors which exist 9.4
assuming the most reactive RCCA to be stuck in its fully withdrawn
position. The core is ultimately shut down by the boric acid 9.5
injection delivered by the safety injection system.

The analysis of a main steam line rupture is performed to demonstrate 9.6
that the following criteria are satisfied.

the most limiting

1. Assuming the most limiting single failure of a stuck RCCA 9.8
   with or without offsite power, and assuming a single failure 9.9
   in the engineered safety features, the core remains in place 9.11
   and intact. Radiation doses do not exceed the guidelines of 9.12
   10CFR100.

2. Although DNB and possible clad perforation following a steam 9.13
   pipe rupture are not necessarily unacceptable, the following 9.14
   analysis, in fact, shows that the DNB design basis is met
   for any rupture assuming the most reactive assembly stuck in 9.15

its fully withdrawn position. The DNB design basis is 9.17
discussed in Section 4.4.

3. A major steam line rupture is classified as an ANS 9.18
Condition IV event. See Section 15.0.1 for a discussion of 9.19
Condition IV events.

4. Effects of minor secondary system pipe breaks are bounded by 9.20
the analysis presented in this section. Minor secondary 9.21
system pipe breaks are classified as Condition III events,
as described in Section 15.0.1.3.

5. The major rupture of a steam line is the most limiting 9.22
cooldown transient and is analyzed at zero power with no
decay heat. Decay heat would retard the cooldown, thereby 9.25
reducing the return to power. A detailed analysis of this 9.26
transient with the most limiting break size, a double ended
rupture, is presented here.

The following functions provide the protection for a steam line 9.28
rupture.

1. Safety injection system actuation from any of the following: 9.31

   a. two out of four low pressurizer pressure signals; 9.33

   b. two out of three hi-1 containment pressure signals; 9.34

   c. one low steam line pressure signal in any two loops. 9.36

2. The overpower reactor trips (neutron flux and ΔT) and the ~~9.38~~
reactor trip occurring in conjunction with receipt of the 9.39
safety injection signal

3. Isolation of the main feedwater lines, via closure of the 9.41
redundant main feedwater isolation valves 9.42

   Sustained high feedwater flow would cause additional 9.44
   cooldown. Therefore, in addition to the normal control 9.46
   action which will close the main feedwater valves, a safety
   injection signal will rapidly close all feedwater control 9.47
   valves and back up feedwater isolation valves, and trip the
   main feedwater pumps. 9.49

4. Trip of the fast acting steam line stop valves (designed to 9.51
close in less than 5 seconds) on:

   a. hi-2 containment pressure; 9.53

   b. safety injection system actuation derived from two out 9.54
   of three low steam line pressure signal in any loop;

c. high negative steam pressure rate indication from two out of three signals in any loop (below Permissive P-11).

Fast-acting isolation valves are provided in each steam line; these valves will fully close within 10 seconds of a large break in the steam line. For breaks downstream of the isolation valves, closure of all valves would completely terminate the blowdown. For any break, in any location, no more than one steam generator would experience an uncontrolled blowdown even if one of the isolation valves fails to close. A description of steam line isolation is included in Chapter 10.

Steam flow is measured by monitoring dynamic head in nozzles located in the throat of the steam generator. The effective throat area of the nozzles is 1.4 square feet, which is considerably less than the main steam pipe area; thus, the nozzles also serve to limit the maximum steam flow for break at any location.

Table 15.1-2 lists the equipment required in the recovery from a high energy line rupture. Not all equipment is required for any one particular break, since the requirements will vary depending upon postulated break locations and details of balance of plant design and pipe rupture criteria as discussed elsewhere in this application. Design criteria and methods of protection of safety related equipment form the dynamic effects of postulated piping ruptures are provided in Section 3.6.

15.1.5.2 Analysis of Effects and Consequences

<u>Method of Analysis</u>

The analysis of the steam pipe rupture has been performed to determine:

1. The core heat flux and RCS temperature and pressure resulting from the cooldown following the steam line break. The LOFTRAN Code (WCAP-7907) has been used.

2. The thermal and hydraulic behavior of the core following a steam line break. A detailed thermal and hydraulic digital-computer code, THINC, has been used to determine if DNB occurs for the core conditions computed in WCAP-7907.

The analsysis has been performed with four reactor coolant loops in operation and with three loops in operation (N-1 loop).

The following conditions were assumed to exist at the time of a main steam line break accident.

1. End-of-life shutdown margin at no-load, equilibrium xenon conditions, and the most reactive RCCA stuck in its fully withdrawn position. Operation of the control rod banks during core burnup is restricted in such a way that addition

of positive reactivity in a steam line break accident will 10.33
not lead to a more adverse condition than the case analyzed.

2. A negative moderator coefficient corresponding to the end- 10.34
of-life rodded core with the most reactive RCCA in the fully 10.35
withdrawn position. The variation of the coefficient with 10.36
temperature and pressure has been included. The K_{eff} 10.37
versus temperature at 1,000 psi corresponding to the
negative moderator temperature coefficient used is shown on 10.39
Figure 15.1-11. The effect of power generation in the core 10.40
on overall reactivity is shown on Figure 15.1-14.

The core properties associated with the sector nearest the 10.42
affected steam generator and those associated with the
remaining sector were conservatively combined to obtain 10.43
average core properties for reactivity feedback
calculations. Further, it was conservatively assumed that 10.44
the core power distribution was uniform. These two 10.45
conditions cause underprediction of the reactivity feedback
in the high power region near the stuck rod. To verify the 10.46
conservatism of this method, the reactivity as well as the
power distribution was checked for the limiting state points 10.47
for the cases analyzed. This core analysis considered to 10.48
Doppler reactivity from the high fuel temperature near the
stuck RCCA, moderator feedback from the high water enthalpy 10.49
near the stuck RCCA, power redistribution and nonuniform
core inlet temperature effects. For cases in which steam 10.51
generation occurs in the high flux regions of the core, the
effect of void formation was also included. It was 10.53
determined that the reactivity employed in the kinetics
analysis was always larger than the reactivity calculated 10.54
including the above local effects for the state points.
These results verify conservatism; i.e., underprediction of 10.55
negative reactivity feedback from power generation.

3. Minimum capability for injection of high concentration boric 10.57
acid (1,950 ppm) solution corresponding to the most
restrictive single failure in the safety injection system. 10.58
The emergency core cooling system consists of three systems: 10.59
(1) the passive accumulators, (2) the residual heat removal 10.60
system, and (3) the safety injection system. Only the 11.1
safety injection system is modeled for the steam line break
accident analysis.

The actual modeling of the safety injection system in 11.3
LOFTRAN is described in WCAP-7907. The flow corresponds to 11.4
that delivered by one charging/high-head safety injection
pump delivering its full flow to the cold leg header. No 11.6
credit has been taken for the low concentration borated
water, which must be swept from the lines downstream of the
RWST prior to the delivery of high concentration boric acid 11.7
to the reactor coolant loops.

For the cases where offsite power is assumed, the sequence 11.8
of events in the safety injection system is the following.
After the generation of the safety injection signal 11.9
(appropriate delays for instrumentation, logic, and signal
transport included), the appropriate valves begin to operate 11.10
and the charging/high-head safety injection pump starts. In 11.11
12 seconds, the valves are assumed to be in their final
position and the pump is assumed to be at full speed. The 11.12
volume containing the low concentration borated water is
swept before the 1,950 ppm borated water reaches the core. 11.13
This delay, described above, is inherently included in the 11.14
modeling.

In cases where offsite power is not available, an additional 11.15
10 second delay is assumed to start the diesels and to load 11.16
the necessary safety injection equipment onto them.

4. Design value of the steam generator heat transfer 11.18
coefficient including allowance for fouling factor

5. Since the steam generators are provided with integral flow 11.19
restrictors at the main steam nozzles with a 1.4 square foot 11.20
throat area, any rupture in a steam line with a break area 11.21
greater than 1.4 square feet, regardless of location, would
have the same effect on the NSSS as the 1.4 square foot 11.23
break. The following cases have been considered in 11.24
determining the core power and RCS transients.

    a. Complete severance of a pipe, with the plant initially 11.26
    at no-load conditions, full reactor coolant flow with
    offsite power available. 11.27

    b. Case (a) with loss of offsite power simultaneous with 11.28
    the steam line break and initiation of the safety
    injection signal. Loss of offsite power results in 11.30
    reactor coolant pump coastdown.

    c. Cases (a) and (b) with one reactor coolant loop out of 11.31
    service.

6. Power peaking factors corresponding to one stuck RCCA and 11.33
nonuniform core inlet coolant temperatures are determined at 11.34
end of core life.

The coldest core inlet temperatures are assumed to occur in 11.36
the sector with the stuck rod. The power peaking factors 11.37
account for the effect of the local void in the region of
the stuck control assembly during the return to power phase 11.38
following the steam line break. This void, in conjunction 11.39
with the large negative moderator coefficient, partially
offsets the effect of the stuck assembly. The power peaking 11.41
factors depend upon the core power, temperature, pressure,
and flow, and, thus, are different for each case studied. 11.42

The core parameters used for each of the two cases 11.43
correspond to values determined from the respective
transient analysis. 11.44

The analysis assumes initial hot shutdown conditions at time 11.45
zero since this represents the most pessimistic initial
condition. Should the reactor be just critical or operating 11.47
at power at the time of a steam line break, the reactor will
be tripped by the normal overpower protection system when 11.48
power level reaches a trip point. Following a trip at 11.49
power, the RCS contains more stored energy than at no-load,
the average coolant temperature is higher than at no-load 11.50
and there is appreciable energy stored in the fuel. Thus, 11.51
the additional stored energy is removed via the cooldown
caused by the steam line break before the no-load conditions 11.52
of RCS temperature and shutdown margin assumed in the
analysis are reached. After the additional stored energy 11.53
has been removed, the cooldown and reactivity insertions
proceed in the same manner as cooldown and reactivity 11.54
insertions proceed in the same manner as in the analysis
which assumes no-load condition at time zero. 11.55

7. In computing the steam flow during a steam line break, the 11.57
Moody Curve (Moody 1965) for fL/D = 0 is used.

Results 11.59

The calculated sequence of events for all four cases analyzed is 12.1
shown in Table 15.1-1. The results presented are a conservative 12.3
indication of the events which would occur assuming a steam line
rupture since it is postulated that all of the conditions described 12.4
above occur simultaneously.

Core Power and Reactor Coolant System Transient 12.7

Figures 15.1-15 through 15.1-17 show the RCS transient and core heat 12.9
flux following a main steam line rupture (complete severance of a 12.11
pipe) at initial no-load condition (case a).

Offsite power is assumed available so that full reactor coolant flow 12.12
exists. The transient shown assumes an uncontrolled steam release 12.13
from only one steam generator. Should the core be critical at or 12.14
near zero power when the rupture occurs, the initiation of safety
injection by low steam line pressure will trip the reactor. Steam 12.17
release from more than one steam generator will be prevented by
automatic trip of the fast acting isolation valves in the steam lines 12.18
by high containment pressure signals or by low steam line pressure
signals. Even with the failure of one valve, release is limited to 12.19
no more than 10 seconds for the other steam generators while the one 12.20
generator blows down. The steam line stop valves are designed to be 12.21
fully closed in less than 5 seconds from receipt of a closure signal.

As shown on Figure 15.1-15, the core attains criticality with the 12.22
RCCA's inserted (with the design shutdown assuming one stuck RCCA) 12.23

before boron solution at 1,950 ppm enters the RCS. A peak core power 12.24
less than the nominal full power value is attained.

The calculation assumes the boric acid is mixed with, and diluted by, 12.25
the water flowing in the RCS prior to entering the reactor core. The 12.27
concentration after mixing depends upon the relative flow rates in
the RCS and in the safety injection system. The variation of mass 12.28
flow rate in the RCS due to water density changes is included in the
calculation as is the variation of flow rate in the safety injection 12.29
system due to changes in the RCS pressure. The safety injection 12.30
system flow calculation includes the line losses in the system as
well as the pump head curve.

Figures 15.1-15A through 15.1-17A show relevant parameters for Case 12.31
(a) assuming one reactor coolant loop is out of service when the 12.33
steam line rupture occurs. Offsite power is available throughout the 12.34
transient. As in Case (a) with all loops in operation, steam is 12.35
released from only one steam generator due to tripping of the fast
acting steam line isolation valves by high containment pressure 12.36
signals or low steam line pressure. The core attains criticality 12.37
with the RCCA inserted (with the design shutdown assuming one stuck
assembly) before boron solution at 1,950 ppm boron enters the RCS 12.38
from the SIS. The actuation and transport delays are taken into 12.39
account assuming all loops in operation.

Based on the results of a generic study (WCAP-9227), the case with 12.41
loss of offsite power (case b) is less severe than that described
above. The case without offsite power affects the transient in 12.42
basically two ways. First, the delay to start safety injection is 12.43
increased by the time required to start the diesel, i.e., 10 seconds.
Second, the loss of the main coolant pump reduces the flow in the 12.44
reactor coolant system. The additional 10 seconds does not 12.45
significantly change the conclusions. The loss of forced 12.46
circulation, however, makes the transient less severe than the case
with forced circulation. This is due to: 1) the core and steam 12.47
generator are more decoupled, hence the time to go critical is much
longer, 2) without forced coolant, the rate of heat transfer is 12.48
reduced in the steam generator, thereby reducing the core cooldown 12.49
rate, and 3) local density feedback effects minimize the power level
following the assumed steam break. All of these effects result in 12.50
the case with forced coolant flow being much more severe than the
case without offsite power for the potential of clad damage. 12.51

It should be noted that following a steam line break, only one steam 12.52
generator blows down completely. Thus, the remaining steam 12.53
generators in operation are still available for dissipation of decay 12.54
heat after the initial transient is over. In the case of loss of 12.56
offsite power, this heat is removed to the atmosphere via the steam
line safety valves, and power-operated relief valves. 12.57

Margin to Critical Heat Flux 12.60

A DNB analysis was performed for the limiting case with offsite power 13.2
available. It was found that the DNB design basis, as stated in 13.4
Section 4.4, was met for both cases.

15.1.5.3 Conclusions 13.6

The analysis has shown that the criteria stated earlier in 13.7
Section 15.1.5.1 are satisfied.

Although DNB and possible clad perforation following a steam pipe 13.9
rupture are not necessarily unacceptable and not precluded by the 13.10
criteria, the above analysis, in fact, shows that no DNB occurs for
any rupture assuming the most reactive RCCA stuck in its fully 13.11
withdrawn position. The radiological consequences of this limiting 13.12
event are within the acceptable criteria of 10CFR Part 100.

15.1.5.4 Radiological Consequences 13.14

The main steam line break is postulated to occur in a main steam line 13.15
outside the containment. The main steam, fast acting, isolation 13.17
valves shut automatically, forcing the steam to be released through 13.18
the main steam relief line safety valves. The plant is assumed to 13.19
have operated with Technical Specification fuel defects and primary
to secondary leakage. A 1 gpm technical specification primary to 13.20
secondary leakage is assumed to occur such that 0.347 gpm leakage 13.21
takes place in the affect~~ive~~ steam generator and 0.653 gpm leakage Y affected
takes place evenly in the other steam generators. 13.22

Associated with this accident is the assumption that 1 percent of the 13.23
fuel rods in the core fail at the onset of the accident. The gap 13.25
activity associated with the damaged fuel rods is uniformly mixed
with the primary coolant and is therefore available for release to 13.26
the atmosphere via primary to secondary leakage.

The iodine partition factor is assumed to be 1.0 for the initial 13.27
steam release from the defective steam generator. An iodine 13.28
partition factor of 0.01 is used for the long term release from the
defective steam generator and for initial and long term releases from 13.29
the non-defective steam generators. Offsite power is also assumed to 13.30
be lost, thus making the condensers unavailable for steam dump. The 13.31
steam released from all steam generators is assumed to be released
directly to the environment at ground level.

Two cases are analyzed and reported for both N-loop and N-1 loop 13.32
plant operating conditions. The first case assumes fuel failure to 13.33
occur without a concurrent iodine spike condition. The fuel failure 13.34
is presumed to result in the release of 1 percent of the fuel gap
fission product activity to the reactor coolant. The second case 13.36
includes a preaccident iodine spike condition when the steam line
break occurs.

The radiological consequences of a main steam line break (all cases) 13.37
are reported in Table 15.0-8. The assumptions used to perform this 13.38
evaluation are summarized in Table 15.1-3. Additionally, for the 13.39
N-1 loop operation analyses, it was assumed that the plant had been
operating at full power with all four loops sufficiently long for core 13.40 loops
activities and coolant concentrations to reach equilibrium. The 13.41
plant then began N-1 loop operation, shortly after which the main
steam line break occurred. The use of these assumptions results in 13.42
the calculated releases to the environment listed in Table 15.1-4. 13.43
These releases, together with the atmospheric dispersion values 13.44
listed in Table 15.0-11, are used to compute the doses to the EAB 13.46
(0-2 hr) and LPZ (0-8 hr) as reported in Table 15.0-8. x as

The resulting doses to the EAB and LPZ for the Case 1 conditions do 13.47
not exceed the guideline values of 10CFR100. The resulting doses to 13.48
the EAB and LPZ assuming Case 2 conditions do not exceed a small
fraction of 10CFR100.

15.1.6 References for Section 15.1 13.50

American National Standard Source Term Specification, N237. 1976. 13.52

Moody, F.S. 1965. Transactions of the ASME, Journal of Heat 13.55
Transfer. Figure 3, page 134. 13.56

WCAP-7907, 1972. Burnett, T.W.T. et al. LOFTRAN Code Description. 13.57

WCAP-8330, 1974. Westinghouse Anticipated Transients Without Trip 13.59
Analysis.

WCAP-9227, 1978. Reactor Core Response to Excessive Secondary Steam 14.1
Release.

TABLE 15.1-1

TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH CAUSE AN INCREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

Accident	Event	N-Loop Time (sec)	N-1 Loop Time (sec)
Feedwater system malfunctions that result in an increase in feedwater flow:	One main feedwater control valve fails fully open	0.0	0.0
	Minimum DNBR occurs	66.0	23.0
	High-high steam generator water level signal generated	80.3	20.5
	Turbine trip occurs due to high-high steam generator level	82.8	22.5
	Reactor trip occurs	84.3	22.6
	Feedwater isolation valves close automatically	87.3	27.5
Excessive increase in secondary steam flow:			
1. Manual reactor control (minimum moderator feedback)	10 percent step load increase	0.0	
	Equilibrium conditions reached (approximate time only)	150	
2. Manual reactor control (maximum moderator feedback)	10 percent step load increase	0.0	

TABLE 15.1-1 (Cont)

Accident	Event	N-Loop Time (sec)	N-1 Loop Time (sec)	
	Equilibrium conditions reached (approximate time only)	100		2.11 2.12 2.13
3. Automatic reactor control (minimum moderator feedback)	10 percent step load increase	0.0		2.16 2.17 2.18 2.19 2.20
	Equilibrium conditions reached (approximate time only)	200		2.22 2.23 2.24
4. Automatic reactor control (maximum moderator feedback)	10 percent step load increase	0.0		2.26 2.27 2.28 2.29 2.30
	Equilibrium conditions reached (approximate time only)	100		2.32 2.33 2.34
Inadvertent opening of a steam generator relief or safety valve	Inadvertent opening of one main steam safety or relief valve	0.0	0.0	2.36 2.37 2.38 2.39
	Pressurizer empties	156	154	2.41
	1,950 ppm boron reaches core	107.50	226.0	2.43 2.44
Steam system piping failure:				2.46 2.47
1. Case a	Steam line ruptures	0	0	2.49
	Pressurizer empty	13.4	11.4	2.51
	Criticality attained	23.6	22.2	2.53
	1,950 ppm boron reaches core	44.20	48.2	2.55 2.56
2. Case b	Steam line ruptures	0	0	2.58

TABLE 15.1-1 (Cont)

Accident	Event	N-Loop Time (sec)	N-1 Loop Time (sec)	
	Pressurizer empty	15.4	12.6	2.60
	Criticality attained	27.0	27.2	3.2
	1,950 ppm boron reaches core	54.20	58.2	3.4 3.5

TABLE 15.1-3

ASSUMPTIONS USED IN MAIN STEAM LINE BREAK ANALYSIS

	Expected	Analysis Input Parameters N-Loop	Analysis Input Parameters N-1 Loop
1. Reactor thermal power level (MWt)	3,411	3,636[1]	3,636[1]
2. Fuel defects (percent)	0.12	0.29[2]	0.29[2]
3. Primary coolant concentrations	Table 11.1-2	Table 15.0-10	Table 15.0-10
4. Primary to secondary leak rate (gpm)	0.009	1.0	1.0
5. Secondary coolant concentrations	Tables 11.1-6 & 11.1-7	Table 15.0-10	Table 15.0-10
6. Activity released to primary coolant from failed fuel and available for release (percent of gap inventory)			
a. Noble gases	0.0	1.0	1.0
b. Iodines	0.0	1.0	1.0
7. Core and gap activities	Table 15.0-7	Table 15.0-7	Table 15.0-7
8. Iodine partition factors in defective steam generator			
a. 0-30 minutes	1.0	1.0	1.0
b. 30 minutes-8 hours	0.01	0.01	0.01
9. Iodine partition factors in nondefective steam generators throughout accident	0.01	0.01	0.01
10. Time to isolate defective steam generator (hours)	8	8	8

TABLE 15.1-3 (Cont)

	Expected	Analysis Input Parameters N-Loop	Analysis Input Parameters N-1 Loop
11. Initial steam and water release from defective steam generator over the first 30 minutes of the accident (lb)	167,000	167,000	167,000
12. Long term (0-8 hour) steam release from the defective steam generator (lb)	12	1,300	1,300
13. Steam release from three nondefective steam generators (lb)			
a. 0-2 hours	417,000	417,000	433,000
b. 2-8 hours	912,000	912,000	912,000
14. Offsite power	Available	Lost	Lost
15. Steam generator contents (lb/sg)			
a. Steam	8,000	8,000	7,600
b. Liquid	103,000	103,000	104,000
16. Feedwater flow to nondefective steam generator (lb)			
a. 0-2 hours	589,000	589,000	589,000
b. 2-8 hours	1,019,000	1,019,000	1,019,000
17. Iodine spike factor	No spike	Pre-accident iodine concentrations shown in Table 15.0-12	Pre-accident iodine concentrations shown in Table 15.0-12

NOTES:

1. Fuel gap activities are based on 3,636 MWt reactor power.
2. Based on 1.0 μCi/gm I-131 dose equivalent.

TABLE 15.1-4

MAIN STEAM LINE BREAK RELEASES
(To the Environment)

N-Loop

Isotope	Case 1 - 1 Percent Fuel Cladding Failure (Curies Released) (0-2 hr) EAB	Case 1 - 1 Percent Fuel Cladding Failure (Curies Released) (0-8 hr) LPZ	Case 2 - Preaccident Iodine Spike (Curies Released) (0-2 hr) EAB	Case 2 - Preaccident Iodine Spike (Curies Released) (0-8 hr) LPZ
Kr-83m	2.18E+01[1]	3.94E+01	4.32E-02	7.70E-02
85m	6.61E+01	1.75E+02	2.00E-01	5.20E-01
85	1.72E+00	6.86E+00	4.65E-03	1.82E-02
87	9.08E+01	1.34E+02	1.03E-01	1.50E-01
88	1.67E+02	3.68E+02	3.63E-01	7.86E-01
89	1.04E+01	1.04E+01	8.69E-04	8.69E-04
Xe-131m	1.57E-01	6.22E-01	1.52E-03	6.07E-03
133m	9.45E+00	3.63E+01	8.19E-02	3.13E-01
133	3.96E+02	1.56E+03	3.54E+00	1.37E+01
135m	2.49E+01	2.88E+01	2.84E-01	5.03E-01
135	1.01E+02	3.29E+02	6.36E-01	2.06E+00
137	1.63E+01	1.63E+01	1.53E-03	1.53E-03
138	5.95E+01	5.97E+01	1.53E-02	1.53E-02
I-131	2.04E+01	3.07E+01	5.73E+00	7.21E+00
132	2.27E+01	2.57E+01	1.87E+00	1.99E+00
133	4.21E+01	6.15E+01	8.43E+00	1.04E+01
134	3.44E+01	3.52E+01	5.66E-01	5.75E-01
135	3.54E+01	4.76E+01	3.96E+00	4.66E+00

NOTE:

1. 2.18E+01 = 2.18×10^{1}.

TABLE 15.1-4 (Cont)

N-1 Loop

Isotope	Case 1 - 1 Percent Fuel Cladding Failure (Curies Released) (0-2 hr) EAB	Case 1 - 1 Percent Fuel Cladding Failure (Curies Released) (0-8 hr) LPZ	Case 2 - Preaccident Iodine Spike (Curies Released) (0-2 hr) EAB	Case 2 - Preaccident Iodine Spike (Curies Released) (0-8 hr) LPZ
Kr-83m	3.20E+01[1]	5.79E+01	4.30E-02	7.70E-02
85m	9.72E+01	2.56E+02	1.98E-01	5.19E-01
85	2.50E+00	9.95E+00	4.61E-03	1.82E-02
87	1.32E+02	1.96E+02	1.01E-01	1.47E-01
88	2.45E+02	5.41E+02	3.65E-01	7.95E-01
89	1.52E+01	1.52E+01	7.69E-04	7.69E-04
Xe-131 m	2.29E-01	9.08E-01	1.51E-03	5.97E-03
133m	1.37E+01	5.26E+01	8.01E-02	3.05E-01
133	5.63E+02	2.22E+03	3.45E+00	1.34E+01
135m	3.65E+01	3.97E+01	2.89E-01	4.25E-01
135	1.46E+02	4.75E+02	5.77E-01	1.93E+00
137	2.34E+01	2.34E+01	4.73E-03	4.73E-03
138	8.67E+01	8.71E+01	1.48E-02	1.49E-02
I-131	2.85E+01	3.64E+01	5.83E+00	6.74E+00
132	3.33E+01	3.61E+01	1.91E+00	1.99E+00
133	5.92E+01	7.39E+01	8.70E+00	9.93E+01
134	5.17E+01	5.26E+01	5.73E-01	5.79E-01
135	5.23E+01	6.22E+01	4.02E+00	4.46E+00

NOTE:

1. 3.20E+01 = 3.20×10^1.

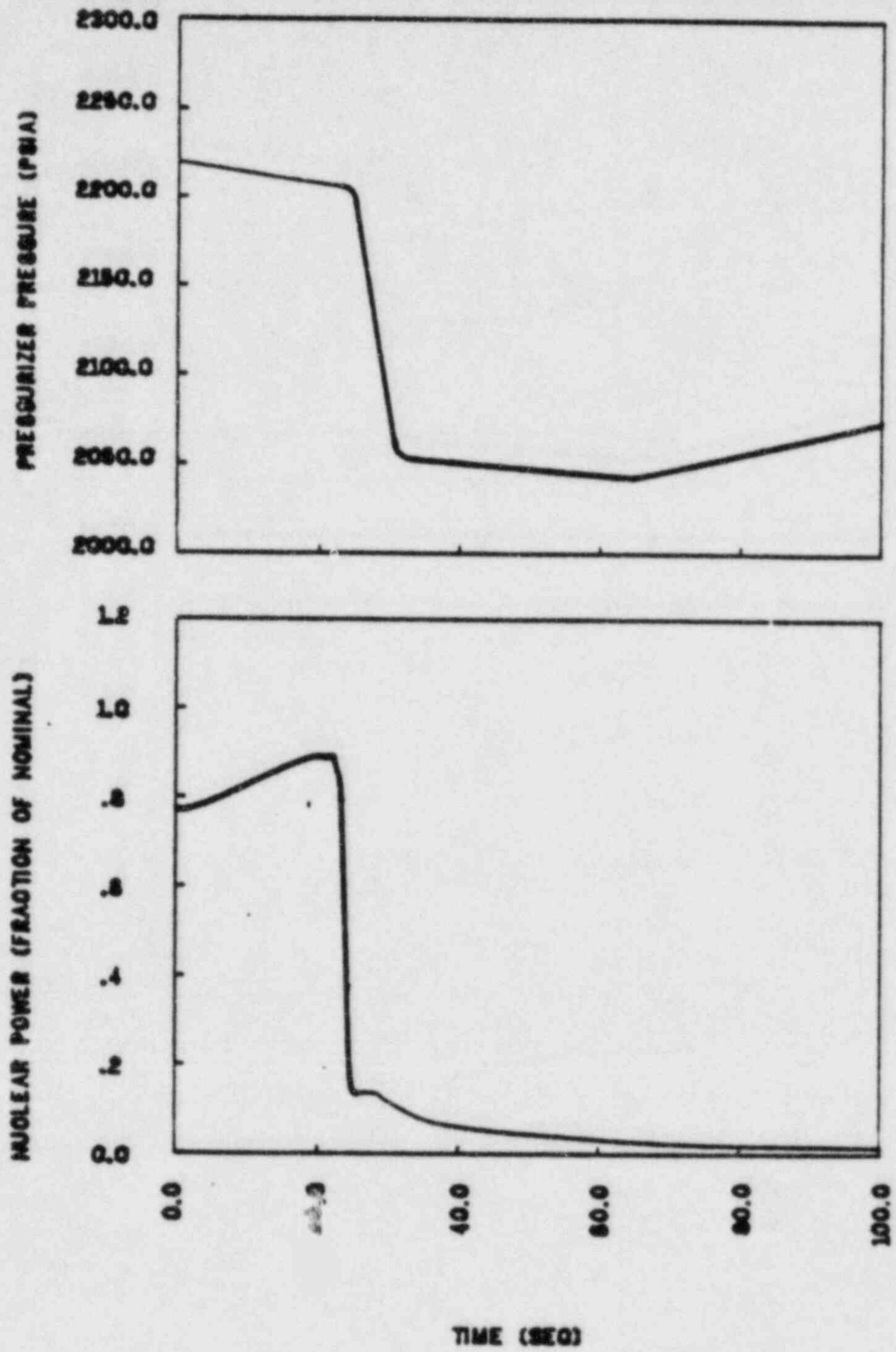


FIGURE 15.1-2b /a

FEEDWATER CONTROL VALVE
MALFUNCTION. REACTOR
AND TURBINE TRIPS
(N-1 LOOP OPERATION)

INTERACTIVE GRAPHICS
D.CHA.STOU.0751AXX.XXX.01.003.08298

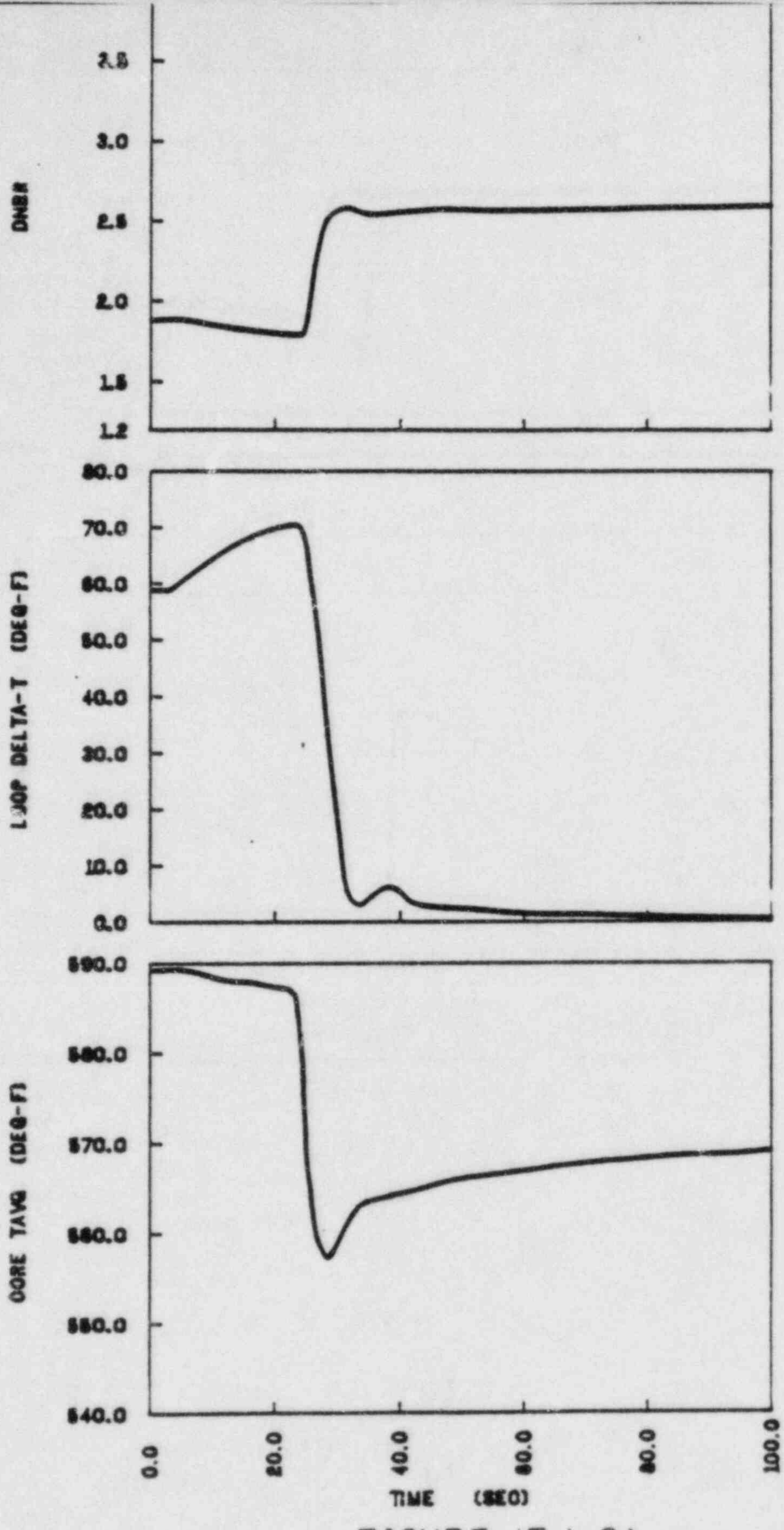


FIGURE 15.1-2A

FEEDWATER CONTROL VALVE MALFUNCTION. REACTOR AND TURBINE TRIPS N-1 LOOP

INTERACTIVE GRAPHICS
D.CHA.HE IP.0751AXX.XXX.01.110.08298

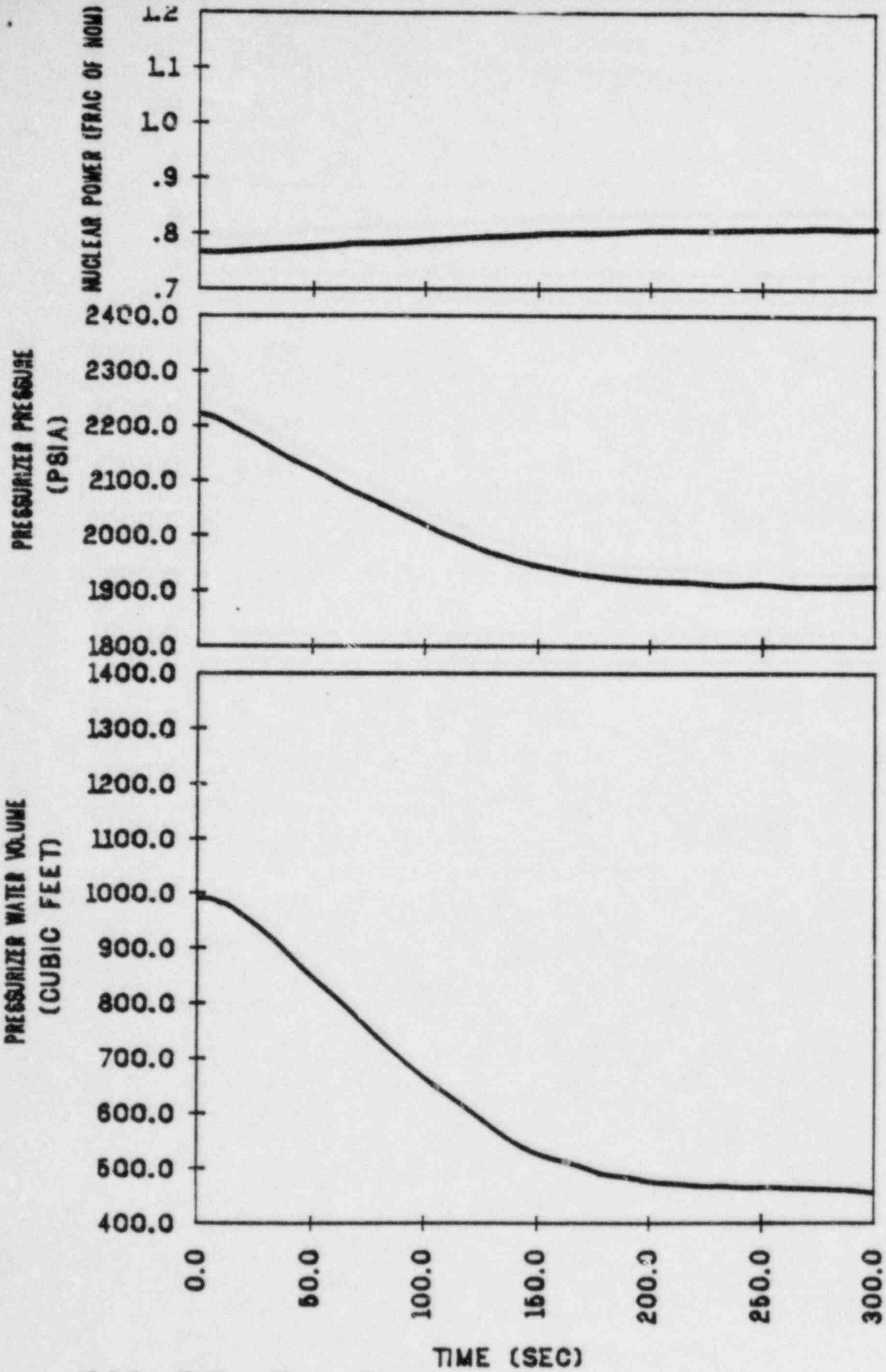


FIGURE 15.1-3A

TEN PERCENT STEP LOAD INCREASE.
MINIMUM MODERATOR FEEDBACK.
MANUAL REACTOR CONTROL
(N-1 /LOOP OPERATION)

INTERACTIVE GRAPHICS
D.CHA.GEOR.0751AXX.XXX.01.072.083183

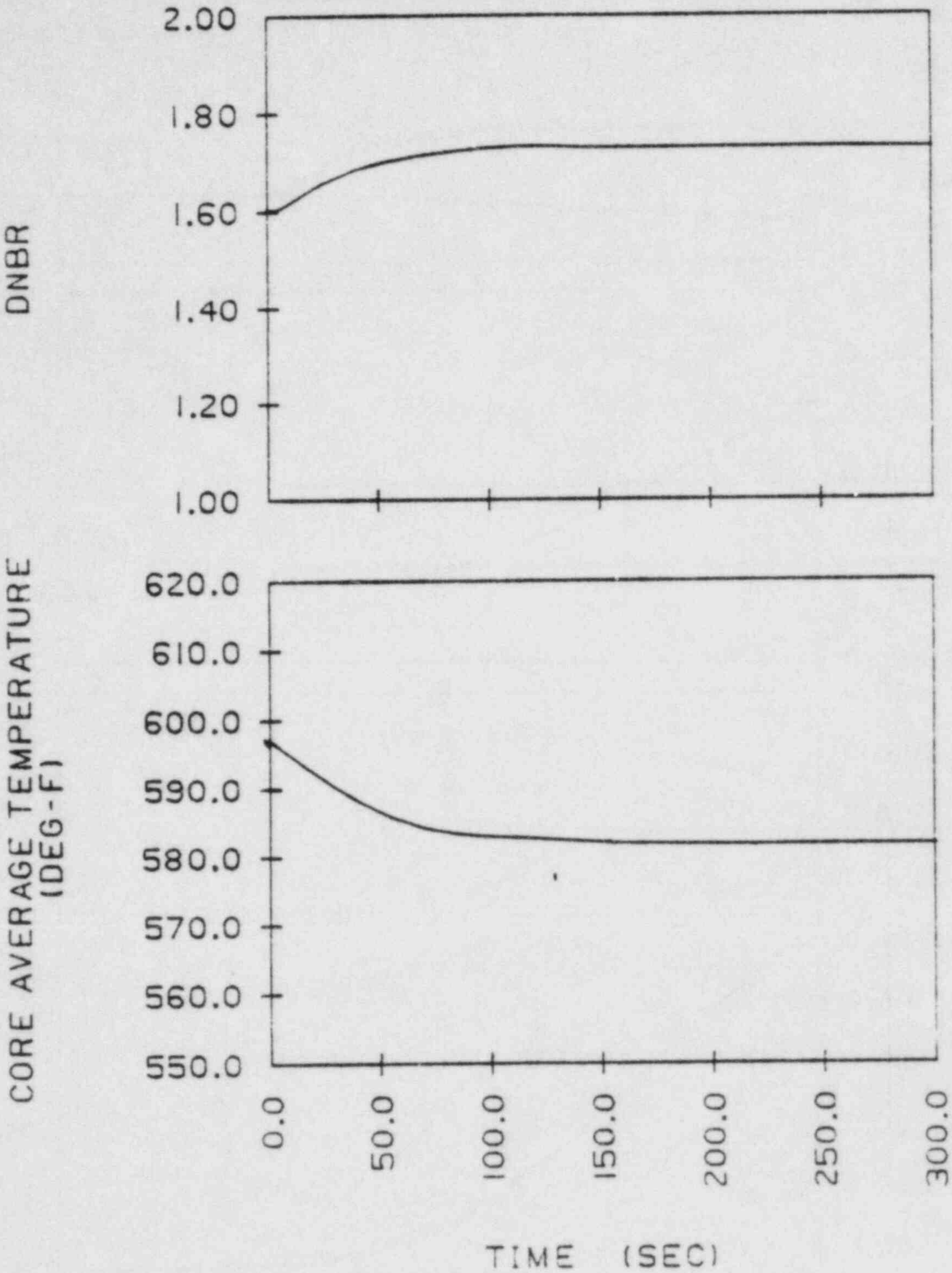


FIGURE 15.1-4
TEN PERCENT STEP LOAD INCREASE,
MINIMUM REACTIVITY FEEDBACK,
MANUAL NO REACTOR CONTROL
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

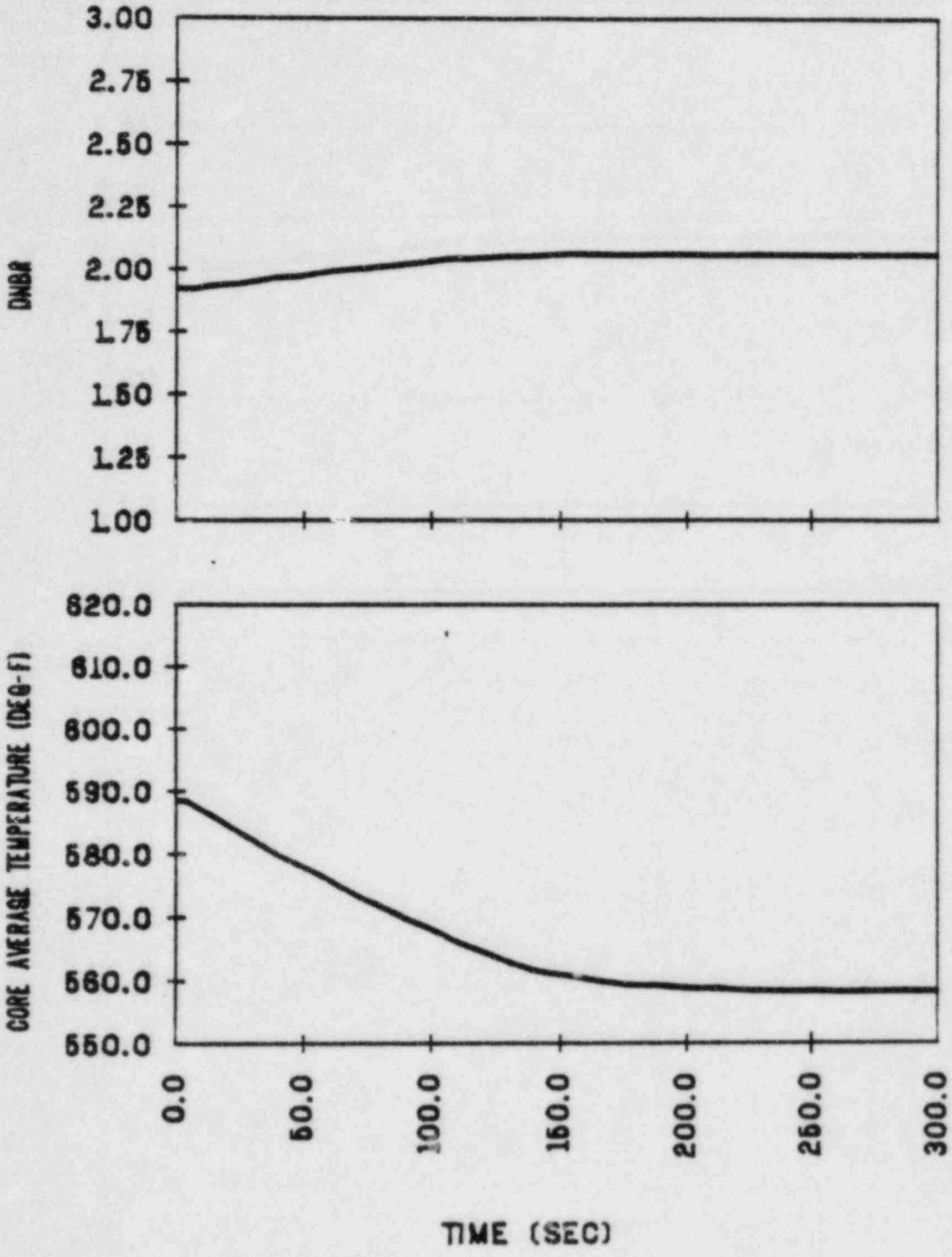


FIGURE 15.1-4A
TEN PERCENT STEP LOAD INCREASE.
MINIMUM REACTIVITY FEEDBACK.
Manual REACTOR CONTROL
(N-1 LOOP OPERATION)

INTERACTIVE GRAPHICS
D.CHA.GEOR.0751AXX..XX.01.045.083183

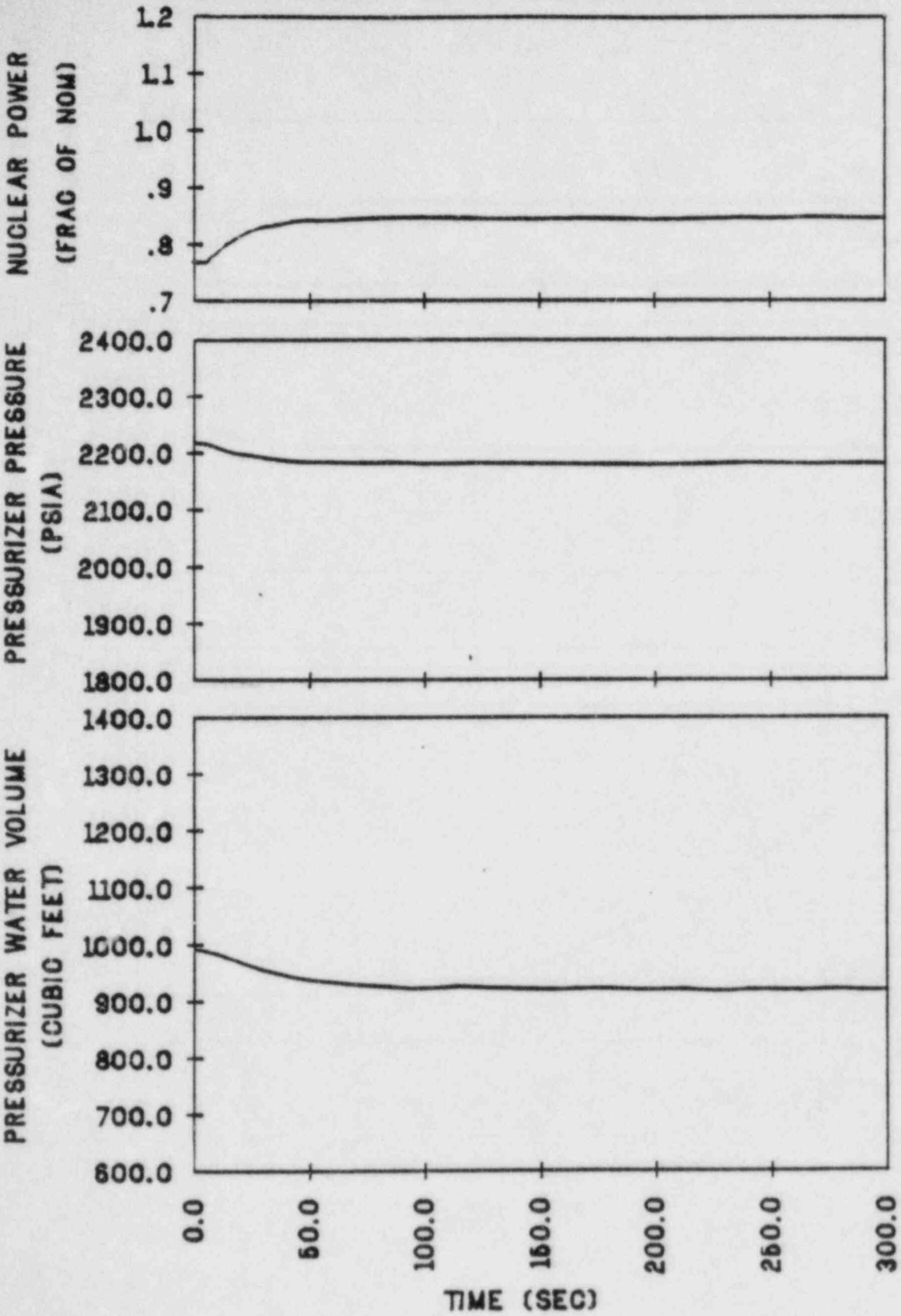


FIGURE 15.1-5A

TEN PERCENT STEP LOAD INCREASE.
MAXIMUM REACTIVITY FEEDBACK.
NO REACTOR CONTROL
N-1 LOOP OPERATION

INTERACTIVE GRAPHICS
D.CHA.GE OR.0751AXX.XXX.01.073.083183

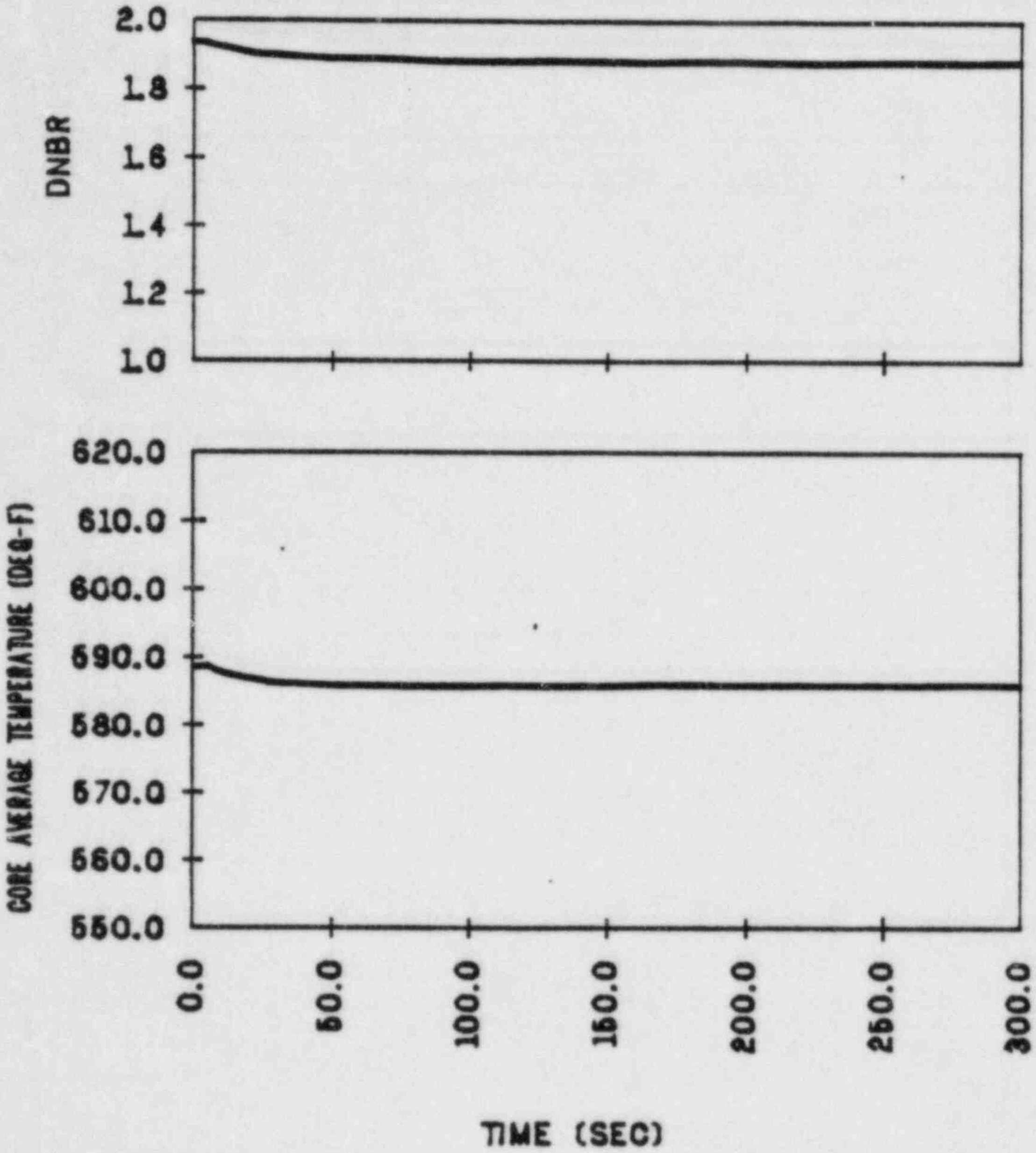


FIGURE 15.1-6A
TEN PERCENT STEP LOAD INCREASE.
MAXIMUM REACTIVITY FEEDBACK.
NO REACTOR CONTROL
(N-1 /LOOP OPERATION)

INTERACTIVE GRAPHICS
D.CHA.GEOR.0751AXX.XXX.01.049.083183

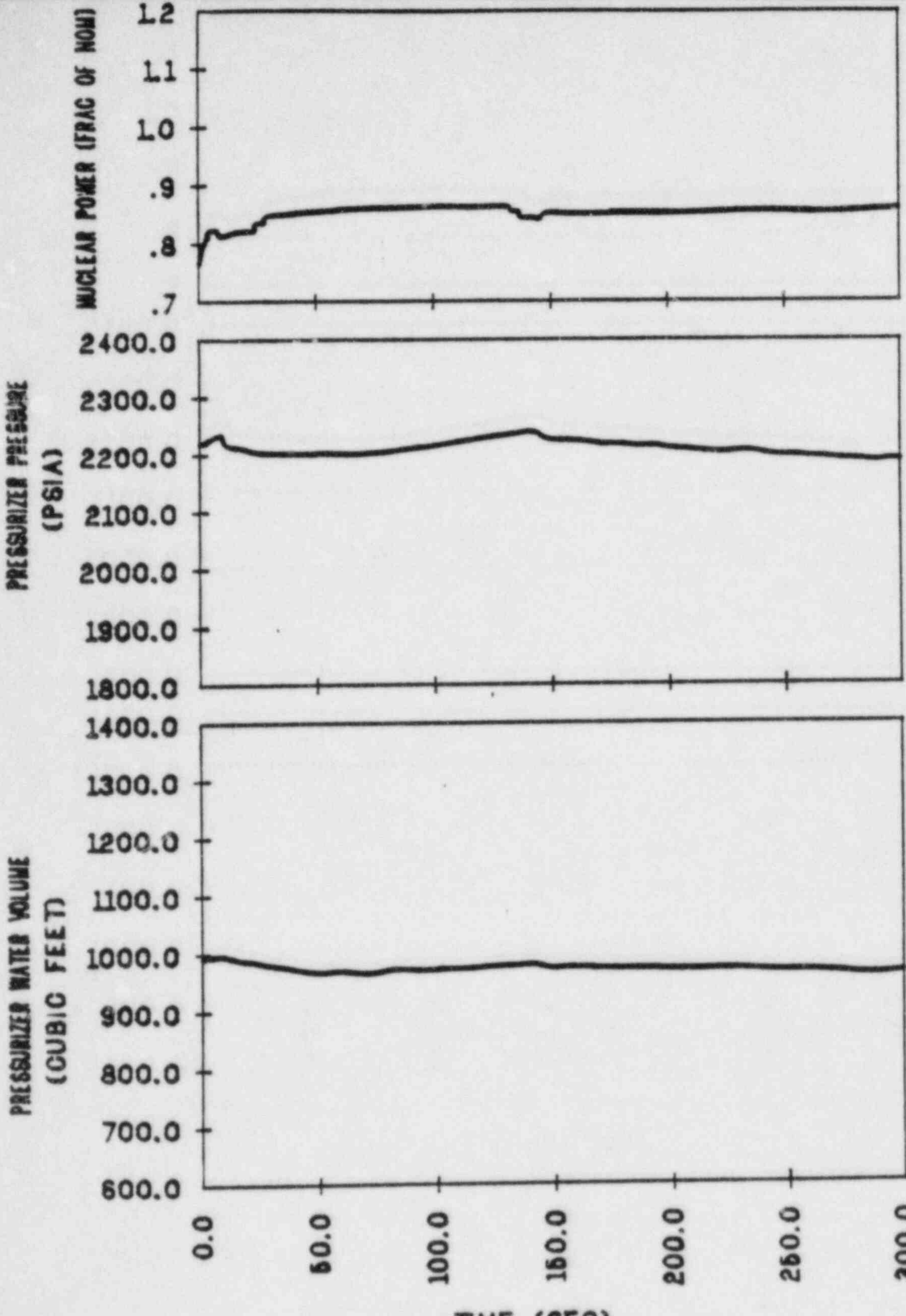


FIGURE 15.1-7A
TEN PERCENT STEP LOAD INCREASE.
MINIMUM REACTIVITY FEEDBACK.
AUTOMATIC REACTOR CONTROL
(N-1 /LOOP OPERATION)

INTERACTIVE GRAPHICS
D.CHA.GEOR.0751AXX.XXX.01.057.083183

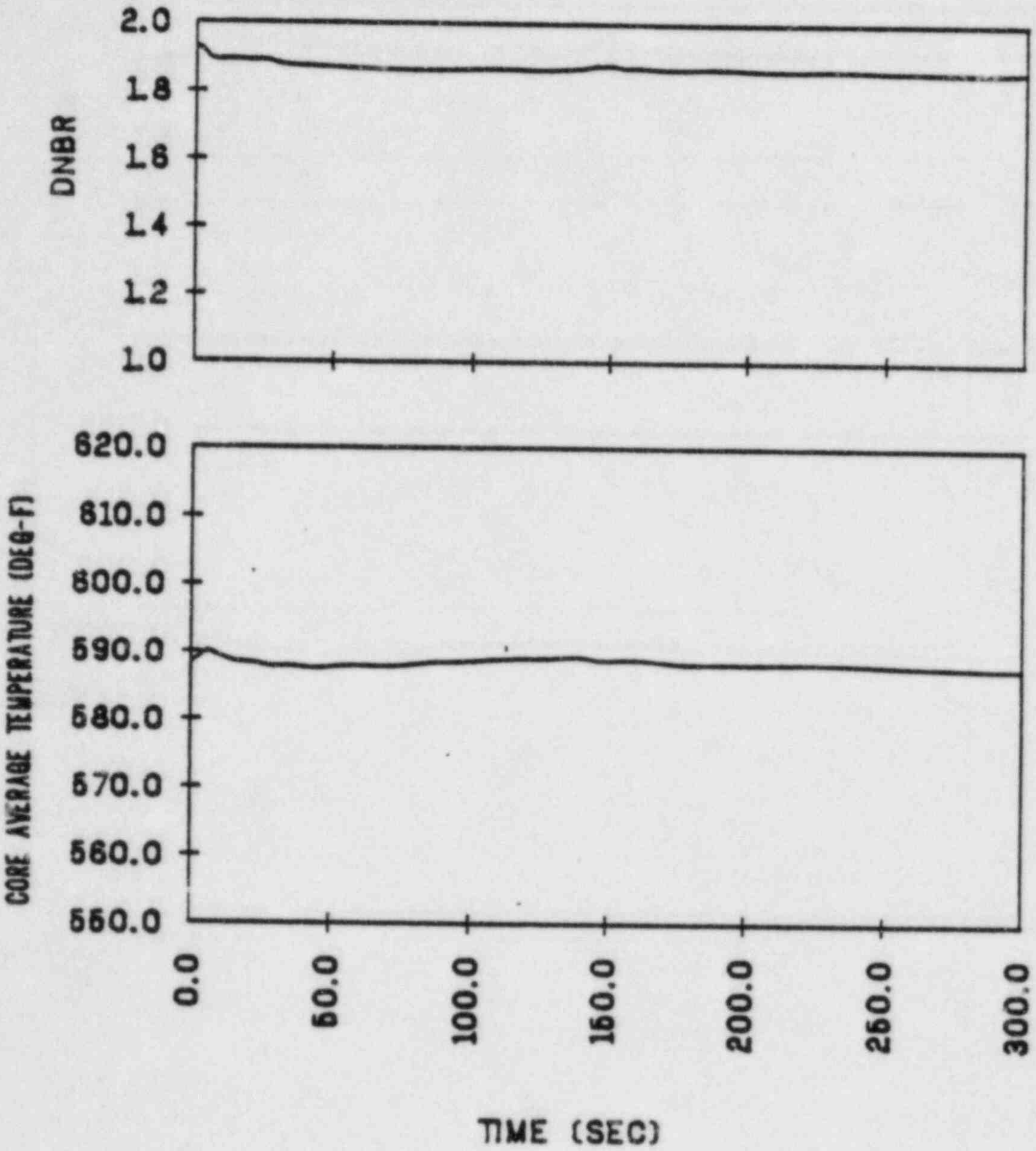


FIGURE 15.1-8A

TEN PERCENT STEP LOAD INCREASE.
MINIMUM REACTIVITY FEEDBACK.
AUTOMATIC REACTOR CONTROL
N-1 /LOOP OPERATION

INTERACTIVE GRAPHICS
D.CHA.GEOR.075!AXX.XXX.01.047.083:83

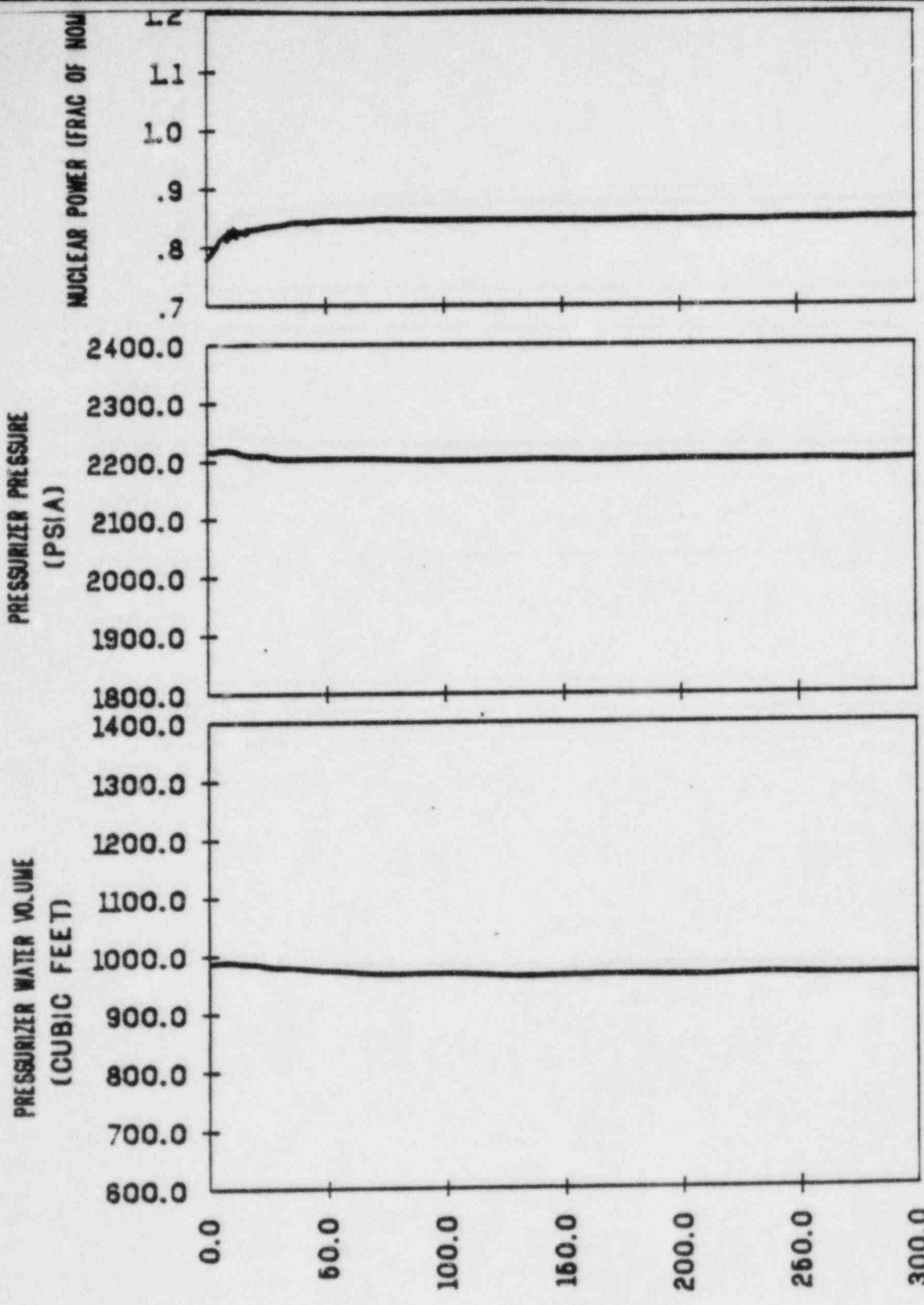


FIGURE 15.1-9A
TEN PERCENT STEP LOAD INCREASE.
MAXIMUM REACTIVITY FEEDBACK.
AUTOMATIC REACTOR CONTROL
(N-1 /LOOP OPERATION)

INTERACTIVE GRAPHICS
D.CHA.GEOR.0751AXX.XXX.01.056.083183

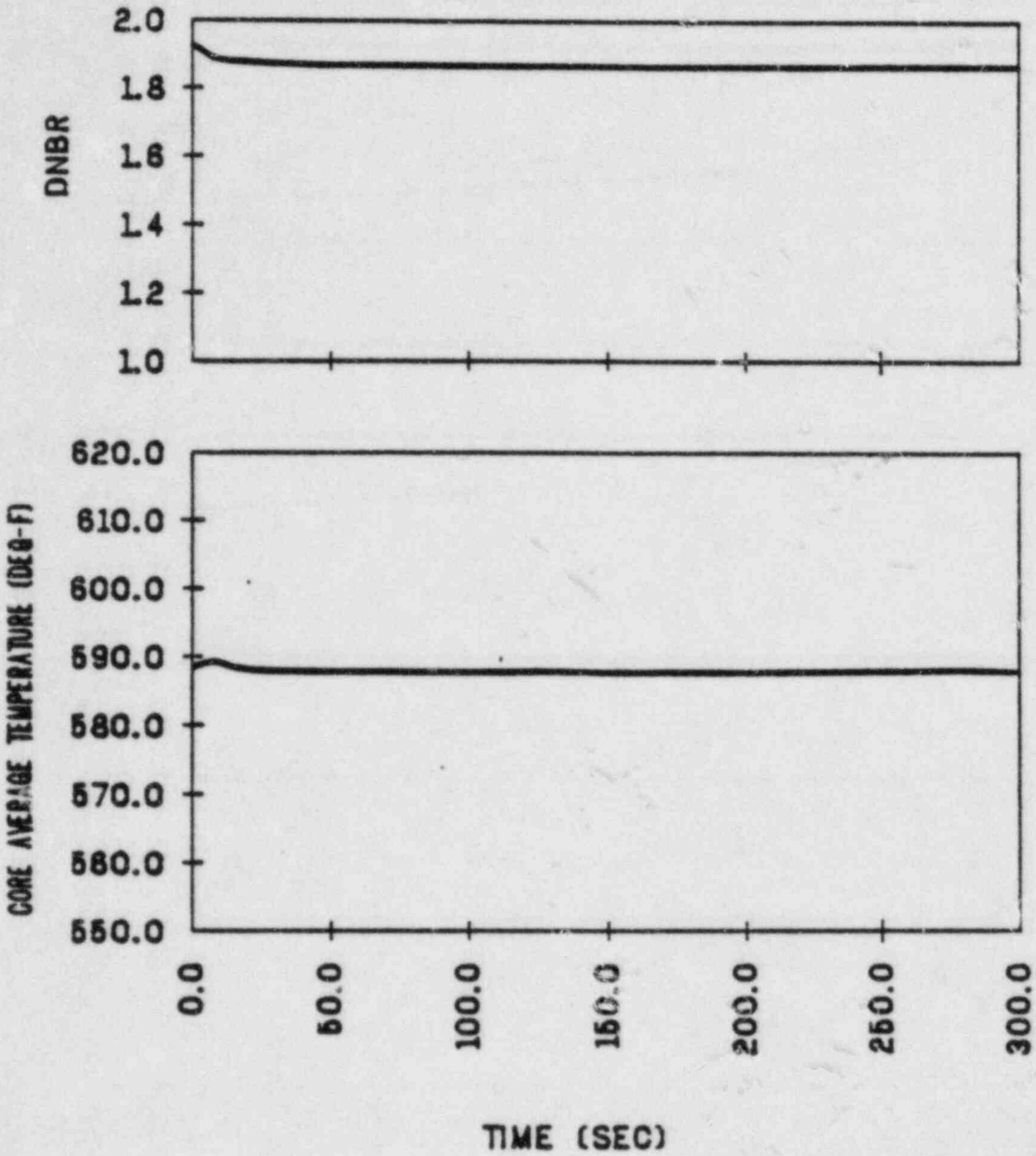


FIGURE 15.1-10A
TEN PERCENT STEP LOAD INCREASE.
MAXIMUM REACTIVITY FEEDBACK.
AUTOMATIC REACTOR CONTROL
(N-1 /LOOP OPERATION)

INTERACTIVE GRAPHICS
D.CHA.GEOR.0751AXX.XXX.01.043.083183

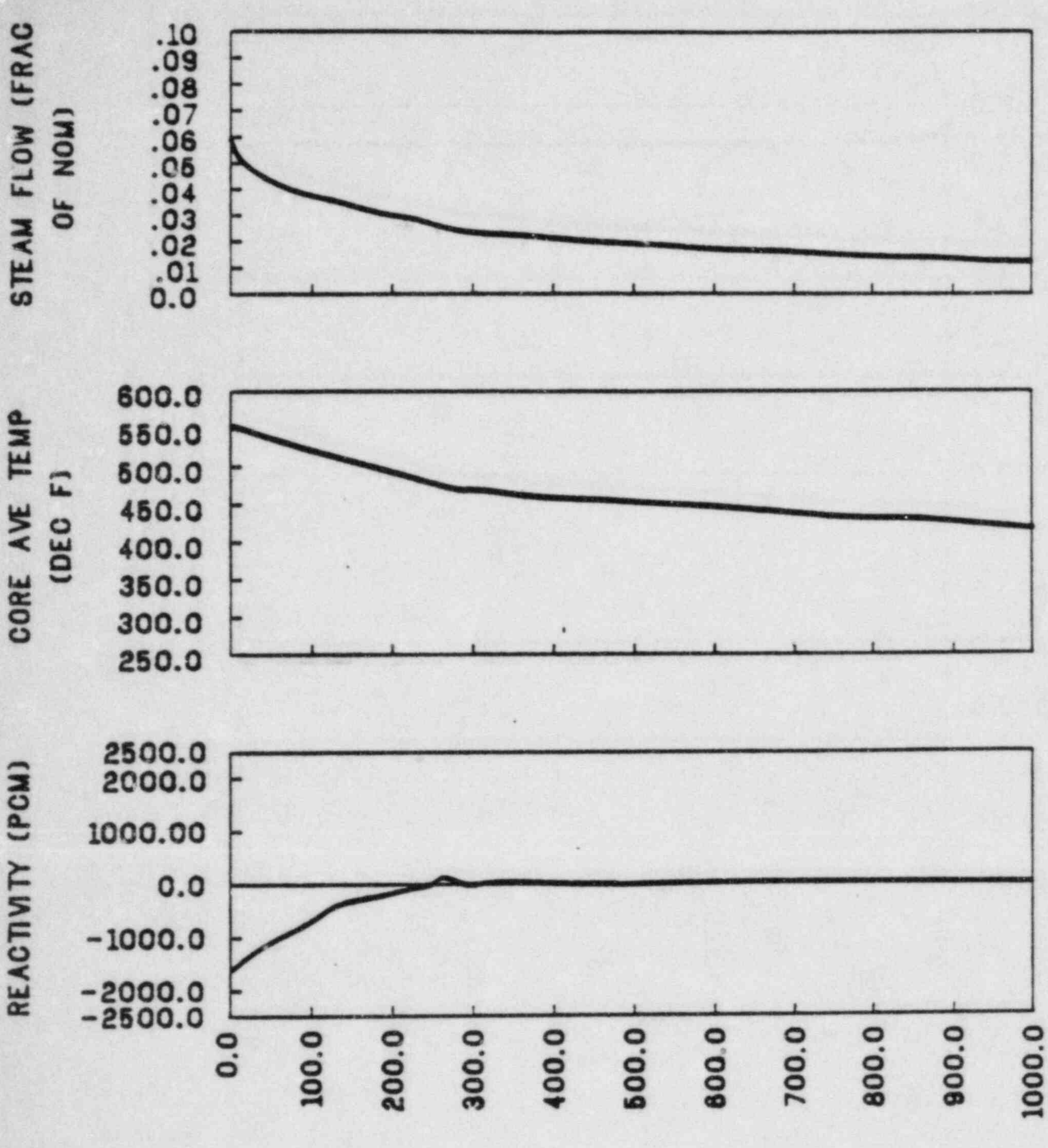


FIGURE 15.1-12A
FAILURE OF STEAM GENERATOR
SAFETY OR DUMP VALVE
N-1 LOOP OPERATION

INTERACTIVE GRAPHICS

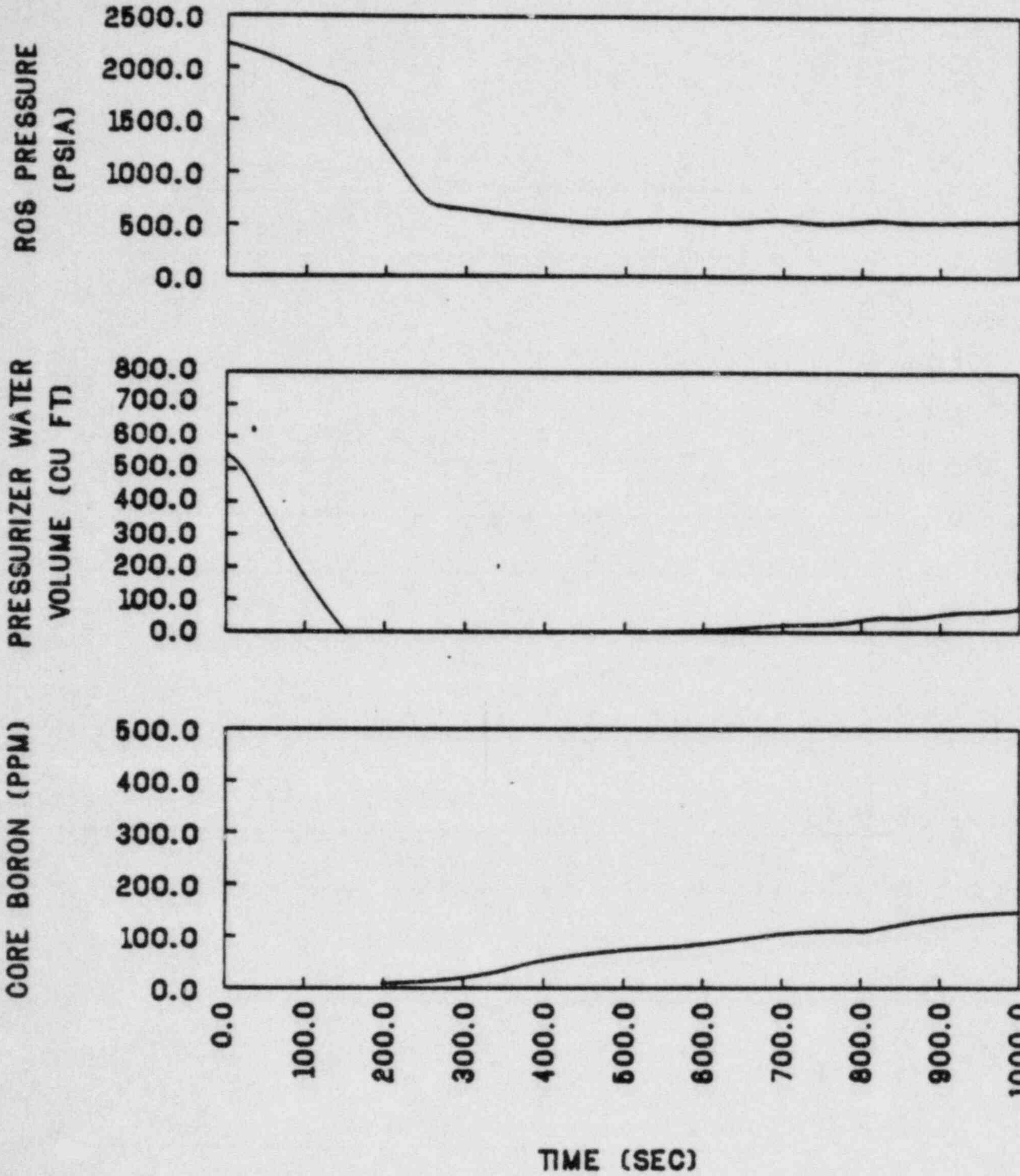


FIGURE 15.1-13A
FAILURE OF STEAM GENERATOR
SAFETY OR DUMP VALVE
N-1 LOOP OPERATION

INTERACTIVE GRAPHICS
D.CHA.PALM.0751AXX.XXX.01.067.082983

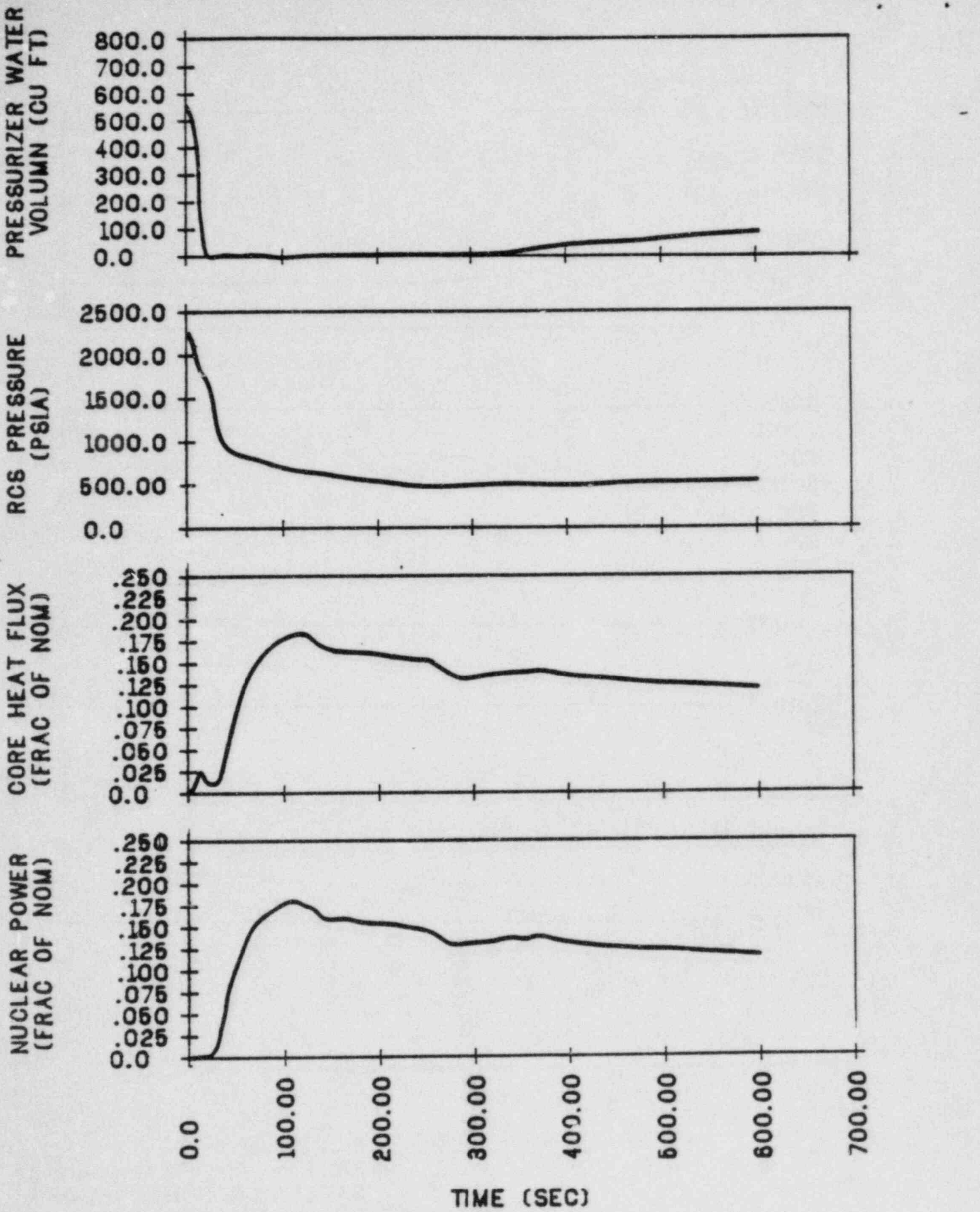


FIGURE 15.1-15a

1.4 FT² STEAMLINE RUPTURE
OFFSITE POWER AVAILABLE
N-1 LOOP OPERATION

INTERACTIVE GRAPHICS
D.CHA.FLAC.0751AXX.XXX.01.069.0829

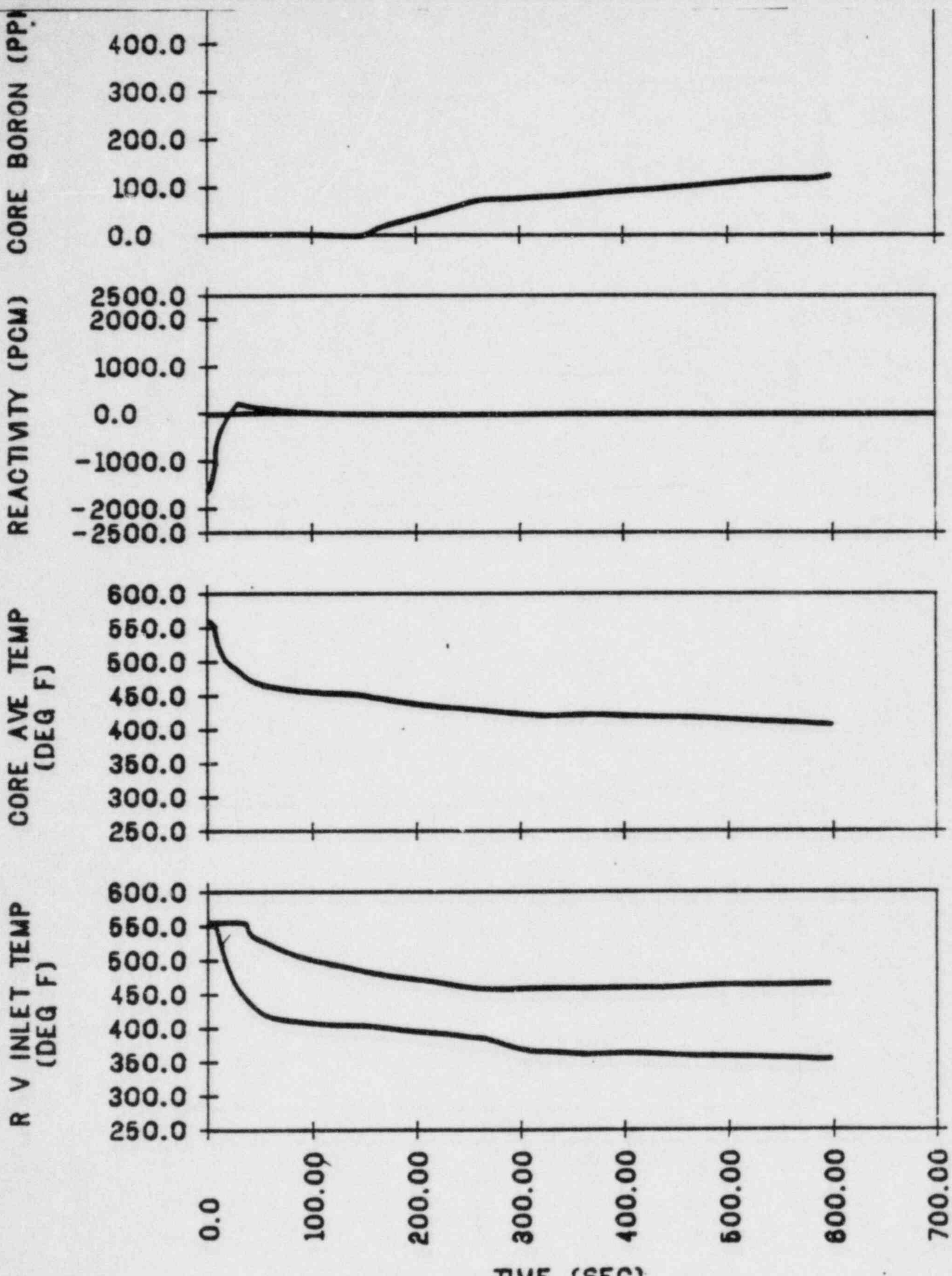


FIGURE 15.1-16a
1.4 FT^2 STEAMLINE RUPTURE
OFFSITE POWER AVAILABLE
N-1 LOOP OPERATION

INTERACTIVE GRAPHICS
D.CHA.FLAC.0751AXX.XXX.01.119.0829

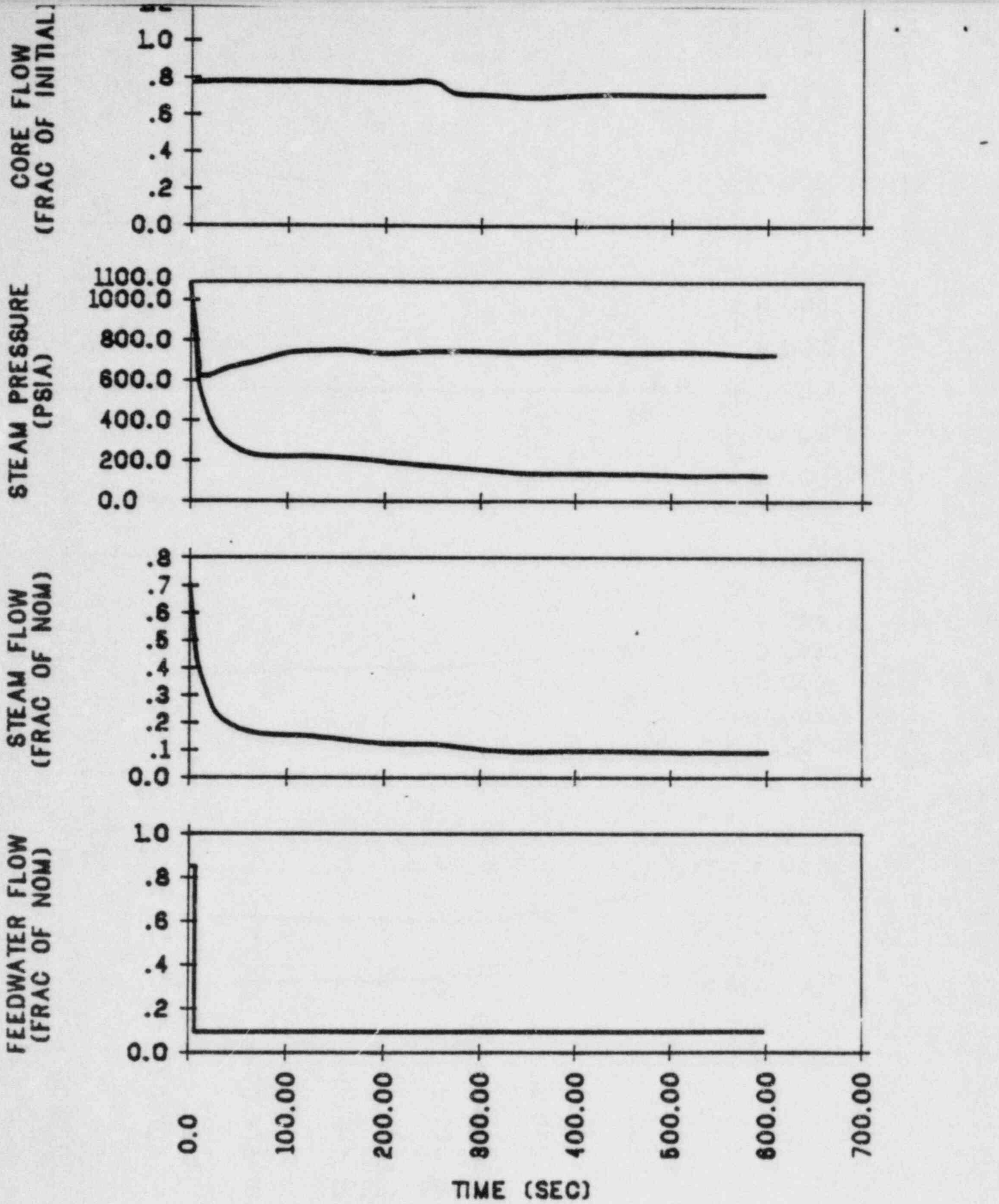


FIGURE 15.1-17a
1.4 FT² STEAMLINE RUPTURE
OFFSITE POWER AVAILABLE
N-1 LOOP OPERATION

INTERACTIVE GRAPHICS
D.CHA.FLAC.0751AXX.XXX.01.120.0829

15.2 DECREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM 1.9

A number of transients and accidents have been postulated which could 1.10
result in a reduction of the capacity of the secondary system to
remove heat generated in the reactor coolant system (RCS). Detailed 1.11
analyses are presented in this section for several such events which
have been identified as more limiting than the others.

Discussions of the following RCS coolant heatup events are presented 1.12
in Section 15.2.

1. Steam pressure regulator malfunction or failure that results 1.14
in decreasing steam flow.
2. Loss of external electrical load. 1.15
3. Turbine trip. 1.16
4. Inadvertent closure of main steam isolation valves. 1.17
5. Loss of condenser vacuum and other events resulting in 1.18
turbine trip.
6. Loss of nonemergency ac power to the station auxiliaries. 1.19
7. Loss of normal feedwater flow. 1.20
8. Feedwater system pipe break. 1.21

The above items are considered to be American Nuclear Society (ANS) 1.23
Condition II events, with the exception of a feedwater system pipe
break, which is considered to be an ANS Condition IV event.
Section 15.0.1 contains a discussion of ANS classifications and 1.24
applicable acceptance criteria.

15.2.1 Steam Pressure Regulator Malfunction or Failure that Results in Decreasing Steam Flow 1.28

There are no steam pressure regulators in Millstone 3 whose failure 1.31
or malfunction could cause a steam flow transient. 1.33

15.2.2 Loss of External Electrical Load 1.35

15.2.2.1 Identification of Causes and Accident Description 1.36

A major load loss on the plant can result from loss of external 1.37
electrical load due to some electrical system disturbance. Offsite 1.39
alternating current (ac) power remains available to operate plant
components such as the reactor coolant pumps; as a result, the onsite
emergency diesel generators are not required to function for this
event. Following the loss of generator load, an immediate fast 1.40
closure of the turbine control valves will occur. This will cause a 1.41
sudden reduction in steam flow, resulting in an increase in pressure
and temperature in the steam generator shell. As a result, the heat 1.42

transfer rate in the steam generator is reduced, causing the reactor coolant temperature to rise, which in turn causes coolant expansion, pressurizer insurge, and RCS pressure rise.

For a loss of external electrical load without subsequent turbine 1.43
trip, no direct reactor trip signal would be generated. The plant 1.44
would be expected to trip from the reactor protection system if a
safety limit were approached. A continued steam load of 1.45
approximately 5 percent would exist after total loss of external
electrical load because of the steam demand of plant auxiliaries.

In the event that a safety limit is approached, protection would be 1.46
provided by the high pressurizer pressure and overtemperature 1.47
ΔT trips. Following a complete loss of load, the maximum turbine 1.48
overspeed would be approximately 8 to 9 percent, resulting in an 1.49
overfrequency of less than 6 Hz. This resulting overfrequency is not 1.50
expected to damage the sensors in any way. However, it is noted that 1.51
frequent testing of this equipment is required by the Technical
Specifications. Any degradation in their performance could be 1.52
ascertained at that time. Any increased frequency to the reactor 1.53
coolant pump motors will result in slightly increased flow rate and
subsequent additional margin to safety limits. For postulated loss 1.55
of load and subsequent turbine generator overspeed, any overfrequency
condition is not seen by other safety-related pump motors, reactor 1.56
protection system equipment, or other safeguards loads. Safeguards 1.57
loads are supplied from offsite power or, alternatively, from
emergency diesel generators. Reactor protection system equipment is 1.58
supplied from the 120 V ac instrument power supply system, which in
turn is supplied from the inverters; the inverters are supplied from 1.59
a Class IE 125 V direct current (dc) bus energized from batteries or 1.60
by a rectified Class IE ac voltage from safeguards buses.

In the event the steam dump valves fail to open following a large 2.1
loss of load, the steam generator safety valves may lift and the 2.2
reactor may be tripped by the high pressurizer pressure signal, the
high pressurizer water level signal, or the 2.3
overtemperature ΔT signal. The steam generator shell side pressure 2.4
and reactor coolant temperatures will increase rapidly. However, the 2.5
pressurizer safety valves and steam generator safety valves are sized
to protect the RCS and steam generator against overpressure for all 2.6
load losses without assuming the operation of the steam dump system, 2.7
pressurizer spray, pressurizer power-operated relief valves,
automatic rod cluster control assembly control or direct reactor trip 2.8
on turbine trip.

The steam generator safety valve capacity is sized to remove the 2.9
steam flow at the engineered safety features rating (105 percent of 2.10
steam flow at rated power) from the steam generator without exceeding
110 percent of the steam system design pressure. The pressurizer 2.12
safety valve capacity is sized based on a complete loss of heat sink
with the plant initially operating at the maximum calculated turbine 2.13
load along with operation of the steam generator safety valves. The 2.14
pressurizer safety valves are then able to relieve sufficient steam

to maintain the RCS pressure within 110 percent of the RCS design 2.15
pressure.

A more complete discussion of overpressure protection can be found in 2.16
WCAP-7769.

A loss of external load is classified as an ANS Condition II event, 2.17
fault of moderate frequency. See Section 15.0.1 for a discussion of 2.18
Condition II events.

A loss of external load event results in a nuclear steam supply 2.19
(NSSS) system transient that is less severe than a turbine trip event
(see Section 15.2.3). Therefore, a detailed transient analysis is 2.20
not presented for the loss of external load.

The primary side transient is caused by a decrease in heat transfer 2.21
capability from primary to secondary due to a rapid termination of
steam flow to the turbine, accompanied by an automatic reduction of
feedwater flow (should feed flow not be reduced, a larger heat sink 2.22
would be available and the transient would be less severe).
Termination of steam flow to the turbine following a loss of external 2.23
load occurs due to automatic fast closure of the turbine control
valves in approximately 0.3 seconds. Following a turbine trip event, 2.24
termination of a steam flow occurs via turbine stop valve closure,
which occurs in approximately 0.1 seconds. Therefore, the transient 2.25
in primary pressure, temperature, and water volume will be less
severe for the loss of external load than for the turbine trip due to
a slightly slower loss of heat transfer capability.

The protection available to mitigate the consequences of a loss of 2.26
external load is the same as that for a turbine trip, as listed in
Table 15.0-6.

15.2.2.2 Analysis of Effects and Consequences 2.28

Refer to Section 15.2.3.2 for the method used to analyze the limiting 2.29
transient (turbine trip) in this grouping of events. The results of 2.32
the turbine trip event analysis are more severe than those expected
for the loss of external load, as discussed in Section 15.2.2.1. 2.33

Normal reactor control systems and engineered safety systems are not 2.34
required to function during a loss of external load. 2.35

The reactor protection system may be required to function following a 2.36
complete loss of external load to terminate core heat input and
prevent departure from nucleate boiling (DNB). Depending on the 2.37
magnitude of the load loss, pressurizer safety valves and/or steam
generator safety valves may be required to open to maintain system
pressure below allowable limits. No single active failure will 2.38
prevent operation of any system required to function. See WCAP-8330 2.39
for a discussion of anticipated transients without trip (ATWT)
considerations.

15.2.2.3 Conclusions 2.41

Based on results obtained for the turbine trip event (Section 15.2.3) 2.42
and considerations described in Section 15.2.2.1, the applicable 2.44
acceptance criteria for a loss of external load event are met.

15.2.2.4 Radiological Consequences 2.46

Loss of external load from full power would result in the operation 2.47
of the turbine bypass system. Operation of the turbine bypass system 2.49
results in bypassing steam to the condenser and atmosphere. Since no 2.50
fuel damage is postulated for this transient the radiological
releases are less severe than those for the steam line break accident 2.51
analyzed in Section 15.1.5.

15.2.3 Turbine Trip 2.53

15.2.3.1 Identification of Causes and Accident Description 2.54

For a turbine trip event, the reactor would be tripped directly 2.55
(unless below approximately 50 percent power) from a signal derived 2.57
from the turbine stop emergency trip fluid pressure and turbine stop
valves. The turbine stop valves close rapidly (typically 2.58
0.1 seconds) on loss of trip fluid pressure actuated by one of a
number of possible turbine trip signals. Turbine trip initiation 2.59
signals include:

- generator trip 3.1
- low condenser vacuum 3.2
- loss of lubricating oil 3.3
- turbine thrust bearing failure 3.4
- turbine overspeed 3.5
- main steam reheat high level 3.6
- manual trip 3.7

Upon initiation of stop valve closure, steam flow to the turbine 3.9
stops abruptly. Sensors on the stop valves detect the turbine trip 3.10
and initiate steam dump and, if above 50 percent power, a reactor
trip. The loss of steam flow results in an almost immediate rise in 3.11
secondary system temperature and pressure with a resultant primary
system transient as described in Section 15.2.2.1 for the loss of
external load event. A slightly more severe transient occurs for the 3.12
turbine trip event due to the more rapid loss of steam flow caused by
the more rapid valve closure.

The automatic steam dump system would normally accommodate the excess 3.13
steam generation. Reactor coolant temperatures and pressure do not 3.14
significantly increase if the steam dump system and pressurizer

pressure control system are functioning properly. If the turbine 3.15
condenser was not available, the excess steam generation would be
dumped to the atmosphere and main feedwater flow would be lost, and .
causing steam generator water level to fall. For this situation 3.16
feedwater flow would be maintained by the auxiliary feedwater system
to ensure adequate residual and decay heat removal capability.
Should the steam dump system fail to operate, the steam generator 3.17
safety valves may lift to provide pressure control. See 3.18
Section 15.2.2.1 for a further discussion of the transient.

A turbine trip is classified as an ANS Condition II event, fault of 3.19
moderate frequency. See Section 15.0.1 for a discussion of 3.20
Condition II events.

A turbine trip event is more limiting than loss of external load, 3.21
loss of condenser vacuum, and other turbine trip events. As such, 3.22
this event has been analyzed in detail. Results and discussion of 3.23
the analysis are presented in Section 15.2.3.2.

The plant systems and equipment available to mitigate the 3.24
consequences of a turbine trip are discussed in Section 15.0.8 and
listed in Table 15.0-6.

15.2.3.2 Analysis of Effects and Consequences 3.26

<u>Method of Analysis</u> 3.28

In this analysis, the behavior of the unit is evaluated for a 3.30
complete loss of steam load from 102 percent of full load power 3.32
without direct reactor trip primarily to show the adequacy of the
pressure relieving devices and also to demonstrate core protection
margins; that is, the turbine is assumed to trip without actuating 3.33
all the sensors for reactor trip on the turbine stop valves. The 3.34
assumption delays reactor trip until conditions in the RCS result in
a trip due to other signals. Thus, the analysis assumes a worst 3.35
transient. In addition, no credit is taken for steam dump. Main 3.37
feedwater flow is terminated at the same time of turbine trip, with
no credit taken for auxiliary feedwater to mitigate the consequences
of the transient.

The turbine trip transients are analyzed by employing the detailed 3.38
digital computer program LOFTRAN (WCAP-7907). The program simulates 3.39
the neutron kinetics, RCS, pressurizer, pressurizer relief and safety
valves, pressurizer spray, steam generator, and steam generator 3.40
safety valves. The program computes pertinent plant variables 3.41
including temperatures, pressures, and power level.

Initial operating conditions are assumed at values consistent with 3.42
steady state N and N-1 loop operation. Plant characteristics and 3.43
initial conditions are discussed in Section 15.0.3.

Major assumptions are summarized below. 3.44

1. Initial operating conditions 3.47

The initial reactor power and RCS temperatures are assumed 3.49
at their maximum values consistent with the steady state 3.51
full power operation including allowances for calibration
and instrument errors. The initial RCS pressure is assumed 3.52
at a minimum value consistent with the steady state full
power operation including allowances for calibration and 3.53
instrument errors. This results in the maximum power 3.54
difference for the load loss, and the minimum margin to core
protection limits at the initiation of the accident. Cases 3.56
with four loops in operation and with three loops, based on
77 percent of four-loop full power, in operation are 3.57
considered.

2. Moderator and Doppler coefficients of reactivity 3.60

The turbine trip is analyzed with a least negative moderator 4.2
temperature, a large negative moderator temperature and 4.4
Doppler power coefficients. (Figure 15.0-2). 4.5

3. Reactor control 4.8

From the standpoint of the maximum pressures attained it is 4.10
conservative to assume that the reactor is in manual
control. If the reactor were in automatic control, the 4.12
control rod banks would move prior to trip and reduce the
severity of the transient.

4. Steam release 4.15

No credit is taken for the operation of the steam dump 4.17
system or steam generator power operated relief valves. The 4.19
steam generator pressure rises to the safety valve setpoint
where steam release through safety valves limits secondary
steam pressure at the setpoint value.

5. Pressurizer spray and power operated relief valves 4.22

Two cases for both the minimum and maximum moderator 4.24
feedback cases are analyzed.

a. Full credit is taken for the effect of pressurizer 4.27
spray and power operated relief valves in reducing or
limiting the coolant pressure. Safety valves are also 4.28
available.

b. No credit is taken for the effect of pressurizer spray 4.29
and power operated relief valves in reducing or
limiting the coolant pressure. Safety valves are 4.30
operable.

6. Feedwater flow 4.33

Main feedwater flow to the steam generators is assumed to be 4.35
lost at the time of turbine trip. No credit is taken for 4.37
auxiliary feedwater flow since a stabilized plant condition
will be reached before auxiliary feedwater initiation is
normally assumed to occur; however, the auxiliary feedwater
pumps would be expected to start on low-low steam generator 4.38
water level. The auxiliary feedwater flow would remove core 4.39
decay heat following plant stabilization.

7. Reactor trip is actuated by the first reactor protection 4.41
system trip setpoint reached with no credit taken for the
direct reactor trip on the turbine trip. Trip signals are 4.42
expected due to high pressurizer pressure,
overtemperature ΔT, high pressurizer water level, and low- —
low steam generator water level.

Plant characteristics and initial conditions are further discussed in 4.44
Section 15.0.3.

Except as discussed above, normal reactor control system and 4.45
engineered safety systems are not required to function. Several 4.46
cases are presented in which pressurizer spray and power operated
relief valves are assumed, but the more limiting cases where these
functions are not assumed are also presented.

The reactor protection system may be required to function following a 4.47
turbine trip. Pressurizer safety valves and/or steam generator 4.48
safety valves may be required to open to maintain system pressures
below allowable limits. No single active failure will prevent 4.49
operation of any system required to function. A discussion of ATWT 4.50
considerations is presented in WCAP-8330.

<u>Results</u> 4.53

The transient responses for a turbine trip from 102 percent of full 4.55
power operation are shown for four cases: two cases for minimum 4.56
moderator feedback and two cases for maximum moderator feedback 4.58
(Figures 15.2-1 through 15.2-8). The calculated sequence of events 4.59
for the accident is shown in Table 15.2-1.

Figures 15.2-1 and 15.2-2 show the transient responses for the total 4.60
loss of steam load with a least negative moderator temperature 5.1
coefficient assuming full credit for the pressurizer spray and
pressurizer power operated relief valves. No credit is taken for the 5.3
steam dump. The reactor is tripped by the high pressurizer pressure 5.4
trip channel. The minimum DNBR remains well above the 1.30 limit. 5.5
The pressurizer safety valves are actuated, and maintain primary 5.6
system pressure below 110 percent of the design value. The steam 5.7
generator safety valves limit the secondary steam conditions to
saturation at the safety valve setpoint.

Figures 15.2-3 through 15.2-4 show the responses for the total loss 5.8
of steam load with a large negative moderator temperature 5.9
coefficient. All other plant parameters are the same as the above. 5.10
The magnitude of the reactivity coefficients used in the analysis in 5.11
conjunction with the pressurizer pressure control assumed cause this 5.12
case to differ from the previous case. The pressure control limits 5.13
peak RCS pressure, initially via pressurizer relief valves, and the
large reactivity feedback causes the nuclear power to be reduced due 5.14
to the rapid rise in average temperature. High pressurizer pressure 5.15
reactor trip is prevented by the relief valve action followed by the
reduction in nuclear power due to reactivity feedback effects. If 5.17
less conservative reactivity coefficients were used, the reactor trip
would occur earlier, on overpressure, as in the previous case. 5.18

The system stabilizes at about 75 percent of nominal nuclear power 5.19
due to the combined effects of the reactivity coefficients and the
steam flow through the steam generator safety valves. This 5.20
stabilized condition continues until sufficient secondary side mass
is lost through the steam generator safety valves to cause a reactor
trip on low-low steam generator water level. This is a highly 5.21
conservative analysis because, in actuality, the level in the steam
generator downcomer would rapidly decrease to the low-low level trip
setpoint within the first 10 seconds due to shrinkage in the
secondary fluid caused by the increase in secondary side pressure and 5.22
the rapid primary side power reduction.

The DNBR does not drop below its initial value at any time during the 5.23
transient. Pressurizer relief valves and spray prevent primary 5.24
system overpressurization as described above; steam generator safety
valves prevent overpressurization in the secondary side. The 5.25
pressurizer safety valves are not actuated for this case.

The turbine trip accident was also studied assuming the plant to be 5.26
initially operating at 102 percent of full power with no credit taken 5.27
for the pressurizer spray, pressurizer power operated relief valves,
or steam dump. The reactor is tripped on the high pressurizer 5.29
pressure signal. Figures 15.2-5 and 15.2-6 show the transients with 5.30
a least negative moderator coefficient. The neutron flux remains 5.31
essentially constant at 102 percent of full power until the reactor
is tripped. The DNBR increases throughout the transient. In this 5.33
case the pressurizer safety valves are actuated, and maintain system
pressure below 110 percent of the design value. 5.34

Figures 15.2-7 and 15.2-8 are the transients with maximum reactivity 5.35
feedback with the other assumptions being the same as in the
preceding case. Again, the DNBR increases throughout the transient 5.36
and the pressurizer safety valves are actuated to limit primary
pressure. 5.37

Figures 15.2-1A through 15.2-8A show the transient response for the 5.38
conditions previously described, assuming three loops in operation. 5.40

WCAP-7769 presents additional results of analysis for a complete loss 5.41
of heat sink including loss of main feedwater. This analysis shows 5.43

the overpressure protection that is afforded by the pressurizer and steam generator safety valves.

15.2.3.3 Conclusions 5.45

Results of the analysis, including those in WCAP-7769 show that the 5.46
plant design is such that a turbine trip without a direct or 5.50
immediate reactor trip presents no hazard to the integrity of the RCS
or the main steam system. Pressure relieving devices incorporated in 5.51
the two systems are adequate to limit the maximum pressures to within
the design limits. 5.52

The integrity of the core is maintained by operation of the reactor 5.53
protection system, i.e., the DNBR will be maintained above the 5.54
1.30 value. The DNBR design basis is described in Section 4.4. 5.55
Applicable acceptance criteria as listed in Section 15.0.1 have been 5.56
met. The above analysis demonstrates the ability of the nuclear 5.57
steam supply system to safely withstand a full load rejection. 5.58

15.2.3.4 Radiological Consequences 5.60

The turbine trip transient and steam released for this event are 6.1
similar to the loss of load transient described in Section 15.2.2.4. 6.2

There are only minimal radiological consequences associated with this 6.4
event; therefore, this event is not limiting. The radiological 6.5
consequences resulting from atmospheric steam dump are less severe
than those of the steam line break event analyzed in Section 15.1.5 6.6
since no fuel damage is postulated to occur.

15.2.4 Inadvertent Closure of Main Steam Isolation Valves 6.8

Inadvertent closure of the main steam isolation valves would result 6.9
in a turbine trip. Turbine trips are discussed in Section 15.2.3. 6.11

15.2.5 Loss of Condenser Vacuum and Other Events Resulting in 6.15
Turbine Trip

Loss of condenser vacuum is one of the events that can cause a 6.18
turbine trip. Turbine trip initiating events are described in 6.20
Section 15.2.3. A loss of condenser vacuum would preclude the use of 6.21
steam dump to the condenser; however, since steam dump is assumed not
to be available in the turbine trip analysis, no additional adverse
effects would result if the turbine trip were caused by loss of 6.22
condenser vacuum. Therefore, the analysis results and conclusions 6.23
contained in Section 15.2.3 apply to loss of condenser vacuum. In 6.24
addition, analyses for the other possible causes of a turbine trip,
as listed in Section 15.2.3.1, are covered by Section 15.2.3.
Possible overfrequency effects due to a turbine overspeed condition 6.25
are discussed in Section 15.2.2.1 and are not a concern for this type
of event.

15.2.6 Loss of Nonemergency AC Power to the Station Auxiliaries 6.27

15.2.6.1 Identification of Causes and Accident Description 6.28

A complete loss of emergency ac power may result in the loss of all 6.29
power to the station auxiliaries; i.e., the reactor coolant pumps, 6.32
condensate pumps, etc. The loss of power may be caused by a complete 6.33
loss of the offsite grid accompanied by a turbine generator trip at
the station, or by a loss of the onsite ac distribution system. 6.34

This transient is more severe than the turbine trip event analyzed in 6.35
Section 15.2.3 because for this case the decrease in heat removal by 6.36
the secondary is accompanied by a flow coastdown which further
reduces the capacity of the primary coolant to remove heat from the 6.37
core. The reactor will trip: 6.38

1. due to turbine trip; 6.40

2. upon reaching one of the trip setpoints in the primary and 6.41
secondary systems as a result of the flow coastdown and 6.42
decrease in secondary heat removal; or

3. due to loss of power to the control rod drive mechanisms as 6.43
a result of the loss of power to the plant.

Following a loss of ac power with turbine and reactor trips, the 6.45
sequence described below will occur.

1. Plant vital instruments are supplied from emergency dc power 6.47
sources.

2. As the steam system pressure rises following the trip, the 6.48
steam generator power operated relief valves may be
automatically opened to the atmosphere. Steam dump to the 6.49
condenser is assumed not to be available. If the steam flow 6.50
rate through the power relief valves is not available, the
steam generator self actuated safety valves may lift to
dissipate the sensible heat of the fuel and coolant plus the
residual decay heat produced in the reactor. 6.51

3. As the no-load temperature is approached, the steam 6.52
generator power operated relief valves (or the self actuated
safety valves, if the power operated relief valves are not
available) are used to dissipate the residual decay heat and 6.53
to maintain the plant at the hot shutdown condition.

4. The emergency diesel generators, started on loss of voltage 6.54
on the plant emergency buses, begin to supply plant vital
loads.

The auxiliary feedwater system is started automatically as follows. 6.56

Two motor-driven auxiliary feedwater pumps are started on any of 6.58
the following:

1. low-low level in any steam generator; 6.60

2. any safety injection signal; 7.1

3. loss of offsite power; or 7.2

4. manual actuation. 7.3

One turbine driven auxiliary feedwater pump is started on any of 7.5
the following:

1. low-low level in any two steam generators; 7.7

2. loss of offsite power; or 7.8

3. manual actuation. 7.9

Refer to Section 10.4.9 for a description of the auxiliary feedwater 7.11
system.

The motor-driven auxiliary feedwater pumps are supplied power by the 7.12
diesels and the turbine-driven pump utilizes steam from the secondary 7.13
system. Both type pumps are designed to start within 1 minute even 7.14
if a loss of all ac power occures simultaneously with loss of normal 7.15
feedwater. The turbine exhausts the secondary steam to the 7.16
atmosphere. The auxiliary pumps take suction from the demineralized 7.17
water tank (DWST) for delivery to the steam generators.

Upon the loss of power to the reactor coolant pumps, coolant flow 7.18
necessary for core cooling and the removal of residual heat is 7.19
maintained by natural circulation in the reactor coolant loops. A 7.20
loss of nonemergency ac power to the station auxiliaries is
classified as an ANS Condition II event, fault of moderate frequency. 7.21
See Section 15.0.1 for a discussion of Condition II events. 7.22

A loss of ac power event, as described above, is a more limiting 7.23
event than the turbine trip initiated decrease in secondary heat 7.24
removal without loss of ac power, which was analyzed in
Section 15.2.3. However, a loss of ac power to the station 7.25
auxiliaries as postulated above could also result in a loss of normal
feedwater if the condensate pumps lose power to operate. A loss of 7.27
normal feedwater caused by a loss of ac power is the most limiting
Condition II event in the decrease in secondary heat removal 7.28
category, and is analyzed in Section 15.2.7. Therefore, detailed 7.29
analytical results for a loss of ac power transient will not be
presented here. The results of the analysis in Section 15.2.7 are 7.30
applicable to the loss of ac power event.

Following the reactor coolant pump coastdown caused by the loss of ac 7.31
power, the natural circulation capability of the RCS will remove
residual and decay heat from the core, aided by auxiliary feedwater
in the secondary system. An analysis is presented here to show that 7.33
the natural circulation flow in the RCS following a loss of ac power
event is sufficient to remove residual heat from the core.

The plant systems and equipment available to mitigate the consequences of a loss of ac power event are discussed in Section 15.0.8 and listed in Table 15.0-6. 7.34

15.2.6.2 Analysis of Effects and Consequences 7.36

<u>Method of Analysis</u> 7.38

A detailed analysis using the LOFTRAN Code (WCAP-7907) is done to 7.42
obtain the natural circulation flow following a station blackout.
The simulation describes the plant thermal kinetics, RCS including 7.43
the natural circulation, pressurizer, steam generators and feedwater 7.44
system. The digital program computes pertinent variables including 7.45
the steam generator level, pressurizer water level, and reactor 7.46
coolant average temperature.

The assumptions used in the analysis are as follows. 7.47

1. The plant is initially operating at 102 percent of the engineered safety features design rating. 7.49

2. A conservative core residual heat generation based upon long term operation at the initial power level preceding the trip. 7.50

3. As the no-load temperature is approached, the steam generator power operated relief valves (or the self actuated safety valves, if the power operated relief valves are not 7.51
available) are used to dissipate the residual decay heat and to maintain the plant at the hot shutdown condition. 7.52

4. The emergency generators started on loss of voltage on the plant emergency buses begin to supply plant vital loads. 7.53

A loss of normal feedwater is classified as an ANS Condition II 7.55
event, fault of moderate frequency. See Section 15.0.1 for a 7.56
discussion of Condition II events.

The following provide the necessary protection against a loss of normal feedwater. 7.57

1. Reactor trip on low-low water level in any steam generator. 7.59

2. High pressurizer pressure. 7.60

3. Reactor trip on overtemperature ΔT. 8~~.1~~

4. High pressurizer water level. 8.2

The auxiliary feedwater system is started automatically as discussed 8.4
in Section 15.2.6.1. The steam-driven auxiliary feedwater pump 8.5
utilizes steam from the secondary system and exhausts to the
atmosphere. The motor-driven auxiliary feedwater pumps are supplied 8.6

by power from the emergency generators. The pumps take suction 8.7
directly from the DWST tank for delivery to the steam generators.

Upon the loss of power to the reactor coolant pumps, coolant flow 8.8
necessary for core cooling and the removal of residual heat is
maintained by natural circulation in the reactor coolant loops. The 8.9
analysis presented in Section 15.2.6 demonstrates the natural
circulation capability of the RCS.

A heat transfer coeffficient in the steam generator associated with 8.10
RCS natural circulation.

Plant characteristics and initial conditions are further discussed in 8.11
Section 15.0.3.

Steady state cases are run at a number of power levels consistent 8.12
with the decay heat generation rates expected. Equilibrium 8.13
conditions are established and the natural circulation flow through
the core is recorded for each power level.

Results 8.16

The transient response of the RCS following a loss of ac power is 8.19
less severe than for the loss of normal feedwater event analyzed in
Section 15.2.7, and the results are not reproduced here. 8.21

The first few seconds of the transient will closely resemble a 8.22
simulation of the complete loss of flow incident (see
Section 15.3.2), i.e., core damage due to rapidly increasing core
temperatures is prevented by promptly tripping the reactor. After 8.24
the reactor trip, stored and residual decay heat must be removed to
prevent damage to either the RCS or the core.

The natural circulation flow as a function of residual reactor power 8.25
is presented in Table 15.2-2. The results show that the natural 8.26
circulation flow available is sufficient to provide adequate core
decay heat removal following reactor trip and reactor coolant pump
coastdown.

15.2.6.3 Conclusions 8.28

Analysis of the natural circulation capability of the RCS has 8.29
demonstrated that sufficient heat removal capability exists following 8.31
reactor coolant pump coastdown to prevent fuel or clad damage.

The loss of ac power transient is less severe than the loss of normal 8.32
feedwater event as analyzed in Section 15.2.7.

15.2.6.4 Radiological Consequences 8.34

A loss of nonemergency ac power to the plant auxiliaries would result 8.35
in a turbine and reactor trip and loss of condenser vacuum. Heat 8.38
removal from the secondary system would occur through the steam
generator power relief valves or safety valves. Since no fuel damage 8.39

is postulated to occur from this transient, the radiological
consequences are less severe than the steam line break accident 8.40
analyzed in Section 15.1.5.

15.2.7 Loss of Normal Feedwater Flow 8.42

15.2.7.1 Identification of Causes and Accident Description 8.43

A loss of normal feedwater (from pump failures, valve malfunctions, 8.44
or loss of offsite ac power) results in a reduction in capability of 8.46
the secondary system to remove the heat generated in the reactor
core. If an alternative supply of feedwater were not supplied to the 8.47
plant, core residual heat following reactor trip would heat the
primary system water to the point where water relief from the
pressurizer would occur, resulting in a substanital loss of water 8.48
from the RCS. Since the plant is tripped well before the steam 8.49
generator heat transfer capability is reduced, the primary system
variables never approach a DNB condition.

The worst postulated loss of normal feedwater event is one initiated 8.50
by a loss of offsite ac power as described in Section 15.2.6. This 8.51
is due to the decreased capability of the reactor coolant to remove
residual core heat as a result of the reactor coolant pump coastdown. 8.52

As stated in Section 15.2.6.1, the following events occur upon loss 8.53
of ac power.

1. Plant vital instruments are supplied from emergency dc power 8.55
sources.

2. As the steam system pressure rises following the trip, the 8.56
steam generator power operated relief valves are
automatically opened to the atmosphere. Steam dump to the 8.57
condenser is assumed not to be available. If the steam flow 8.58
rate through the power relief valves is not available, the
steam generator self actuated safety valves may lift to
dissipate the sensible heat of the fuel and coolant plus the
residual decay heat produced in the reactor. 8.59

A loss of normal feedwater caused by a loss of offsite ac power is 9.1
the most limiting Condition II event in the decrease in secondary
heat removal category, for the reasons presented in Section 15.2.7.1.
Therefore, a full analysis of the system transient is presented below 9.2
to show that following a loss of normal feedwater, the auxiliary
feedwater system is capable of removing the stored and residual heat,
thus preventing either overpressurization of the RCS or loss of water 9.3
from the reactor core, and returning the plant to a safe condition.

15.2.7.2 Analysis of Effects and Consequences 9.5

<u>Method of Analysis</u> 9.7

A detailed analysis using the LOFTRAN Code (WCAP-7907) is performed 9.9
in order to obtain the plant transient following a loss of normal 9.10
feedwater. The simulation describes the plant thermal kinetics, RCS 9.12
including the natural circulation, pressurizer, steam generators and 9.13
feedwater system. The digital program computes pertinent variables 9.14
including the steam generator level, pressurizer water level, and
reactor coolant average temperature. 9.15

Assumptions made in the analysis are as follows. 9.16

1. The plant is initially operating at 102 percent of the engineered safety features design rating. 9.18

2. A conservative core residual heat generation based upon long 9.19
term operation at the initial power level preceding the 9.20
trip.

3. A heat transfer coefficient in the steam generator 9.21
associated with RCS natural circulation.

4. Reactor trip occurs on steam generator low-low level. 9.22

5. The worst single failure in the auxiliary feedwater system 9.23
occurs.

6. Auxiliary feedwater is delivered to two steam generators for 9.24
N-loop analysis. The most limiting auxiliary feedwater 9.25
system configuration for N-1 loop operation is for auxiliary
feedwater to be supplied to one active steam generator. 9.26

7. Secondary system steam relief is achieved through the self- 9.27
actuated safety valves. Note that steam relief will, in 9.28
fact be through the power-operated relief valves or
condenser dump valves for most cases of loss of normal
feedwater. However, for the sake of analysis these have 9.29
been assumed unavailable.

8. The initial reactor coolant average temperature is 6.5°F 9.30
higher than the nominal value since this results in a
greater expansion of the RCS water during the transient and,
thus, in a higher water level in the pressurizer. 9.31

The loss of normal feedwater analysis is performed to demonstrate the 9.33
adequacy of the reactor protection and engineered safeguards systems 9.34
(e.g., the auxiliary feedwater system) in removing long term decay
heat and preventing excessive heatup of the RCS with possible 9.35
resultant RCS overpressurization or loss of RCS water.

As such, the assumptions used in this analysis are designed to 9.36
minimize the energy removal capability of the system and to maximize

the possibility of water relief from the coolant system by maximizing
the coolant system expansion, as noted in the assumptions listed 9.37
above.

One such assumption is the loss of external (offsite) ac power. This 9.39
assumption results in coolant flow decay down to natural circulation
conditions and a corresponding reduction in the steam generator heat
transfer coefficient. Following a loss of offsite ac power, the 9.40
first few seconds of a loss of normal feedwater transient will be
virtually identical to the transient response (including DNBR and
neutron flux versus time) presented in Section 15.3.2 for the 9.41
complete loss of forced reactor coolant flow.

If ac power were not lost for this incident the reactor coolant flow 9.42
would remain at its normal value and the reactor would trip via the 9.43
low-low steam generator level trip with no change in DNBR below the
value at the start of the transient. The reactor coolant pumps would 9.45
be manually tripped at some later time to reduce heat addition to the
RCS. The auxiliary feedwater system has sufficient capacity, even 9.46
assuming the worst single failure, to preclude filling the
pressurizer should the pumps not be tripped. 9.47

Plant characteristics and initial conditions are further discussed in 9.48
Section 15.0.3. Plant systems and equipment which are available to 9.49
mitigate the effects of a loss of normal feedwater accident are
discussed in Section 15.0.8 and listed in Table 15.0-6. Normal 9.50
reactor control systems are not required to function during this
transient. The reactor protection system is required to function 9.51
following a loss of normal feedwater as analyzed here. The auxiliary 9.52
feedwater system is required to deliver a minimum auxiliary feedwater
flow rate. No single active failure will prevent operation of any 9.53
system required to function. A discussion of ATWT considerations is 9.54
presented in WCAP-8330.

Results 9.57

Figures 15.2-9 and 15.2-10 show the significant plant parameter 9.59
transients following a loss of normal feedwater with four loops in 10.1
operation initially. Figures 15.2-9A and 15.2-10A are for three 10.2
loops in operation initially.

Following the reactor and turbine trip from full load, the water 10.3
level in the steam generators will fall due to the reduction of steam
generator void fraction and because steam flow through the safety
valves continues to dissipate the stored and generated heat. One 10.5
minute following the initiation of the low-low level trip, at least
one auxiliary feedwater pump is automatically started, reducing the
rate of water level decrease.

The capacity of the auxiliary feedwater pumps is such that the water 10.6
level in the steam generator being fed does not recede below the
lowest level at which sufficient heat transfer area is available to
dissipate core residual heat without water relief from the RCS relief 10.7
or safety valves. From Figure 15.2-10 it can be seen that at no time 10.8

is the tubesheet uncovered in the steam generators receiving auxiliary feedwater flow and that at no time is there water relief from the pressurizer.

The calculated sequence of events for this accident is listed in 10.9
Table 15.2-1. As shown on Figures 15.2-9 and 15.2-10, the plant 10.10
approaches a stabilized condition following reactor trip and
auxiliary feedwater initiation. Standard plant shutdown procedures 10.12
may be followed to further cool down the plant.

15.2.7.3 Conclusions 10.14

Results of the analysis show that a loss of normal feedwater does not 10.15
adversely affect the core, the RCS, or the steam system since the 10.17
auxiliary feedwater capacity is such that reactor coolant water is
not relieved from the pressurizer relief or safety valves, and the
water level in all steam generators receiving auxiliary feedwater is 10.18
maintained above the tubesheets.

15.2.7.4 Radiological Consequences 10.20

The steam release and resulting radiological consequences from this 10.21
transient would be the same as that for the loss of offsite ac power; 10.22
and, similarly, radiological consequences resulting from this 10.24
transient are less severe than those of the steam line break accident 10.25
analyzed in Section 15.1.5.

15.2.8 Feedwater System Pipe Break 10.27

15.2.8.1 Identification of Causes and Accident Description 10.28

A major feedwater line rupture is defined as a break in a feedwater 10.29
line large enough to prevent the addition of sufficient feedwater to
the steam generators to maintain shell side fluid inventory in the 10.31
steam generators. If the break is postulated in a feedline between 10.32
the check valve and the steam generator, fluid from the steam
generator may also be discharged through the break. (A break 10.33
upstream of the feedline check valve would affect the nuclear steam
supply system only as a loss of feedwater. This case is covered by 10.34
the evaluation in Section 15.2.7).

Depending upon the size of the break and the plant operating 10.35
conditions at the time of the break, the break could cause either a
RCS cooldown (by excessive energy discharge through the break) or a
RCS heatup. Potential RCS cooldown resulting from a secondary pipe 10.36
rupture is evaluated in Section 15.1.5. Therefore, only the RCS 10.37
heatup effects are evaluated for a feedwater line rupture.

A feedwater line rupture reduces the ability to remove heat generated 10.38
by the core from the RCS for the following reasons. 10.39

1. Feedwater flow to the steam generators is reduced. Since 10.42
feedwater is subcooled, its loss may cause reactor coolant
temperatures to increase prior to reactor trip.

2. Fluid in the steam generator may be discharged through the break, and would then not be available for decay heat removal after trip.

3. The break may be large enough to prevent the addition of any main feedwater after trip.

An auxiliary feedwater system is provided to assure that adequate feedwater will be available such that:

1. no substantial overpressurization of the RCS shall occur, and

2. decay heat is removed in order to maintain sufficient liquid in the RCS to keep the reactor core covered.

Refer to Section 10.4.9 for a description of the auxiliary feedwater system interfaces.

A major feedwater line rupture is classified as an ANS Condition IV event. See Section 15.0.1 for a discussion of Condition IV events.

A main feedwater line rupture is the most limiting event in the decrease in secondary heat removal category. Therefore, a full transient analysis is presented here.

The severity of the feedwater line rupture transient depends on a number of system parameters including break size, initial reactor power, and credit taken for the functioning of various control and safety systems. A number of cases of feedwater line break have been analyzed. Based on these analyses, it has been shown that the most limiting feedwater line rupture is a double ended rupture of the largest feedwater line, occurring at full power with and without loss of offsite power, and at 77 percent power for three loops in operation with and without loss of offsite power with no credit taken for pressurizer pressure control. These cases are analyzed below.

The following provides the necessary protection for a main feedwater rupture.

1. A reactor trip on any of the following conditions:

   a. high pressurizer pressure;

   b. overtemperature ΔT;

   c. low-low steam generator water level in any steam generator; or

   d. safety injection signals from either of the following:

      1) low steam line pressure, or

      2) high containment pressure (hi-1)

(Refer to Chapter 7 for a description of the actuation 11.16
system).

2. An auxiliary feedwater system to provide an assured source 11.18
of feedwater to the steam generators for decay heat removal.
(Refer to Section 10.4.9 for a description of the auxiliary 11.19
feedwater system).

15.2.8.2 Analysis of Effects and Consequences 11.22

<u>Method of Analysis</u> 11.24

A detailed analysis using the LOFTRAN Code (WCAP-7907) is performed 11.26
in order to determine the plant transient following a feedwater line
rupture. The code describes the plant thermal kinetics, RCS 11.28
including natural circulation, pressurizer, steam generators, and
feedwater system, and computes pertinent variables including the 11.29
pressurizer pressure, pressurizer water level, and reactor coolant 11.30
average temperature.

The cases analyzed assume a double ended rupture of the largest 11.31
feedwater pipe at full power. Major assumptions made in the analyses 11.32
are as follows.

1. The plant is initially operating at 102 percent of 11.34
engineered safety features design rating (77 percent power
for three loops in operation). 11.35

2. Initial reactor coolant average temperature is 6.5°F above 11.36
the nominal value, and the initial pressurizer pressure is 11.37
30 psi above its nominal value.

3. No credit is taken for the pressurizer power operated relief 11.38
valves or pressurizer spray.

4. Initial pressurizer level is at the nominal programmed value 11.39
plus 2 percent (error); initial steam generator water level 11.40
is at the nominal value.

5. No credit is taken for the high pressurizer pressure reactor 11.41
trip. Note: This assumption is made for calculational 11.42
convenience. Pressurizer power operated relief valves and 11.43
spray could act to delay the high pressure trip.
Assumptions 3 and 5 permit evaluation of one hypothetical, 11.44
limiting case rather than two possible cases: one with a
high pressure trip and no pressure control and one with 11.45
pressure control but not high pressure trip.

6. Main feedwater flow to all steam generators is assumed to be 11.46
lost at the time the break occurs (all main feedwater spills 11.47
out through the break).

7. The worst possible break area is assumed. This maximizes 11.49
the blowdown discharge rate following the time of trip,
which maximizes the resultant heatup of the reactor coolant. 11.50

8. A conservative feedline break discharge quality is assumed 11.51
prior to the time the reactor trip occurs, thereby
maximizing the time the trip setpoint is reached. After the 11.53
trip occurs, a saturated liquid discharge is assumed until
all the water inventory is discharged from the affected
steam generator. This minimizes the heat removal capability 11.54
of the affected steam generator.

9. Reactor trip is assumed to be actuated by low-low level in 11.55
the affected steam generator when the water level falls
below the top of the U-tubes of any steam generator. 11.56

10. The auxiliary feedwater system is actuated by the low-low 11.57
steam generator water level signal. The auxiliary feedwater 11.58 480
system is assumed to supply a total of 850 gpm to three 375
unaffected steam generators. (600 gpm to two intact steam 11.59
generators for three-loop operation.) A 60-second delay was 11.60
assumed following the low-low level signal to allow time for 12.1
startup of the emergency diesel generators and the auxiliary
feedwater pumps. 12.2

11. No credit is taken for heat energy deposited in RCS metal 12.3
during the RCS heatup.

12. No credit is taken for charging or letdown. 12.4

13. Steam generator heat transfer area is assumed to decrease as 12.5
the shell side liquid inventory decreases.

14. Conservative core residual heat generation is assumed based 12.6
upon long term operation at the initial power level
preceding the trip. 12.7

15. No credit is taken for the following potential protection 12.8
logic signals to mitigate the consequences of the accident: 12.9

    a. high pressurizer pressure 12.11

    b. overtemperature ΔT 12.12

    c. high pressurizer level 12.13

    d. high containment pressure 12.14

Receipt of a low-low steam generator water level signal in at least 12.16
one steam generator starts the motor-driven auxiliary feedwater 12.17
pumps, which then deliver auxiliary feedwater flow to the steam
generators. The turbine-driven auxiliary feedwater pump is initiated 12.18
if the low-low steam generator water level signal is reached in at 12.19
least two steam generators. Similiarly, receipt of a low steam line 12.20

pressure signal in at least one steam line initiates a steam line isolation signal which closes the main steam line isolation valves in all steam lines. This signal also gives a safety injection signal 12.21
which initiates flow of borated water into the RCS. The amount of 12.22
safety injection flow is a function of RCS pressure.

Emergency operating procedures following a secondary system line 12.23
rupture will call for the following actions to be taken by the 12.24
reactor operator.

1. Isolate feedwater flow spilling out the break of ruptured 12.26
steam generator and align system so level in intact steam generators recovers.

2. Turn off all reactor coolant pumps (if offsite power is 12.27
still available).

3. Stop high head safety injection pumps if water level in the 12.28
pressurizer is recovering, and the intact steam generators 12.29
are at the safety valve setpoint, and water level in intact
steam generators is in the narrow range span. 12.30

Shutting off the reactor coolant pumps (action 2, above) serves to 12.32
decrease the addition of energy (approximately 3.54 MW per pump) to 12.33
the RCS. Isolating feedwater flow through the break allows 12.34
additional auxiliary feedwater flow to be diverted to the intact
steam generators (see Assumption 10, above). 12.35

Subsequent to recovery of level in the intact steam generators, the 12.36
high head safety injection pumps will be turned off and plant operating procedures will be followed in cooling the plant to hot
shutdown conditions. Plant characteristics and initial conditions 12.37
are further discussed in Section 15.0.3.

No reactor control systems are assumed to function. The reactor 12.39
protection system is required to function following a feedwater line
rupture as analyzed here. No single active failure will prevent 12.40
operation of this system.

The engineered safety systems assumed to function are the auxiliary 12.41
feedwater system and the safety injection system. For the auxiliary 12.42
feedwater system, passive flow limiting devices (cavitating venturis) limit flow to the faulted steam generator and permits all intact steam generators to receive auxiliary feedwater following the break.
The turbine-driven auxiliary feedwater pump has been assumed to fail; 12.43
flow from the two motor-driven pumps delivers 450 gpm to the three 42-
intact steam generators (375 gpm in the case of N-1 loop operation).

Following the trip of the reactor coolant pumps, there will be a flow 12.44
coastdown until flow in the loops reaches the natural circulation
value. The natural circulation capability of the RCS has been shown 12.45
in Section 15.2.6., for the loss of ac power transient, to be
sufficient to remove core decay heat following reactor trip. Pump 12.46
coastdown characteristics are demonstrated in Sections 15.3.1 and

15.3.2 for single and multiple reactor coolant pump trips, 12.47
respectively.

A detailed description and analysis of the safety injection system is 12.48
provided in Section 6.3. The auxiliary feedwater system is described 12.49
in Section 10.4.9.

Results 12.52

Calculated plant parameters following a major feedwater line rupture 12.54
are shown on Figures 15.2-11 through 15.2-20 4 for four loops in 12.57
operation initially, and Figures 15.2-11A through 15.2-20A 4 for three
loops in operation initially. Results for the case with offsite 12.59
power available are presented on Figures 15.2-11 through 15.2-18. 17
Results for the case where offsite power is lost are presented on 12.60
Figures 15.2-19 18 through 15.2-26. 24 The calculated sequence of events 13.1
for both cases analyzed are listed in Table 15.2-1.

The system response following the feedwater line rupture is similar 13.2
for both cases analyzed. Results presented on Figures 15.2-12 and 13.3
15.2-17 16 (with offsite power available) and Figures 15.2-20 19 and
15.2-25 23 (without offsite power) show that pressures in the RCS and 13.4
main steam system remain below 110 percent of the respective design 13.5
pressures. Pressurizer pressure increases until reactor trip occurs 13.6
on low-low steam generator water level. Pressure then decreases, due 13.7
to the loss of heat input, until the safety injection system is
actuated on low steam line pressure in the ruptured loop. Coolant 13.8
expansion occurs due to reduced heat transfer capability in the steam
generators; the pressurizer safety valves open to maintain primary
pressure at an acceptable value. Addition of the safety injection 13.9
flow aids in cooling down the primary and helps to ensure that
sufficient fluid exists to keep the core covered with water.

Figures 15.2-11 and 15.2-19 18 show that following reactor trip, the 13.10
core remains subcritical except for a brief return to criticality 13.12
following a feedline break with offsite power available. This is due 13.13
to the cooldown caused by the steam generator blowdown. This 13.14 440.47
condition is terminated when boron from the safety injection system
reaches the core at approximately 212 seconds. DNBR remains above 13.16
1.30 at all times during the transients, as shown on Figures 15.2-18 17
and 15.2-26 24; the DNBR design basis is discussed in Section 4.4. 13.17
Release of radioactivity due to the steam generator blowdown is less 13.18
than that calculated for the steam line rupture, analyzed in 13.19
Section 15.1.5.
172

RCS pressure will be maintained at the safety valve setpoint until 13.20
safety injection flow is terminated by the operator, as mentioned 13.21
above. The reactor core remains covered with water throughout the 13.22
transient, as there is no water relief from the pressurizer.

The major difference between the two cases analyzed can be seen in 13.23
the plots of hot and cold leg temperatures, Figures 15.2-14 through 13.24
15.2-16 15 (with offsite power available) and Figures 15.2-22 through
15.2-24 23 (without offsite power). It is apparent from the initial 13.26

portion of the transient (300 seconds), that the case without offsite
power results in higher temperatures in the hot leg. For longer 13.28
times, however, the case with offsite power results in a more severe
rise in temperature until the coolant pumps are turned off and the 13.29
auxiliary feedwater system is realigned. The pressurizer does not 13.30
fill for either case, and the core remains covered with water.

15.2.8.3 Conclusions 13.32

Results of the analyses show that for the postulated feedwater line 13.33
rupture, the assumed auxiliary feedwater system capacity is adequate
to remove decay heat, to prevent overpressurizing the RCS, and to 13.35
prevent uncovering the reactor core. Radioactivity doses from the 13.36
postulated feedwater line rupture are less than those previously
presented for the postulated steam line break. All applicable 13.37
acceptance criteria are therefore met.

15.2.8.4 Radiological Consequences 13.39

The feedwater line break with the most significant consequences would 13.40
be one that occurred inside the containment between a steam generator 13.41
and the feedwater check valve. In this case, the contents of the 13.43
steam generator would be released to the containment. Since no fuel 13.44
failures are postulated, the radioactivity released is less than that
from the steam line break. Furthermore, automatic isolation of the 13.45
containment would further reduce any radiological consequences from
this postulated event. 13.46

15.2.9 References for Section 15.2 13.48

WCAP-7769, 1971. Mangan, M.A. Overpressure Protection for 13.50
Westinghouse Pressurized Water Reactors.

WCAP-7907, 1972. Burnett, T.W.T. et al. LOFTRAN Code Description. 13.54

WCAP-7908, 1972. Hargrove, H.G. FACTRAN-A FORTRAN-IV Code for 13.56
Thermal Transients in a UO_2 Fuel Rod.

WCAP-8330, 1974. Westinghouse Anticipated Transients Without Trip 13.58
Analysis.

TABLE 15.2-1

TIME SEQUENCE OF EVENTS FOR INCIDENTS
WHICH CAUSE A DECREASE IN HEAT
REMOVAL BY THE SECONDARY SYSTEM

Accident	Event	N-Loop Time (sec)	N-1 Loop Time (sec)
Turbine trip			
1. With pressurizer control (minimum moderator feedback)	Turbine trip, loss of main feed flow	0.0	0.0
	Initiation of steam release from steam generator safety valves	7.0	9.0
	High pressurizer pressure reactor trip point reached	7.1	11.3
	Rods begin to drop	9.1	13.3
	Minimum DNBR occurs	10.0	15.0
	Peak pressurizer pressure occurs	11.0	14.0
2. With pressurizer control (maximum moderator feedback)	Turbine trip, loss of main feed flow	0.0	0.0
	Initiation of steam release from steam generator safety valves	7.0	9.0
	Peak pressurizer pressure occurs	7.0	6.0
	Overtemperature ΔT reactor trip setpoint reached	9.9	48.8
	Rods begin to drop	11.9	50.8
	Minimum DNBR occurs	(1)	(1)

TABLE 15.2-1 (Cont)

Accident	Event	N-Loop Time (sec)	N-1 Loop Time (sec)	
3. Without pressurizer control (minimum moderator feedback)	Turbine trip, loss of main feed flow	0.0	0.0	2.12 2.13 2.14 2.15
	High pressurizer pressure reactor trip point reached	5.2	6.5	2.18 2.19 2.20
	Initiation of steam release from steam generator safety valves	7.0	9.0	2.22 2.23 2.24
	Rods begin to drop	7.2	8.5	2.26
	Peak pressurizer pressure occurs	8.0	10.0	2.28 2.29
	Minimum DNBR occurs	[1]	[1]	2.31
4. Without pressurizer control (maximum moderator feedback)	Turbine trip, loss of main feed flow	0.0	0.0	2.34 2.35 2.36 2.37
	High pressurizer pressure reactor trip point reached	5.2	6.5	2.40 2.41 2.42
	Initiation of steam release from steam generator safety valves	7.0	9.0	2.44 2.45 2.46 X
	Rods begin to drop	7.2	8.5	2.48
	Peak pressurizer pressure occurs	8.0	9.0	2.50 2.51
	Minimum DNBR occurs	[1]	[1]	2.53
Loss of normal feedwater flow	Main feedwater flow stops	10.0	10.0	2.55 2.56
	Low steam generator water level trip	65.2	67.4	2.58 2.59
	Rods begin to drop	67.2	69.4	3.1

TABLE 15.2-1 (Cont)

Accident	Event	N-Loop Time (sec)	N-1 Loop Time (sec)
	Reactor coolant pumps begin to coastdown	67.2	69.4
	Peak water level in pressurizer occurs	70.0	-
	Two steam generators begin to receive 550 gpm of auxiliary feed from one motor-driven auxiliary feedwater pump	126.0	-
	One steam generator begins to receive 484 gpm of auxiliary feedwater from one auxiliary feedwater pump	-	126.7
	Peak water level in pressurizer occurs	-	1740
	Core decay heat decreases to auxiliary feedwater heat removal capacity	~2000	~1800
Feedwater system pipe break			
1. With offsite power available	Main feedline rupture occurs	10	10
	Low-low steam generator level reactor trip setpoint reached in ruptured steam generator	16.6 15.4	17.0 16.4
	Rods begin to drop	17.4 18.6	18.4 19.0
	Auxiliary feedwater is delivered to three intact steam generators	76.6 75.4	77.0 76.4
INSERT after line 4.2	Low steam line pressure setpoint reached in ruptured steam generator	210.1 207.7	~~[illegible]~~ 77.4

TABLE 15.2-1 (Cont)

Accident	Event	N-Loop Time (sec)	N-1 Loop Time (sec)	
	Borated safety injection flow enters cold legs	~~212.1~~ 168.3	~~76.5~~ 79.9	4.1 4.2
	Insert from line 3.55			
	All main steam line isolation valves close	~~217.1~~ 214.1	~~81.5~~ 84.9	4.5 4.6
	Pressurizer safety valve setpoint reached	~~790~~ 612.1	~~357~~ 313.2	4.8 4.9
	Steam generator safety valve setpoint reached intact steam generators	~~962~~ 698	~~420~~ 402	4.12 4.13 4.14
	INSERT 2'			
2. Without offsite power	Main feedline rupture occurs	10	10	4.24 4.25
	Low-low steam generator level reactor trip setpoint reached in ruptured steam generator	~~16.6~~ 15.4	~~17.0~~ 16.4	4.29 4.30 4.31 4.32
	Rods begin to drop, power lost to the reactor coolant pumps	17.4 ~~18.6~~	18.4 ~~19.0~~	4.36 4.37 4.38
	Auxiliary feedwater is delivered to three intact steam generators	75.4 ~~76.6~~	76.4 ~~77.0~~	4.42 4.43 4.44
	Low steam line pressure setpoint reached in ruptured steam generator	76.6 ~~76.8~~	37.2 ~~38.9~~	4.48 4.49 4.50
	Borated safety injection flow enters cold legs	~~78.6~~ ~~78.0~~	39.8 ~~50.9~~	4.54 4.55
	All main steam line isolation valves close	83.6 ~~83.8~~	44.8 ~~45.6~~	4.59 4.60
	Pressurizer safety valve setpoint reached	282 ~~317~~	137 ~~160~~	5.4 5.5
	Steam generator safety valve setpoint reached in intact steam generators	358 ~~540~~	139 ~~147~~	5.9 5.10 5.11
	INSERT '2'			

INSERT '2'

	N-loop Time (sec)	N-1 loop Time (sec)
Operator decreases auxiliary feedwater flow when intact steam generators attain nominal level	1800	1800
Operator turns off safety injection pumps and begins plant cooldown following appropriate operating procedure	>1800	>1800

TABLE 15.2-1 (Cont)

NOTE: 5.15

1. DNBR does not decrease below its initial value. 5.19

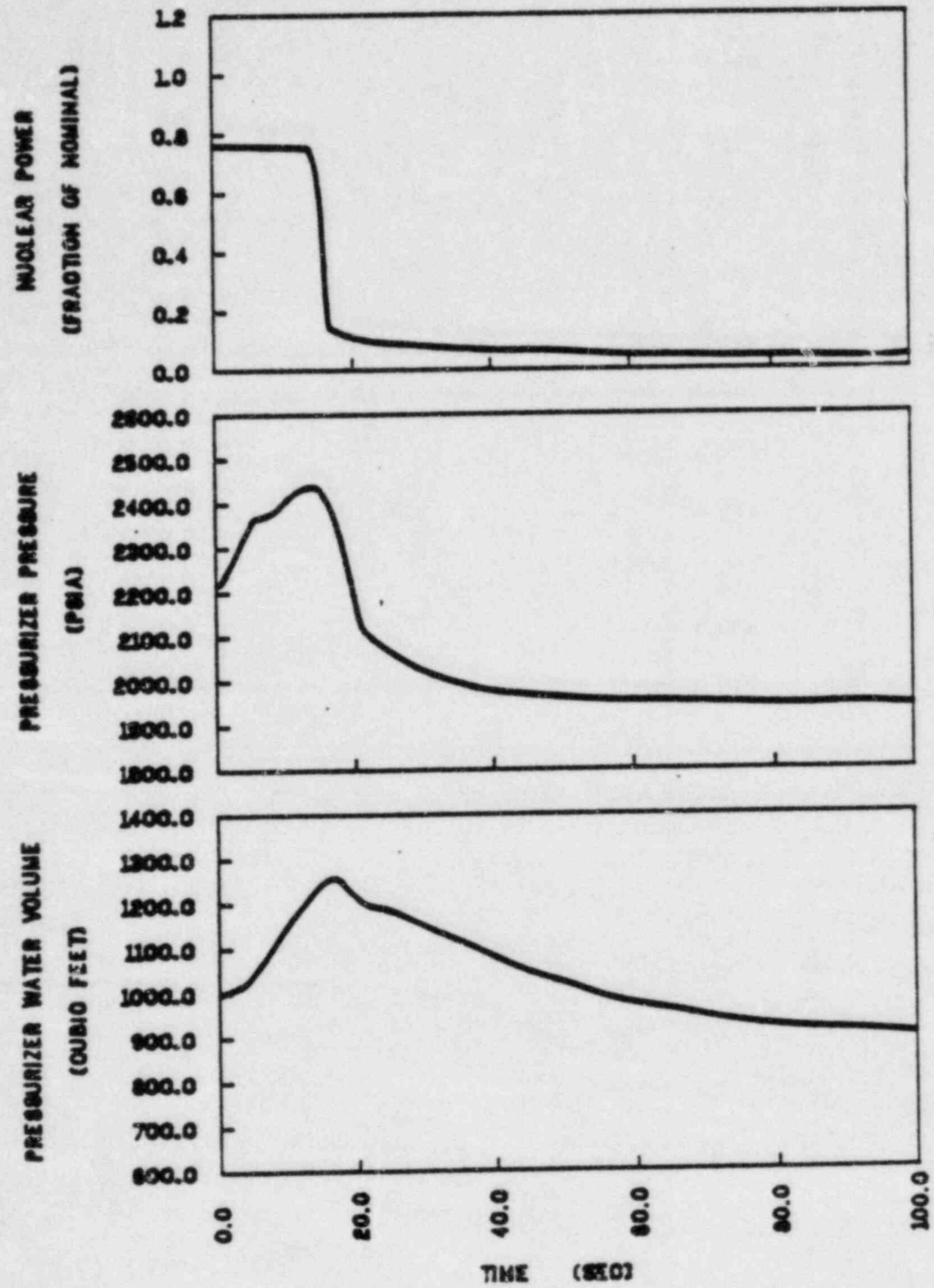


FIGURE 15.2-1A
TURBINE TRIP EVENT WITH PRESSURIZER SPRAY AND POWER OPERATED RELIEF VALVES. MINIMUM MODERATOR FEEDBACK N-1 LOOP

INTERACTIVE GRAPHICS
D.CHA.HEIF 0751AXX.XXX.01.111.08298

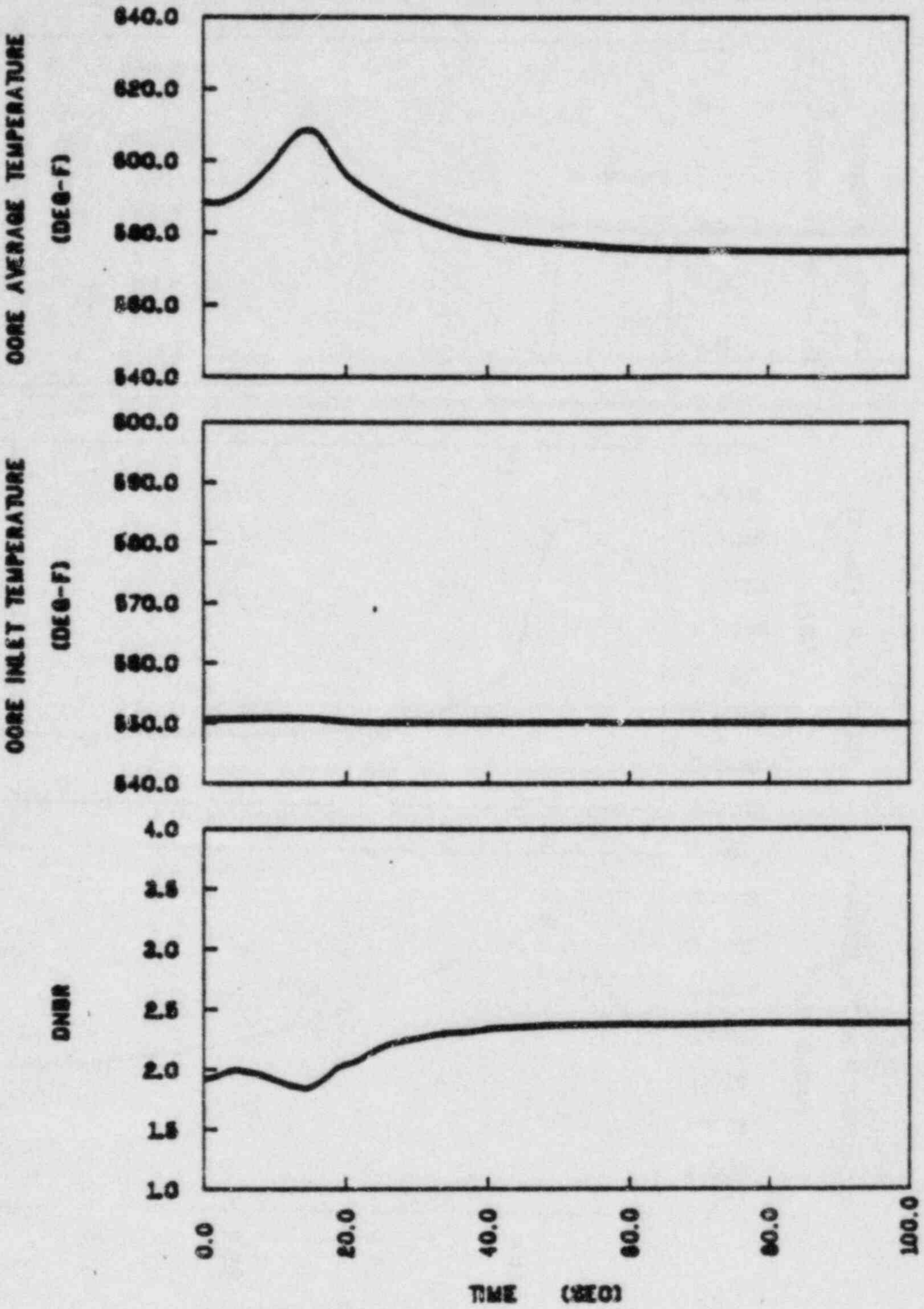


FIGURE 15.2-2A

TURBINE TRIP EVENT WITH PRESSURIZER SPRAY AND POWER OPERATED RELIEF VALVES. MINIMUM MODERATOR FEEDBACK N-1 LOOP

INTERACTIVE GRAPHICS

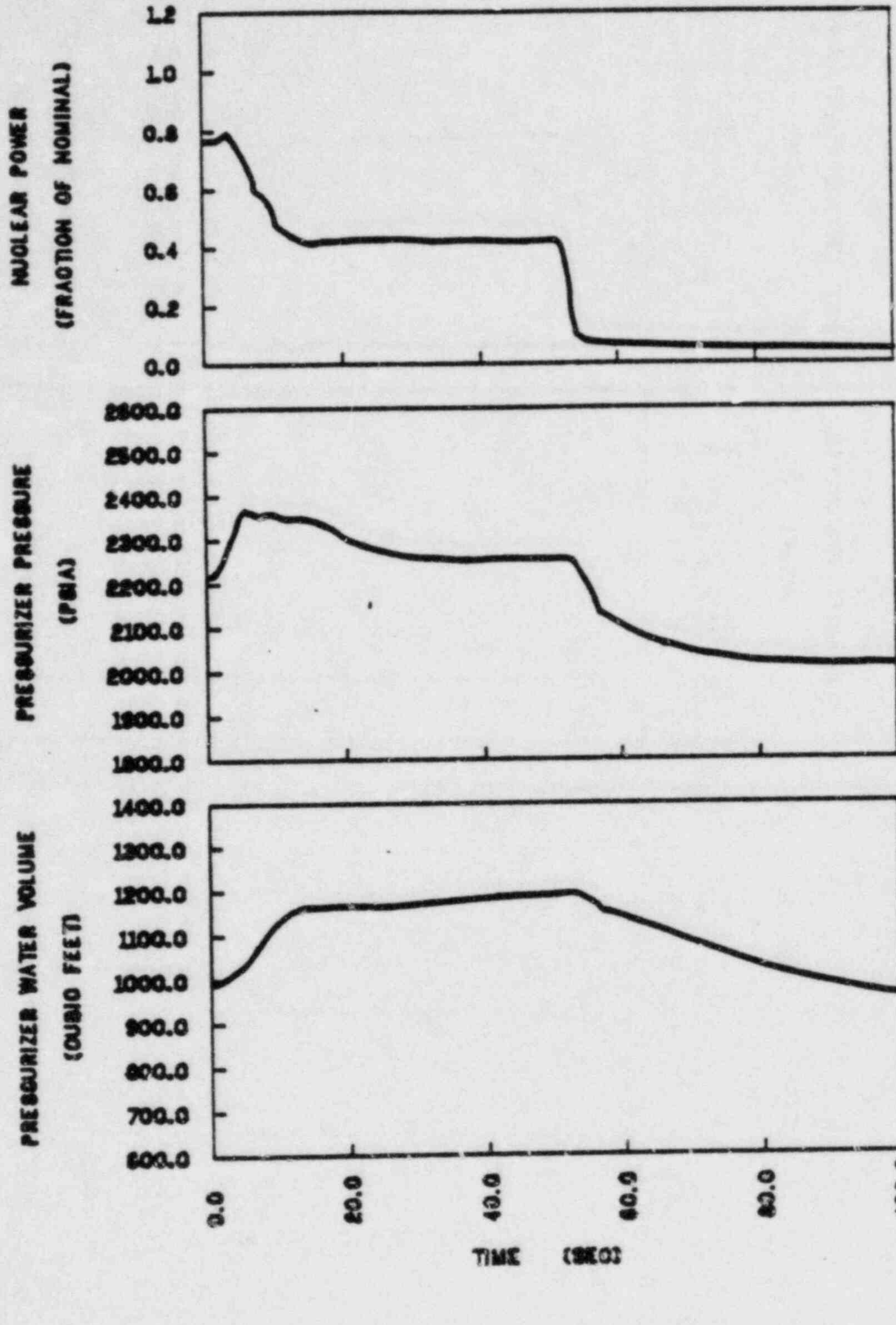


FIGURE 15.2-3A
TURBINE TRIP WITH
PRESSURIZER PRESSURE CONTROL
MAXIMUM REACTIVITY FEEDBACK
N-1 LOOP

INTERACTIVE GRAPHICS
D.CHA.HE IP.0751AXX.XXX.01.113.0829

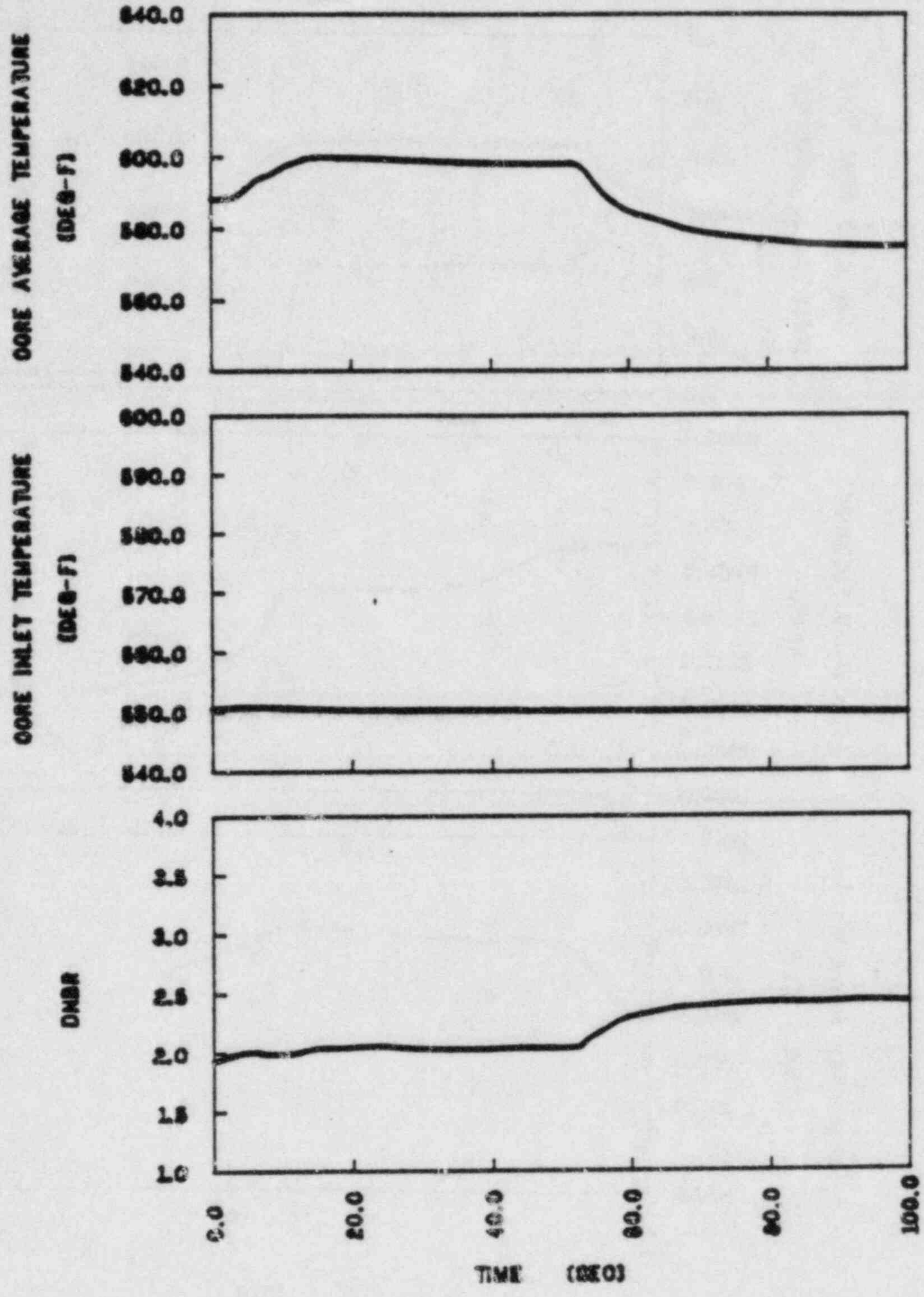


FIGURE 15.2-4A
TURBINE TRIP WITH
PRESSURIZER PRESSURE CONTROL.
MAXIMUM REACTIVITY FEEDBACK
N-1 LOOP

INTERACTIVE GRAPHICS
D.CHA.HE IP.0751AXX.XXX.01 I14.082'

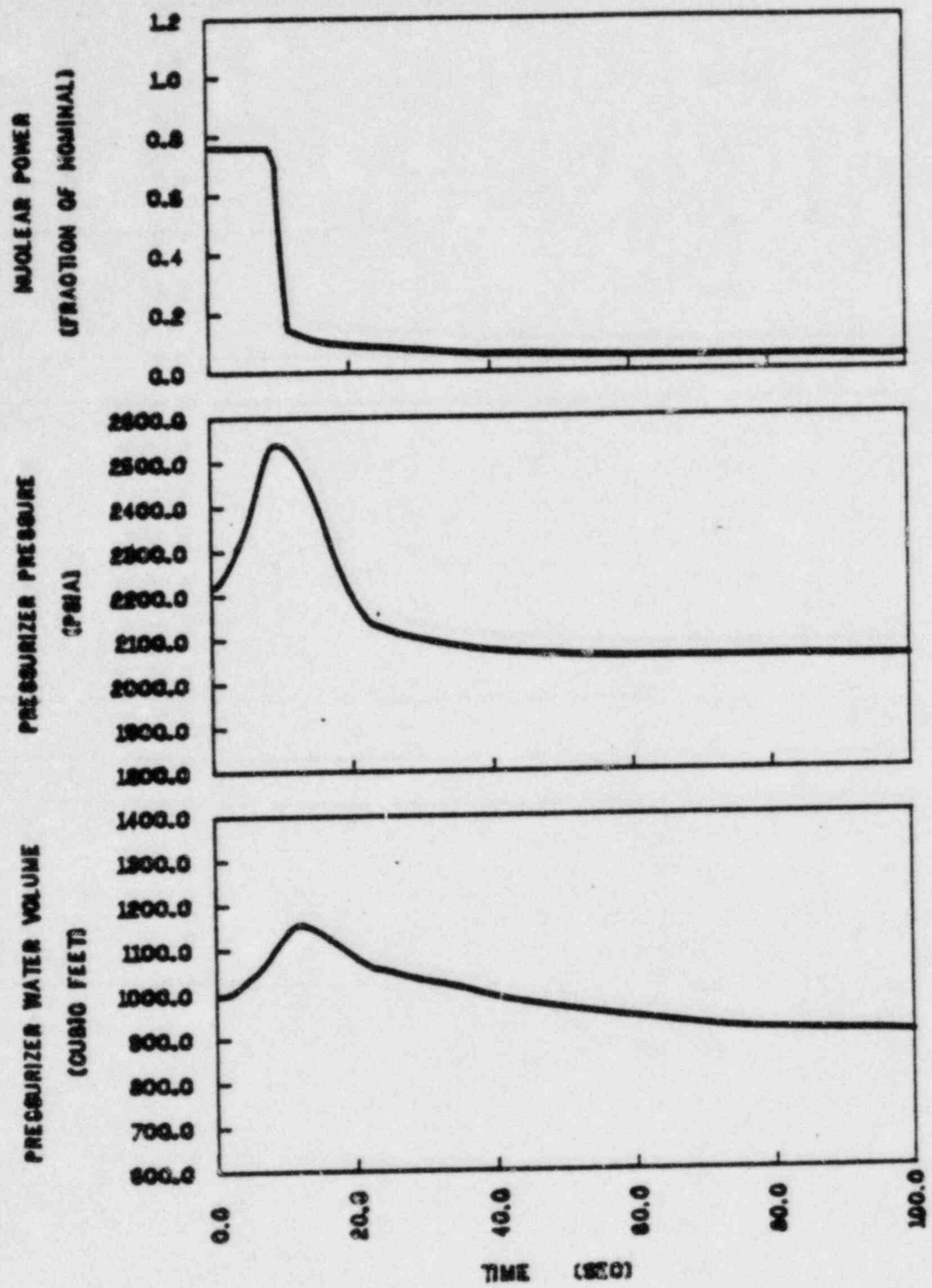


FIGURE 15.2-5A
TURBINE TRIP EVENT WITHOUT
PRESSURIZER SPRAY CONTROL
MINIMUM MODERATOR FEEDBACK
N-1 LOOP

INTERACTIVE GRAPHICS
D.CHA.HE IP.0751AXX.XXX.01.076.0825

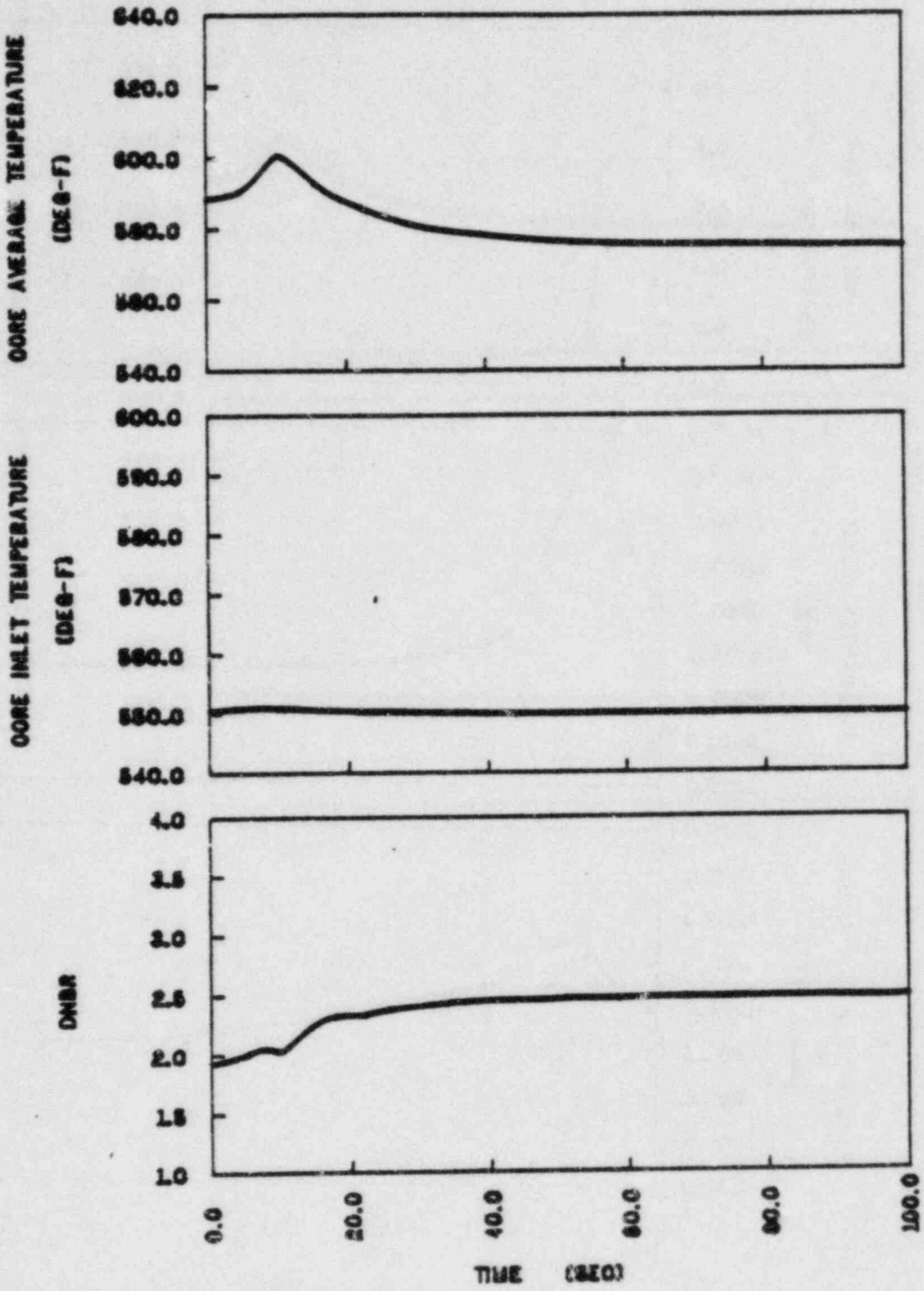


FIGURE 15.2-6A
TURBINE TRIP WITHOUT
PRESSURIZER PRESSURE CONTROL
MINIMUM REACTIVITY FEEDBACK
N-1 LOOP

INTERACTIVE GRAPHICS

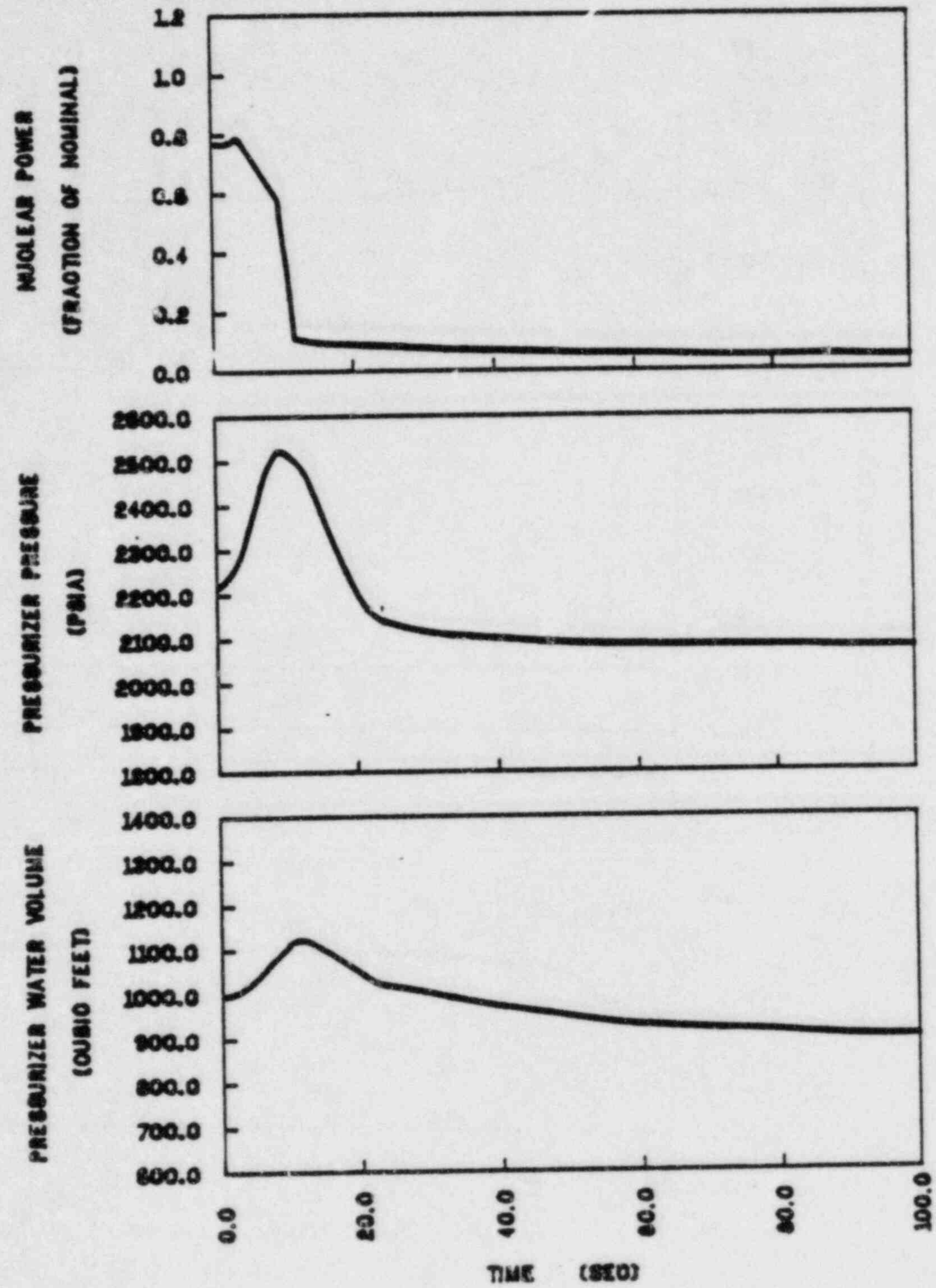


FIGURE 15.2-7A
TURBINE TRIP EVENT WITHOUT
PRESSURIZER SPRAY OR POWER
OPERATED RELIEF VALVES.
MAXIMUM MODERATOR FEEDBACK
N-1 LOOP

INTERACTIVE GRAPHICS
D.CHA.HE IP.0751AXX.XXX.01.078.0829

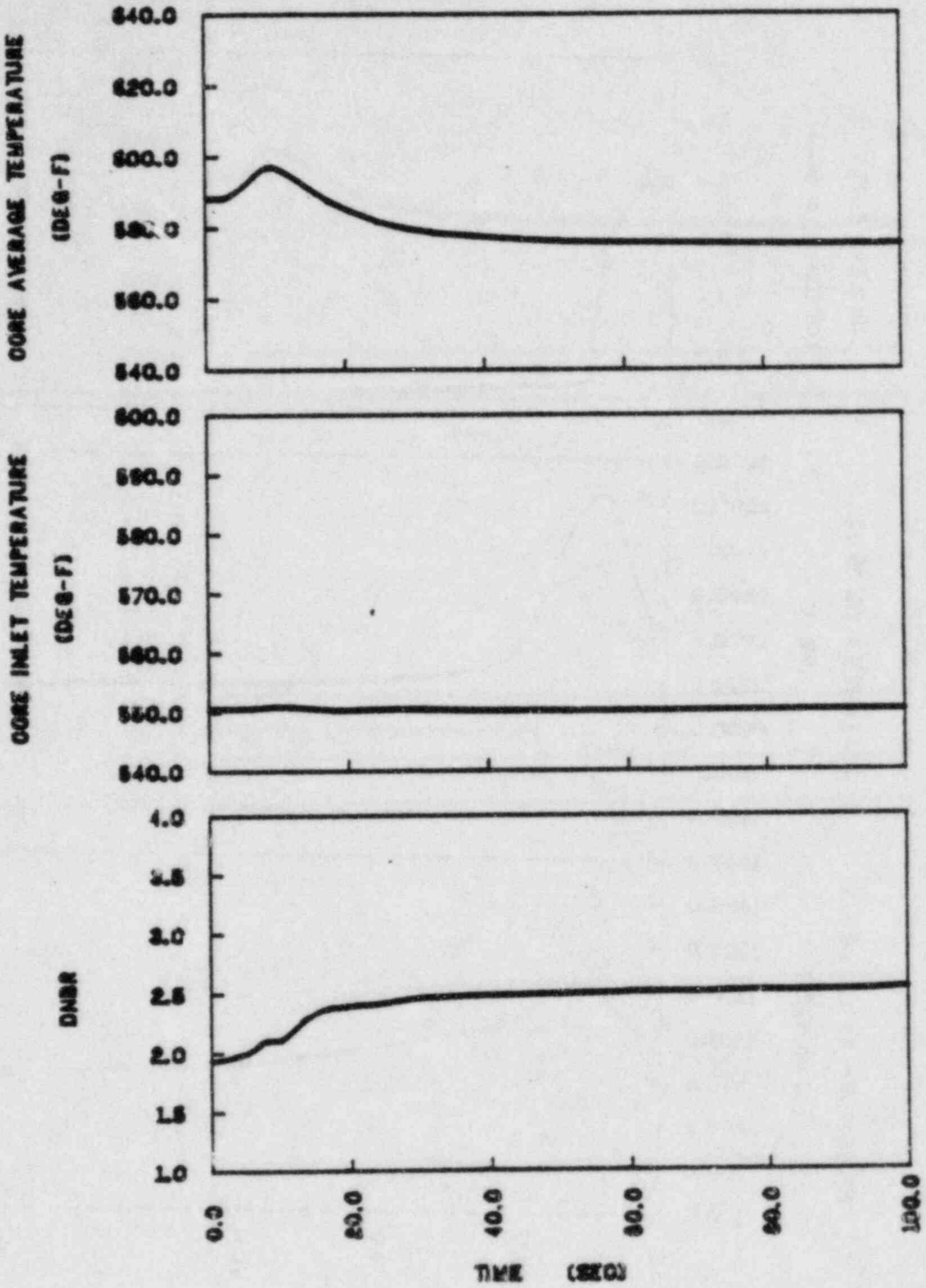


FIGURE 15.2-8A
TURBINE TRIP EVENT WITHOUT
PRESSURIZER SPRAY OR POWER
OPERATED RELIEF VALVES.
MAXIMUM MODERATOR FEEDBACK
N-1 LOOP

INTERACTIVE GRAPHICS
D.CHA.HE IP.0751AXX.XXX.01.079.0825

Pressurizer ~~pressure~~ water volume transient for loss of normal feedwater

N-1 loop operation

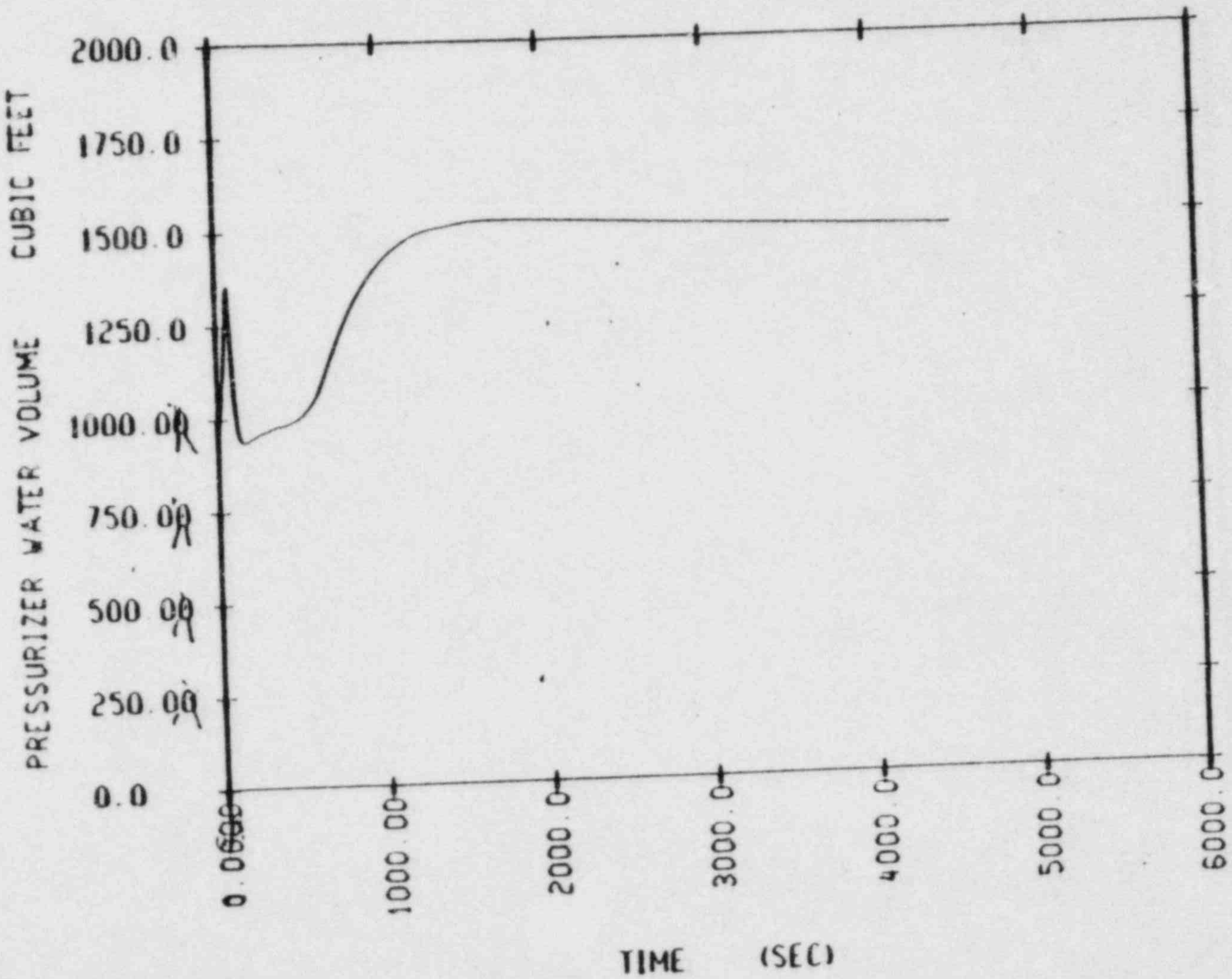
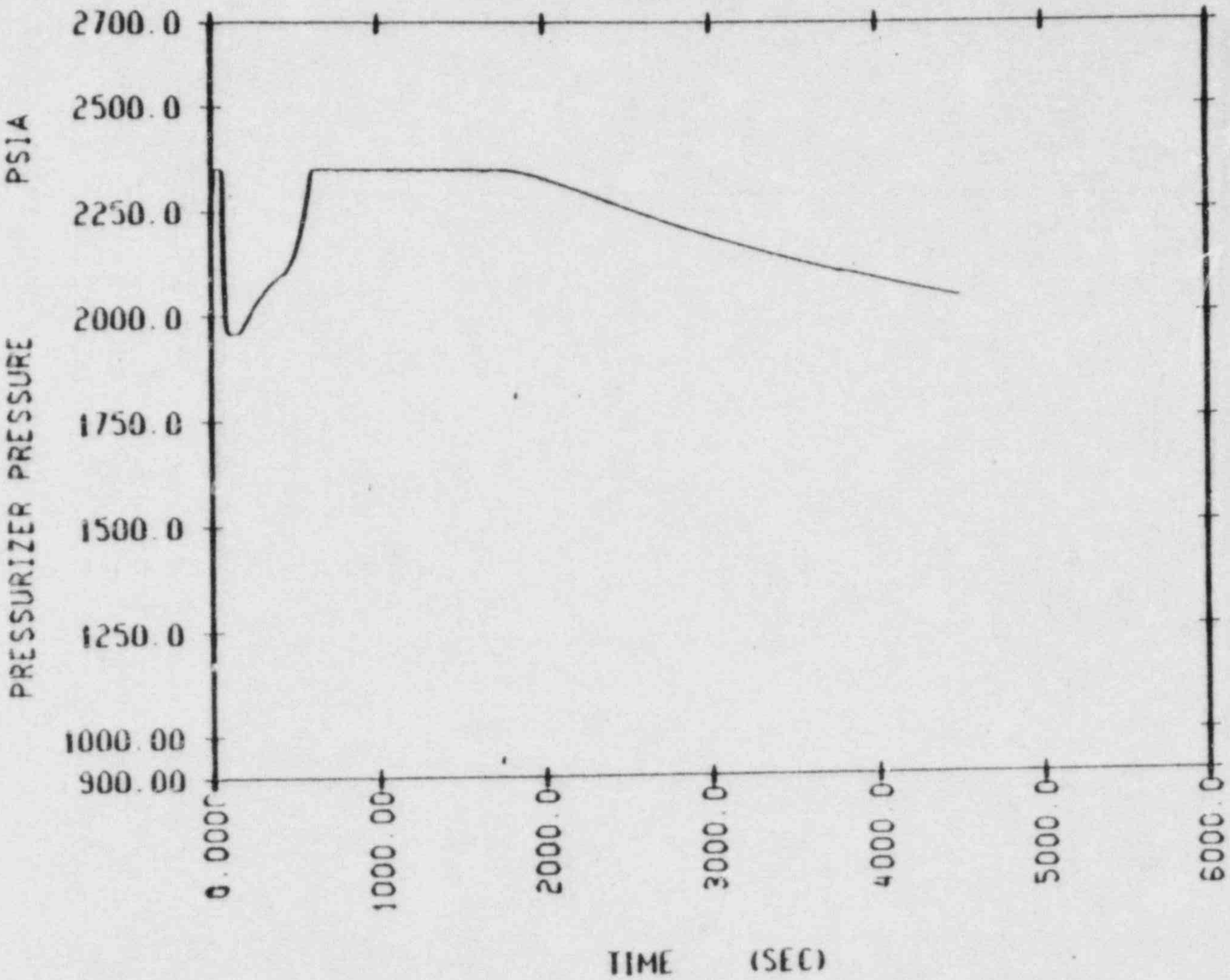


Figure 15.2-9a
Top



For Figure 15.2-9a
Bottom

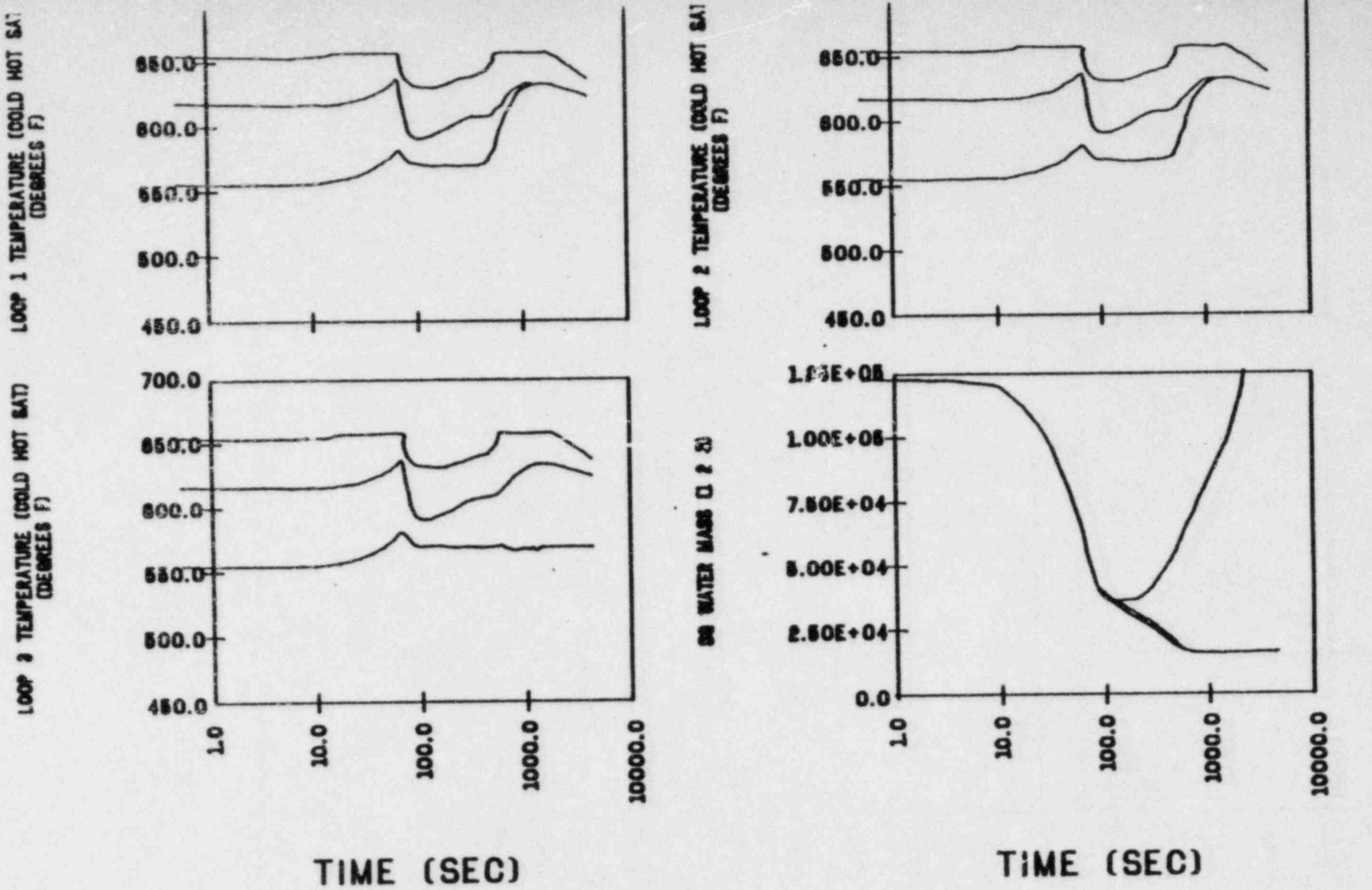


FIGURE 15.2-10A
CORE AVERAGE TEMPERATURE TRANSIENT AND STEAM GENERATOR WATER VOLUME TRANSIENT FOR LOSS OF NORMAL FEEDWA
N-1 LOOP OPERATION

ITERACTIVE GRAPHICS

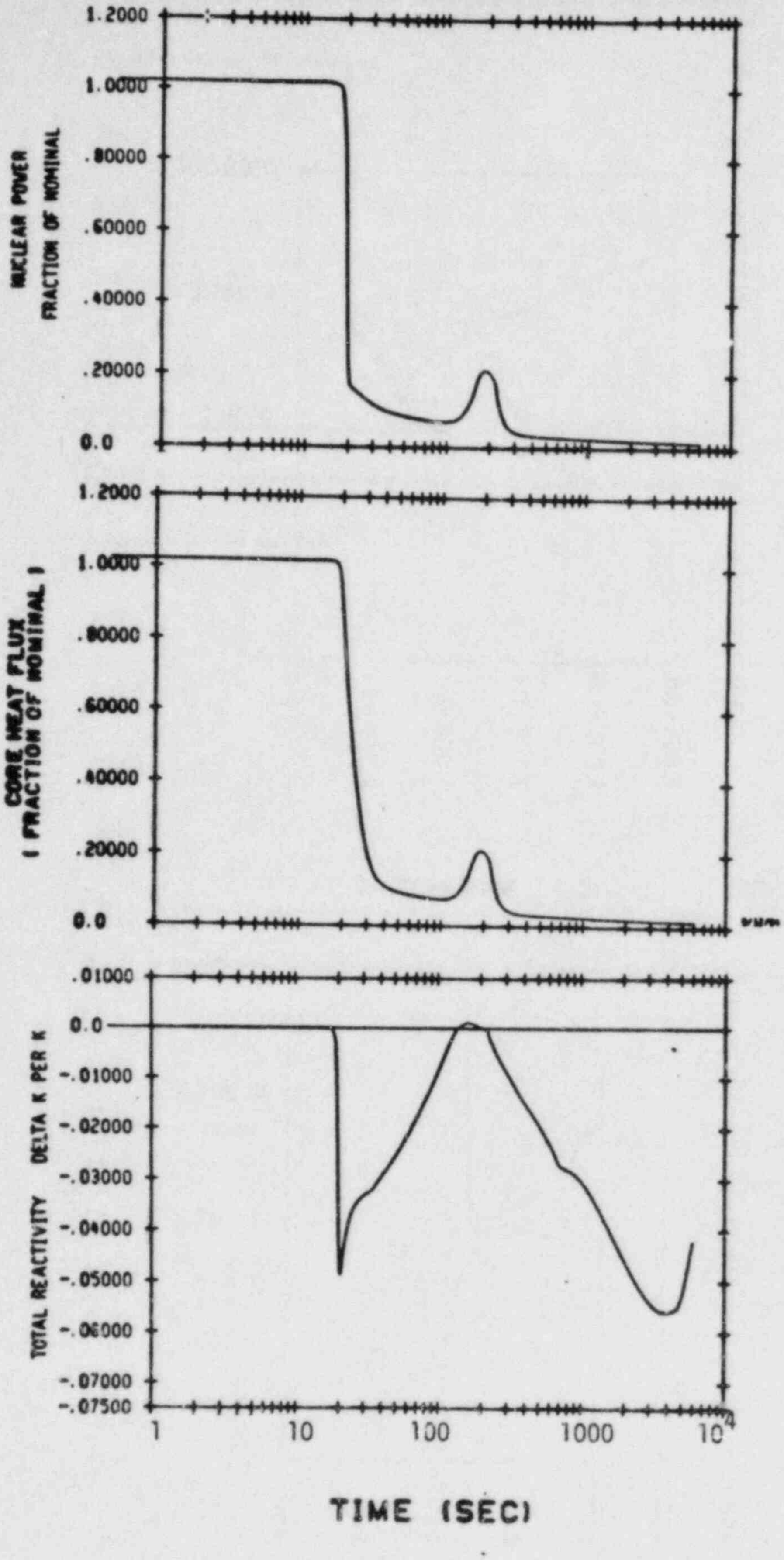


FIGURE 15.2-11
MAIN FEEDLINE RUPTURE WITH
OFFSITE POWER AVAILABLE
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

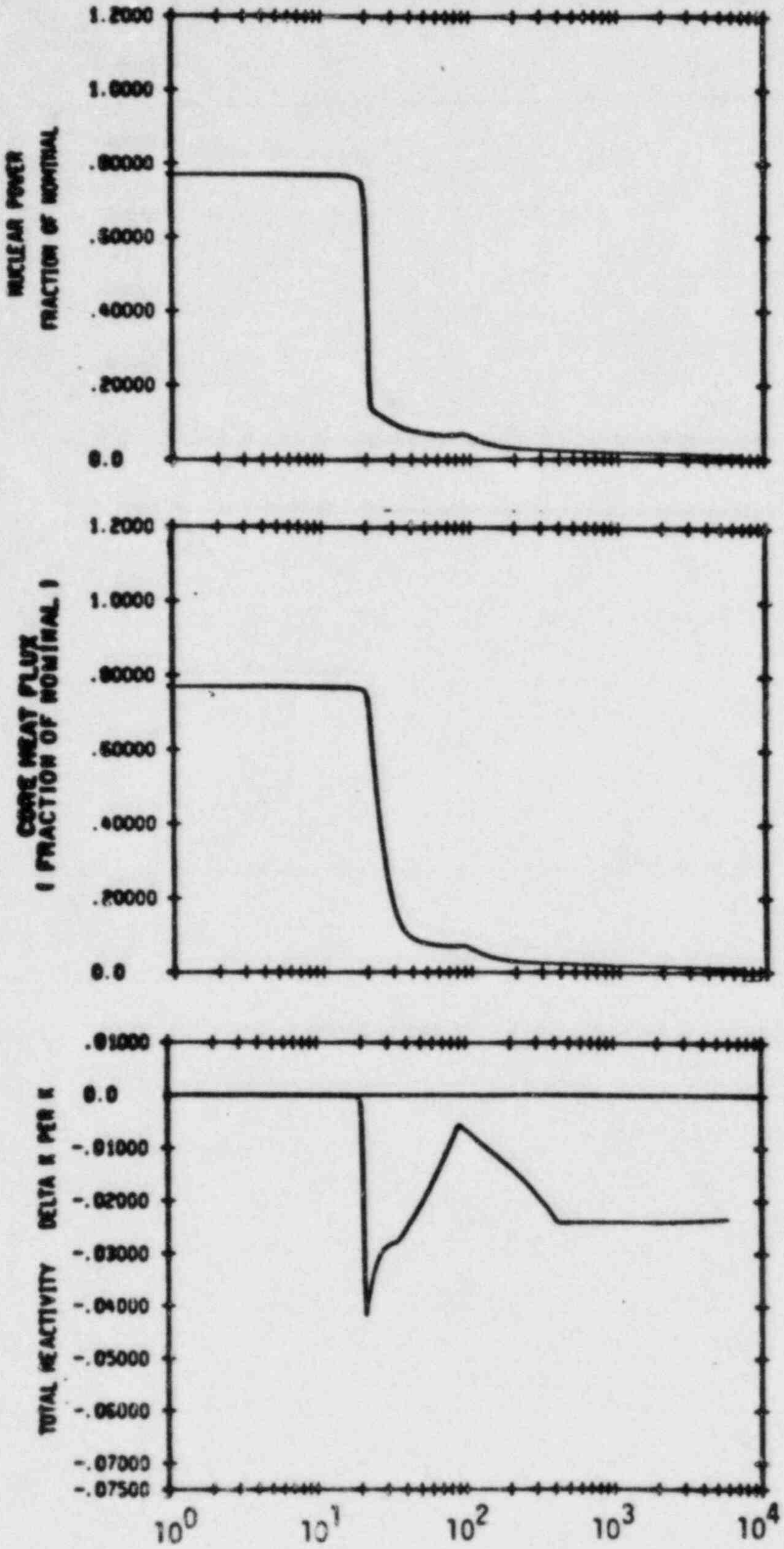


FIGURE 15.2-11A
MAIN FEEDLINE RUPTURE WITH
OFFSITE POWER AVAILABLE
MILLSTONE NUCLEAR POWER STATI
UNIT 3
FINAL SAFETY ANALYSIS REPORT
(N-1 LOOP OPERATION)

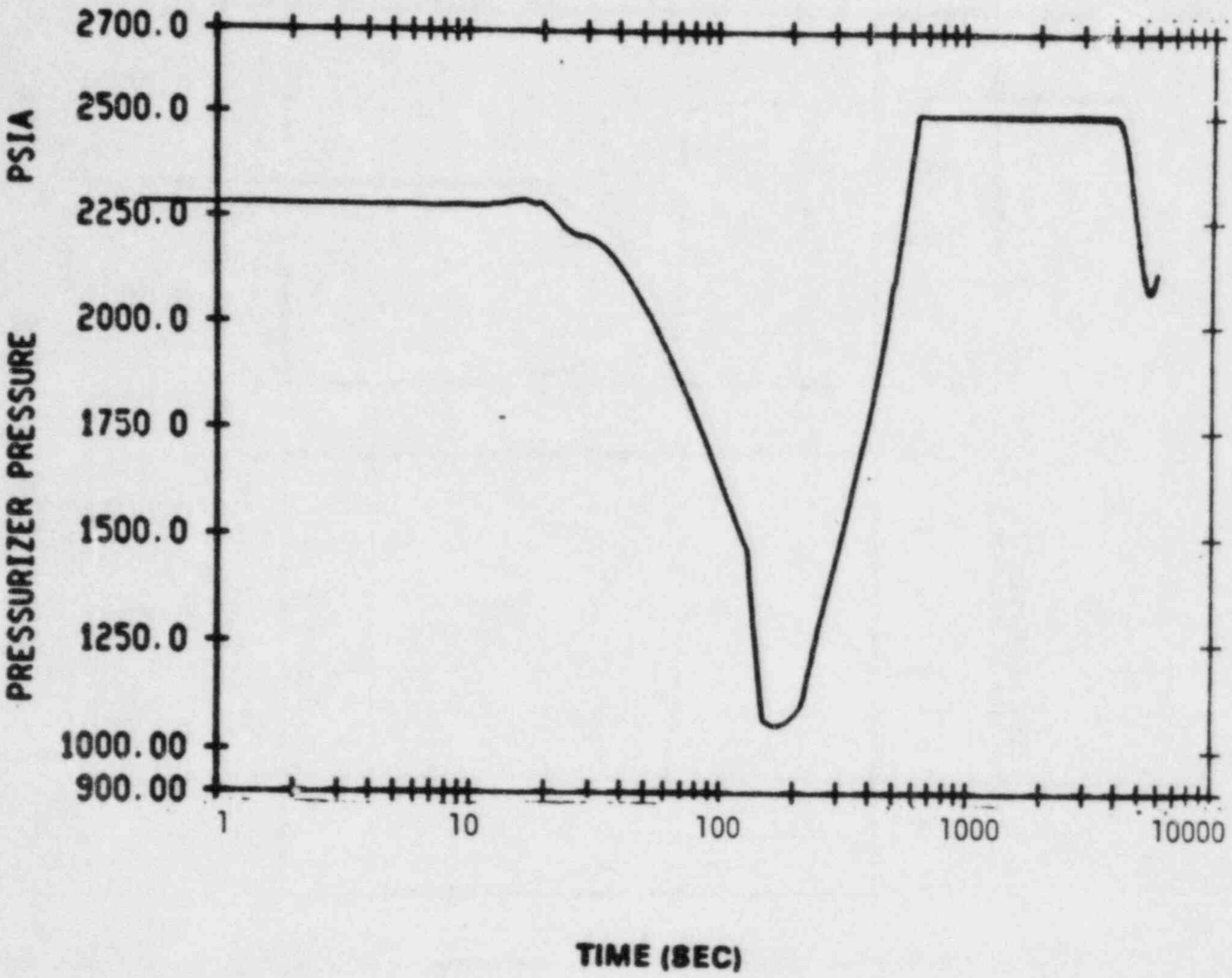


FIGURE 15.2-12
MAIN FEEDLINE RUPTURE WITH
OFFSITE POWER AVAILABLE
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

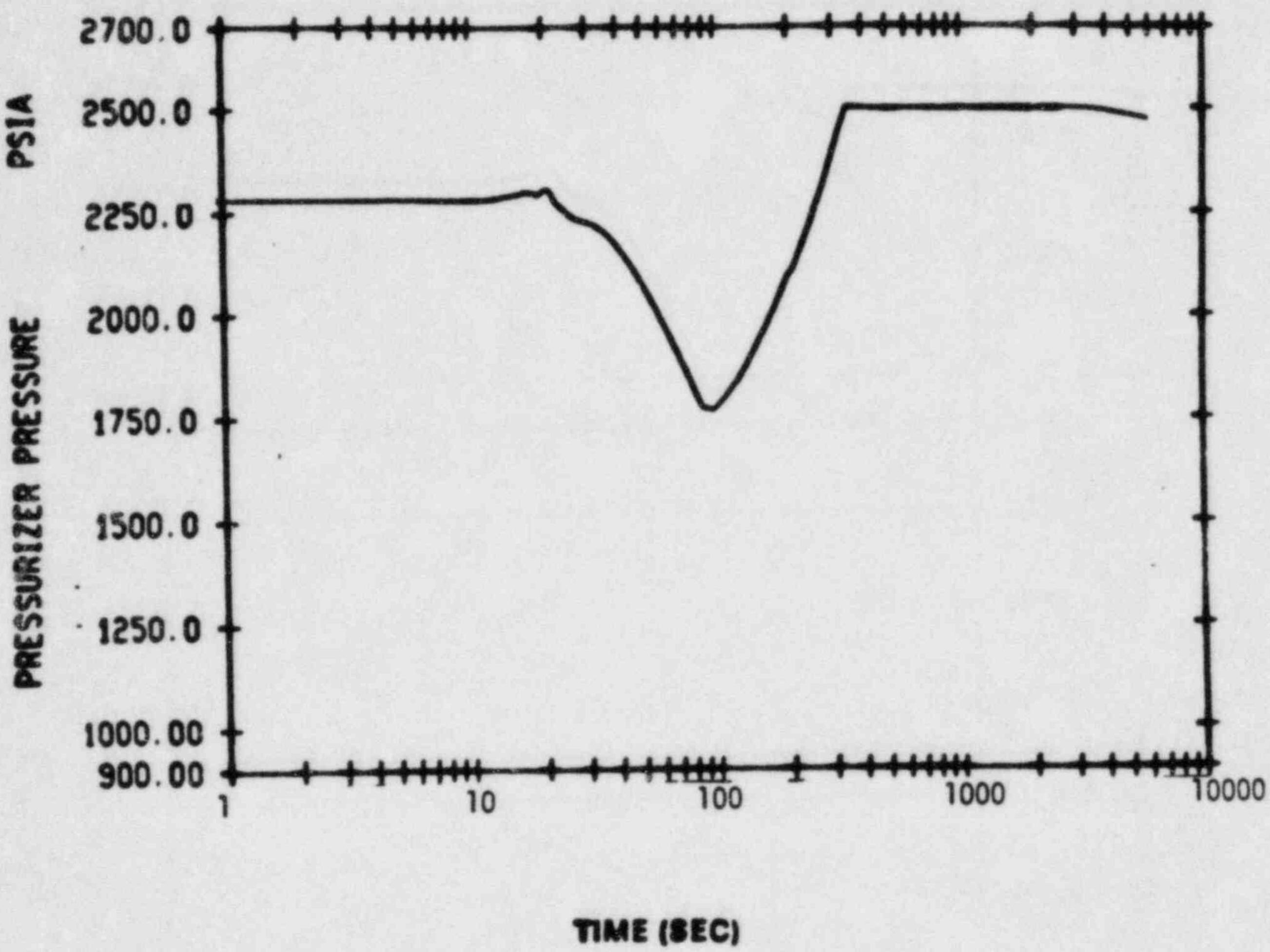


FIGURE 15.2-12A
MAIN FEEDLINE RUPTURE WITH OFFSITE POWER AVAILABLE
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT
(N-1 LOOP OPERATION)

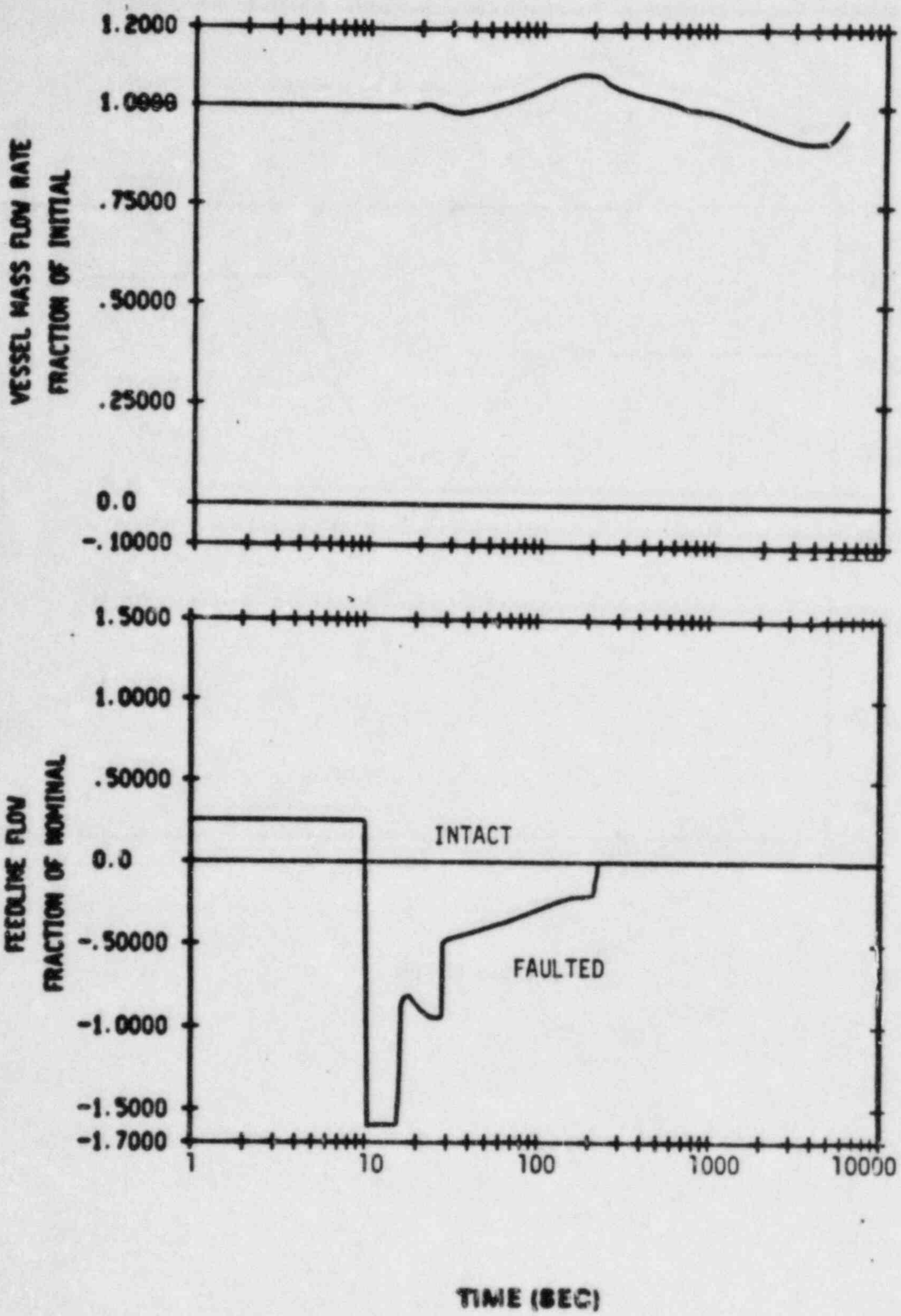


FIGURE 15.2-13
MAIN FEEDLINE RUPTURE WITH
OFFSITE POWER AVAILABLE
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

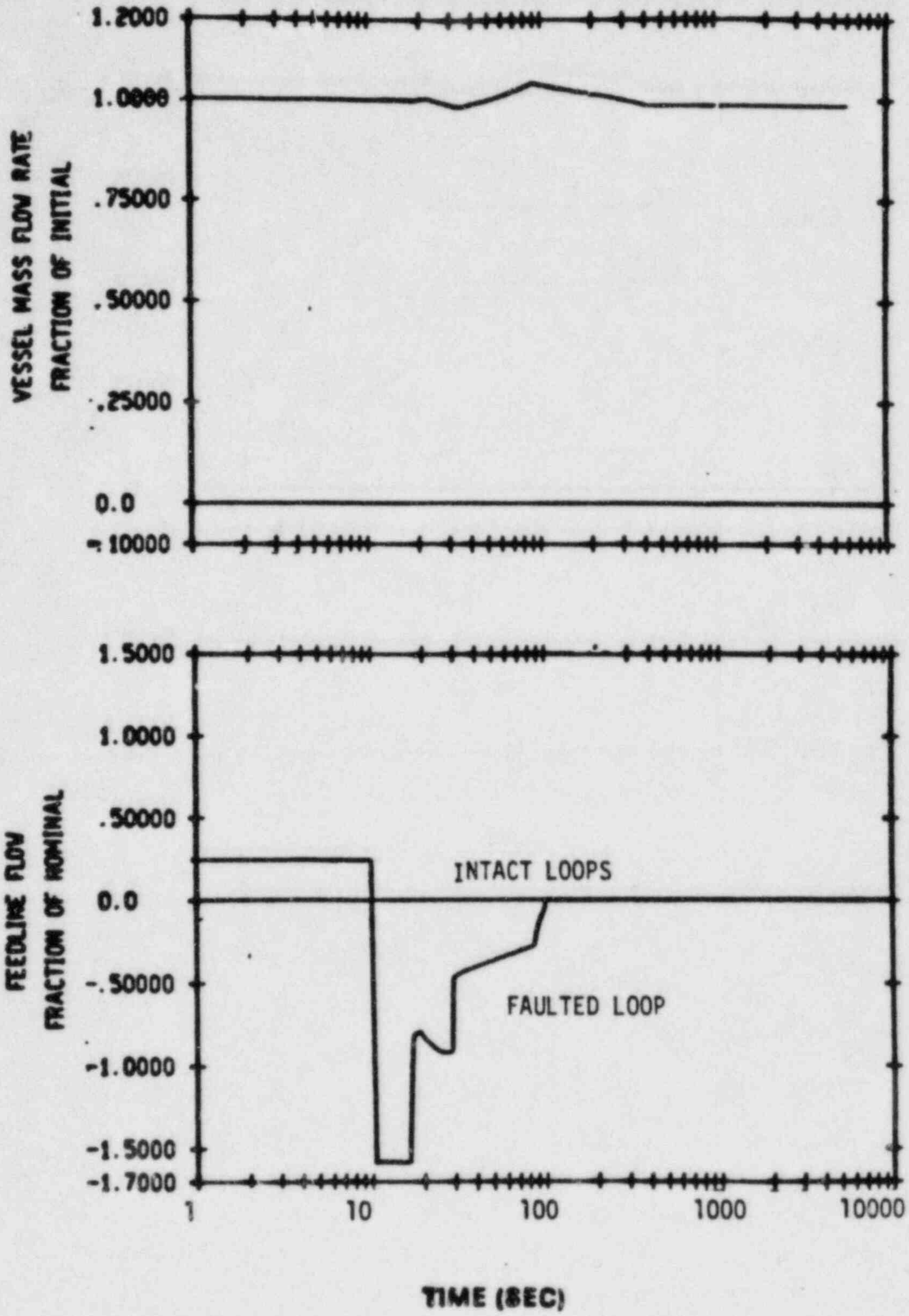


FIGURE 15.2-13A
MAIN FEEDLINE RUPTURE WITH OFFSITE POWER AVAILABLE
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT
(N-1 LOOP OPERATION)

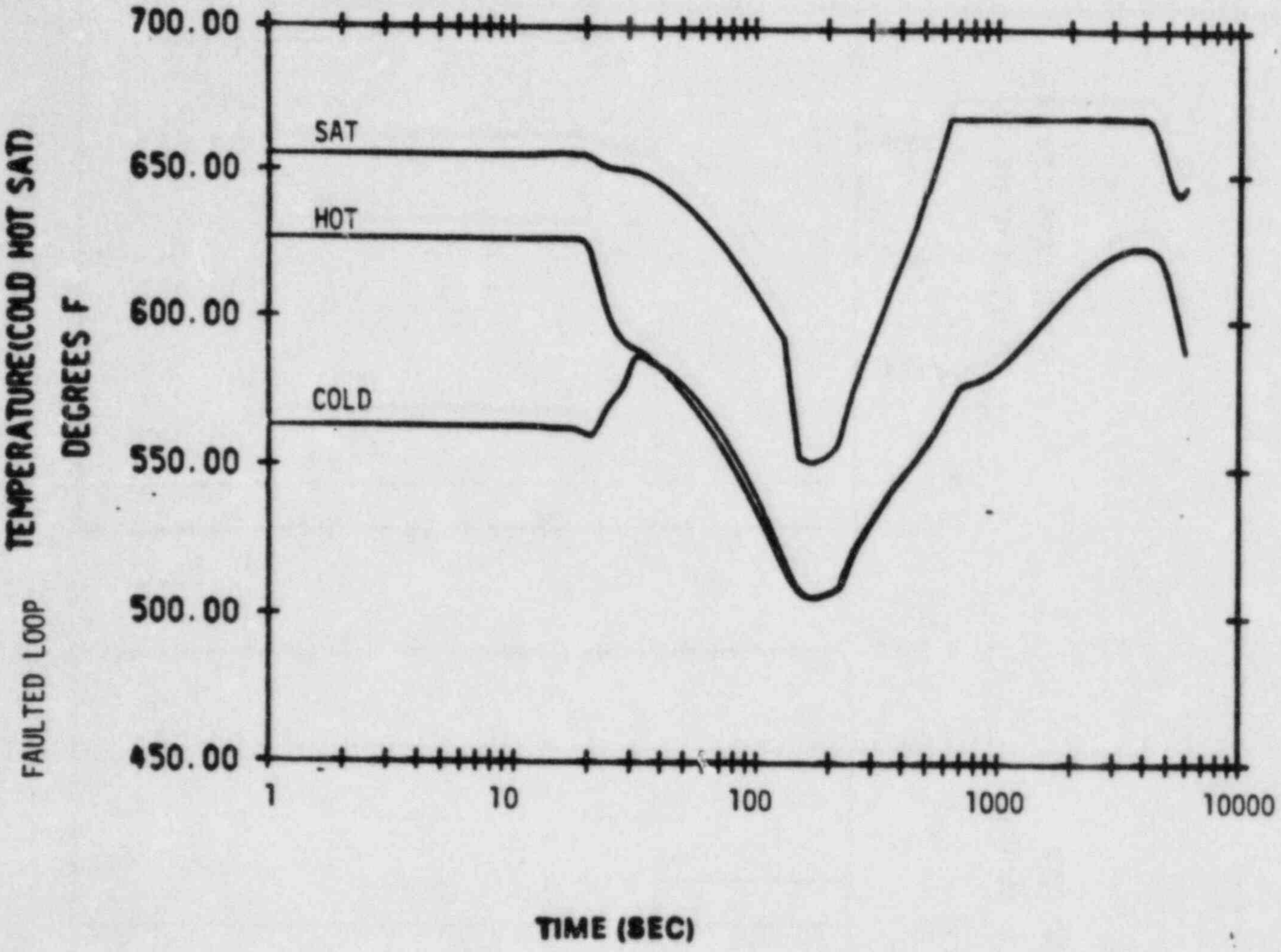


FIGURE 15.2-14
MAIN FEEDLINE RUPTURE WITH
OFFSITE POWER AVAILABLE
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

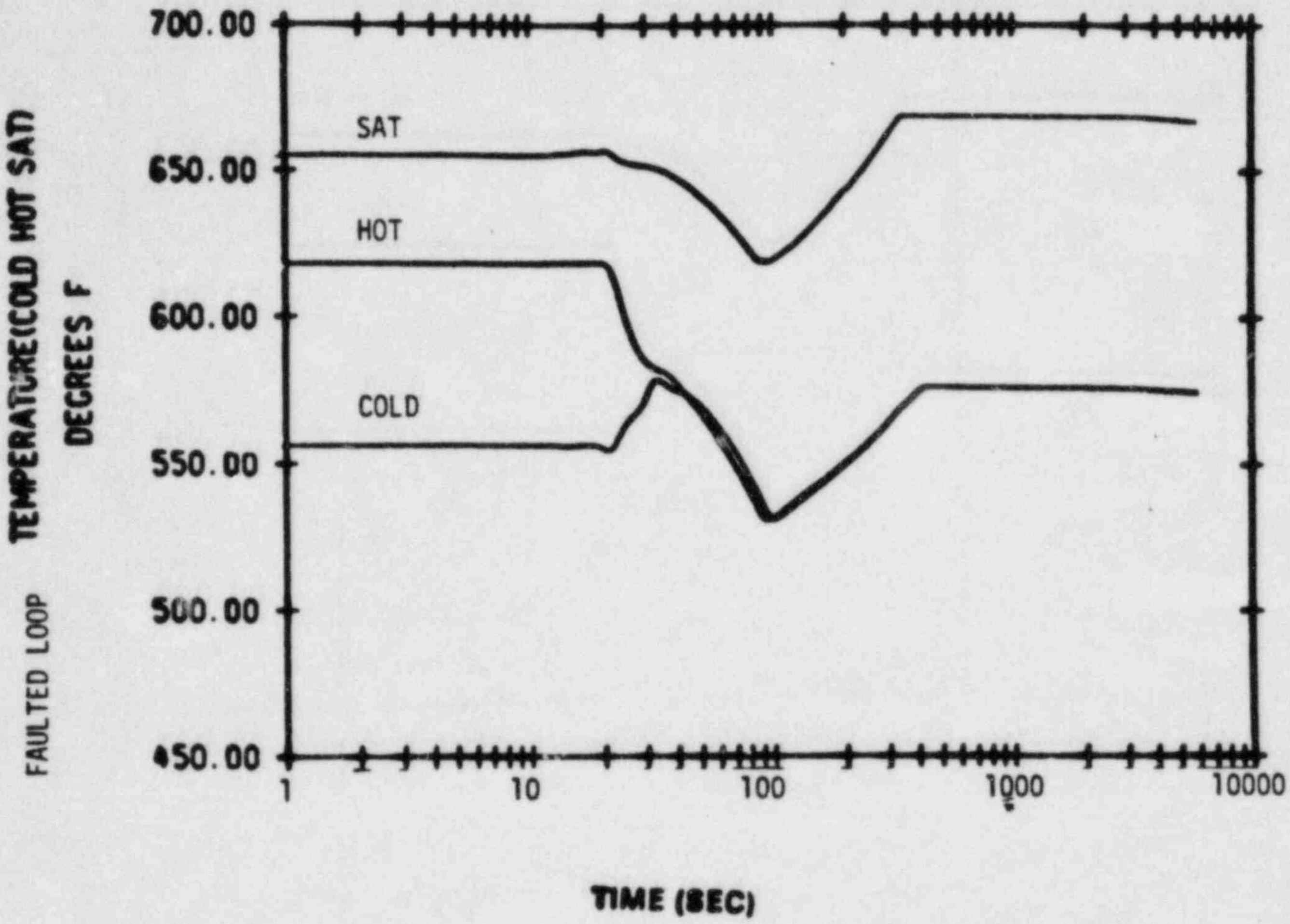


FIGURE 15.2-14A
MAIN FEEDLINE RUPTURE WITH
OFFSITE POWER AVAILABLE
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT
(N-1 LOOP OPERATION)

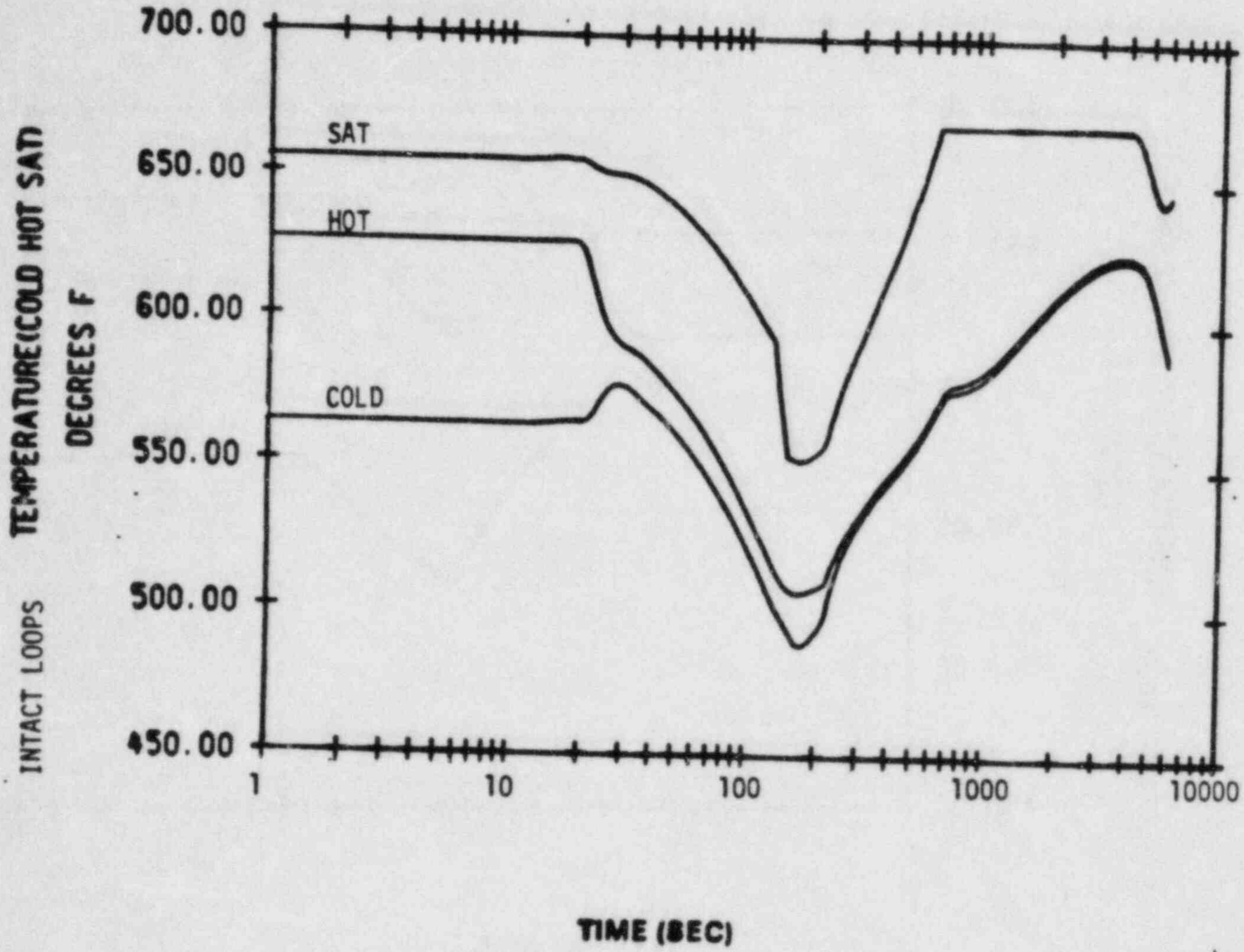


FIGURE 15.2-15
MAIN FEEDLINE RUPTURE WITH
OFFSITE POWER AVAILABLE
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

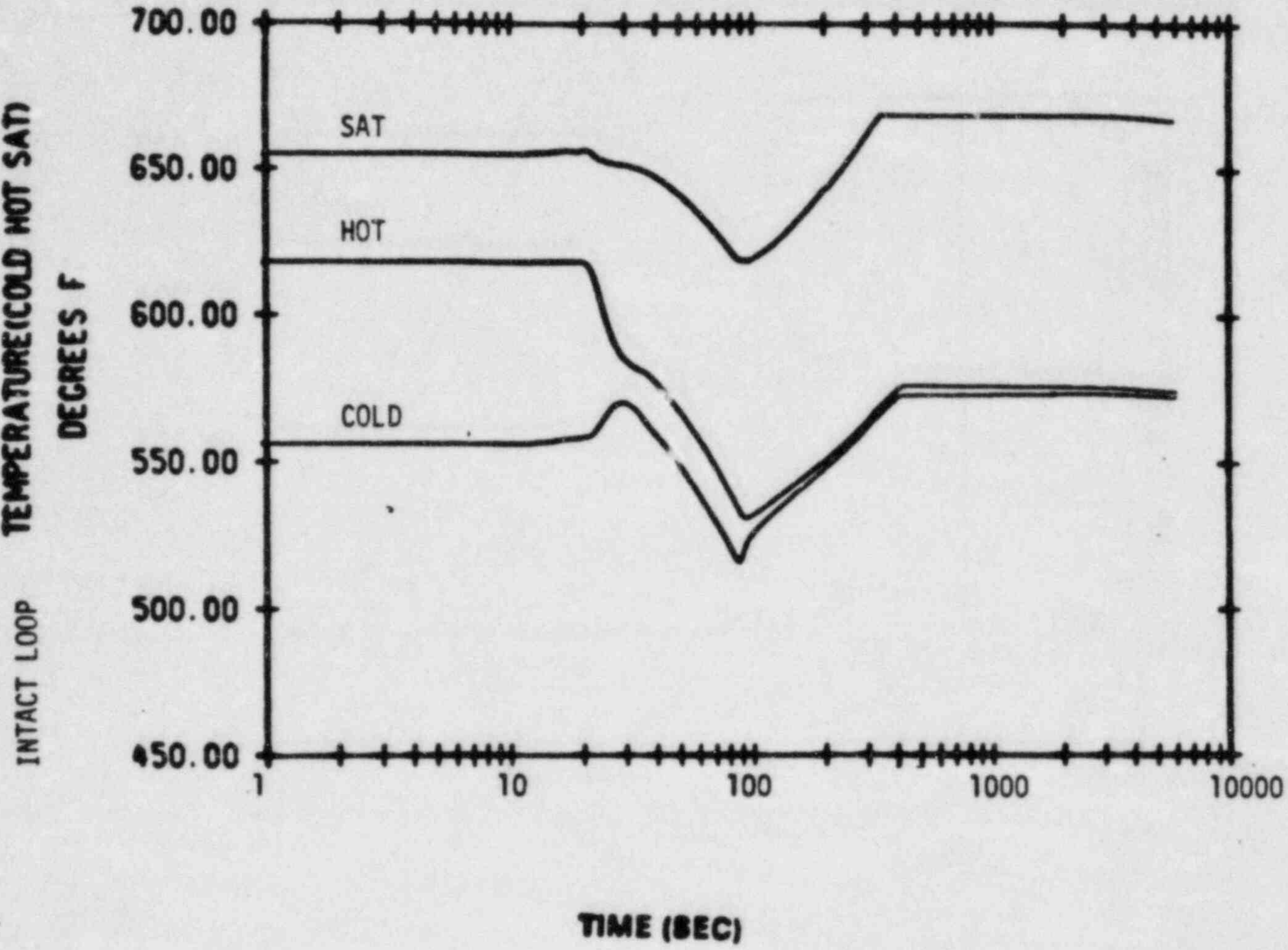


FIGURE 15.2-15A
MAIN FEEDLINE RUPTURE WITH
OFFSITE POWER AVAILABLE
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT
(N-1 LOOP OPERATION)

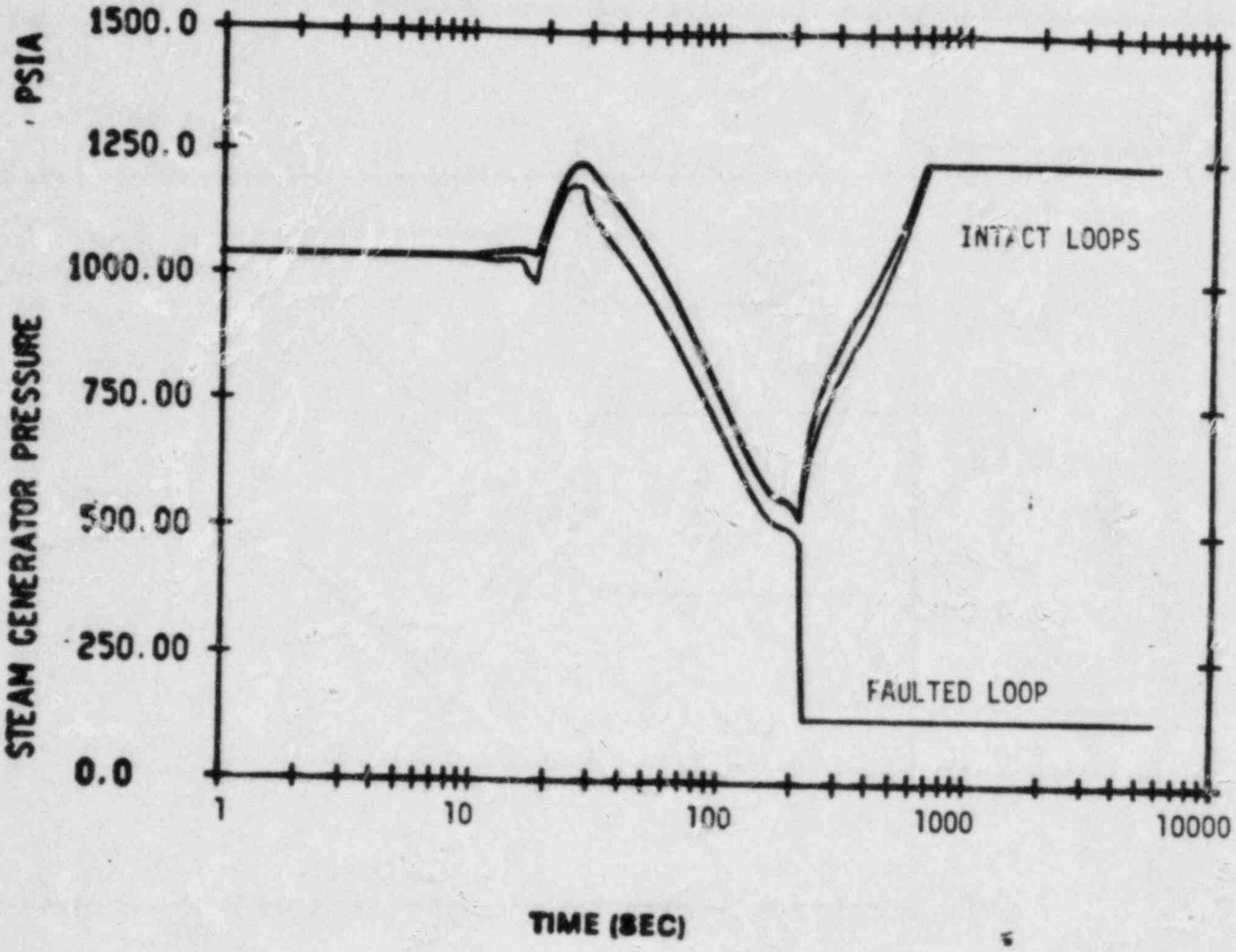


FIGURE 15.2 - 16
MAIN FEEDLINE RUPTURE WITH OFFSITE POWER AVAILABLE
MILLSTONE NUCLEAR POWER STATION UNIT 3
FINAL SAFETY ANALYSIS REPORT

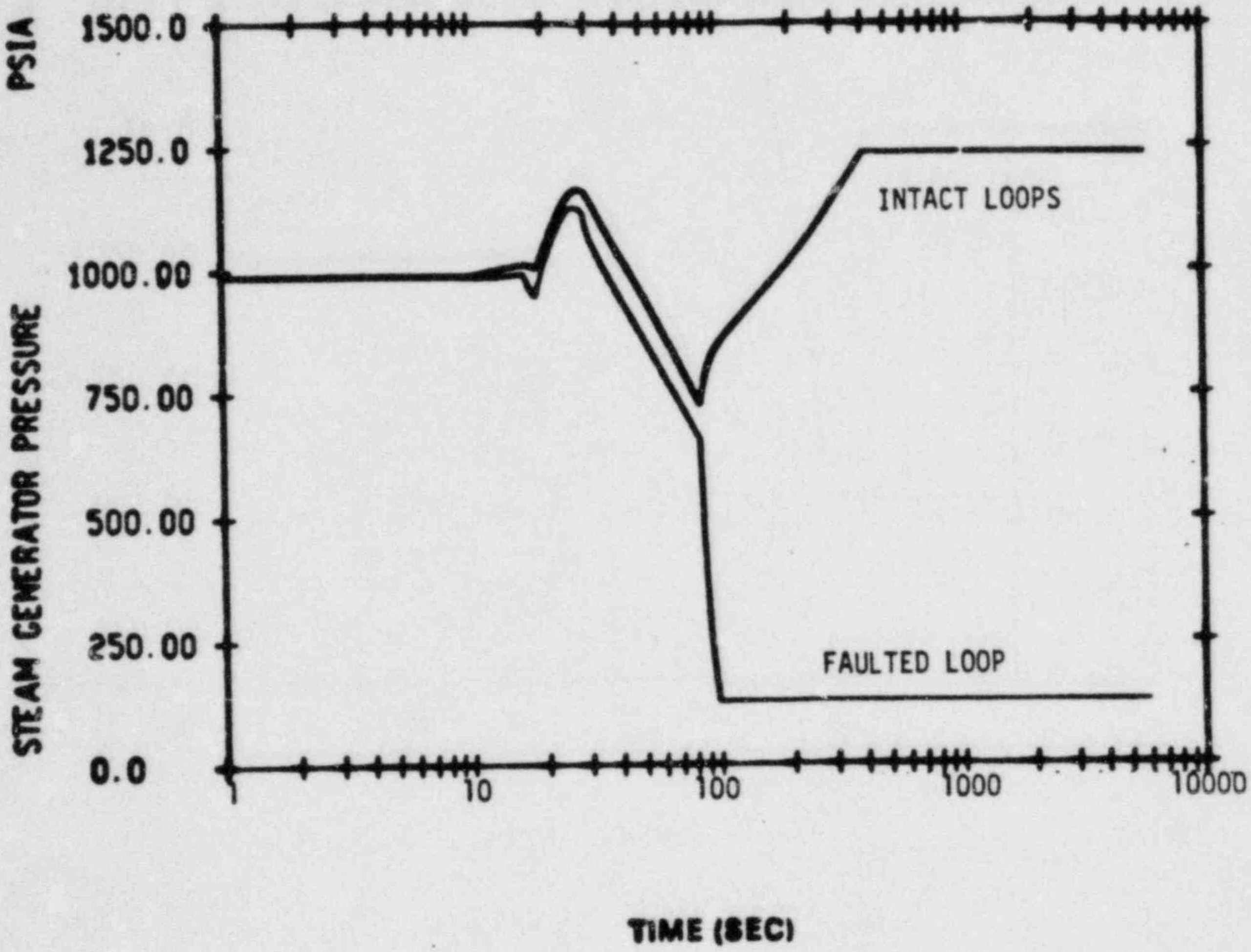


FIGURE 15.2-16A
MAIN FEEDLINE RUPTURE WITH OFFSITE POWER AVAILABLE
MILLSTONE NUCLEAR POWER STATIO
UNIT 3
FINAL SAFETY ANALYSIS REPORT
(N-1 LOOP OPERATION)

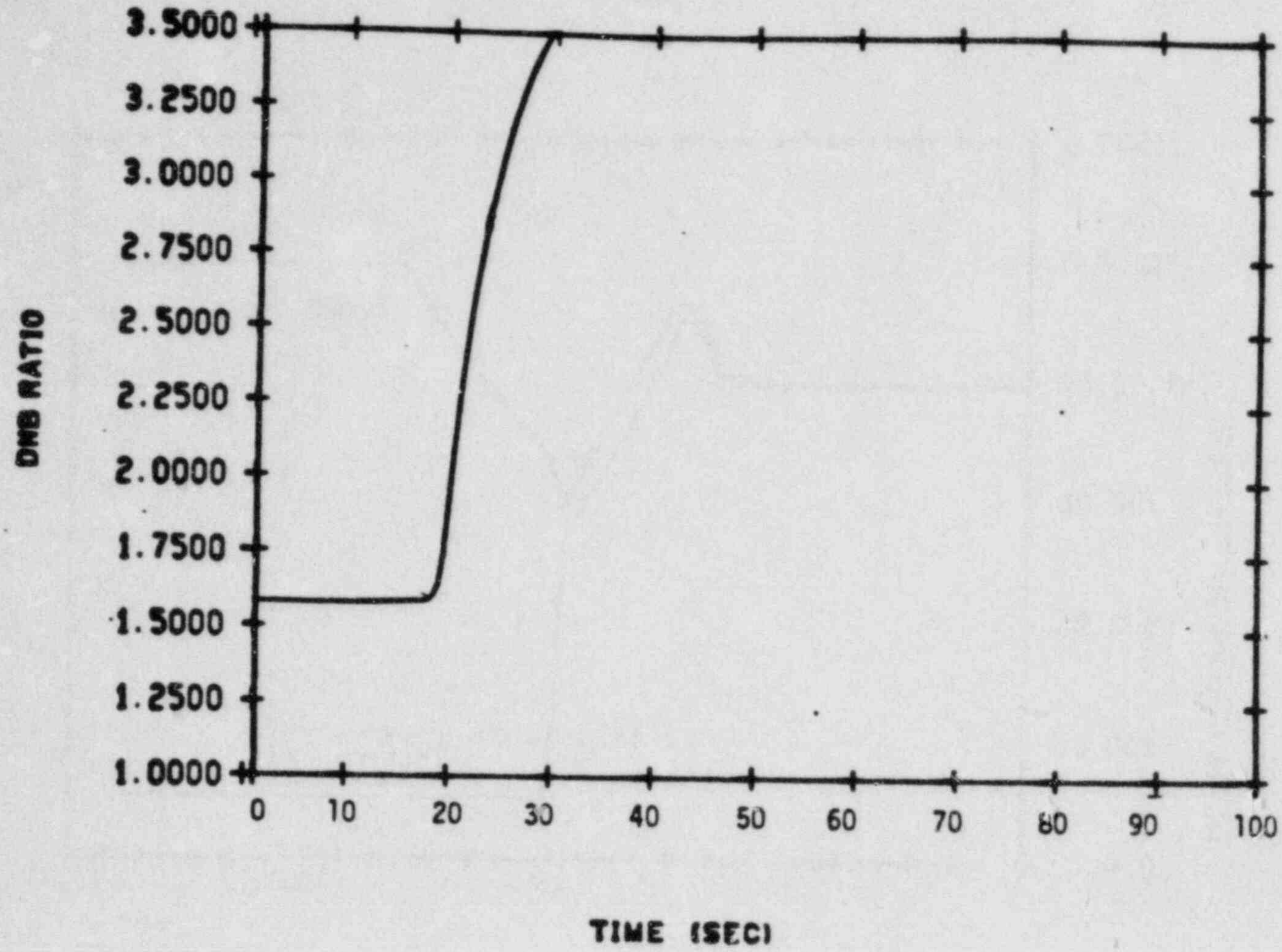


FIGURE 15.2-17
MAIN FEEDLINE RUPTURE WITH
OFFSITE POWER AVAILABLE
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

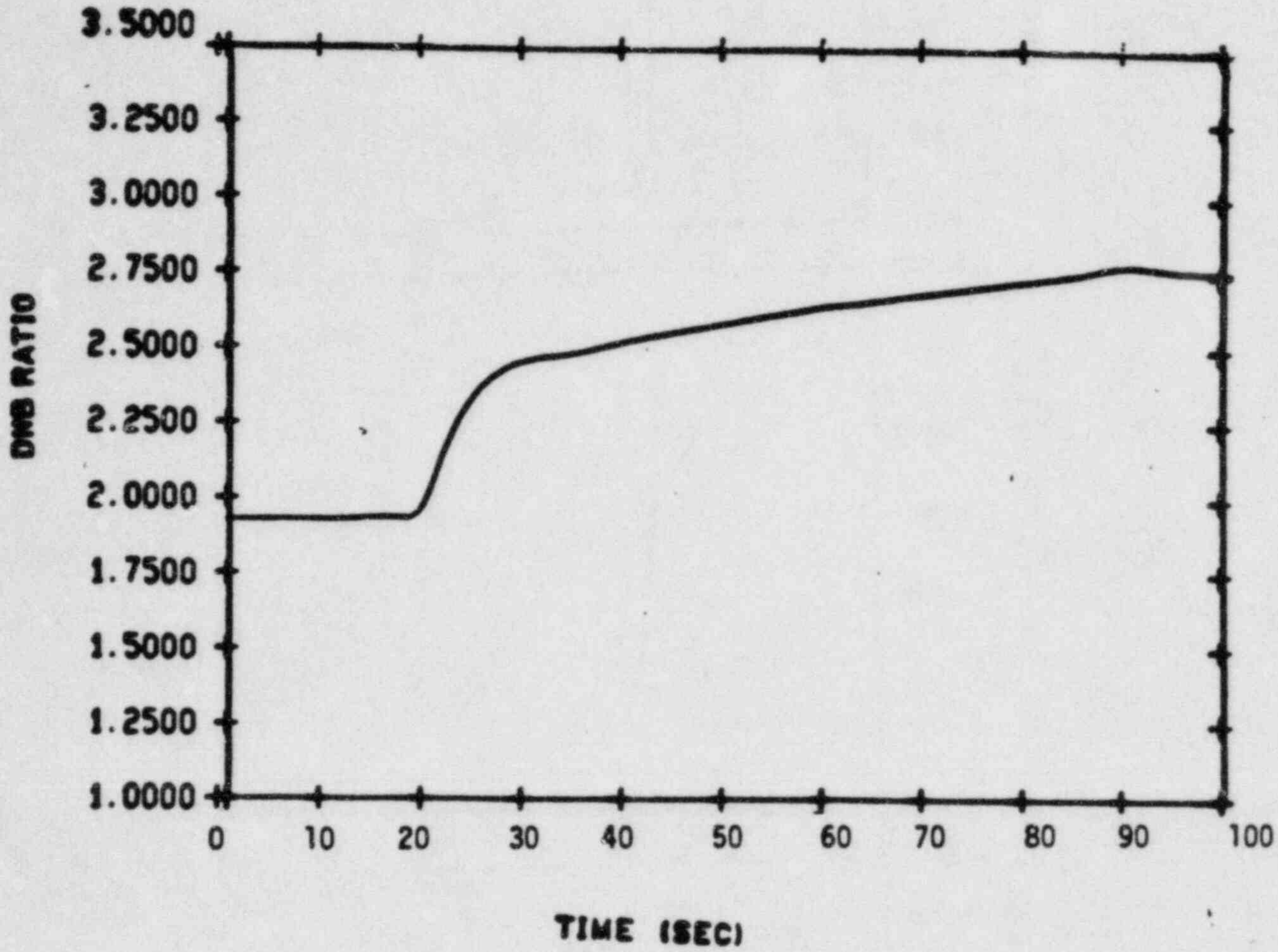


FIGURE 15.2-17A
MAIN FEEDLINE RUPTURE WITH
OFFSITE POWER AVAILABLE
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT
(N-1 LOOP OPERATION)

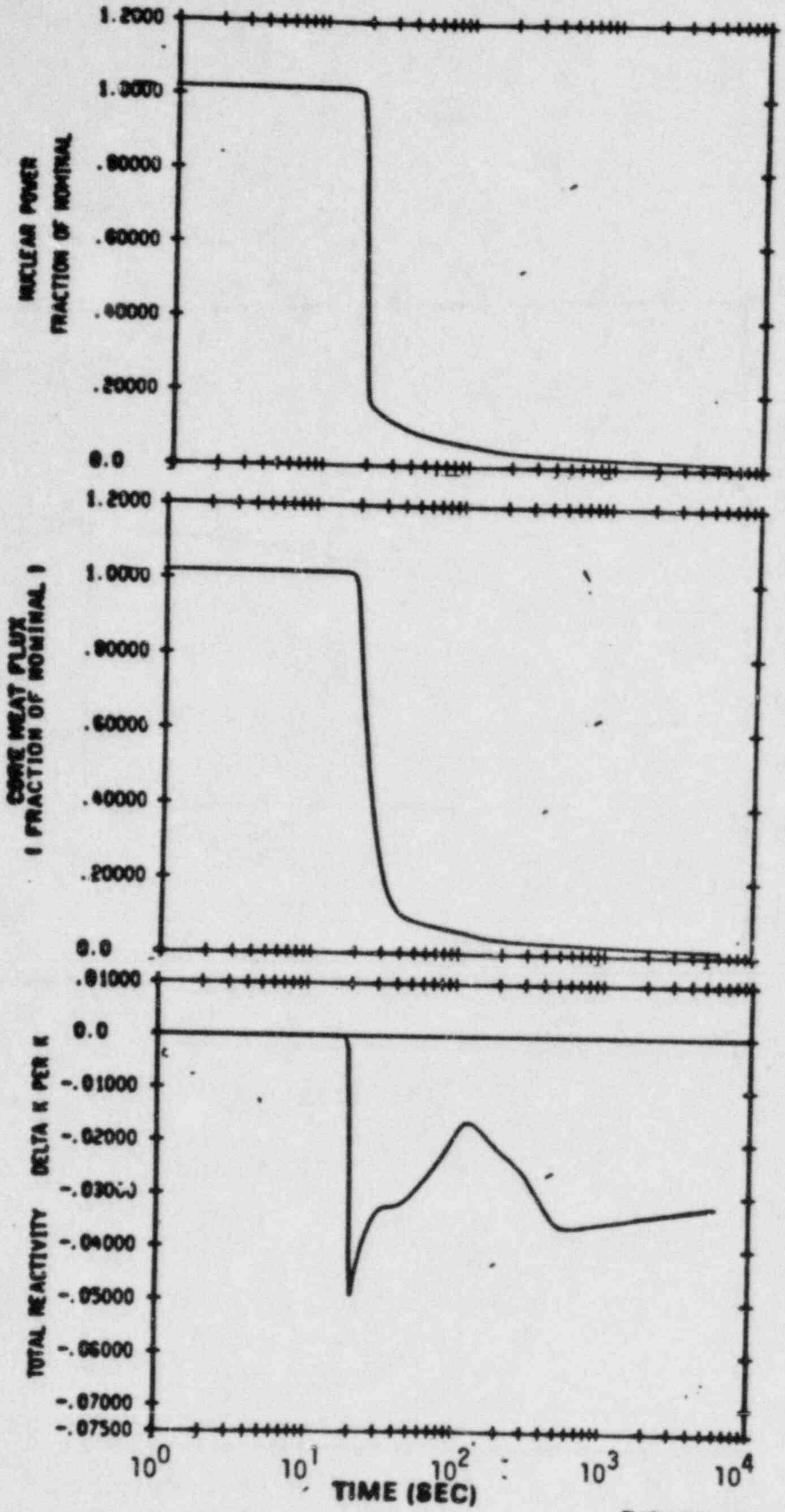


FIGURE 15.2- 18
MAIN FEEDLINE RUPTURE
WITHOUT OFFSITE POWER
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

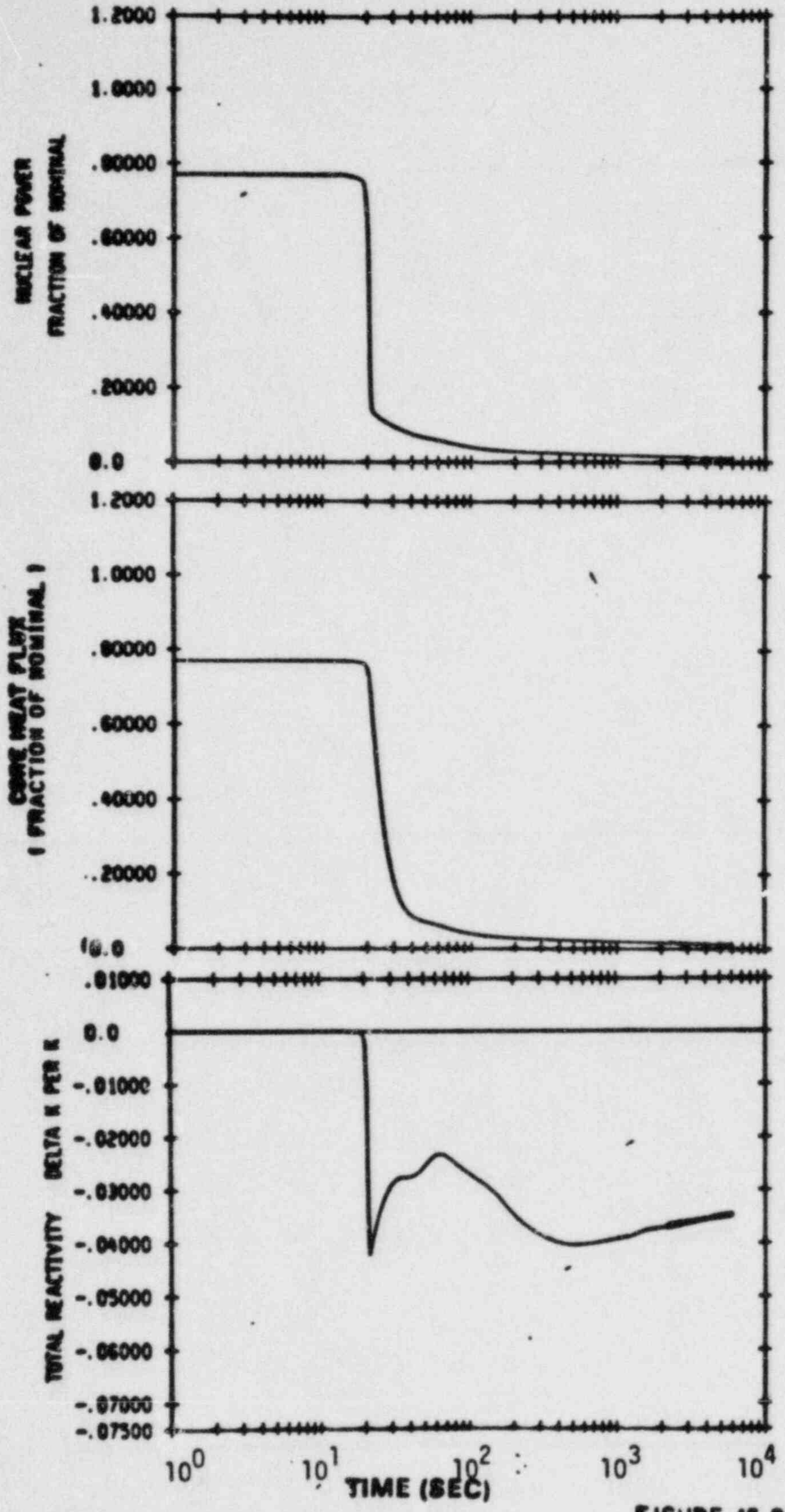


FIGURE 15.2-18A
MAIN FEEDLINE RUPTURE
WITHOUT OFFSITE POWER
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT
(N-1 Loop Operation)

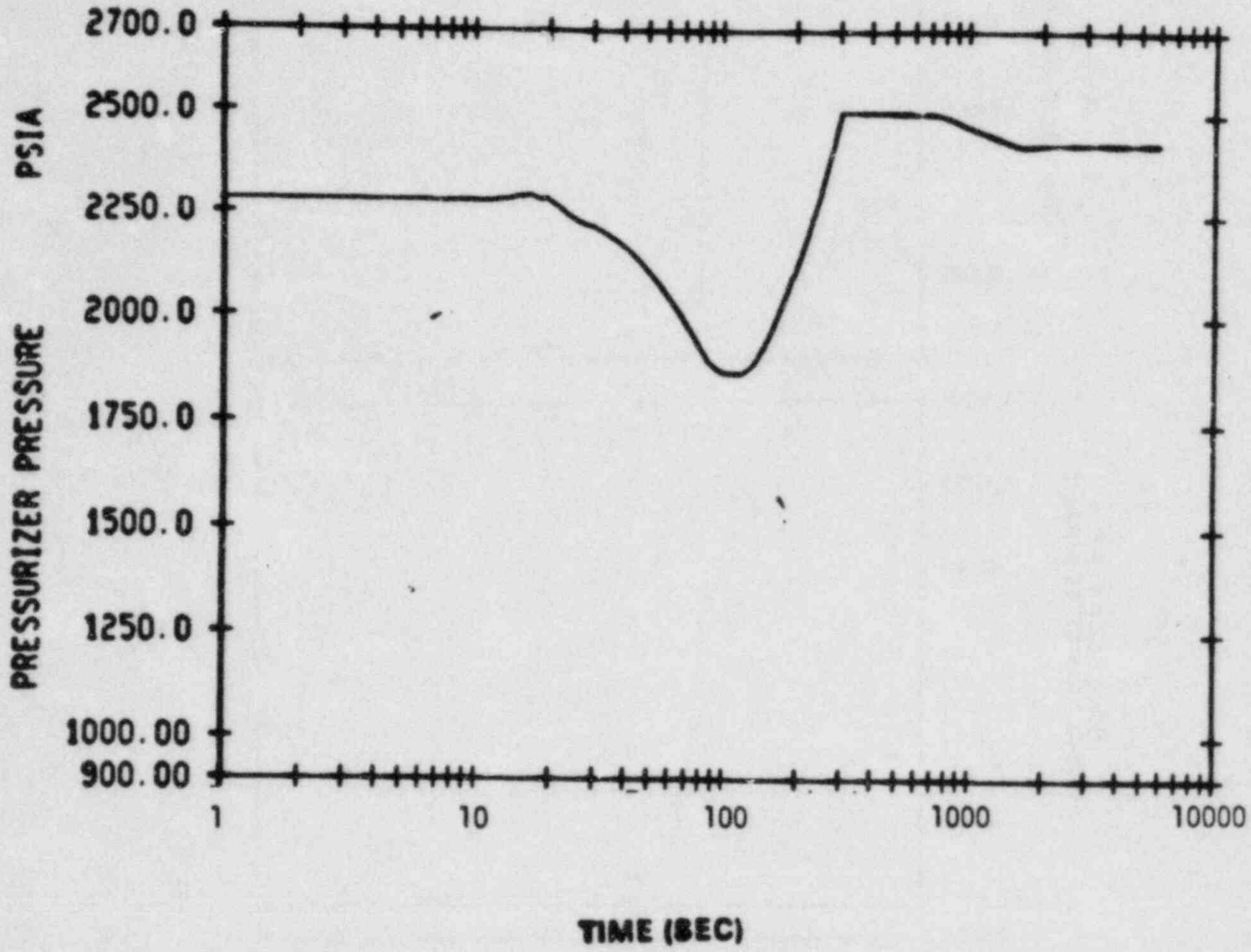


FIGURE 15.2-19
MAIN FEEDLINE RUPTURE
WITHOUT OFFSITE POWER
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

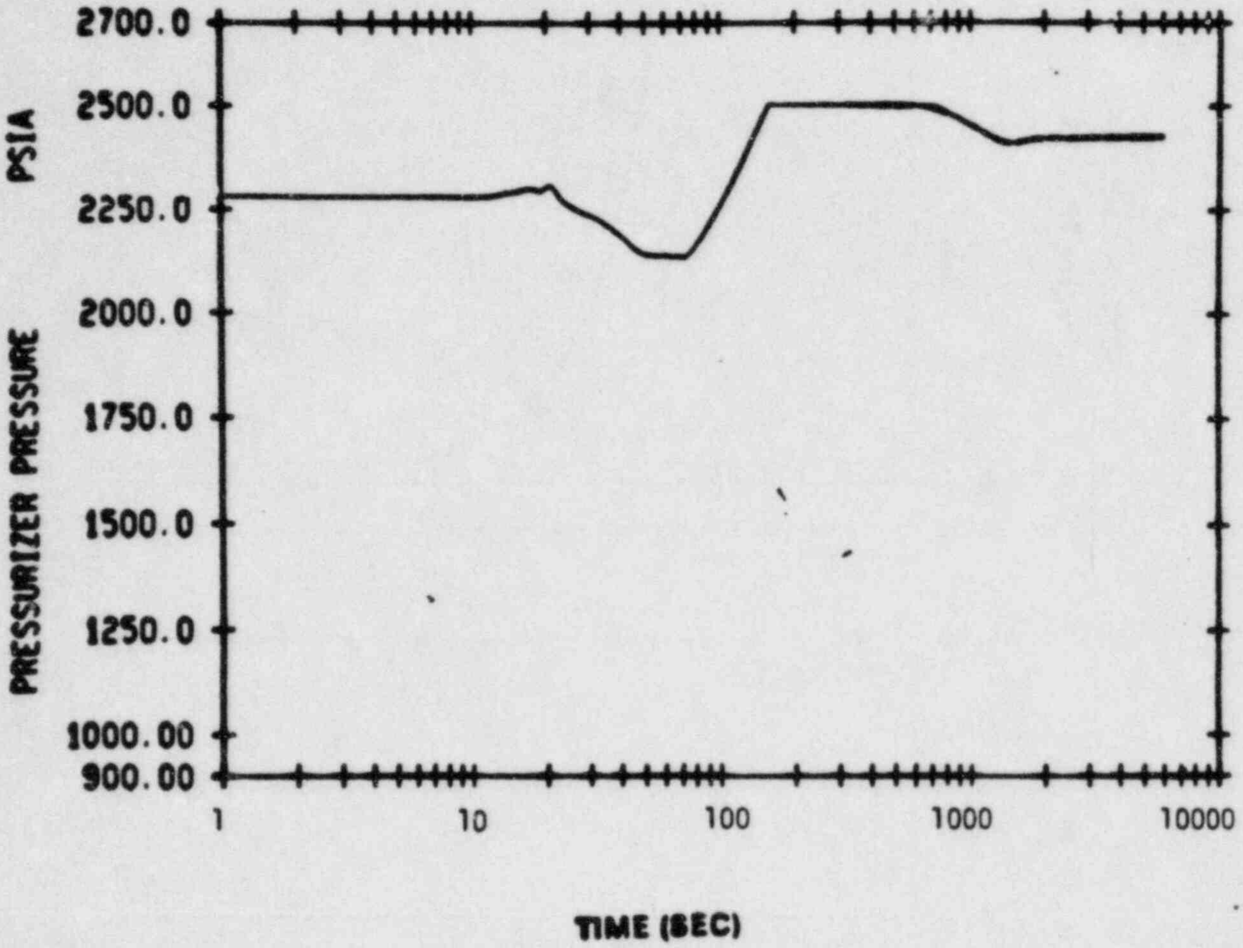


FIGURE 15.2-19A
MAIN FEEDLINE RUPTURE
WITHOUT OFFSITE POWER
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT
(N-1 Loop Operation)

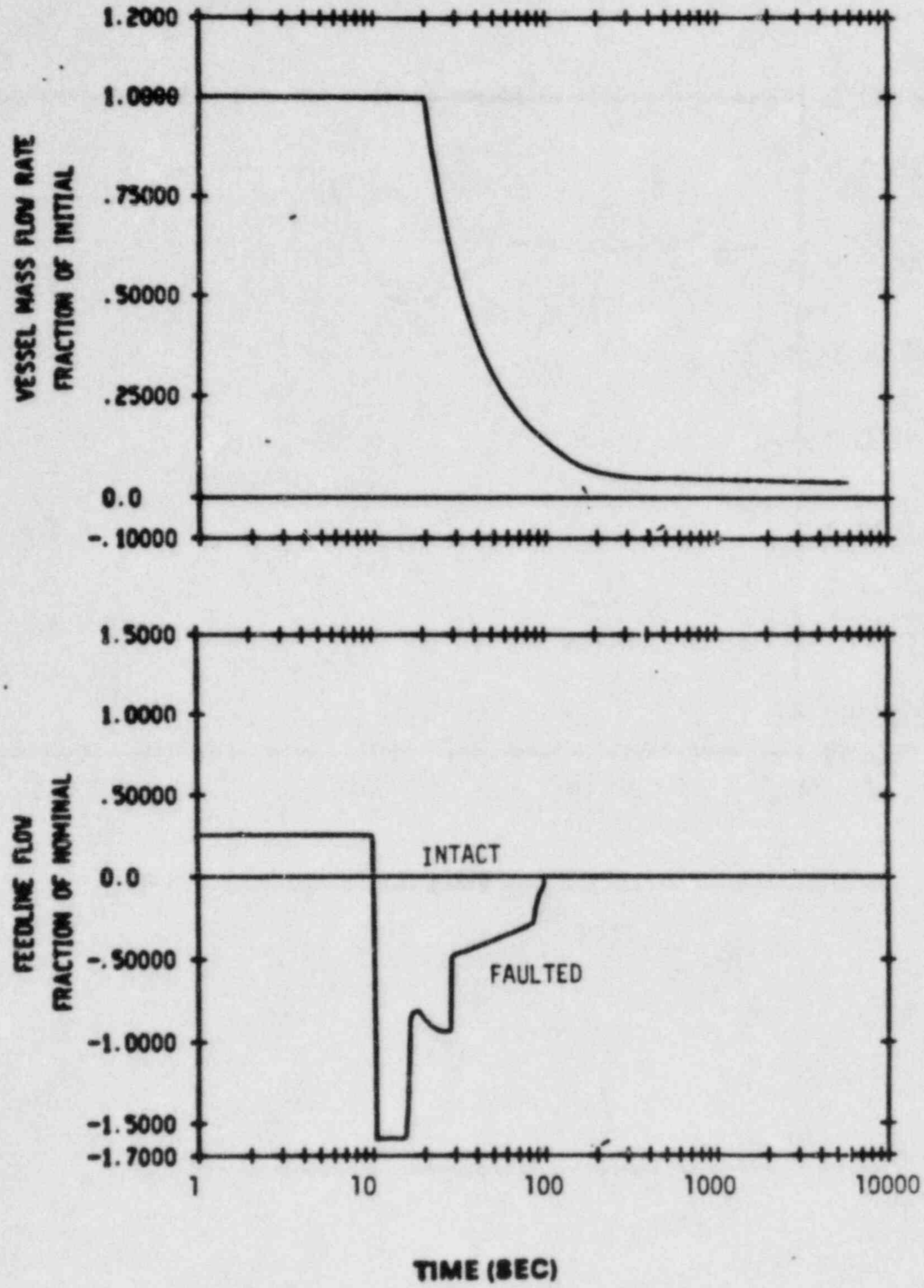


FIGURE 15.2-20
MAIN FEEDLINE RUPTURE
WITHOUT OFFSITE POWER
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

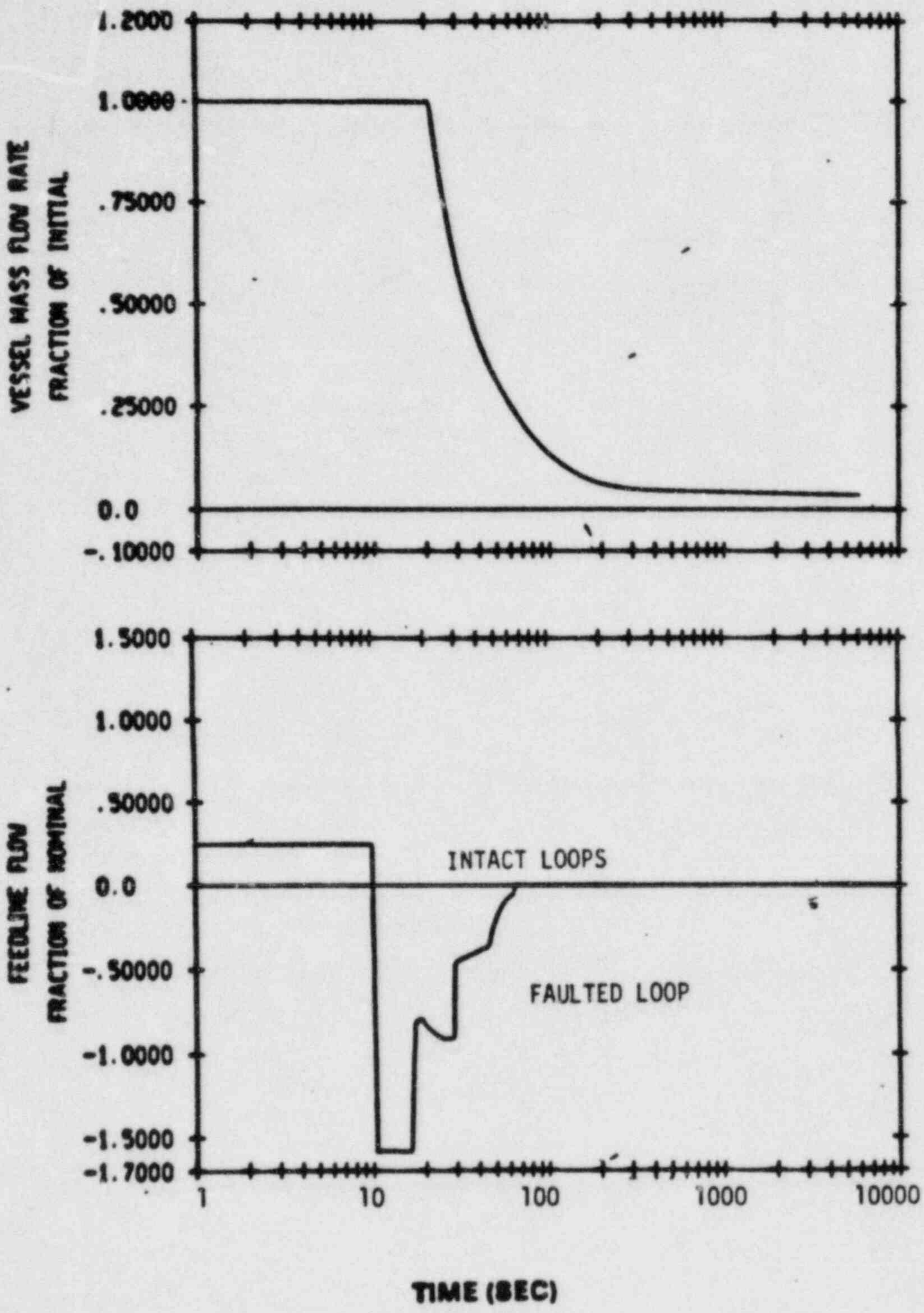


FIGURE 15.2-~~10A~~ 20
MAIN FEEDLINE RUPTURE
WITHOUT OFFSITE POWER
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT
(N-1 Loop Operation)

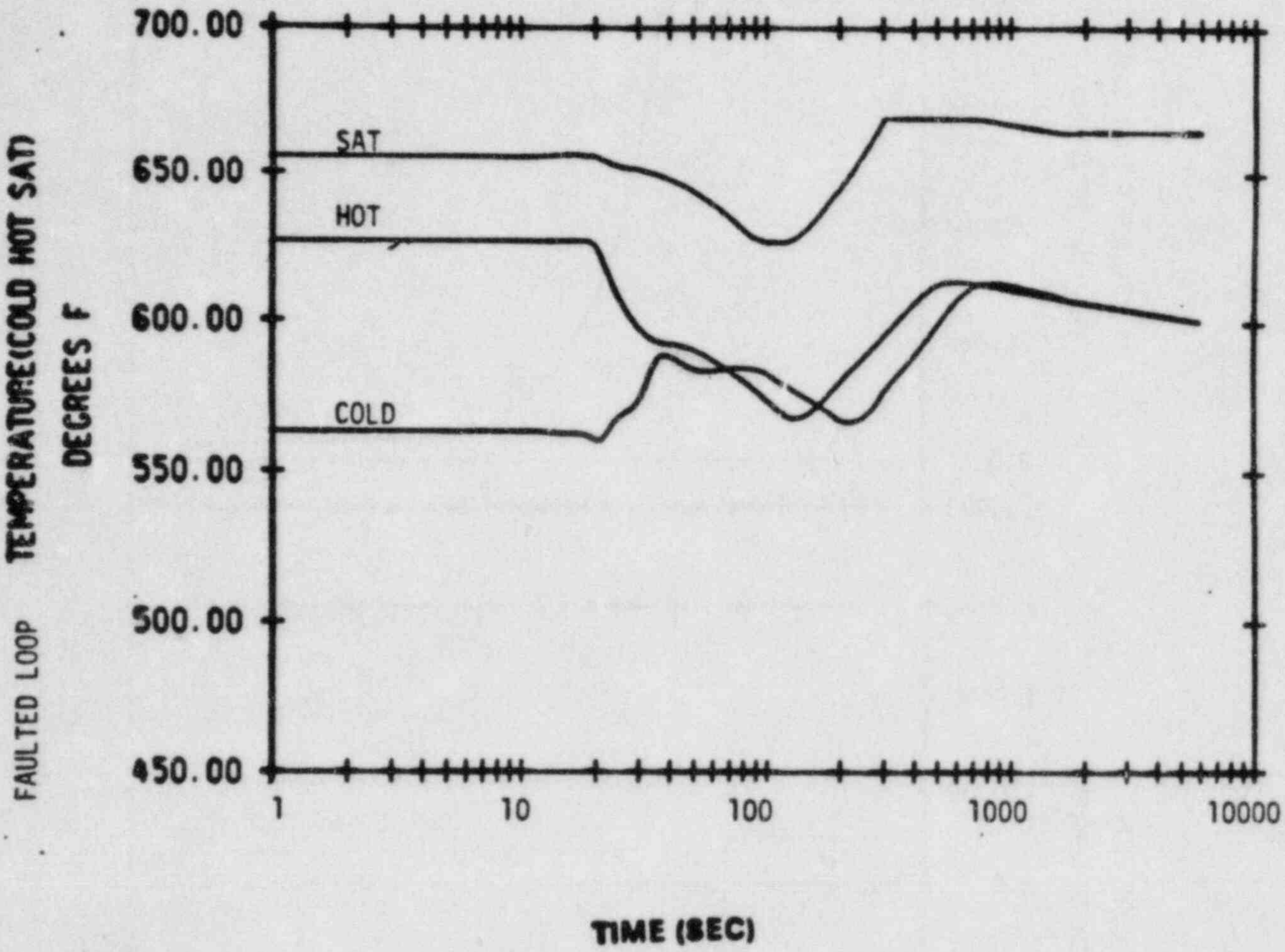


FIGURE 15.2-21
MAIN FEEDLINE RUPTURE
WITHOUT OFFSITE POWER
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

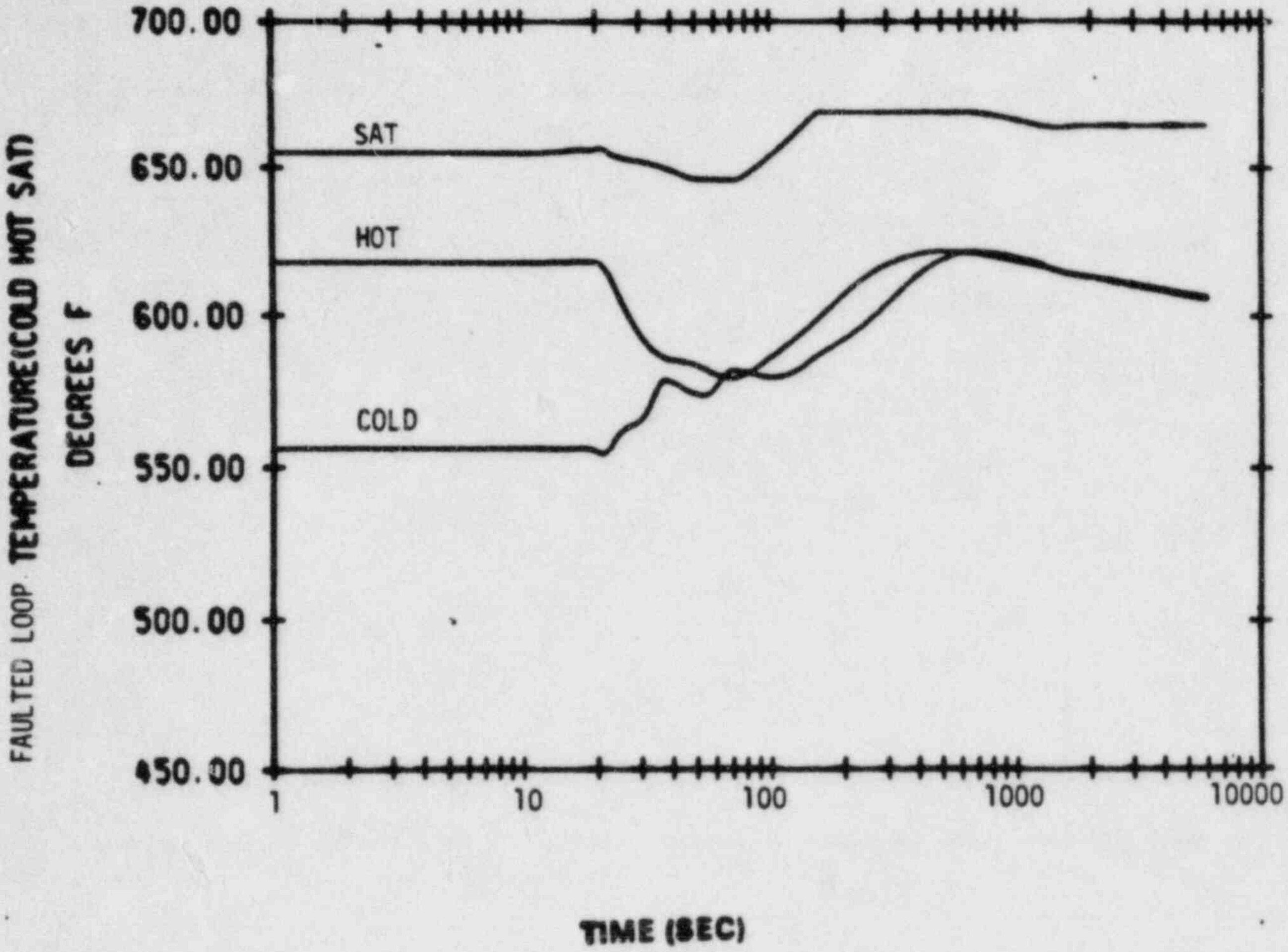


FIGURE 15.2-21A
MAIN FEEDLINE RUPTURE
WITHOUT OFFSITE POWER
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT
(N-1 Loop Operation)

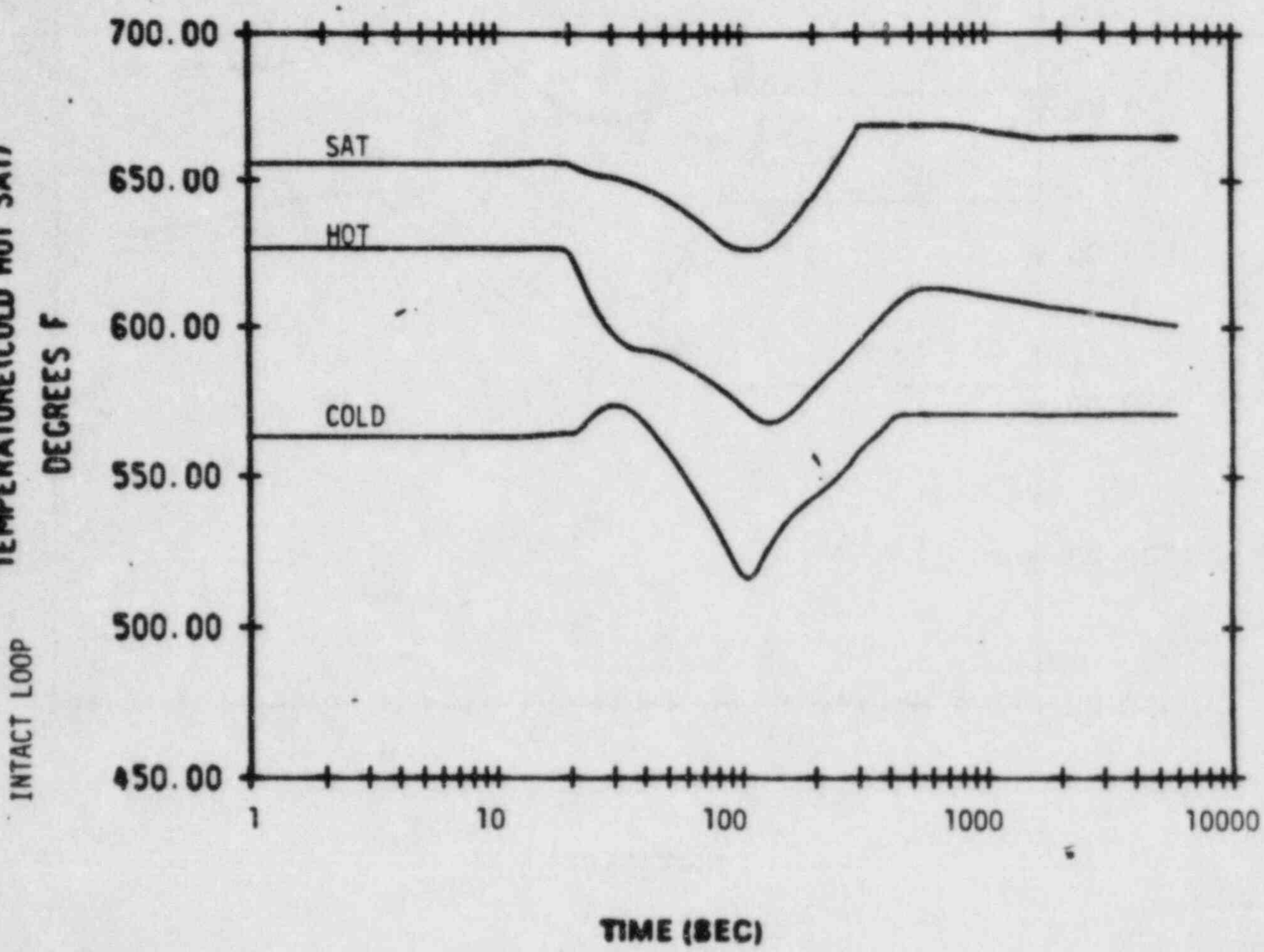


FIGURE 15.2- 22
MAIN FEEDLINE RUPTURE
WITHOUT OFFSITE POWER
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

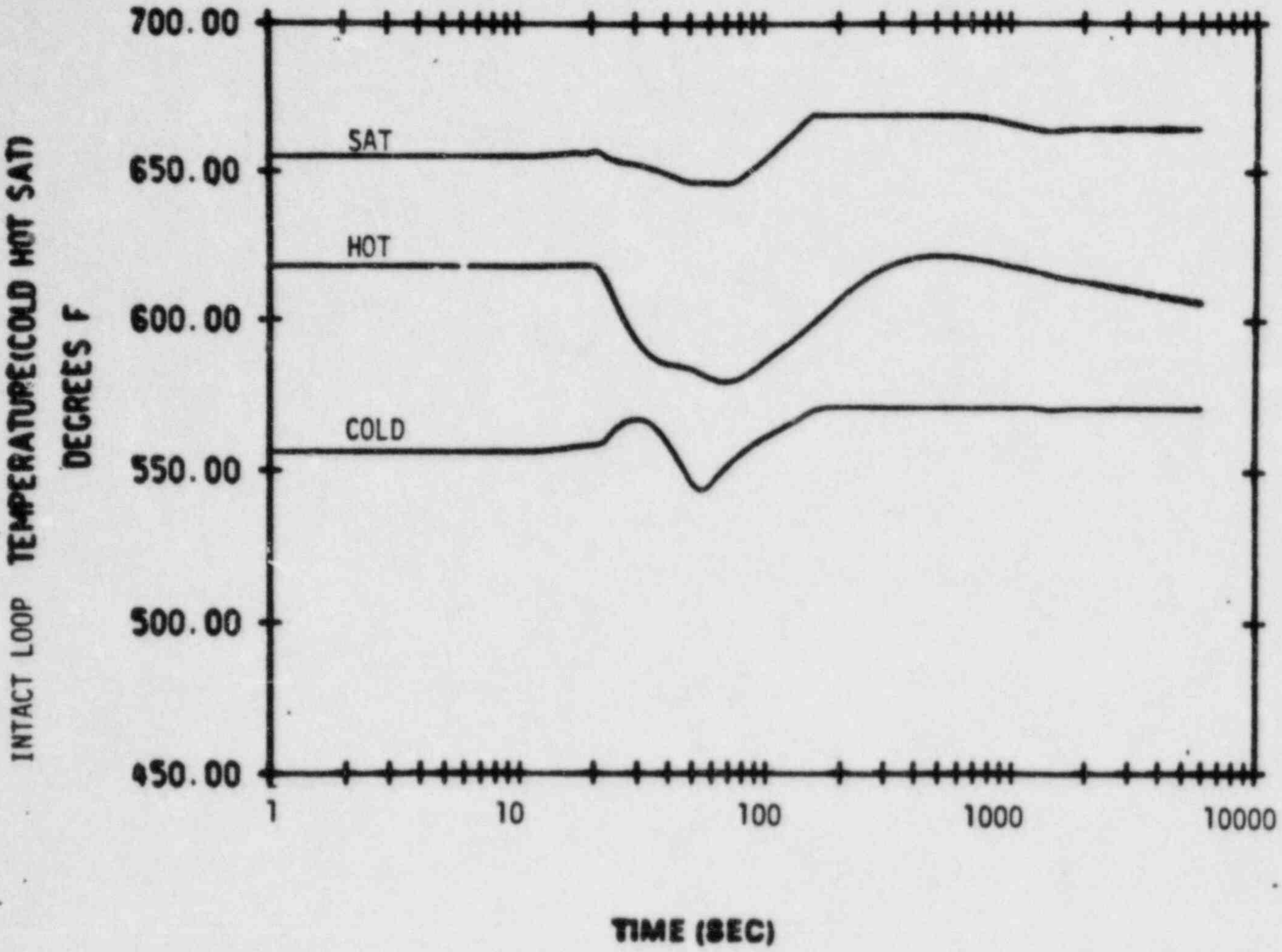


FIGURE 15.2-22A
MAIN FEEDLINE RUPTURE
WITHOUT OFFSITE POWER
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

(N-1 Loop Operation)

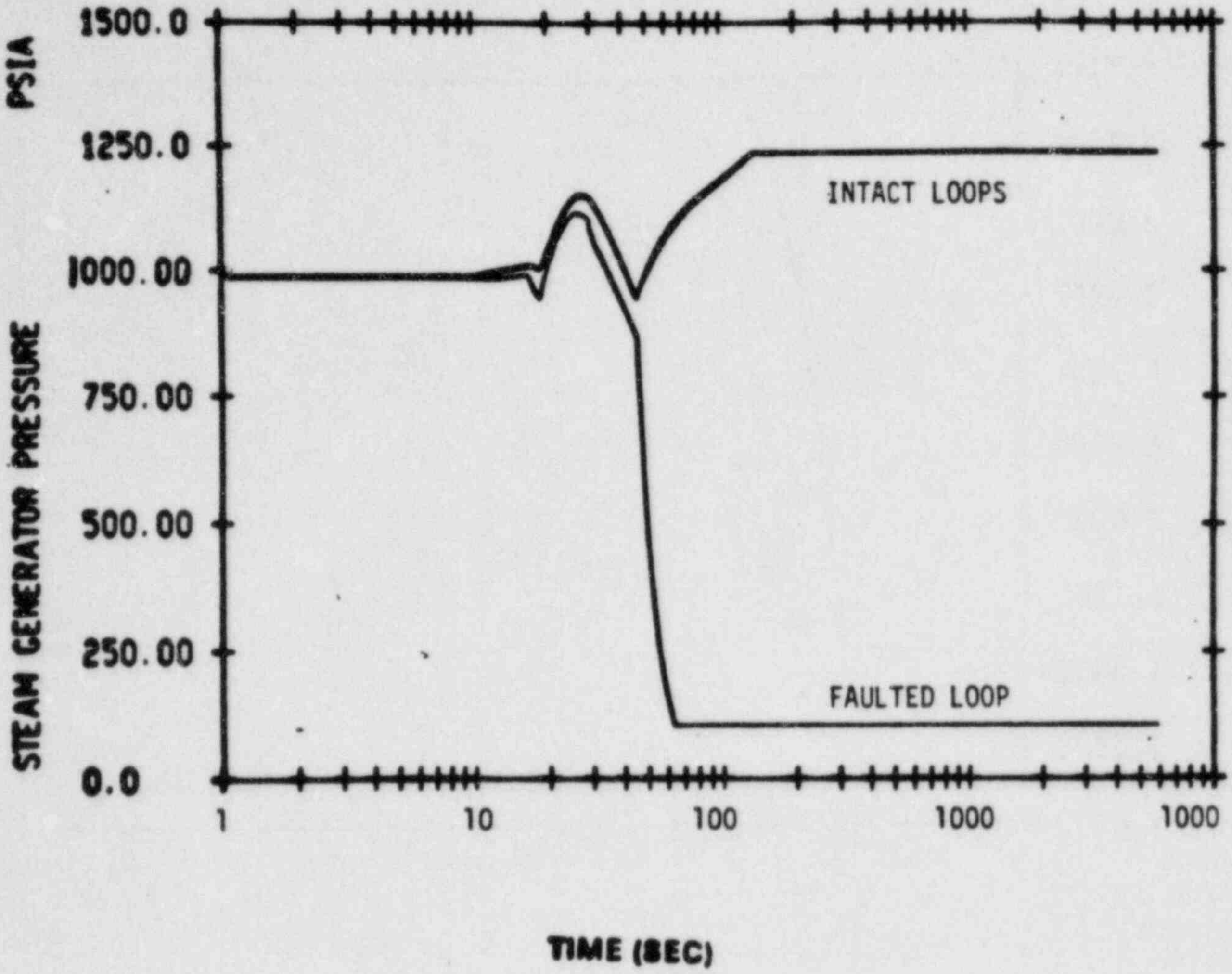


FIGURE 15.2-23A
MAIN FEEDLINE RUPTURE
WITHOUT OFFSITE POWER
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

(N-1 Loop Operation)

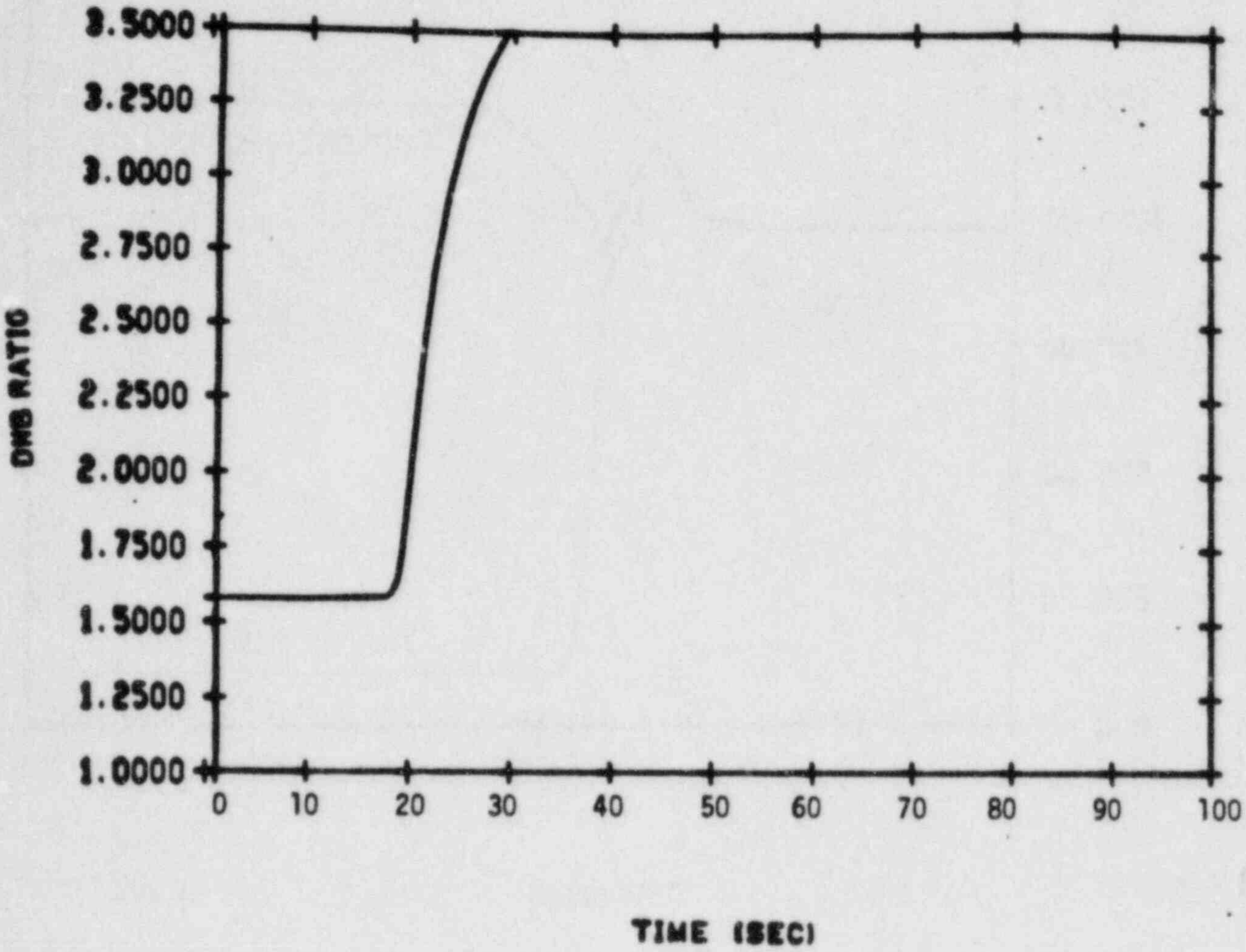


FIGURE 15.2-24
MAIN FEEDLINE RUPTURE
WITHOUT OFFSITE POWER
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

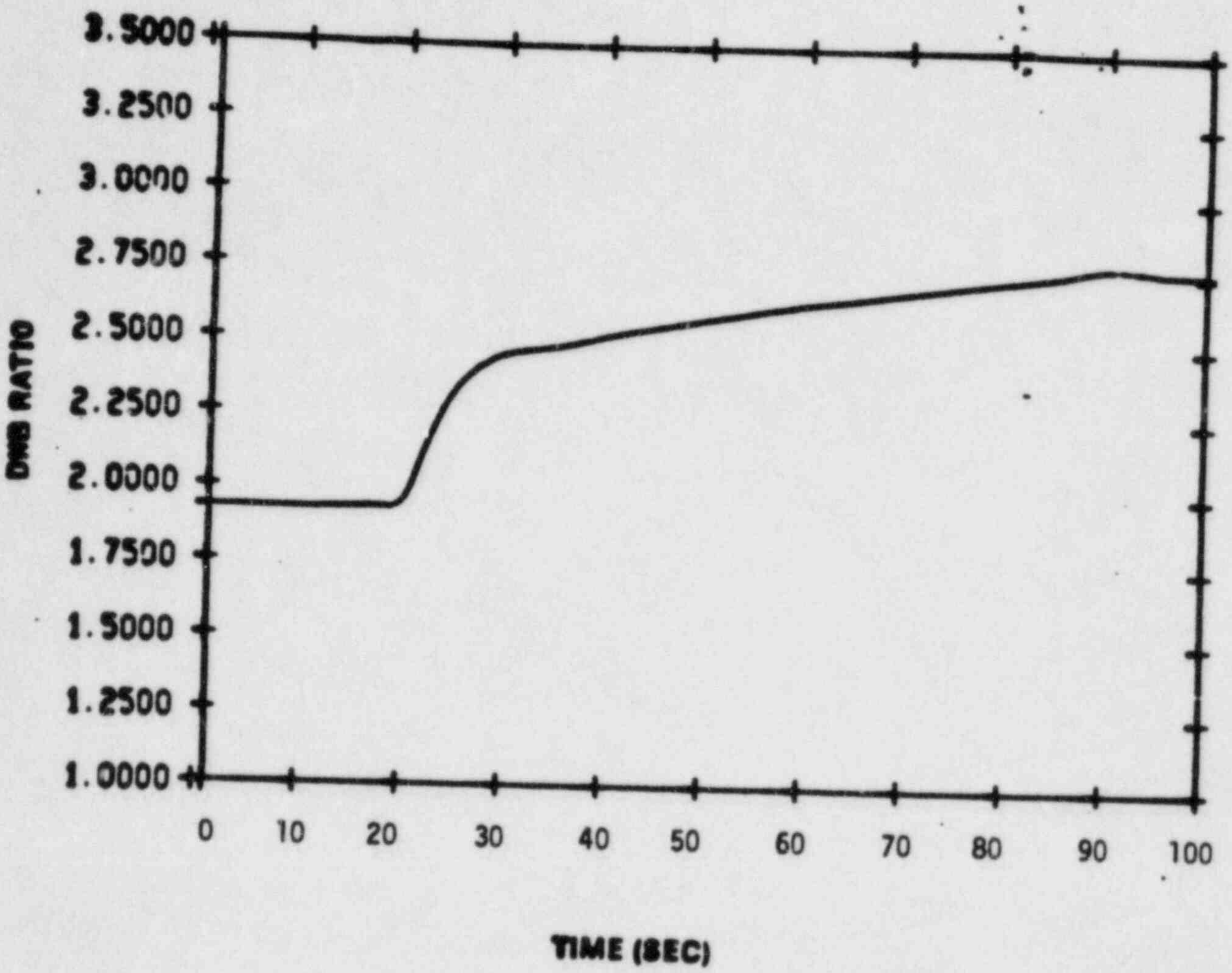


FIGURE 15.2-24A
MAIN FEEDLINE RUPTURE
WITHOUT OFFSITE POWER
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

(N-1 Loop Operation)

15.3 DECREASE IN REACTOR COOLANT SYSTEM FLOW RATE 1.10

A number of faults are postulated which could result in a decrease in 1.11
reactor coolant system flow rate. These events are discussed in this 1.12
section. Detailed analyses are presented for the most limiting of 1.13
these events.

Discussions of the following flow decrease events are presented in 1.14
Section 15.3.

1. Partial loss of forced reactor coolant flow. 1.16
2. Complete loss of forced reactor coolant flow. 1.17
3. Reactor coolant pump shaft seizure (locked rotor). 1.18
4. Reactor coolant pump shaft break. 1.19

Item 1 above is considered to be an ANS Condition II event, Item 2 an 1.21
ANS Condition III event, and Items 3 and 4 ANS Condition IV events. 1.22
Section 15.0.1 contains a discussion of ANS classifications. 1.23

15.3.1 Partial Loss of Forced Reactor Coolant Flow 1.25

15.3.1.1 Identification of Causes and Accident Description 1.26

A partial loss-of-coolant flow accident can result from a mechanical 1.27
or electrical failure in a reactor coolant pump, or from a fault in 1.29
the power supply to the pump or pumps supplied by a reactor coolant
pump bus. If the reactor is at power at the time of the accident, 1.30
the immediate effect of loss-of-coolant flow is a rapid increase in 1.31
the coolant temperature. This increase could result in DNB with 1.32
subsequent fuel damage if the reactor is not tripped promptly.

Normal power for the pumps is supplied through individual buses 1.33
connected to the generator. When a turbine or generator trip occurs, 1.34
the buses are automatically transferred to a transformer supplied
from external power lines, and the pumps will continue to supply 1.35
coolant flow to the core.

This event is classified as an ANS Condition II incident (an incident 1.36
of moderate frequency) as defined in Section 15.0.1. 1.37

The necessary protection against a partial loss-of-coolant flow 1.38
accident is provided by the low primary coolant flow reactor trip 1.39
signal which is actuated in any reactor coolant loop by two out of
three low flow signals. Above Permissive 8, low flow in any loop 1.40
will actuate a reactor trip. Between approximately 10-percent power 1.41
(Permissive 7) and the power level corresponding to Permissive 8, low
flow in any two loops will actuate a reactor trip. 1.42

15.3.1.2 Analysis of Effects and Consenquences 1.44

Method of Analysis 1.46

Two cases have been analyzed: 1.48

1. loss of pump with four loops in operation, and 1.50

2. loss of pump with three loops in operation. 1.51

This transient is analyzed by three digital computer codes. First, 1.55
the LOFTRAN (WCAP-7907, 1972) Code is used to calculate the loop and
core flow during the transient, the time of reactor trip based on the 1.57
calculated flows, the nuclear power transient, and the primary system
pressure and temperature transients. The FACTRAN (WCAP-7908, 1972) 1.59
Code is then used to calculate the heat flux transient based on the
nuclear power and flow from LOFTRAN. Finally, the THINC Code 2.2
(Section 4.4) is used to calculate the DNBR during the transient
based on the heat flux from FACTRAN and flow from LOFTRAN. The "R" 2.4
grid spacer factor is applied to the W-3 correlation. The DNBR 2.5
transients presented represent the minimum of the typical or thimble
cell.

Initial Conditions 2.8

Plant characteristics and initial conditions are discussed in 2.10
Section 15.0.3. Initial operating conditions assumed for this event 2.12
are the most adverse with respect to the margin to DNB, i.e., maximum 2.13
guaranteed steady state thermal power, minimum steady stat pressure,
and maximum steady state coolant average temperature and minimum 2.14
(thermal design) RCS flow rate.

Reactivity Coefficients 2.17

A conservatively large absolute value of Doppler-only power 2.19
coefficient is used (Figure 15.0-2). This is equivalent to a total 2.21
integrated Doppler reactivity from 0 to 100-percent power of
0.016 Δk. —

The least negative moderator temperature coefficient (Figure 15.0-3) 2.22
is assumed since this results in the maximum core power during the 2.23
initial part of the transient when the minimum DNBR is reached.

Flow Coastdown 2.26

The flow coastdown analysis is based on a momentum balance around 2.28
each reactor coolant loop and across the reactor core. This momentum 2.31
balance is combined with the continuity equations, a pump momentum
balance, and the pump characteristics and is based on high estimates 2.32
of system pressure losses.

Plant systems and equipment which are necessary to mitigate the 2.33
effects of the accident are discussed in Section 15.0.8 and listed in 2.34

Table 15.0-6. No single active failure in any of these systems or 2.35
equipment will adversely affect the consequences of the accident.

Results 2.38

Figures 15.3-1 through 15.3-4 show the transient response for the 2.40
loss of one reactor coolant pump with four loops in operation. 2.41
Figure 15.3-4 shows the DNBR to be always greater than 1.30. 2.43

Figures 15.3-1A through 15.3-4A show the transient response for the 2.44
loss of reactor coolant pump with three loops in operation. The 2.46
minimum DNBR is greater than 1.30, as shown in Figure 15.3-4A.

For both cases analyzed, since DNB does not occur, the ability of the 2.47
primary coolant to remove heat from the fuel rod is not greatly 2.48
reduced. Thus, the average fuel and clad temperatures do not 2.49
increase significantly above their respective initial values.

The calculated sequence of events tables for the two cases analyzed 2.50
are shown in Table 15.3-1. The affected reactor coolant pump will 2.51
continue to coastdown, and the core flow will reach a new equilibrium
value corresponding to the number of pumps still in operation. With 2.53
the reactor tripped, a stable plant condition will eventually be
obtained. Normal plant shutdown may then proceed. 2.54

15.3.1.3 Conclusions 2.56

The analysis shows that the DNBR will not decrease below the limiting 2.57
value of 1.30 at any time during the transient. Thus, no fuel or 2.59
clad damage is predicted, and all applicable acceptance criteria are
met.

15.3.1.4 Radiological Consequences 3.1

A partial loss of reactor coolant flow from full load would result in 3.3
a reactor and turbine trip. Assuming in addition that the condenser 3.5
is not available, atmospheric steam dump may be required.

There are only minimal radiological consequences associated with this 3.6
event. The radiological consequences resulting from atmospheric 3.7
steam dump are less severe than the steam line break event analyzed 3.8
in Section 15.1.5 since fuel damage as a result of this transient is
not postulated. Therefore, this event is not limiting. 3.9

15.3.2 Complete Loss of Forced Reactor Coolant Flow 3.11

15.3.2.1 Identification of Causes and Accident Description 3.12

A complete loss of forced reactor coolant flow may result from a 3.13
simultaneous loss of electrical supplies to all reactor coolant 3.14
pumps. If the reactor is at power at the time of the accident, the 3.16
immediate effect of loss-of-coolant flow is a rapid increase in the 3.17
coolant temperature. This increase could result in DNB with 3.18
subsequent fuel damage if the reactor were not tripped promptly.

Normal power for the reactor coolant pumps is supplied through buses 3.19
from a transformer connected to the generator. When a turbine or 3.20
generator trip occurs, the buses are automatically transferred to a
transformer supplied from external power lines, and the pumps will 3.21
continue to supply coolant flow to the core.

This event is classified as an ANS Condition III incident (an 3.22
infrequent incident), as defined in Section 15.0.1.

The following signals provide the necessary protection against a 3.23
complete loss of flow accident:

1. reactor coolant pump underspeed, and 3.25

2. low reactor coolant loop flow 3.26

The reactor trip on reactor coolant pump underspeed is provided to 3.28
protect against conditions which can cause a loss of voltage to all 3.29
reactor coolant pumps, i.e., station blackout. This function is 3.30
blocked below approximately 10-percent power (Permissive 7).

The reactor trip on reactor coolant pump underspeed is also provided 3.31
to trip the reactor for an underfrequency condition, resulting from 3.32
frequency disturbances on the power grid. If the maximum grid 3.33
frequency decay rate is less than approximately 5 Hertz/second, this
trip will protect the core from underfrequency events (Section 8.2). 3.34
This effect is fully described in WCAP-8424, Revision 1 (1975). 3.35

The reactor trip on low primary coolant loop flow is provided to 3.36
protect against loss of flow conditions which affect only one reactor 3.37
coolant loop. This function is generated by two out of three low 3.38
flow signals per reactor coolant loop. Above Permissive 8, low flow 3.39
in any loop will actuate a reactor trip. Between approximately 3.40
10-percent power (Permissive 7) and the power level corresponding to
Permissive 8, low flow in any two loops will actuate a reactor trip. 3.41

15.3.2.2 Analysis of Effects and Consequences 3.43

Two cases have been analyzed: 3.44

1. loss of four pumps with four loops in operation, and 3.46

2. loss of three pumps with three loops in operation. 3.47

This transient is analyzed by three digital computer codes. First, 3.51
the LOFTRAN (WCAP-7907, 1972) Code is used to calculate the loop and
core flow during the transient, the time of reactor trip based on the 3.52
calculated flows, the nuclear power transient, and the primary system
pressure and temperature transients. The FACTRAN (WCAP-7908, 1972) 3.54
Code is then used to calculate the heat flux transient based on the
nuclear power and flow from LOFTRAN. Finally, the THINC Code 3.57
(Section 4.4) is used to calculate the DNBR during the transient
based on the heat flux from FACTRAN and flow from LOFTRAN. The "R" 3.59
grid spacer factor is applied to the W-3 correlation. The DNBR 3.60

transients presented represent the minimum of the typical or thimble
cell.

The method of analysis and the assumptions made regarding initial 4.1
operating conditions and reactivity coefficients are identical to 4.2
those discussed in Section 15.3.1, except that following the loss of
power supply to all pumps at power, a reactor trip is actuated by 4.3
reactor coolant pump underspeed.

Results 4.6

Figures 15.3-5 through 15.3-8 show the transient response for the 4.8
loss of power to all reactor coolant pumps with four loops in 4.9
operation. The reactor is assumed to be tripped on an underspeed 4.11
signal. Figure 15.3-8 shows the DNBR to be always greater than 1.30. 4.12 | 440.52

Figures 15.3-5A through 15.3-8A show the transient response for the 4.13
loss of power to all reactor coolant pumps with three loops in 4.14
operation. The reactor is again assumed to be tripped on an 4.15
underspeed signal. The minimum DNBR is greater than 1.30, as shown 4.16
in Figure 15.3-12A.

For both cases analyzed, since DNB does not occur, the ability of the 4.17
primary coolant to remove heat from the fuel rod is not greatly 4.18
reduced. Thus, the average fuel and clad temperatures do not 4.19
increase significantly above their respective initial values.

The calculated sequence of events is shown in Table 15.3-1. The 4.21
reactor coolant pumps will continue to coastdown, and natural
circulation flow will eventually be established, as demonstrated in 4.22
Section 15.2.6. With the reactor tripped, a stable plant condition 4.23
would be attained. Normal plant shutdown may then proceed. 4.24

15.3.2.3 Conclusions 4.26

The analysis performed has demonstrated that for the complete loss of 4.27
forced reactor coolant flow, the DNBR does not decrease below 1.30 at 4.29
any time during the transient. Thus, no fuel or clad damage is 4.30
predicted, and all applicable acceptance criteria are met.

15.3.2.4 Radiological Consequences 4.32

A complete loss of reactor coolant flow from full load results in a 4.33
reactor and turbine trip. Assuming in addition that the condenser is 4.35
not available, atmospheric steam dump would be required. The 4.36
quantity of steam released would be the same as for a loss of offsite
power incident.

There are only minimal radiological consequences associated with this 4.37
event. Since fuel damage is not postulated, the radiological 4.38
consequences resulting from atmospheric steam dump are less severe 4.39
than the steam line break analyzed in Section 15.1.5. Therefore, 4.40
this event is not limiting.

15.3.3 Reactor Coolant Pump Shaft Seizure (Locked Rotor) 4.42

15.3.3.1 Identification of Causes and Accident Description 4.43

The accident postulated is an instantaneous seizure of a reactor 4.44
coolant pump rotor such as is discussed in Section 5.4. Flow through 4.47
the affected reactor coolant loop is rapidly reduced, leading to an
initiation of a reactor trip on a low flow signal. 4.48

Following initiation of the reactor trip, heat stored in the fuel 4.49
rods continues to be transferred to the coolant causing the coolant 4.50
to expand. At the same time, heat transfer to the shell side of the 4.51
steam generators is reduced, first because the reduced flow results 4.52
in a decreased tube side film coefficient and then because the
reactor coolant in the tubes cools down while the shell side 4.54
temperature increases (turbine steam flow is reduced to zero upon
plant trip). The rapid expansion of the coolant in the reactor core, 4.55
combined with reduced heat transfer in the steam generators, causes 4.56
an insurge into the pressurizer and a pressure increase throughout
the reactor coolant system. The insurge into the pressurizer 4.57
compresses the steam volume, actuates the automatic spray system,
opens the power-operated relief valves, and opens the pressurizer 4.58
safety valves, in that sequence. The two power-operated relief 4.59
valves are designed for reliable operation and would be expected to
function properly during the accident. However, for conservatism, 5.1
their pressure reducing effect, as well as the pressure reducing
effect of the spray, are not included in the analysis. 5.3

This event is classified as an ANS Condition IV incident (a limiting 5.4
fault) as defined in Section 15.0.1.

15.3.3.2 Analysis of Effects and Consequences 5.6

<u>Method of Analysis</u> 5.8

Two digital-computer codes are used to analyze this transient. The 5.12
LOFTRAN Code (WCAP-7907, 1972) is used to calculate the resulting
loop and core flow transients following the pump seizure, the time of 5.13
reactor trip based on the loop flow transients, the nuclear power
following reactor trip, and to determine the peak pressure. The 5.15
thermal behavior of the fuel located at the core hot spot is
investigated using the FACTRAN Code (WCAP-7907, 1972), which uses the 5.16
core flow and the nuclear power calculated by LOFTRAN. The FACTRAN 5.17
Code includes a film boiling heat transfer coefficient.

Two cases have been analyzed: 5.18

1. four loops in operation, one locked rotor, and 5.20

2. three loops in operation, one locked rotor. 5.21

At the beginning of the postulated locked rotor accident, i.e., at 5.23
the time the shaft in one of the reactor coolant pumps is assumed to 5.24
seize, the plant is assumed to be in operation under the most adverse

steady state operating conditions, i.e., the maximum guaranteed 5.25
steady state thermal power, maximum steady state pressure, and
maximum steady state coolant average temperature. Plant 5.27
characteristics and initial conditions are further discussed in
Section 15.0.3. The accident is evaluated with offsite power 5.28
available, without offsite power available, and with the loss of one
protection train. For the case without offsite power available, 5.30
power is lost to the unaffected pumps 2 seconds after reactor trip.
With three loops operating, the maximum power level (including 5.31
errors) allowed in that mode of operation is assumed.

440.50

For the peak pressure evaluation, the initial pressure is 5.32
conservatively estimated as 30 psi above nominal pressure 5.33
(2,250 psia) to allow for errors in the pressurizer pressure
measurement and control channels. This is done to obtain the highest 5.34
possible rise in the coolant pressure during the transient. To 5.35
obtain the maximum pressure in the primary side, conservatively high
loop pressure drops are added to the calculated pressurizer pressure. 5.36
The pressure responses, shown on Figures 15.3-10 and 15.3-10A are the 5.37
responses at the point in the reactor coolant system having the 5.38
maximum pressure.

Evaluation of the Pressure Transient 5.41

After pump seizure, the neutron flux is rapidly reduced by control 5.43
rod insertion. Rod motion is assumed to begin one second after the 5.45
flow in the affected loop reaches 87 percent of nominal flow. No 5.46
credit is taken for the pressure reducing effect of the pressurizer
relief valves, pressurizer spray, steam dump, or controlled feedwater 5.47
flow after plant trip.

Although these operations are expected to occur and would result in a 5.48
lower peak pressure, an additional degree of conservatism is provided 5.49
by ignoring their effect.

The pressurizer safety valves are full open at 2,575 psia and their 5.50
capacity for steam relief is as described in Section 5.4. 5.51

Evaluation of DNB in the Core during the Accident 5.54

For this accident, DNB is assumed to occur in core, and therefore, an 5.56
evaluation of the consequences, with respect to fuel rod thermal 5.57
transients, is performed. Results obtained for analysis of this "hot 5.59
spot" condition represent the upper limit with respect to clad
temperature and zirconium water reaction. 5.60

In the evaluation, the rod power at the hot spot is assumed to be 6.1
2.5 times the average rod power (i.e., $F = 2.5$) at the initial core 6.2
power level. For three loops in operation, $F_Q = 3.0$. 6.3

Film Boiling Coefficient 6.6

The film boiling coefficient is calculated in the FACTRAN Code using 6.8
the Bishop-Sandberg-Tong film boiling correlation. The fluid 6.10

properties are evaluated at film temperature (average between wall and bulk temperatures). The program calculates the film coefficient at every time step, based upon the actual heat transfer conditions at the time. The neutron flux, system pressure, bulk density, and mass flow rate as a function of time are used as program input.

For this analysis, the initial values of the pressure and the bulk density are used throughout the transient since they are the most conservative, with respect to clad temperature response. For conservatism, DNB was assumed to start at the beginning of the accident.

<u>Fuel Clad Gap Coefficient</u>

The magnitude and time dependence of the heat transfer coefficient between fuel and clad (gap coefficient) has a pronounced influence on the thermal results. The larger the value of the gap coefficient, the more heat is transferred between pellet and clad. Based on investigations on the effect of the gap coefficient upon the maximum clad temperature during the transient, the gap coefficient was assumed to increase from a steady state value consistent with initial fuel temperature to 10,000 Btu/hr-ft^2-°F at the initiation of the transient. Thus, the large amount of energy stored in the fuel because of the small initial value is released to the clad at the initiation of the transient.

<u>Zirconium Steam Reaction</u>

The zirconium steam reaction can become significant above 1,800°F (clad temperature). The Baker-Just parabolic rate equation shown below is used to define the rate of the zirconium steam reaction.

$$\frac{d(w^2)}{dt} = 33.3 \times 10^6 \exp\left(-\frac{45,500}{1.986\,T}\right)$$

where:

w = amount reacted (mg/cm^2)

t = time (sec)

T = temperature (°F)

The reaction heat is 1,510 cal/gm.

The effect of zirconium-steam reaction is included in the calculation of the "hot spot" clad temperature transient.

Plant systems and equipment which are available to mitigate the effects of the accident are discussed in Section 15.0.8 and listed in Table 15.0-6. No single active failure in any of these systems or equipment will adversely affect the consequences of the accident.

Results 7.1

1. Locked Rotor with Four Loops Operating 7.3

The transient results for this case are shown on 7.5
Figures 15.3-9 through 15.3-12. The results of these 7.7
calculations are also summarized in Table 15.3-2. The peak 7.8
reactor coolant system pressure reached during the transient
is less than that which would cause stresses to exceed the 7.9
faulted condition stress limits. Also, the peak clad 7.10
surface temperature is considerably less than 2,700°F. It 7.11
should be noted that the clad temperature was conservatively
calculated assuming that DNB occurs at the initiation of the 7.12
transient.

2. Locked Rotor with Three Loops Operating 7.15

The transient results for this case are shown on Figures 7.17
15.3-9A through 15.3-12A. The results of these calculations 7.19
are also summarized in Table 15.3-2. The peak RCS pressure 7.20
is slightly higher than for the previous case, but is still
less than that which would cause stresses to exceed the 7.21
faulted condition stress limits. The cladding temperature 7.22
transient is still well below the 2,700°F limit.

15.3.3.3 Conclusions 7.25

1. Since the peak reactor coolant system pressure reached 7.27
during any of the transients is less than that which would
cause stresses to exceed the faulted condition stress 7.30
limits, the integrity of the primary coolant system is not
endangered.

2. Since the peak clad surface temperature calculated for the 7.31
hot spot during the worst transient remains considerably
less than 2,700°F, the core will remain in place and intact 7.32
with no loss of core cooling capability.

15.3.3.4 Radiological Consequences 7.35

It is postualted that if a locked rotor accident occurs, the activity 7.36
in the gap of the fuel rods suffering clad damage will be released to 7.37
the reactor coolant. The released activity and the Technical 7.39
Specification activity in the primary coolant from prior operation
with fuel defects leak to the secondary side of the steam generator 7.41
at a Technical Specification leak rate of 1 gpm. Six percent of the 7.42
fuel rods in the core are postulated to have clad damage. Secondary 7.43
coolant activities are assumed to be at Technical Specification
concentration. A partition factor of 0.01 for iodine occurs between 7.44
the water and steam phases in the steam generators during the course 7.45
of the accident. The release to the environment continues until 7.46
primary side pressure and secondary side pressures are equalized.

Offsite power is assumed to be lost, thereby making the condenser 7.47
unavailable for steam dump. Activity released from the steam 7.48
generators to the environment occurs at ground level.

The radiological consequences of a postulated locked rotor accident 7.49
for both N-loop and N-1 loop operation are reported in Table 15.0-8. 7.51
The assumptions used to perform this evaluation are summarized in 7.52
Table 15.3-3. Technical Specification limits were used. For the 7.54 | 440.53
N-1 loop operation case, it was assumed that the plant had been
operating at full power, with N-loops, sufficiently long for the gap 7.55
activity and coolant concentrations to reach equilibrium. One loop 7.56
was then taken out of service, after which the accident was assumed
to occur. The releases to the environment are listed in 7.57
Table 15.3-4. The releases together with the atmospheric dispersion 7.58
factors listed in Table 15.0-11 are used to compute the doses to the 7.59
EAB (0-2 hr) and LPZ (0-8 hr) reported in Table 15.0-8. The 7.60
conservatism inherent in the accident evaluation described in
Section 15.3.3.3 notwithstanding, it is assumed that the gap activity 8.1 | 440.53
is released during the transient from fuel failure. The gap activity 8.2
release is the predominant source for the calculated dose.

The radiological consequences of this accident are a small fraction 8.3
of the guidelines of 10CFR100; i.e., less than 30 Rem to the thyroid 8.4
and 2.5 Rem to the whole body.

15.3.4 Reactor Coolant Pump Shaft Break 8.6

15.3.4.1 Identification of Causes and Accident Description 8.7

The accident is postulated as an instantaneous failure of a reactor 8.8
coolant pump shaft, such as discussed in Section 5.4. Flow through 8.11
the affected reactor coolant loop is rapidly reduced, though the
initial rate of reduction of coolant flow is greater for the reactor 8.12
coolant pump rotor seizure event. Reactor trip is initiated on a low 8.13
flow signal in the affected loop.

Following initiation of the reactor trip, heat stored in the fuel 8.14
rods continues to be transferred to the coolant causing the coolant 8.15
to expand. At the same time, heat transfer to the shell side of the 8.16
steam generators is reduced, first because the reduced flow results 8.17
in a decreased tube side film coefficient and then because the
reactor coolant in the tubes cools down while the shell side 8.18
temperature increases (turbine steam flow is reduced to zero upon
plant trip). The rapid expansion of the coolant in the reactor core, 8.19
combined with reduced heat transfer in the steam generators causes an 8.20
insurge into the pressurizer and a pressure increase throughout the
reactor coolant system. The insurge into the pressurizer compresses 8.21
the steam volume, actuates the automatic spray system, opens the
power-operated relief valves, and opens the pressurizer safety 8.22
valves, in that sequence. The two power-operated relief valves are 8.23
designed for reliable operation and would be expected to function
properly during the accident. However, for conservatism, their 8.25
pressure reducing effect, as well as the pressure reducing effect of
the spray, is not included in the analysis. 8.26

This event is classified as an ANS Condition IV incident (a limiting fault) as defined in Section 15.0.1. 8.27

15.3.4.2 Conclusions 8.29

The radiological consequences of a reactor coolant pump shaft break 8.30
are no worse than those calculated for the locked rotor incident 8.32
(Section 15.3.3). With a failed shaft, the impeller could 8.34
conceivably be free to spin in a reverse direction as opposed to
being fixed in position, as assumed in the locked rotor analysis. 8.35
However, the net effect on core flow is negligible, resulting in only 8.36
a slight decrease in the end point (steady state) core flow. For 8.38
both the shaft break and locked rotor incidents, reactor trip occurs
very early in the transient. In addition, the locked rotor analysis 8.39
conservatively assumes that DNB occurs at the beginning of the
transient.

15.3.5 References for Section 15.3 8.41

WCAP-7907, 1972. Burnett, T.W.T. et al. LOFTRAN Code Description. 8.43

WCAP-7908, 1972. Hargrove, H.G. FACTRAN - A Fortran-IV Code for Thermal Transients in a UO_2 Fuel Rod. 8.47

WCAP-8424, Revision 1, 1975. Baldwin, M.S.; Merrian, M.M.; Schenkel, 8.49
H.S.; and Van DeWalle, D.J. An Evaluation of Loss of Flow Accidents 8.50
Caused by Power System Frequency Transients in Westinghouse PWRs.

WCAP-8846, 1976. Skaritka, J. Hybrid B_4C Absorber Control Rod Evaluation Report. 8.51

TABLE 15.3-1

TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH RESULT IN A DECREASE IN REACTOR COOLANT SYSTEM FLOW

Accident	Event	N-Loop Time (sec)	N-1 Loop Time (sec)
Partial loss of forced reactor coolant flow			
Four loops operating, one pump coasting down	Coastdown begins	0.0	0.0
	Low flow reactor trip	1.5	2.5
	Rods begin to drop	2.5	3.5
	Minimum DNBR occurs	3.3	4.3
Complete loss of forced reactor coolant flow			
	All operating pumps lose power and begin coasting down	0.0	0.0
	Reactor coolant pump underspeed trip point reached	0.8	0.8
	Rods begin to drop	1.4	1.4
	Minimum DNBR occurs	3.0	2.8
Reactor coolant pump shaft seizure (locked rotor)			
	Rotor on one pump locks	0.0	0.0
	Low flow trip point reached	0.6	0.012
	Rods begin to drop	1.06	~~0~~1.012
	Maximum clad temperature	3.1	3.2
	Maximum RCS pressure occurs	3.2	3.0

TABLE 15.3-2

SUMMARY OF RESULTS FOR LOCKED ROTOR TRANSIENTS

	Four Loops Operating Initially	Three Loops Operating Initially
Maximum Reactor Coolant System Pressure (psia)	2,548	2,593
Maximum Clad Temperature (°F) Core Hot Spot	1,762	1,716
$Zr-H_2O$ Reaction at Core Hot Spot (percent by weight)	0.24	0.2

TABLE 15.3-3

ASSUMPTIONS USED FOR THE LOCKED ROTOR ACCIDENT

	Expected	Analysis Input Parameters N-Loop	N-1 Loop
1. Power MWt	3,411	3,636(1)	3,636(1)
2. Fuel Defects (%)	0.12	0.29(2)	0.29(2)
3. Primary Coolant Concentrations	Table 11.1-2	Table 15.0-10	Table 15.0-10
4. Primary to Secondary Leak Rate (gpm)	0.009	1.0	1.0
5. Secondary Coolant Concentrations	Tables 11.1-6 and 11.1-7	Table 15.0-10	Table 15.0-10
6. Activity Released to Reactor Coolant from Failed Fuel			
a. Noble Gas	0.0	6% of gap inventory	6% of gap inventory
b. Iodine	0.0	6% of gap inventory	6% of gap inventory
7. Core and Gap Activities	Table 15.0-7	Table 15.0-7	Table 15.0-7
8. Iodine Partition Factor Prior to and during the Accident	0.01	0.01	0.01
9. Duration of Plant Cooldown by Secondary System after Accident (hr)	8	8	8
10. Steam Release from Steam Generators (lb)			
a. (0-2 hr)	543,000	543,000	543,000
b. (2-8 hr)	1,363,000	1,363,000	1,363,000

TABLE 15.3-3 (Cont)

	Expected	Analysis Input Parameters N-Loop	Analysis Input Parameters N-1 Loop
11. Feedwater Flow to Steam Generators (lb)			
a. (0-2 hr)	771,000	771,000	771,000
b. (2-8 hr)	1,505,000	1,505,000	1,505,000
12. Steam Generator Inventory prior to Accident (lb/SG)			
a. Liquid	103,000	103,000	104,000
b. Steam	8,000	8,000	7,600
13. Primary coolant mass (lb)	520,000	520,000	350,000[3]

NOTES:

1. Fuel gap activities are based on reactor power of 3,636 MWt.

2. Based on 1.0 μCi/gm I-131 dose equivalent.

3. In the N-1 loop operation analysis, the pressurizer volume has been conservatively excluded from the primary coolant.

TABLE 15.3-4

LOCKED ROTOR ACCIDENT RELEASES
(To the Environment)

Isotope	Total Release (Ci)			
	N-Loop		N-1 Loop	
	0-2 hr	0-8 hr	0-2 hr	0-8 hr
I-131	7.12E+00[1]	9.21E+01	1.30E+01	1.80E+02
I-132	6.71E+00	3.20E+01	1.30E+01	6.10E+01
I-133	1.51E+01	1.75E+02	2.80E+01	3.30E+02
I-134	6.77E+00	1.28E+01	1.30E+01	2.40E+01
I-135	1.26E+01	1.14E+02	2.40E+01	2.10E+02
Kr-83m	5.19E+01	1.39E+02	9.20E+01	2.30E+02
Kr-85m	1.68E+02	7.29E+02	3.00E+02	1.20E+03
Kr-85	4.57E+00	3.19E+01	7.90E+00	5.00E+01
Kr-87	2.06E+02	4.14E+02	3.60E+02	6.80E+02
Kr-88	4.11E+02	1.42E+03	7.30E+02	2.30E+03
Kr-89	2.99E+00	2.99E+00	5.70E+00	5.70E+00
Xe-131m	4.13E-01	2.87E+00	7.10E-01	4.50E+00
Xe-133m	2.49E+01	1.67E+02	4.30E+01	2.60E+02
Xe-133	1.04E+03	7.16E+03	1.80E+03	1.10E+04
Xe-135m	2.72E+01	5.92E+01	5.00E+01	1.10E+02
Xe-135	2.61E+02	1.47E+03	4.50E+02	2.30E+03
Xe-137	5.61E+00	5.61E+00	1.10E+01	1.10E+01
Xe-138	6.50E+01	6.57E+01	1.20E+02	1.20E+02

NOTE:

1. 7.12E+00 = 7.12×10^0.

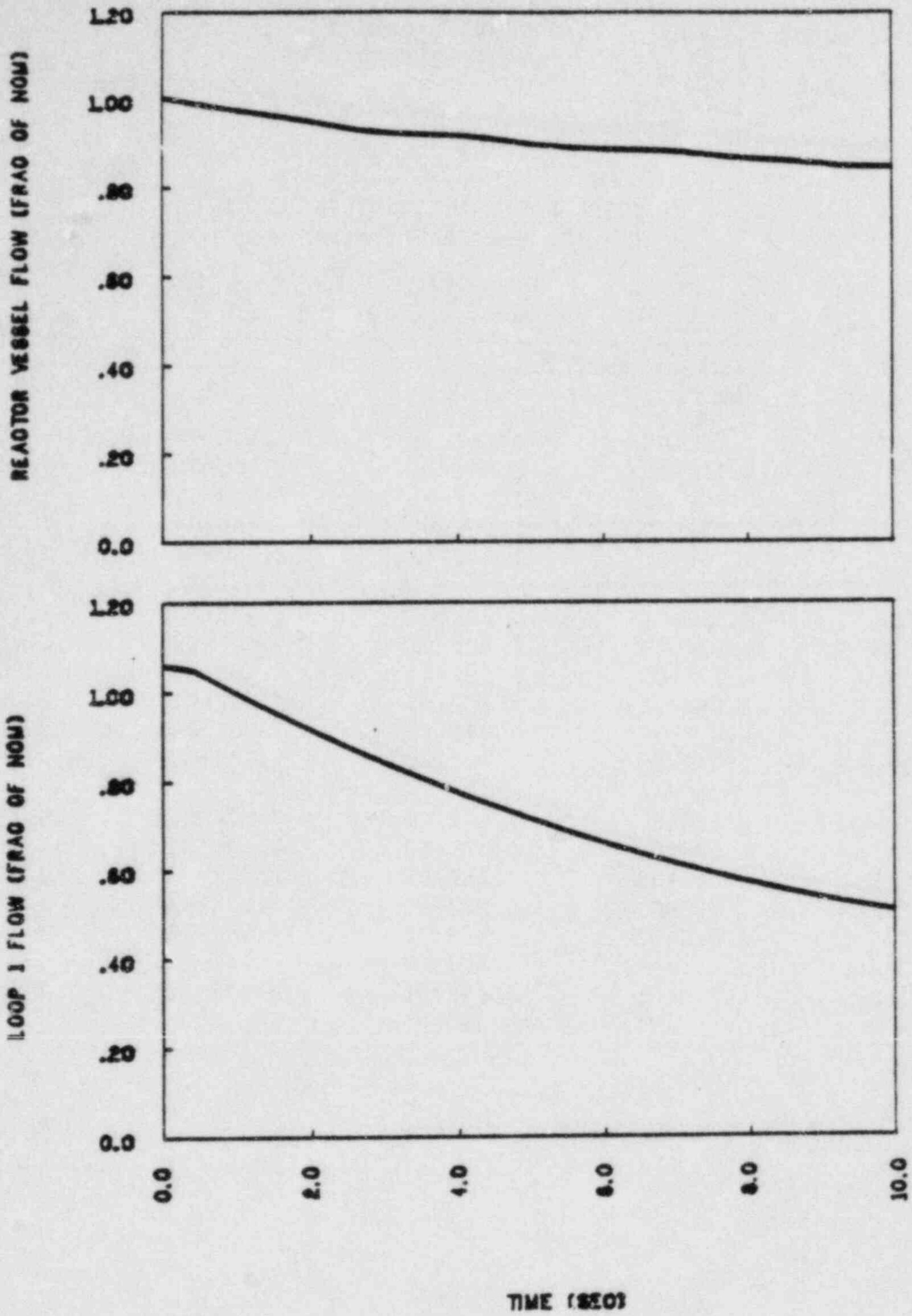


FIGURE 15.3-1a

FLOW TRAIESENTS FOR THREE LOOPS IN OPERATION. ONE PUMP COASTING DOWN N-1 LOOP OPERATION

INTERACTIVE GRAPHICS
D.CHA.STOU.0751AXX.XXX.01.013.082983

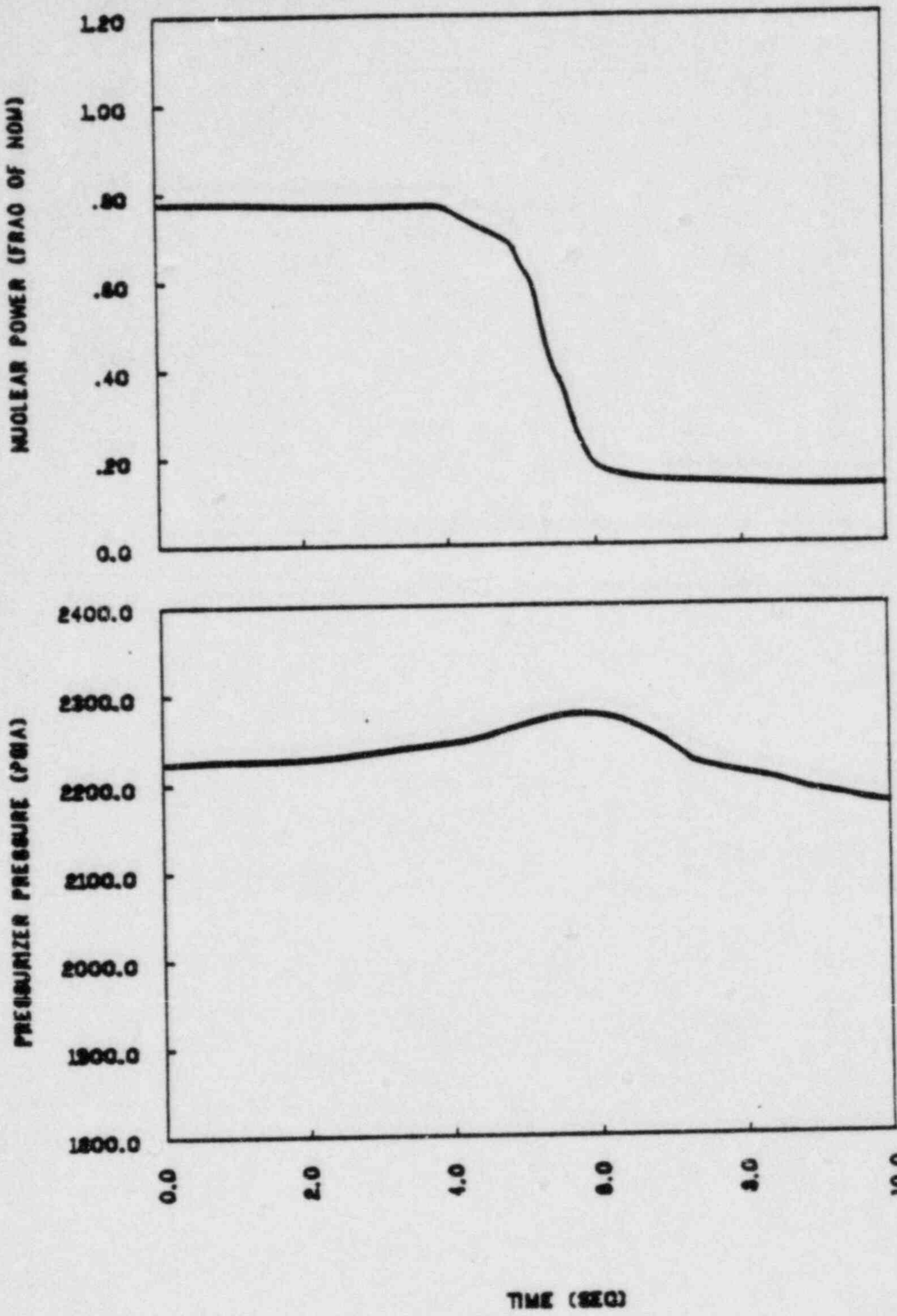


FIGURE 15.3-2a

NUCLEAR POWER AND PRESSURIZER PRESSURE TRANSIENTS FOR THREE LOOPS IN OPERATION. ONE PUMP COASTING DOWN N-1 LOOP OPERATION

INTERACTIVE GRAPHICS
D.CHA.STOU.0751AXX.XXX.01.016.082983

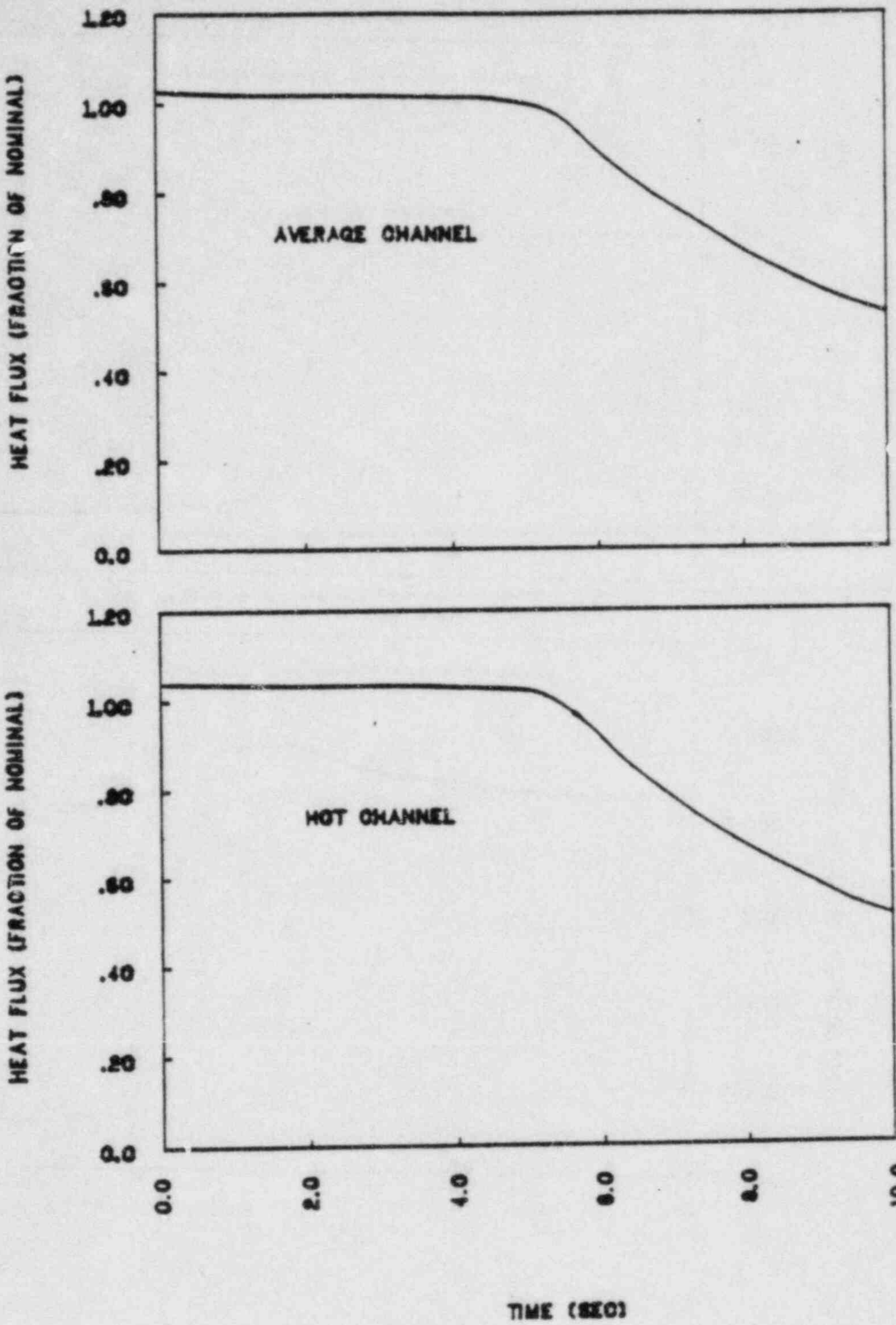


FIGURE 15.3-3a

AVERAGE AND HOT CHANNEL HEAT FLUX TRANSIENTS FOR THREE LOOPS IN OPERATION, ONE PUMP COASTING DOWN (N-1 LOOP OPERATION)

INTERACTIVE GRAPHICS
D.CHA.STOU.0751AXX.XXX.01.011.082983

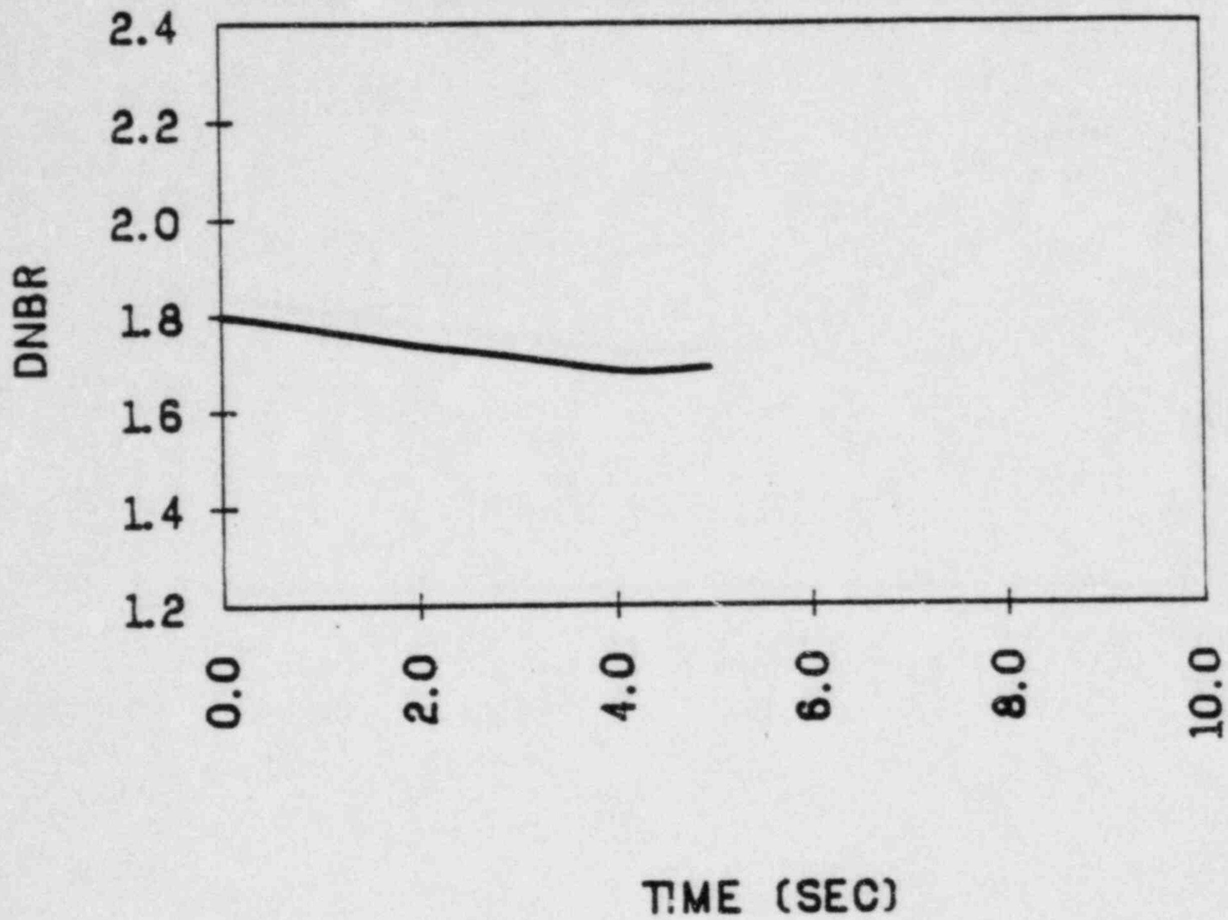


FIGURE 15.3-4.
DNBR VERUS TIME FOR 3 LOOPS IN OPERATION. ONE PUMP COASTING DOWN N-1 /LOOP OPERATION

INTERACTIVE GRAPHICS
D.CHA.MAYF.0751AXX.XXX.01.020.083083

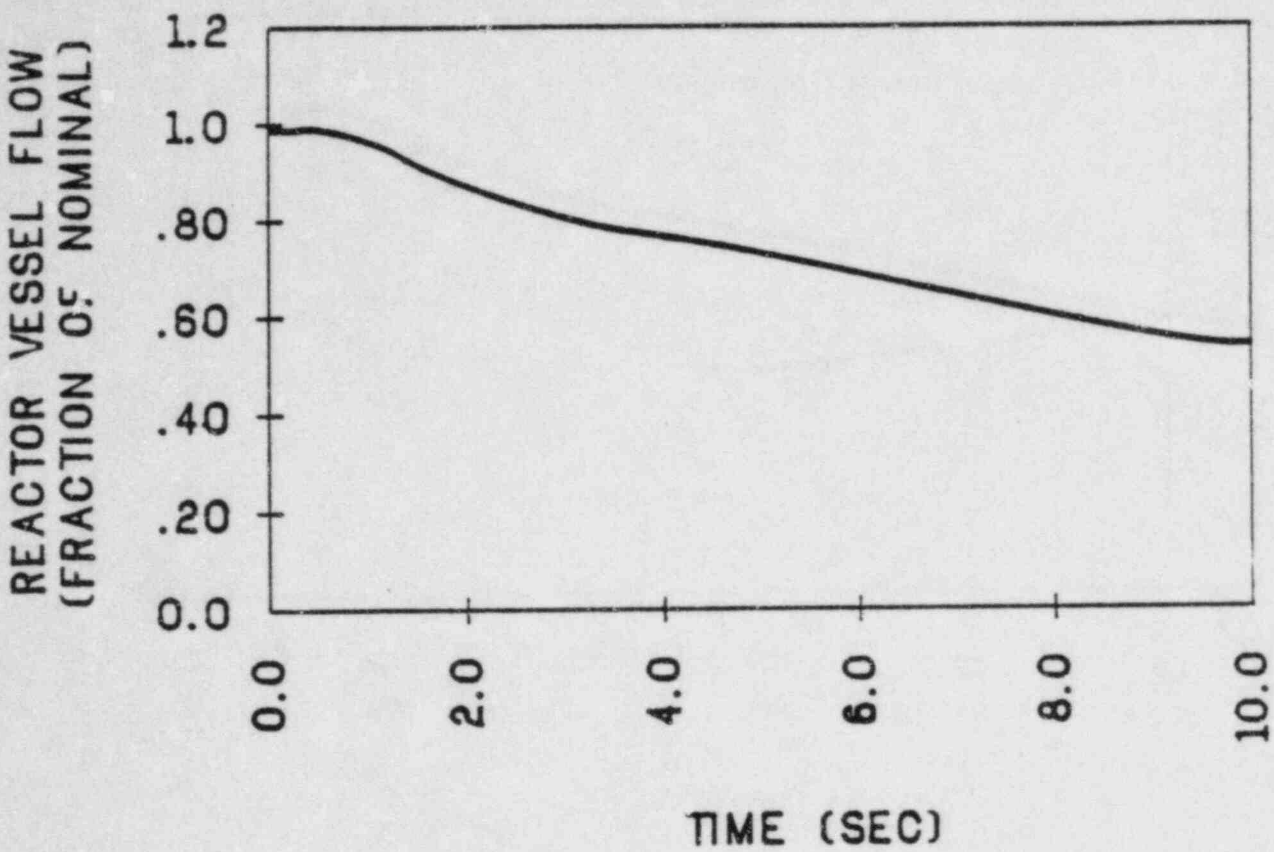


FIGURE 15.3-5A
CORE FLOW COASTDOWN 3 LOOPS IN OPERATION. 3 PUMP COASTING DOWN N-1 /LOOP OPERATION

INTERACTIVE GRAPHICS
D.CHA.MAYF.075 IXX.XXX.01.021.083083

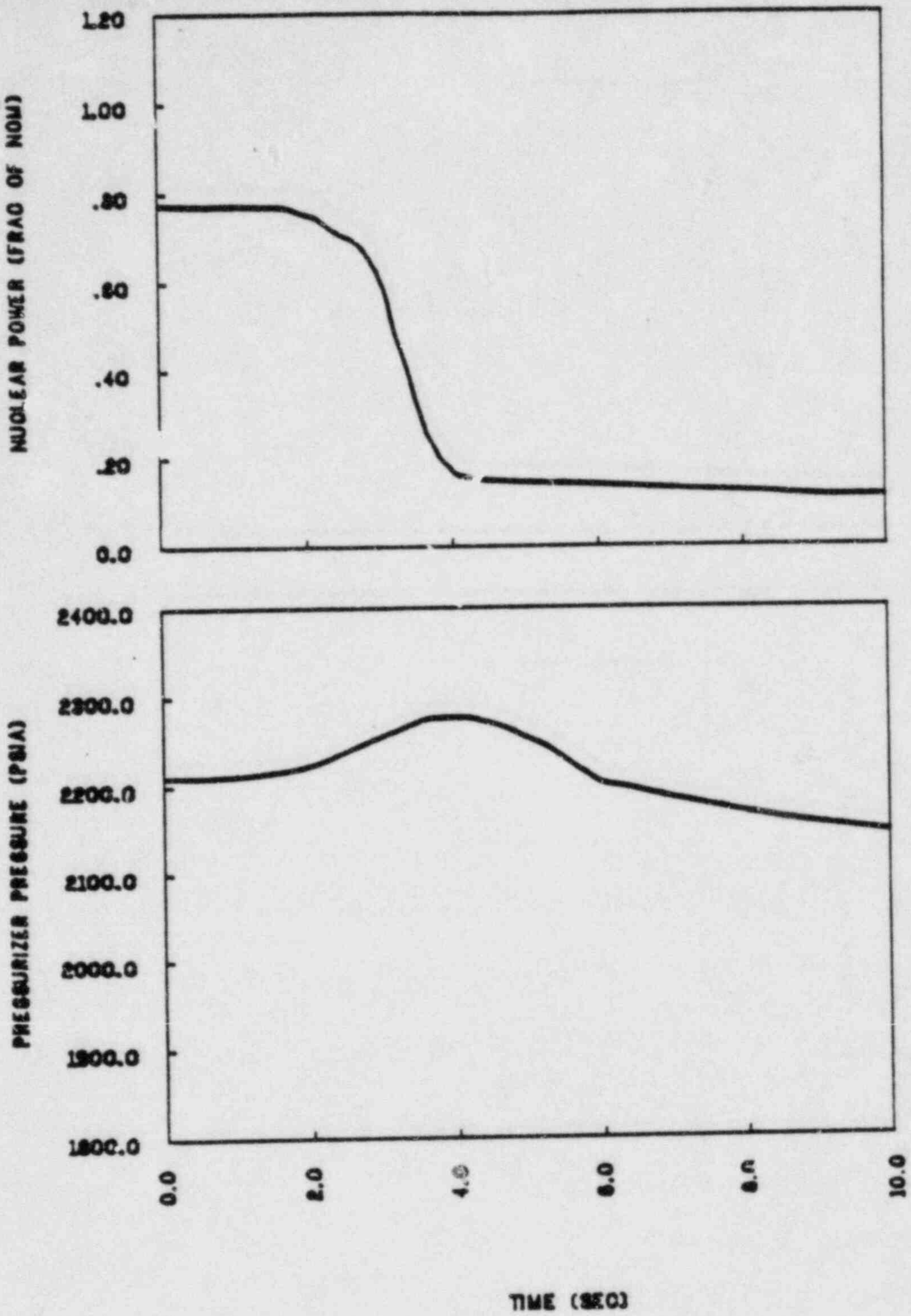


FIGURE 15.3-6a

NUCLEAR POWER AND PRESSURIZER PRESSURE TRANSIENTS FOR THREE LOOPS IN OPERATION. THREE PUMPS COASTING DOWN N-1 LOOP OPERATION

INTERACTIVE GRAPHICS
D.CHA.STOU.0751AXX.XXX.01.015.082983

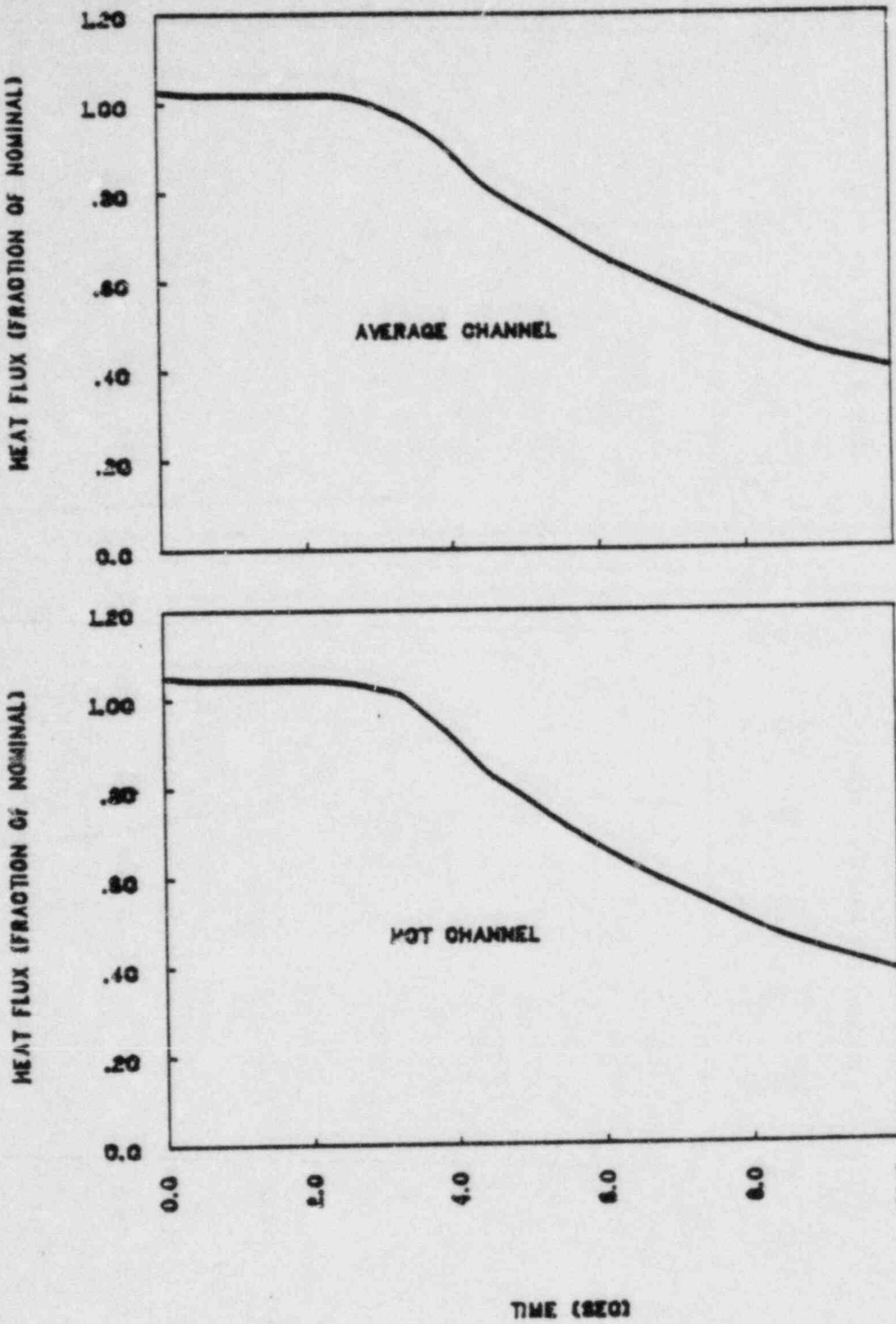


FIGURE 15.3-7a

AVERAGE AND HOT CHANNEL HEAT
HEAT TRANSIENTS FOR THREE
LOOPS IN OPERATION. THREE PUMPS
COASTING DOWN
(N-1 LOOP OPERATION)

INTERACTIVE GRAPHICS
D.CHA.STOU.0751AXX.XXX.01.012.082983

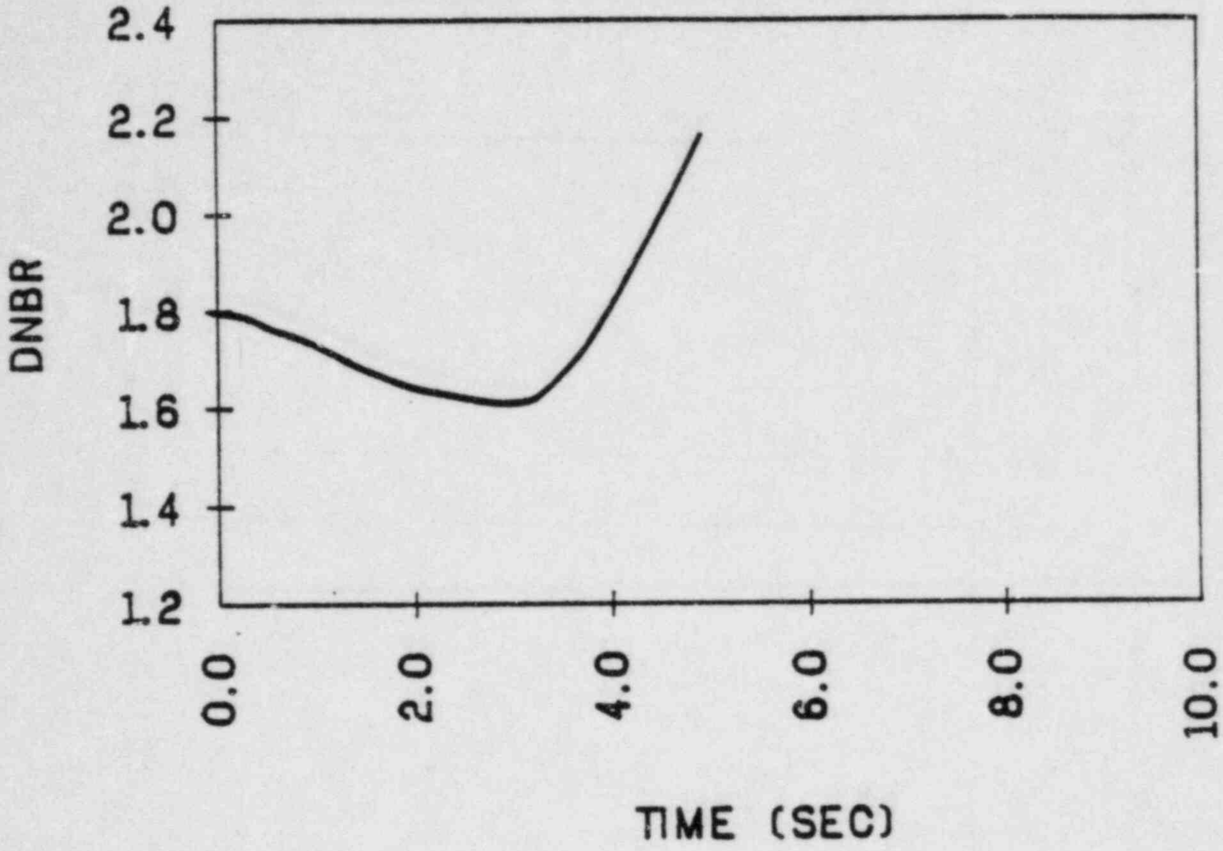


FIGURE 15.3-8A

DNBR VERSUS TIME FOR 3 LOOPS IN OPERATION, 3 PUMP COASTING DOWN N-1 /LOOP OPERATION

INTERACTIVE GRAPHICS
D.CHA.MAYF.0751AXX.XXX.01.022.083083

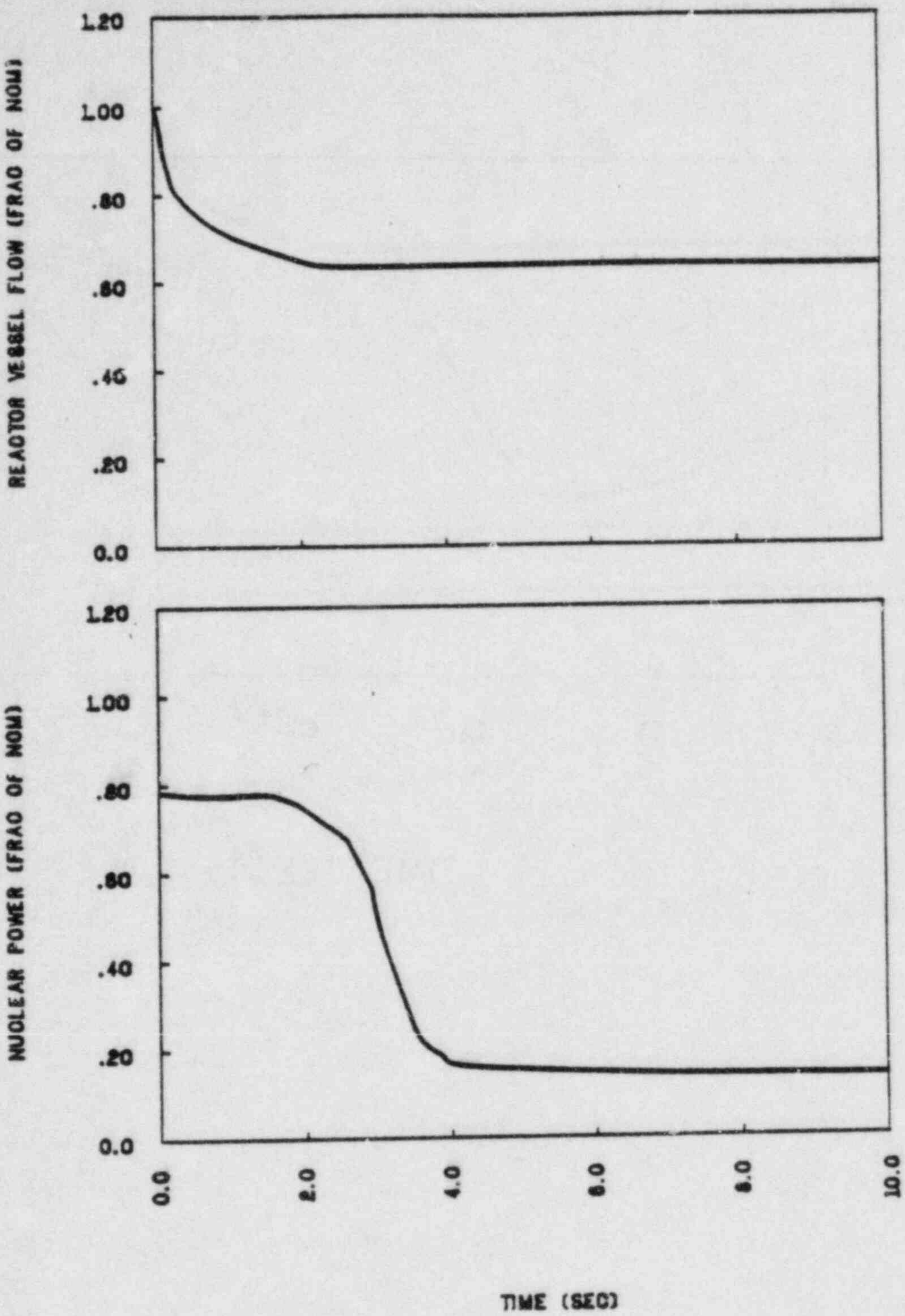


FIGURE 15.3-9a

FLOW TRANSIENTS FOR THREE LOOPS IN OPERATION, ONE LOCKED ROTOR (N-1 LOOP OPERATION)

INTERACTIVE GRAPHICS

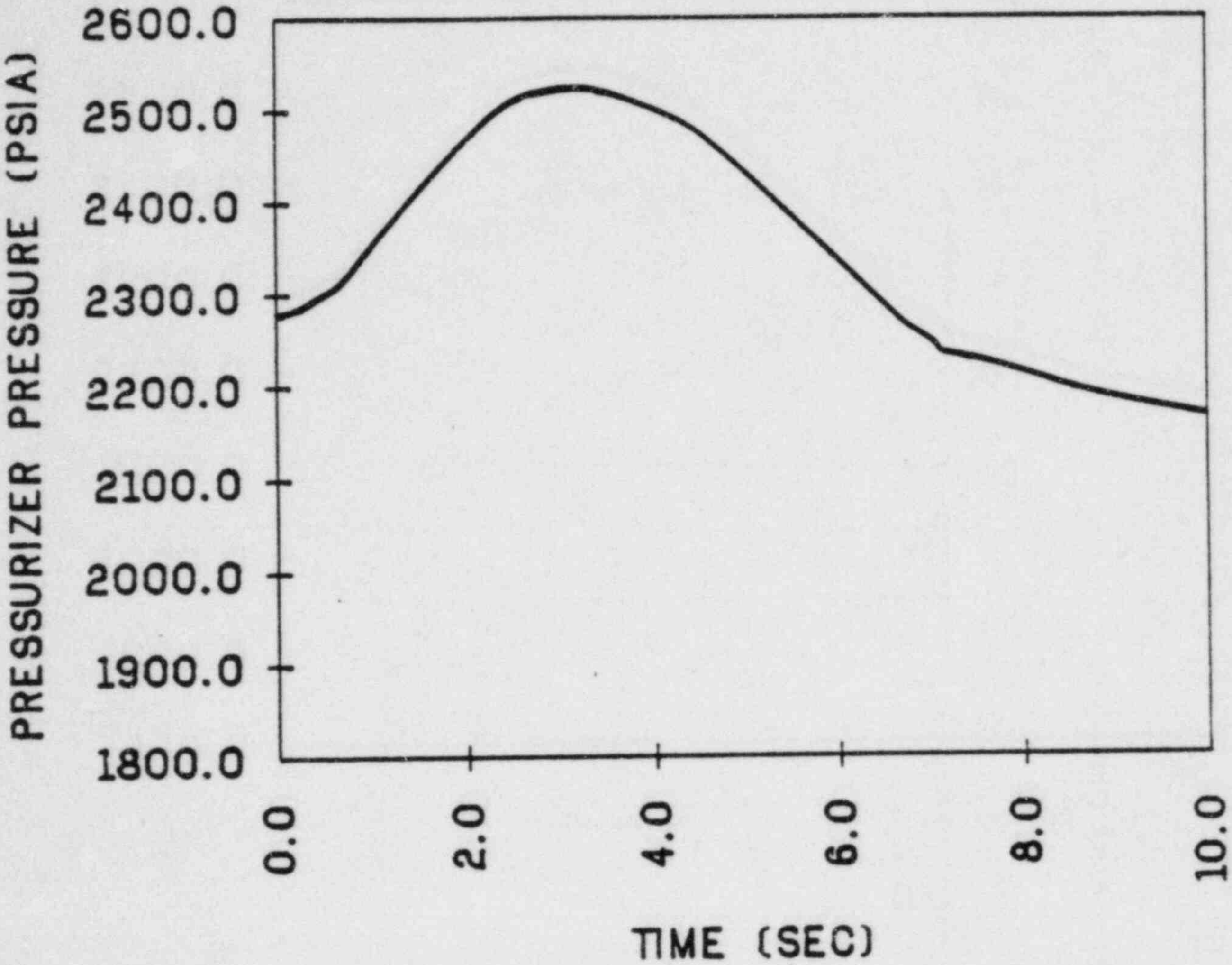


FIGURE 15.3-10A
PEAK REACTOR COOLANT PRESSURE FOR 3 LOOPS IN OPERATION, ONE LOCKED ROTOR
N-1 LOOP OPERATION

INTERACTIVE GRAPHICS
D.CHA.MAYF.0751AXX.XXX.01.024.083083

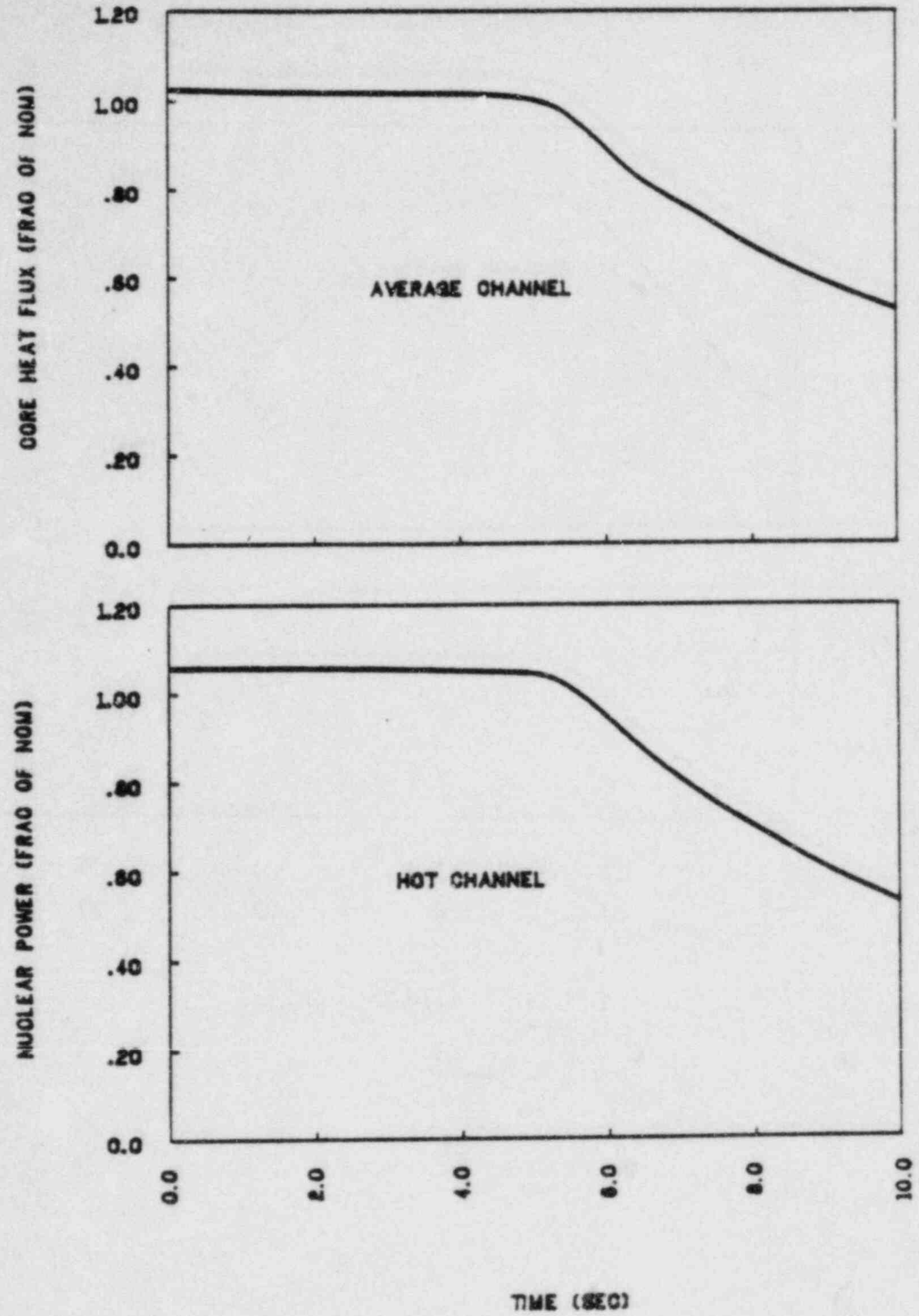


FIGURE 15.3-11a

NUCLEAR POWER AVERAGE CHANNEL AND HOT CHANNEL HEAT FLUX TRANSIENTS FOR THREE LOOPS IN OPERATION. ONE LOCKED ROTOR (N-1 LOOP OPERATION)

INTERACTIVE GRAPHICS
D.CHA.STOU.0751AXX.XXX.01.010.082983

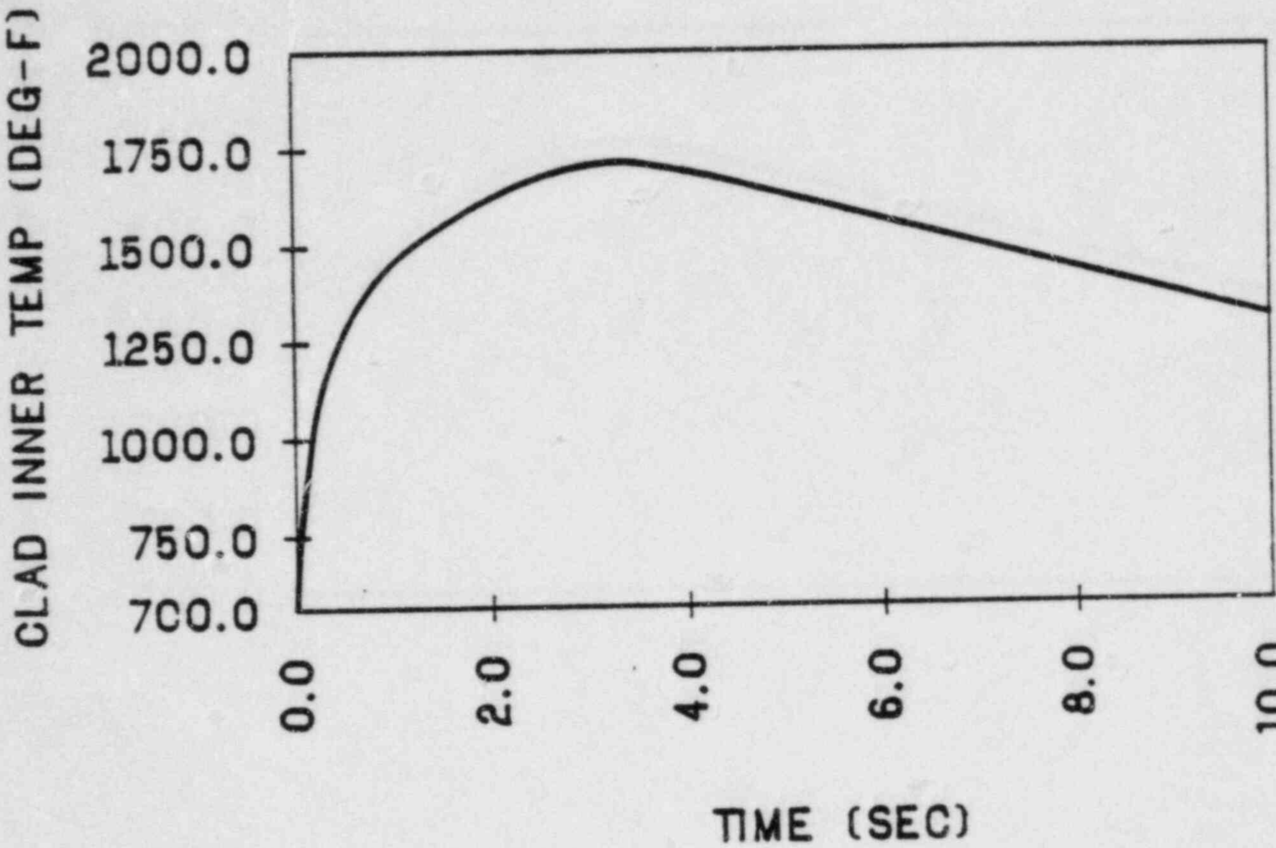


FIGURE 15.3-12A
MAXIMUM CLAD TEMPERATURE AT HOT SPOT FOR 3 LOOPS IN OPERATION. ONE LOCKED ROTOR (N-1 LOOP OPERATION)

INTERACTIVE GRAPHICS
D.CHA.MAYF.0751AXX.XXX.01.023.083083

15.4 REACTIVITY AND POWER DISTRIBUTION ANOMALIES 1.10

A number of faults have been postulated which could result in 1.11
reactivity and power distribution anomalies. Reactivity changes 1.12
could be caused by control rod motion or ejection, boron
concentration changes, or addition of cold water to the reactor
coolant system (RCS). Power distribution changes could be caused by 1.13
control rod motion, misalignment, or ejection, or by static means
such as fuel assembly mislocation. These events are discussed in 1.14
this section. Detailed analyses are presented for the most limiting 1.15
of these events.

Discussions of the following incidents are presented in Section 15.4. 1.16

1. Uncontrolled rod cluster control assembly bank withdrawal 1.18
   from a subcritical or low power startup condition.

2. Uncontrolled rod cluster control assembly bank withdrawal at 1.19
   power

3. Rod cluster control assembly misalignment. 1.20

4. Startup of an inactive reactor coolant pump at an incorrect 1.21
   temperature.

5. A malfunction or failure of the flow controller in a BWR 1.22
   loop that results in an increased reactor coolant flow rate
   (Not applicable to Millstone 3). 1.23

6. Chemical and volume control system malfunction that results 1.24
   in a decrease in the boron concentration in the reactor
   coolant. 1.25

7. Inadvertent loading and operation of a fuel assembly in an 1.26
   improper position.

8. Spectrum of rod cluster control assembly ejection accidents. 1.27

Items 1, 2, 4, and 6 above are considered to be American Nuclear 1.29
Society (ANS) Condition II events, Item 7 is an ANS Condition III 1.30
event, and Item 8 is an ANS Condition IV event. Item 3 entails both 1.31
Condition II and III events. Section 15.0.1 contains a discussion of 1.32
ANS classifications.

15.4.1 Uncontrolled Rod Cluster Control Assembly Bank Withdrawal from a Subcritical or Low Power Startup Condition 1.36

15.4.1.1 Identification of Causes and Accident Description 1.39

A rod cluster control assembly (RCCA) withdrawal accident is defined 1.40
as an uncontrolled addition of reactivity to the reactor core caused 1.42
by withdrawal of RCCAs resulting in a power excursion. Such a 1.43
transient could be caused by a malfunction of the reactor control or
rod control systems. This could occur with the reactor subcritical, 1.44

at hot zero power or at power. The "at power" case is discussed in 1.45
Section 15.4.2.

Although the reactor is normally brought to power from a subcritical 1.46
condition by means of RCCA withdrawal, initial startup procedures
with a clean core call for boron dilution. The maximum rate of 1.47
reactivity increase in the case of boron dilution is less than
assumed in this analysis (Section 15.4.6).

The RCCA drive mechanisms are wired into preselected bank 1.48
configurations which are not altered during reactor life. These 1.49
circuits prevent the RCCAs from being automatically withdrawn in
other than their respective banks. Power supplied to the banks is 1.50
controlled such that no more than two banks can be withdrawn at the
same time and in their proper withdrawal sequence. The RCCA drive 1.51
mechanisms are of the magnetic latch type and coil actuation is
sequenced to provide variable speed travel. The maximum reactivity 1.52
insertion rate analyzed in the detailed plant analysis is that
occurring with the simultaneous withdrawal of the combination of two
sequential control banks having the maximum combined worth at maximum 1.53
speed. 1.54

This event is classified as an ANS Condition II incident (a fault of 1.55
moderate frequency) as defined in Section 15.0.1.

The neutron flux response to a continuous reactivity insertion is 1.56
characterized by a very fast rise terminated by the reactivity
feedback effect of the negative Doppler coefficient. This self 1.57
limitation of the power excursion is of primary importance since it
limits the power to a tolerable level during the delay time for
protective action. Should a continuous RCCA withdrawal accident 1.58
occur, the transient will be terminated by the following automatic
features of the reactor protection system. 1.59

1. Source range high neutron flux reactor trip 2.2

   Acutated when either of two independent source range 2.4
   channels indicates a neutron flux level above a preselected 2.6
   manually adjustable setpoint. This trip function may be 2.7
   manually bypassed only after an intermediate range flux
   channel indicates a flux level above a specified level. It 2.8
   is automatically reinstated when both intermediate range
   channels indicate a flux level below a specified level.

2. Intermediate range high neutron flux reactor trip 2.11

   Actuated when either of two independent intermediate range 2.13
   channels indicates a flux level above a preselected manually 2.15
   adjustable setpoint. This trip function may be manually 2.16
   bypassed only after two out of the four power range channels
   are reading above approximately 10 percent of full power and
   is automatically reinstated when three of the four channels
   indicate a power level below this value. 2.17

3. Power range high neutron flux reactor trip (low setting) 2.20

Actuated when two out of the four power range channels 2.22
indicate a power level above approximately 25 percent of
full power. This trip function may be manually bypassed 2.23
when two out of the four power range channels indicate a 2.25
power level above approximately 10 percent of full power and
is automatically reinstated only after three out of the four
channels indicate a power level below this value. 2.27

4. Power range high neutron flux reactor trip (high setting) 2.30

Actuated when two out of the four power range channels 2.32
indicate a power level above a preset setpoint. This trip 2.33
function is always active.

5. High nuclear flux rate reactor trip 2.37

Actuated when the positive rate of change of neutron flux on 2.39
two out of four nuclear power range channels indicates a
rate above the preset setpoint. This trip function is 2.42
always active.

In addition, control rod stops on high intermediate range 2.43
flux level (one of two) and high power range flux level (one
of four) serve to discontinue rod withdrawal and prevent the
need to actuate the intermediate range flux level trip and 2.44
the power range flux level trip, respectively.

15.4.1.2 Analysis of Effects and Consequences 2.47

<u>Method of Analysis</u> 2.49

The analysis of the uncontrolled RCCA bank withdrawal from 2.51
subcritical accident is performed in three stages: first an average
core nuclear power transient calculation, then an average core heat 2.53
transfer calculation, and finally the departure from nucleate boiling 2.54
ratio (DNBR) calculation. The average core nuclear calculation is 2.55
performed using spatial neutron kinetics methods TWINKLE (WCAP-7979-A 2.56
and WCAP-8028), to determine the average power generation with time
including the various total core feedback effects, i.e., Doppler
reactivity and moderator reactivity. The average heat flux and 2.57
temperature transients are determined by performing a fuel rod
transient heat transfer calculation in FACTRAN (WCAP-7908). The 2.58
average heat flux is next used in THINC (Section 4.4) for transient
DNBR calculation.

Plant characteristics and initial conditions are discussed in 2.59
Section 15.0.3. In order to give conservative results for a startup 2.60
accident, the following assumptions are made.

1. Since the magnitude of the power peak reached during the 3.2
initial part of the transient for any given rate of
reactivity insertion is strongly dependent on the Doppler

coefficient, conservatively low values as a function of 3.4
power are used. See Section 15.0.4 and Table 15.0-2. 3.5

2. Contribution of the moderator reactivity coefficient is 3.6
negligible during the initial part of the transient because
the heat transfer time between the fuel and the moderator is
much longer than the neutron flux response time. However, 3.8
after the initial neutron flux peak, the succeeding rate of
power increase is affected by the moderator reactivity
coefficient. A highly conservative value is used in the 3.9
analysis to yield the maximum peak heat flux.

3. The reactor is assumed to be a hot zero power. This 3.11
assumption is more conservative than that of a lower initial
system temperature. The higher initial system temperature 3.12
yields a larger fuel-water heat transfer coefficient, larger 3.13
specific heats, and a less negative (smaller absolute
magnitude) Doppler coefficient, all of which tend to reduce
the Doppler feedback effect thereby increasing the neutron 3.14
flux peak. The initial effective multiplication factor is 3.15
assumed to be 1.0 since this results in the worst nuclear
power transient.

4. Reactor trip is assumed to be initiated by power range high 3.16
neutron flux (low setting). The most adverse combination of 3.17
instrument and setpoint errors, as well as delays for trip
signal actuation and RCCA release, is taken into account. A 3.18
10 percent increase is assumed for the power range flux trip
setpoint raising it from the nominal value of 25 percent to 3.19
35 percent. Since the rise in the neutron flux is so rapid, 3.20
the effect of errors in the trip setpoint on the actual time
at which the rods are released is negligible. In addition, 3.22
the reactor trip insertion characteristic is based on the
assumption that the highest worth RCCA is stuck in its fully 3.23
withdrawn position. See Section 15.0.5 for RCCA insertion 3.24
characteristics.

5. The maximum positive reactivity insertion rate assumed is 3.25
greater than that for the simultaneous withdrawal of the
combination of the two sequential control banks having the
greatest combined worth at maximum speed (45 inches/minute). 3.26
Control rod drive mechanism design is discussed in 3.27
Section 4.6.

6. The most limiting axial and radial power shapes, associated 3.28
with having the two highest combined worth banks in their
high worth position, are assumed in the DNB analysis.

7. The inital power level was assumed to be below the power 3.29
level expected for any shutdown condition (10^{-9} of nominal
power). The combination of highest reactivity insertion 3.30
rate and lowest initial power produces the highest peak heat
flux.

8. Two reactor coolant pumps are assumed to be in operation, 3.31
since this is conservative with respect to DNB.

Plant systems and equipment which are available to mitigate the 3.33
effects of the accident are discussed in Section 15.0.8 and listed in
Table 15.0-6. No single active failure in any of these systems or 3.34
equipment will adversely affect the consequences of the accident.

Results 3.37

Figures 15.4-1 through 15.4-3 show the transient behavior for the 3.39
uncontrolled RCCA bank withdrawal incident, with the accident 3.41
terminated by reactor trip at 35 percent of nominal power. The 3.43
reactivity insertion rate used is greater than that calculated for
the two highest worth sequential control banks, both assumed to be in 3.44
their highest incremental worth region. It is also greater (by more 3.45
than a factor of 10) than the maximum reactivity insertion rate of
the part length RCCAs.

Figure 15.4-1 shows the neutron flux transient. The neutron flux 3.47
does not overshoot the nominal full power value.

The energy release and the fuel temperature increases are relatively 3.48
small. The thermal flux response, of interest for DNB 3.49
considerations, is shown on Figure 15.4-2. The beneficial effect of 3.50
the inherent thermal lag in the fuel is evidenced by a peak heat flux
much less than the full power nominal value. There is a large margin 3.51
to DNB during the transient since the rod surface heat flux remains
below the design value, and there is a high degree of subcooling at
all times in the core. Figure 15.4-3 shows the response of the 3.52
average fuel and cladding temperature. The average fuel temperature 3.53
increases to a value lower than the nominal full power value. The 3.54
minimum DNBR at all times remains above the limiting value.

The calculated sequence of events for this accident is shown in 3.55
Table 15.4-1. With the reactor tripped, the plant returns to a 3.56
stable condition. The plant may subsequently be cooled down further 3.57
by following normal plant shutdown procedures.

15.4.1.3 Conclusions 3.59

In the event of a RCCA withdrawal accident from the subcritical 3.60
condition, the core and the RCS are not adversely affected, since the 4.2
combination of thermal power and the coolant temperature result in a
DNBR greater than the limiting value of 1.30. The DNBR design basis 4.4
is described in Section 4.4; applicable acceptance criteria have been
met.

15.4.1.4 Radiological Consequences 4.6

There are no radiological consequences associated with an 4.7
uncontrolled rod cluster control assembly bank withdrawal from a 4.10
subcritical or low power startup condition event since radioactivity
is retained within the fuel rods and reactor coolant system. 4.11

15.4.2 Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power 4.15

15.4.2.1 Identification of Causes and Accident Description 4.18

Uncontrolled RCCA bank withdrawal at power results in an increase in 4.19
the core heat flux. Since the heat extraction from the steam 4.21
generator lags behind the core power generation until the steam
generator pressure reaches the relief or safety valve setpoint, there
is a net increase in the reactor coolant temperature. Unless 4.23
terminated by manual or automatic action, the power mismatch and
resultant coolant temperature rise could eventually result in DNB.
Therefore, in order to avert damage to the fuel clad, the reactor 4.24
protection system is designed to terminate any such transient before
the DNBR falls below 1.30.

This event is classified as an ANS Condition II incident (a fault of 4.25
moderate frequency) as defined in Section 15.0.1.

The automatic features of the reactor protection system which prevent 4.26
core damage following the postulated accident include the following:

1. Power range neutron flux instrumentation actuates a reactor 4.28
   trip if two out of four channels exceed an overpower
   setpoint.

2. Reactor trip is actuated if any two out of four channels 4.29
   exceed an overtemperature ΔT setpoint. This setpoint is ~~4.31~~
   automatically varied with axial power imbalance, coolant
   temperature and pressure to protect against DNB.

3. Reactor trip is actuated if any two out of four channels 4.32
   exceed an overpower ΔT setpoint. This setpoint is ~~4.33~~
   automatically varied with axial power imbalance to ensure
   that the allowable heat generation rate (kW/ft) is not
   exceeded.

4. A high pressurizer pressure reactor trip actuated from any 4.34
   two out of four pressure channels, which is set at a fixed
   point. This set pressure is less than the set pressure for 4.35
   the pressurizer safety valves.

5. A high pressurizer water level reactor trip actuated from 4.36
   any two out of three level channels when the reactor power
   is above approximately 10 percent (Permissive-7).

In addition to the above listed reactor trips, there are the 4.38
following RCCA withdrawal blocks:

1. high neutron flux (one out of four power range); 4.40

2. overpower ΔT (two out of four); and ~~4.41~~

3. overtemperature ΔT (two out of four). ~~4.42~~

The manner in which the combination of overpower and overtemperature 4.44
ΔT trips provide protection over the full range of RCS conditions is ~~4.45~~
described in Chapter 7. Figures 15.0-1 and 15.0-1A present allowable 4.46
reactor coolant loop average temperature and ΔT for the design power ~~4.47~~
distribution and flow as a function of primary coolant pressure. The 4.48
boundaries of operation defined by the overpower ΔT trip and the ——
overtemperature ΔT trip are represented as "protection lines" on the ~~4.49~~
diagram. During operation with one loop out of service, the solid 4.50
state protection system setpoints for the overtemperature ΔT trip ~~4.51~~
should be manually reset consistent with the core limits for N-1 loop
operation. The protection lines are drawn to include all adverse 4.52
instrumentation and setpoint errors so that under nominal conditions
trip would occur well within the area bounded by these lines. The 4.54
utility of this diagram is in the fact that the limit imposed by any
given DNBR can be represented as a line. The DNB lines represent the 4.55
locus of conditions for which the DNBR equals 1.30. All points below 4.56
and to the left of a DNB line for a given pressure have a DNBR
greater than 1.30. The diagram shows that DNB is prevented for all 4.57
cases if the area enclosed with the maximum protection lines is not
traversed by the applicable DNBR line at any point.

The area of permissible operation (power, pressure, and temperature) 4.58
is bounded by the combination of reactor trips: high neutron flux
(fixed setpoint); high pressure (fixed setpoint); low pressure (fixed
setpoint); overpower and overtemperature ΔT (variable setpoints). ~~4.59~~

15.4.2.2 Analysis of Effects and Consequences 5.1

<u>Method of Analysis</u> 5.3

This transient is analyzed by the LOFTRAN Code (WCAP-7907). This 5.6
code simulates the neutron kinetics, RCS, pressurizer relief and 5.8
safety valves, pressurizer spray, steam generator, and steam
generator safety valves. The code computes pertinent plant variables 5.9
including temperatures, pressures, and power level. The core limits 5.10
are illustrated on Figure 15.0-1 are used as input to LOFTRAN to
determine the minimum DNBR during the transient. 5.11

Initial operating conditions are assumed at values consistent with 5.12
steady state N and N-1 loop operation. Plant characteristics and 5.13
initial conditions are discussed in Section 15.0.3. In order to 5.14
obtain conservative results for an uncontrolled rod withdrawal at
power accident, the following assumptions are made. 5.15

1. Initial reactor power and RCS temperatures are assumed to be 5.17
   at their maximum values consistent with the steady state 5.18
   full power operation, including allowances for calibration
   and instrument errors. The initial RCS pressure is assumed 5.19
   at a minimum value consistent with steady state full power
   operation, including allowances for calibration and 5.20
   instrument errors. Cases with four loops in operation and 5.21
   with three loops in operation are considered.

2. Reactivity coefficients - two cases are analyzed: 5.22

a. Minimum reactivity feedback 5.24

A least negative moderator coefficient of reactivity is 5.26
assumed corresponding to the beginning of core life. A 5.27
variable Doppler power coefficient with core power is
used in the analysis. A conservatively small (in 5.28
absolute magnitude) value is assumed.

b. Maximum reactivity feedback 5.31

A conservatively large positive moderator density 5.33
coefficient and a large (in absolute magnitude)
negative Doppler power coefficient are assumed. 5.35

3. The reactor trip on high neutron flux is assumed to be 5.37
actuated at a conservative value of 118 percent nominal full 5.38
power. For the N-1 loop cases, reactor trip is actuated 5.39
when neutron flux reaches the P-8 trip point, conservatively
assumed to be at 89 percent of nominal power. The ΔT trips ~~5.41~~
include all adverse instrumentation and setpoint errors; the
delays for trip actuation are assumed to be the maximum 5.42
values.

4. The RCCA trip insertion characteristic is based on the 5.43
assumption that the highest worth assembly is stuck in its
fully withdrawn position.

5. The maximum positive reactivity insertion rate is greater 5.44
than that for the simultaneous withdrawal of the
combinations of the two control banks having the maximum
combined worth at maximum speed. This is also much greater 5.45
than the maximum reactivity insertion rate associated with
withdrawal of a part length RCCA.

6. Cases are analyzed for operation with four loops in service 5.46
and for operation with three loops in service (N-1 loop 5.47
operation).

Plant systems and equipment which are available to mitigate the 5.49
effects of the accident are discussed in Section 15.0.8 and listed in
Table 15.0-6. No single active failure in any of these systems or 5.50
equipment will adversely offset the consequences of the accident. A 5.51
discussion of anticipated transients without trip (ATWT)
considerations is presented in WCAP-8330.

<u>Results</u> 5.54

Figures 15.4-4 through 15.4-6 show the transient response for a rapid 5.56
RCCA withdrawal incident starting from full power. Reactor trip on 5.58
high neutron flux occurs shortly after the start of the accident.
Since this is rapid with respect to the thermal time constants of the 5.60
plant, small changes in T_{avg} and pressure result and margin to DNB is ~~6.1~~
maintained. The design basis for DNBR is described in Section 4.4. 6.2

The transient response for a slow RCCA withdrawal from full power is 6.3
shown on Figures 15.4-7 through 15.4-9. Reactor trip on 6.4
overtemperature ΔT occurs after a longer period and the rise in —
temperature and pressure is consequently larger than for rapid RCCA 6.5
withdrawal. Again, the minimum DNBR is greater than 1.30. 6.6

Figure 15.4-10 shows the minimum DNBR as a function of reactivity 6.7
insertion rate from initial full power operation for minimum and 6.8
maximum reactivity feedback. It can be seen that two reactor trip 6.9
channels provide protection over the whole range of reactivity
insertion rates. These are the high neutron flux and overtemperature 6.10
ΔT channels. The minimum DNBR is never less than 1.30. ~~6.11~~

Figures 15.4-11 and 15.4-12 show the minimum DNBR as a function of 6.12
reactivity insertion rate for RCCA withdrawal incidents starting at 6.14
60 and 10 percent power, respectively. The results are similar to 6.15
the 100 percent power case, except as the initial power is decreased,
the range over which the overtemperature ΔT trip is effective is ~~6.16~~
increased. In neither case does the DNBR fall below 1.30. 6.17

The shape of the curves of minimum DNBR versus reactivity insertion 6.18
rate in the referenced figures is due both to reactor core and 6.19
coolant system transient response and to protection system action in
initiating a reactor trip. 6.20

Figures 15.4-4A through 15.4-11A show similar results for RCCA bank 6.21
withdrawal transients during plant operation with one loop out of 6.22
service (N-1 loop operation).

Referring to Figure 15.4-11, for example, it is noted that: 6.23

1. For high reactivity insertion rates (i.e., between ~ ~~6.25~~
3.0×10^{-4} Δk/sec and 1.0×10^{-3} Δk/sec) reactor trip is ~~6.26~~
initiated by the high neutron flux trip for the minimum
reactivity feedback cases. The neutron flux level in the 6.27
core rises rapidly for these insertion rates while core heat
flux and coolant system temperature lag behind due to the 6.28
thermal capacity of the fuel and coolant system fluid.
Thus, the reactor is tripped prior to significant increase 6.29
in heat flux or water temperature with resultant high
minimum DNBRs during the transient. As reactivity insertion 6.31
rate decreases, core heat flux and coolant temperatures can
remain more nearly in equilibrium with the neutron flux; 6.32
minimum DNBR during the transient thus decreases with
decreasing insertion rate.

2. The Overtemperature ΔT reactor trip circuit initiates a ~~6.33~~
reactor trip when the ΔT power measurement exceeds a —
setpoint based on measured RCS cold leg temperature and 6.34
pressure. This trip circuit is described in detail in 6.35
Chapter 7; however, it is important in this context to note
that the T_{cold} and ΔT power measurements are lead lag ~~6.36~~
compensated in order to account for lags and delays in 6.37
initiation of reactor trip.

3. With further decrease in reactivity insertion rate, the 6.38
overtemperature ΔT and high neutron flux trips become
equally effective in terminating the transient (e.g., at 6.39
$\sim$ 3.0 x 10^{-4} Δk/sec reactivity insertion rate). ~~6.40~~

As the reactivity insertion rate decreases further, nuclear 6.42
power rises more slowly and RCS temperatures increase more
before a reactor trip is initiated. As seen on 6.44
Figure 15.0-1, this results in a trip with greater margin to
the DNB limit due to the slope of the overtemperature ΔT ~~6.45~~
lines. Thus, the minimum DNBR increases for lower insertion 6.46
rates.

Figures 15.4-10 through 15.4-12 illustrate minimum DNBR calculated 6.48
for minimum amd maximum reactivity feedback.

Since the RCCA withdrawal at power incident is an overpower 6.49
transient, the fuel temperatures rise during the transient until 6.50
after reactor trip occurs. For high reactivity insertion rates, the 6.51
overpower transient is fast with respect to the fuel rod thermal time
constant, and the core heat flux lags behind the neutron flux 6.52
response. Due to this lag, the peak core heat flux does not exceed 6.53
118 percent of its nominal value (i.e., the high neutron flux trip 6.54
setpoint assumed in the analysis). Taking into account the effect of 6.55
the RCCA withdrawal on the axial core power distribution, the peak
fuel temperature will still remain below the fuel melting
temperature.

For slow reactivity insertion rates, the core heat flux remains more 6.56
nearly in equilibrium with the neutron flux. The overpower transient 6.57
is terminated by the overtemperature ΔT reactor trip before a DNB
condition is reached. The peak heat flux again is maintained below 6.58
118 percent of its nominal value. Taking into account the effect of 6.59
the RCCA withdrawal on the axial core power distribution, the peak
clad center-line temperature will remain below the fuel melting 6.60
temperature.

The reactor is tripped sufficiently fast during the RCCA withdrawal 7.1
at power transient to ensure that the ability of the primiary coolant 7.2
to remove heat from the fuel rods in not reduced. Thus, the fuel 7.3
cladding temperature does not rise significantly above its initial
value during the transient.

The calculated sequence of events for this accident is shown in 7.4
Table 15.4-1. With the reactor tripped, the plant eventually returns 7.5
to a stable condition. The plant may subsequently be cooled down 7.6
further by following normal plant shutdown procedures.

15.4.2.3 Conclusions 7.8

The high neutron flux and overtemperature ΔT trip channels provide ~~7.9~~
adequate protection over the entire range of possible reactivity 7.11
insertion rates; i.e., the minimum value of DNBR is always larger
than 1.30.

15.4.2.4 Radiological Consequences

There are only minimal radiological consequences associated with an uncontrolled rod cluster control assembly bank withdrawal at power event. The reactor trip causes a turbine trip and heat is removed from the secondary system through the steam generator power relief valves or safety valves. Since no fuel damage is postulated to occur, the radiological consequences associated with atmospheric steam release from this event are less severe than the steam line break event analyzed in Section 15.1.5.

15.4.3 Rod Cluster Control Assembly Misalignment

15.4.3.1 Identification of Causes and Accident Description

RCCA misalignment accidents include:

1. a dropped full length assembly;

2. a dropped full length assembly bank;

3. Statically misaligned full length assembly (Table 15.4-2); and

4. withdrawal of a single full length assembly.

Each RCCA has a position indicator channel which displays position of the assembly. The displays of assembly positions are grouped for the operator's convenience. Fully inserted assemblies are further indicated by a rod at bottom signal, which actuates a local alarm and control room annunciator. Group demand position is also indicated.

Full length RCCAs are always moved in preselected banks, and the banks are always moved in the same preselected sequence. Each bank of RCCAs is divided into two groups. The rods comprising a group operate in paralled through multiplexing thyristors. The two groups in a bank move sequentially such that the first group is always within one step of the second group in the bank. A definite schedule of actuation (or deactuation of the stationary gripper, movable gripper, and lift coils of a mechanism) is required to withdraw the RCCA attached to the mechanism. Since the stationary gripper, movable gripper, and lift coils associated with the four RCCAs of a rod group are driven in parallel, any single failure which would cause rod withdrawal would affect a minimum of one group. Mechanical failures are in the direction of insertion, or immobility.

The dropped assembly, dropped assembly bank, and statically misaligned assembly events are classified as ANS Condition II incidents (faults of moderate frequency) as defined in Section 15.0.1. The single RCCA withdrawal incident is classified as an ANS Condition III event, as discussed below.

No single electrical or mechanical failure in the rod control system could cause the accidental withdrawal of a single RCCA from the

inserted bank at full power operation. The operator could 7.48
deliberately withdraw a single RCCA in the control bank since this
feature is necessary in order to retrieve an assembly should one be
accidentally dropped. The event analyzed must result from multiple 7.49
wiring failures (probability for single random failure is on the
order of 10^{-4}/year (Section 7.7.2.2) or multiple deliberate operator
actions and subsequent and repeated operator disregard of event 7.50
indication. It is the position of Westinghouse that the probability 7.51
of such a combination of conditions is so low that the limiting
consequences may include slight fuel damage.

Thus, consistent with the philosophy and format of ANSI N18.2, the 7.52
event is classified as a Condition III event. By definition 7.53
"Condition III occurrences include incidents, any one of which may
occur during the lifetime of a particular plant," and "shall not
cause more than a small fraction of fuel elements in the reactor to
be damaged..." 7.54

This selection of criterion is not in violation of General Design 7.56
Criterion (GDC) 25 which states, "The protection system shall be
designed to assure that specified acceptable fuel design limits are
not exceeded for any <u>single</u> malfunction of the reactivity control 7.57
systems, such as accidental withdrawal (not ejection or dropout) of
control <u>rods</u>." (Emphasis has been added). It has been shown that 7.59
single failures resulting in RCCA bank withdrawal do not violate
specified fuel design limits. Moreover, no single malfunction can 7.60
result in the withdrawal of a single RCCA. Thus, it is concluded 8.1
that criterion established for the single rod withdrawal at power is
appropriate and in accordance with GDC 25.

A dropped assembly bank is detected by: 8.2

1. sudden drop in the core power level as seen by the nuclear 8.4
   instrumentation system;

2. asymmetric power distribution as seen on out-of-core neutron 8.5
   detectors or core exit thermocouples;

3. rod at bottom signal; 8.6

4. rod deviation alarm; 8.7

5. rod position indication. 8.8

Misaligned assemblies are detected by: 8.10

1. asymmetric power distribution as seen on out-of-core neutron 8.12
   detectors or core exit thermocouples;

2. rod deviation alarm; 8.13

3. rod position indicators. 8.14

The resolution of the rod position indicator channel is ± 5 percent 8.16
of span (± 7.2 inches). Deviation of any assembly from its group by 8.17
twice this distance (10 percent of span, or 14.4 inches) will not
cause power distributions worse than the design limits. The 8.18
deviation alarm alerts the operator to rod deviation with respect to
the group position in excess of 5 percent of span. If the rod 8.19
deviation alarm is not operable, the operator is required to take
action as required by the Technical Specifications.

If one or more rod position indicator channels should be out of 8.20
service, detailed operating instructions shall be followed to assure
the alignment of the nonindicated assemblies. The operator is also 8.21
required to take action as required by the Technical Specifications.

In the extremely unlikely event of simultaneous electrical failures 8.22
which could result in single RCCA withdrawal, rod deviation, and rod
control urgent failure would both be displayed on the plant
annunciator, and the rod position indicators would indicate the 8.23
relative positions of the assemblies in the bank. The urgent failure 8.24
alarm also inhibits automatic rod motion in the group in which it
occurs. Withdrawal of a single RCCA by operator action, whether 8.25
deliberate or by a combination of errors, would result in activation
of the same alarm and the same visual indications. Withdrawal of a 8.26
single RCCA results in both positive reactivity insertion tending to
increase in core power, and an increase in local power density in the
core area associated with the RCCA. Automatic protection for this 8.27
event is provided by the overtemperature ΔT reactor trip, although
due to the increase in local power density it is not possible in all 8.28
cases to provide assurance that the core safety limits will not be 8.29
violated.

Plant systems and equipment which are available to mitigate the 8.30
effects of the various control rod misoperations are discussed in
Section 15.0.8 and listed in Table 15.0-6. No single active failure 8.31
in any of these systems or equipment will adversely affect the
consequences of the accident.

15.4.3.2 Analysis of Effects and Consequences 8.33

1. Dropped assembly, dropped assembly bank, and statically 8.35
misaligned assembly.

8.39
<u>Method of Analysis</u>

a. One or more dropped assemblies from the same group. 8.41

For evaluation of the dropped assembly event, the 8.43
transient system response is calculated using the 8.44
LOFTRAN code. The code simulates the neutron kinetics, 8.46
reactor coolant system, pressurizer, pressurizer relief
and safety valves, pressurizer spray, steam generator, 8.47
and steam generator safety valves. The code computes 8.48
pertinent plant variables including temperatures,
pressures, and power level.

Statepoints are calculated and nuclear models are used 8.49
to obtain a hot channel factor consistent with the
primary system conditions and reactor power. By 8.51
incorporating the primary conditions from the transient
and the hot channel factor from the nuclear analysis,
the DNB design basis is shown to be met using the THINC 8.52
code. The transient response, nuclear peaking factor 8.53
analysis, and DNB design basis confirmation are
performed in accordance with the methodology described 8.54
in WCAP-10298A, Dropped Rod Methodology for Negative
Flux Rate Trip Plants.

b. Statically ~~Statistically~~ Misaligned RCCA 8.57X

Steady state power distribution are analyzed using the 8.59
computer codes as described in Table 4.1-2. The 9.1
peaking factors are then used as input to the THINC
code to calculate the DNBR.

Results 9.4

a. One or more dropped RCCAs Assemblys from the Same Group 9.6

Single or multiple dropped RCCAs within the same group 9.8
result in a negative reactivity insertion which may be 9.11
detected by the power range negative neutron flux rate
trip circuitry. If detected, the reactor is tripped 9.12
within approximately 2.5 seconds following the drop of
an RCCA. The core is not adversely affected during 9.13
this period, since power is decreasing rapidly.
Following reactor trip, normal shutdown procedures are 9.14
followed. The operator may manually retrieve the RCCA 9.15
by following approved operating procedures.

For those dropped RCCAs which do not result in a 9.16
reactor trip, power may be reestablished either by
reactivity feedback or control bank withdrawal. 9.18
Following a dropped rod event in manual rod control, 9.19
the plant will establish a new equilibrium condition.
The equilibrium process without control system 9.20
interaction is monotonic, thus removing power overshoot
as a concern, and establishing the automatic rod 9.21
control mode of operation as the limiting case.

For a dropped RCCA event in the automatic rod control 9.22
mode, the rod control system detects the drop in power
and initiates control bank withdrawal. Power overshoot 9.24
may occur due to this action by the automatic rod
controller after which the control system will insert
the control bank to restore nominal power. 9.25
Figures 15.4-13 ~~(for N-loop operation)~~ and ~~15.4-13A~~ 15.4-14 9.26
~~(for N-1 loop operation)~~ show a typical transient
response to a dropped RCCA (or RCCAs) in automatic 9.27
control for N-loop operation. Figures 15.4-13a and 15.4-14a show a typical transient response to a dropped RCCA (or RCCAs) in automatic control for N-1 loop operation. Uncertainties in the initial condition are 9.28

included in the DNB evaluation as described in WCAP-10298A. In all cases, the minimum DNBR remains 9.29
above the limit value.

b. Dropped RCCA group 9.33

A dropped RCCA group typically results in a reactivity 9.35
insertion of 400 pcm (N-loop operation) which will be 9.38
detected by the power range negative neutron flux rate
trip circuitry. The reactor is tripped within 9.39
approximately 2.5 seconds following the drop of a RCCA
bank. The core is not adversely affected during this 9.40
period, since power is decreasing rapidly. Following 9.41
reactor trip, normal shutdown procedures may
subsequently be followed to further cool down the
plant. Any operator action required to maintain the 9.42
plant in a stabilized condition will be in a time frame
in excess of 10 minutes following the incident. 9.43

b ~~c.~~ Statically misaligned RCCA 9.46

The most severe misalignment situations with respect to 9.49
DNBR at significant power levels arise from cases in
which one RCCA is fully inserted, or where bank D is 9.51
inserted to rod insertion limits with one RCCA fully
withdrawn. Multiple independent alarms, including a 9.52
bank insertion limit alarm, alert the operator well
before the postulated conditions are approached. The 9.54
bank can be inserted to its insertion limit with any
one assembly fully withdrawn without the DNBR failing
below the limiting value. 9.55

The insertion limits in the Technical Specifications 9.56
may vary from time to time depending on a number of
limiting criteria. It is preferable, therefore, to 9.58
analyze the misaligned RCCA case at full power for a
position of the control bank as deeply inserted as the 9.59
criteria on minimum DNBR and power peaking factor
(Section 4.4) will allow. The full power insertion 9.60
limits on control bank D must then be chosen to be
above that position and will usually be dictated by 10.1
other criteria. Detailed results will vary from cycle 10.2
to cycle depending on fuel arrangements.

For the RCCA misalignment shown in Table 15.4-2, with 10.3
bank D inserted to its full power insertion limit and
one RCCA fully withdrawn, DNBR does not fall below the 10.4
limit value. This case was analyzed starting at 10.5
102 percent of full power, nominal RCS pressure 6.1
-30 psia, nominal RCS average temperature +6.5°F, and 10.6 X
full thermal design RCS flow, with the increased radial
peaking factor associated with the misaligned RCCA. 10.9

DNB calculations have not been performed specifically 10.10
for assemblies missing from other banks; however, power
shape calculations have been done as required for the
RCCA ejection analysis. Inspection of the power shapes 10.11
shows that the DNB and peak kW/ft situation is less
severe than the bank D case discussed above, assuming
insertion limits on the other banks equivalent to a
bank D full-in insertion limit.

fall
For RCCA misalignments with one RCCA fully inserted, 10.12
the DNBR does not fall below the limit value. This 10.13
case is analyzed assuming the initial reactor power,
pressure, and RCS temperatures are at their nominal
values, including uncertainties (as given in 10.14
Section 15.0-3) but with the increased radial peaking
factor associated with the misaligned RCCA. 10.15

DNB does not occur for the RCCA misalignment incident 10.16
and thus the ability of the primary coolant to remove
heat from the fuel rod is not reduced. The peak fuel 10.18
temperature corresponds to a linear heat generation
rate based on the radial peaking factor penalty
associated with the misaligned RCCA (as noted in 10.19
Table 15.4-2) and the design axial power distribution.
The resulting linear heat generation is well below that 10.20
which would cause fuel melting.

Following the identification of an RCCA misalignment 10.21
condition by the operator, the operator is required to
take action as required by the plant Technical 10.22
Specifications and operating instructions
(Figures 5.4-13, 5.4-14, and 5.4-15). 10.23

2. Single RCCA Withdrawal 10.26

Method of Analysis 10.28

Power distributions within the core are calculated using the computer 10.30
codes as described in Table 4.1-2. The peaking factors are then used 10.32
by THINC to calculate the minimum DNBR for the event. The case of 10.33
the worst rod withdrawn from bank D inserted at the insertion limit,
with the reactor initially at full power, was analyzed. This 10.35
incident is assumed to occur at beginning-of-life since this results
in the minimum value of moderator temperature coefficient. This 10.37
assumption maximizes the power rise and minimizes the tendency of
increased moderator temperature to flatten the power distribution.

Results 10.40

For the single rod withdrawal event, two cases have been considered. 10.42

(a) If the reactor is in the manual control mode, continuous 10.45
withdrawal of a single RCCA results in both an increase in 10.46
core power and coolant temperature, and an increase in the 10.47

local hot channel factor in the area of the withdrawing
RCCA. In terms of the overall system response, this case is 10.48
similar to those presented in Section 15.4.2; however, the
increased local power peaking in the area of the withdrawn
RCCA results in lower minimum DNBRs than for the withdrawn 10.50
bank cases. Depending on initial bank insertion and 10.51
location of the withdrawn RCCA, automatic reactor trip may
not occur sufficiently fast to prevent the minimum core DNBR
from falling below the limit value. Evaluation of this case 10.52
at the power and coolant conditions at which the
overtemperature ΔT trip would be expected to trip the plant ~~10.53~~
shows that an upper limit for the number of rods with a DNBR
less than 1.30 is 5 percent. 10.54

(b) If the reactor is in the automatic control mode, the 10.55
multiple failures that result in the withdrawal of a single
RCCA will result in the immobility of the other RCCAs in the 10.56
controlling bank. The transient will then proceed in the 10.57
same manner as Case (a).

For the above cases a reactor trip will result, although not 10.59
sufficiently fast in all instances to prevent a minimum DNBR
in the core of less than 1.30. Following reactor trip, 10.60
normal shutdown procedures may be followed to further
cooldown the plant.

15.4.3.3 Conclusions 11.3

For cases of dropped single RCCAs or dropped banks for which the 11.4
reactor is tripped by the power range negative neutron flux rate 11.5
trip, there is no reduction in the margin to core thermal limits and 11.7
the DNB design basis, as described in Section 4.4, is met. It is 11.9
shown for all cases which do not result in reactor trip that the DNBR
remains greater than the limit value; therefore, the DNB design basis 11.10
is met.

For all cases of any RCCA fully inserted, or bank D inserted to its 11.11
rod insertion limits with any single RCCA in that bank fully 11.12
withdrawn (static misalignment), the DNBR remains greater than the
limit value. Thus, the DNB design basis as described is Section 4.4 11.13
is met.

For the case of the accidental withdrawal of a single RCCA, with the 11.14
reactor in the automatic or manual control mode and initially
operating at full power with bank D at the insertion limit, an upper
bound of the number of fuel rods experiencing DNB is 5 percent of the 11.16
total fuel rods in the core.

15.4.3.4 Radiological Consequences 11.18

The most limiting rod cluster control assembly misoperation, 11.19
accidental withdrawal of a single RCCA, is predicted to result in 11.20
limited fuel damage. The subsequent reactor and turbine trip would 11.22
result in atmospheric steam dump, assuming the condenser was not

Case D: 15.1

Case in which a Region 2 fuel assembly instead of a Region 1 assembly 15.2
is loaded near the core periphery (Figure 15.4-25).

15.4.7.3 Conclusions 15.5

Fuel assembly enrichment errors would be prevented by adminstrative 15.6
procedures implemented in fabrication.

In the event that a single pin or pellet has a higher enrichment than 15.8
the nominal value, the consequences in terms of reduced DNBR and 15.9
increased fuel and clad temperatures will be limited to the
incorrectly loaded pin or pins and perhaps the immediately adjacent 15.10
pins.

Fuel assembly loading errors are prevented by administrative 15.11
procedures implemented during core loading. In the unlikely event 15.12
that a loading error occurs, analyses in this section confirm that
resulting power distribution effects will either be readily detected 15.13
by the incore movable detector system or will cause a sufficiently 15.14
small perturbation to be acceptable within the uncertainties allowed
between nominal and design power shapes. 15.15

15.4.7.4 Radiological Consequences 15.17

There are no radiological consequences associated with inadvertent 15.18
loading and operation of a fuel assembly in an improper position 15.20
since activity is retained within the fuel rods and reactor coolant
system.

15.4.8 Spectrum of Rod Cluster Control Assembly Ejection Accidents 15.22

15.4.8.1 Identification of Causes and Accident Description 15.23

This accident is defined as the mechanical failure of a control rod 15.25
mechanism pressure housing, resulting in the ejection of a RCCA and 15.26
drive shaft. The consequence of this mechanical failure is a rapid 15.27
positive reactivity insertion together with an adverse core power
distribution, possibly leading to localized fuel rod damage. 15.28

15.4.8.1.1 Design Precautions and Protection 15.30

Certain features in the Millstone 3 pressurized water reactor are 15.31
intended to preclude the possibility of a rod ejection accident, or 15.32
to limit the consequences if the accident were to occur. These 15.34
include a sound, conservative mechanical design of the rod housings,
together with a thorough quality control (testing) program during
assembly, and a nuclear design which lessens the potential ejection
worth of RCCAs and minimizes the number of assemblies inserted at 15.36
high power levels.

Mechanical Design 15.39

The mechanical design is discussed in Section 4.6. Mechanical design 15.42
and quality control procedures intended to preclude the possibility 15.44
of a RCCA drive mechanism housing failure are listed below.

1. Each full length control rod drive mechanism housing is 15.46
completely assembled and shop tested at 4,100 psi.

2. The mechanism housings are individually hydrotested after 15.47
they are attached to the head adapters in the reactor vessel 15.48
head, and checked during the hydrotest of the completed
reactor coolant system.

3. Stress levels in the mechanism are not affected by 15.49
anticipated system transients at power, or by the the
movement of the coolant loops. Moments induced by the 15.50
design basis earthquake can be accepted within the allowable
primary stress range specified by the American Society of 15.51
Mechanical Engineers (ASME) Code, Section III, for Class 1
components.

4. The latch mechanism housing and rod travel housing are each 15.52
a single length of forged Typed 304 stainless steel. This 15.53
material exhibits excellent notch toughness at all
temperatures which wlll be encountered.

A significant margin of strength in the elastic range together with 15.55
the large energy absorption capability in the plastic range gives
additional assurance that gross failure of the housing will not
occur. The joints between the latch mechanism housing and head 15.57
adapter, and between the latch mechanism housing and rod travel
housing, are threaded joints reinforced by canopy type rod welds. 15.58
Administrative regulations require periodic inspections of these (and 15.59
other) welds.

Nuclear Design 16.2

Even if a rupture of a RCCA drive mechanism housing is postulated, 16.4
the operation of a plant utilizing chemical shim is such that the
severity of an ejected RCCA is inherently limited. In general, the 16.6
reactor is operated with the RCCAs inserted only far enough to permit
load follow. Reactivity changes caused by core depletion and xenon 16.7
transients are compensated by boron changes. Further, the location 16.8
and grouping of control RCCA banks are selected during the nuclear
design to lessen the severity of a RCCA ejection accident.
Therefore, should a RCCA be ejected from its normal position during 16.9
full power operation, only a minor reactivity excursion, at worst,
could be expected to occur. However, it may be occasionally 16.10
desirable to operate with larger than normal insertions. For this 16.11
reason, a rod insertion limit is defined as a function of power
level. Operation with the RCCAs above this limit guarantees adequate 16.12
shutdown capability and acceptable power distribution. The position 16.13
of all RCCAs is continuously indicated in the control room. An alarm 16.14

will occur if a bank of RCCAs approaches its insertion limit or if
one RCCA deviates from its bank. Operating instructions require 16.15
boration at low level alarm and emergency boration at the low-low
alarm.

<u>Reactor Protection</u> 16.18

The reactor protection in the event of a rod ejection accident has 16.20
been described in WCAP-7588. The protection for this accident is 16.23
provided by high neutron flux trip (high and low setting) and high
rate of neutron flux increase trip. These protection functions are 16.24
described in detail in Section 7.2.

<u>Effects on Adjacent Housings</u> 16.27

Disregarding the remote possibility of the occurrence of a RCCA 16.29
mechanism housing failure, investigations have shown that failure of 16.32
a housing due to either longitudinal or circumferential cracking
would not cause damage to adjacent housing. The full length control 16.34
rod drive mechanism is described in Section 3.9.4.

<u>Effects of Rod Travel Housing longitudinal Failures</u> 16.37

If a longitudinal failure of the rod travel housing should occur, the 16.39
region of the position indicator assembly opposite the break would be
stressed by the reactor coolant pressure of 2,250 psia. The most 16.42
probable leakage path would be provided by the radial deformation of
the position indicator coil assembly, resulting in the growth of
axial flow passages between the rod travel housing and the hollow
tube along which the coil assemblies are mounted. 16.43

If failure of the position indicator coil assembly should occur, the 16.44
resulting free radial jet from the failed housing could cause it to
bend and contact adjacent rod housings. If the adjacent housings 16.45
were on the periphery, they might bend outward from their bases. The 16.46
housing material is quite ductile; plastic hinging without cracking
would be expected. Housing adjacent to a failed housing, in 16.47
locations other than the periphery, would not be bent because of the
rigidity of multiple adjacent housings.

<u>Effect of Rod Travel Housing Circumferential Failures</u> 16.50

If circumferential failure of a rod travel housing should occur, the 16.52
broken-off section of the housing would be ejected vertically because
the driving force is vertical and the position indicator coil 16.54
assembly and the drive shaft would tend to guide the broken-off piece 16.55
upwards during its travel. Travel is limited by the missile shield, 16.56
thereby limiting the projectile acceleration. When the projectile 16.57
reached the missile shield it would partially penetrate the shield
and dissipate its kinetic energy. The water jet from the break would 16.58
continue to push the broken-off piece against the missile shield.

If the broken-off piece of the rod travel housing were short enough 16.59
to clear the break when fully ejected, it would rebound after impact

with the missile shield. The top end plates of the position 16.60
indicator coil assemblies would prevent the broken piece from
directly hitting the rod travel housing of a second drive mechanism.
Even if a direct hit by the rebounding piece were to occur, the low 17.1
kinetic energy of the rebounding projectile would not be expected to
cause significant damage.

Possible Consequences 17.4

From the above discussion, the probability of damage to an adjacent 17.6
housing must be considered remote. However, even if damage is 17.9
postulated, it would not be expected to lead to a more severe
transient, since RCCA's are inserted in the core in symmetric
patterns, and control rods immediately adjacent to worst ejected rods
are not in the core when the reactor is critical. Damage to an 17.11
adjacent housing could, at worst, cause that RCCA not to fail on
receiving a trip signal; however, this is already taken into account
in the analysis by assuming a stuck rod adjacent to the ejected rod.

Summary 17.14

The considerations given above lead to the conclusion that failure of 17.16
a control rod housing, due either to longitudinal or circumferential
cracking, would not cause damage to adjacent housings that would 17.18
increase severity of the initial accident. 17.19

15.4.8.1.2 Limiting Criteria 17.21

This event is classified as an ANS Condition IV incident. See 17.23
Section 15.0.1 for a discussion of ANS classifications. Due to the 17.26
extremely low probability of a RCCA ejection accident, some fuel
damage could be considered an acceptable consequence.

Comprehensive studies of the threshold of fuel failure and of the 17.27
threshold of significant conversion of the fuel thermal energy to
mechanical energy, have been carried out as part of the SPERT project
by the Idaho Nuclear Corporation (Taxelius 1970). Extensive tests of 17.29
UO_2 zirconium clad fuel rods representative of those in pressurized
water reactor type cores have demonstrated failure thresholds in the
range of 240 to 257 cal/gm. However, other rods of a slightly 17.30
different design have exhibited failures as low as 225 cal/gm. These 17.31
results differ significantly from the TREAT (Liimataninen and Testa
1966) results, which indicated a failure threshold of 280 cal/gm.
Limited results have indicated that this threshold decreases by about 17.32
10 percent with fuel burnup. The clad failure mechanism appears to 17.33
be melting for zero burnup rods and brittle fracture for irradiated
rods. Also important is the conversion ratio of thermal to 17.34
mechanical energy. This ratio becomes marginally detectable above 17.35
300 cal/gm for unirradiated rods and 200 cal/gm for irradiated rods;
catastrophic failure, (large fuel dispersal, large pressure rise)
even for irradiated rods, did not occur below 300 cal/gm. 17.38

In view of the above experimental results, criteria are applied to 17.39
ensure that there is little or no possibility of fuel dispersal in

the coolant, gross lattice distortion, or severe shock waves. These 17.40
criteria are as follows.

1. Average fuel pellet enthalpy at the hot spot below 17.42
225 cal/gm for unirradiated fuel and 200 cal/gm for
irradiated fuel.

2. Average clad temperature at the hot spot below the 17.43
temperature at which clad embrittlement may be expected
(2700°F).

3. Peak reactor coolant pressure less than that which could 17.44
cause stresses to exceed the faulted condition stress
limits.

4. Fuel melting will be limited to less than 10 percent of the 17.45
fuel volume at the hot spot even if the average fuel pellet
enthalpy is below the limits of criterion 1 above.

15.4.8.2 Analysis of Effects and Consequences 17.48

<u>Method of Analysis</u> 17.50

The calculation of the RCCA ejection transient is performed in two 17.52
stages, first an average core channel calculation and then a hot
region calculation. The average core calculation is performed using 17.54
spatial neutron kinetics methods to determine the average power
generation with time including the various total core feedback 17.55
effects, i.e., Doppler reactivity and moderator reactivity. Enthalpy 17.56
and temperature transients in the hot spot are then determined by
multiplying the average core energy generation by the hot channel
factor and performing a fuel rod transient heat transfer calculation.
The power distribution calculated without feedback is pessimistically 17.57
assumed to persist throughout the transient.

A detailed discussion of the method of analysis can be found in 17.58
WCAP-7588, Rev. 1-A.

<u>Average Core Analysis</u> 18.1

The spatial kinetics computer code, TWINKLE (WCAP-7979-A and 18.3
WCAP-8028-A) is used for the average core transient analysis. This 18.5
code uses cross sections generated by LEOPARD (WCAP-3269-26) to solve
the two group neutron diffusion theory kinetic equation in one, two,
or three spatial dimensions (rectangular coordinates) for six delayed
neutron groups and up to 2,000 spatial points. The computer code 18.9
includes a detailed multiregion, transient fuel-clad-coolant heat
transfer model for calculation of pointwise Doppler and moderator
feedback effects. In this analysis, the code is used as a one 18.10
dimensional axial kinetics code since it allows a more realistic
representation of the spatial effects of axial moderator feedback and
RCCA movement. However, since the radial dimension is missing, it is 18.11
still necessary to employ very conservative methods (described below)

of calculating the ejected rod worth and hot channel factor. Further 18.12
description of TWINKLE appears in Section 15.0.11.

Hot Spot Analysis 18.15

In the hot spot analysis, the initial heat flux is equal to the 18.17
nominal times the design hot channel factor. During the transient, 18.18
the heat flux hot channel factor is linearly increased to the
transient value in 0.1 second, the time for full ejection of the rod.
Therefore, the assumption is made that the hot spot before and after 18.20
ejection are coincident. This is very conservative since the peak 18.21
after ejection will occur in or adjacent to the assembly with the
ejected rod, and prior to ejection the power in this region will
necessarily be depressed.

The hot spot analysis is performed using the detailed fuel and clad 18.22
transient heat transfer computer code, FACTRAN (WCAP-7908). This 18.23
computer code calculates the transient temperature distribution in a
cross section of a metal clad UO_2 fuel rod, and the heat flux at the 18.24
surface of the rod, using as input the nuclear power versus time and
the local coolant conditions. The zirconium-water reaction is 18.25
explicitly represented, and all material properties are represented
as functions of temperature. A parabolic radial power distribution 18.26
is used within the fuel rod.

FACTRAN uses the Dittus-Boelter or Jens-Lottes correlation to 18.27
determine the film heat transfer before DNB, and the Bishop Sandburg-
Tong correlation (Bishop, Sandburg, and Tong 1965) to determine the
film boiling coefficient after DNB. The Bishop-Sandburg-Tong 18.30
correlation is conservatively used assuming zero bulk fluid quality.
The DNBR is not calculated; instead the code is forced into DNB by 18.31
specifying a conservative DNB heat flux. The gap heat transfer 18.32
coefficient can be calculated by the code; however, it is adjusted in
order to force the full power steady state temperature distribution
to agree with the fuel heat transfer design codes. Further 18.34
description of FACTRAN appears in Section 15.0.11.

System Overpressure Analysis 18.37

Because safety limits for fuel damage specified earlier are not 18.39
exceeded, there is little likelihood of fuel dispersal into the
coolant. The pressure surge may therefore be calculated on the basis 18.40
of conventional heat transfer from the fuel and prompt heat
generation in the coolant.

The pressure surge is calculated by first performing the fuel heat 18.42
transfer calculation to determine the average and hot spot heat flux
versus time. Using this heat flux data, a THINC (Section 4.4) 18.43
calculation is conducted to determine the volume surge. Finally, the 18.44
volume surge is simulated in a plant transient computer code. This 18.45
code calculates the pressure transient taking into account fluid
transport in the RCS and heat transfer to the steam generators. No 18.46
credit is taken for the possible pressure reduction caused by the
assumed failure of the control rod pressure housing.

15.4.8.2.2 Calculation of Basic Parameters 18.49

Input parameters for the analysis are conservatively selected on the 18.51
basis of values calculated for this type of core. The more important 18.53
parameters are discussed below. Table 15.4-3 presents the parameters 18.54
used in this analysis.

Ejected Rod Worths and Hot Channel Factors 18.57

The values for ejected rod worths and hot channel factors are 18.59
calculated using either three dimensional static methods or by a 19.1
synthesis method employing one dimensional and two dimensional
calculations. The computer codes as described in Table 4.1-2 are 19.2
used in the analysis. No credit is taken for the flux flattening 19.3
effects of reactivity feedback. The calculation is performed for the 19.4
maximum allowed bank insertion at a given power level, as determined
by the rod insertion limits. Adverse xenon distributions are 19.5
considered in the calculation to provide worst case results. 19.6

Appropriate margins are added to the ejected rod worth and hot 19.7
channel factors to account for any calculational uncertainties,
including an allowance for nuclear power peaking due to
densification.

Power distribution before and after ejection for a "worst case" can 19.8
be found in WCAP-7588, Rev. 1-A. During initial plant startup physics 19.9
testing, ejected rod worths and power distributions are measured in
the zero and full power configurations and compared to values used in
the analysis. It has been found that the ejected rod worth and power 19.10
peaking factors are consistently overpredicted in the analysis.

Reactivity Feedback Weighting Factors 19.13

The largest temperature rises, and hence the largest reactivity 19.15
feedbacks occur in channels where the power is higher than average.
Since the weight of a region is dependent of flux, these regions have 19.17
high weights. This means that the reactivity feedback is larger than 19.18
that indicated by a simple channel analysis. Physics calculations 19.19
have been carried out for temperature changes with a flat temperature
distribution, and with a large number of axial and radial temperature
distributions. Reactivity changes were compared and effective 19.20
weighting multipliers which when applied to single channel feedbacks
correct them to effective whole core feedbacks for the appropriate
flux shape. In this analysis, since a one dimensional (axial) 19.21
spatial kinetics method is employed, axial weighting is not necessary
if the initial condition is made to match the ejected rod
configuration. In addition, no weighting is applied to the moderator 19.22
feedback. A conservative radial weighting factor is applied to the 19.23
transient fuel temperature to obtain an effective fuel temperature as
a function of time accounting for the missing spatial dimension.
These weighting factors have also been shown to be conservative 19.24
compared to three dimensional analysis (Risher 1975).

Moderator and Doppler Coefficient 19.27

The critical boron concentrations at the beginning-of-life and end- 19.29
of-life are adjusted in the nuclear code in order to obtain moderator
density coefficient curves which are conservative compared to actual
design conditions for the plant. As discussed above, no weighting 19.32
factor is applied to these results.

The Doppler reactivity defect is determined as a function of power 19.33
level using a one dimensional steady state computer code with a
Doppler weighting factor of 1.0. The Doppler defect used is given in 19.34
Section 15.0.4. The Doppler weighting factor will increase under 19.35
accident conditions, as discussed above.

Delayed Neutron Fraction, β_{eff} ~~19.38~~

Calculations of the effective delayed neutron fraction (β_{eff}) ~~19.40~~
typically yield values no less than 0.70 percent at beginning-of-life 19.42
and 0.50 percent at end-of-life for the first cycle. The accident is 19.44
sensitive to β_{eff} if the ejected rod worth is equal to or greater
than β_{eff} as in zero power transients. In order to allow for future ~~19.47~~
cycles, pessimistic estimates of β_{eff} of 0.55 percent at beginning of
cycle and 0.44 percent at end of cycle were used in the analysis. 19.48

Trip Reactivity Insertion 19.51

The trip reactivity insertion assumed is given in Table 15.4-3 and 19.53
includes the effect of one stuck RCCA adjacent to the ejected rod. 19.54
These values are reduced by the ejected rod reactivity. The shutdown 19.57
reactivity was simulated by dropping a rod of the required worth into
the core. The start of rod motion occurred 0.5 seconds after the 19.58
high neutron flux trip point was reached. This delay is assumed to 19.59
consist of 0.2 seconds for the instrument channel to produce a
signal, 0.15 seconds for the trip breaker to open and 0.15 seconds
for the coil to release the rods. A curve of trip rod insertion 19.60
versus time was used which assumed that insertion to the dashpot does
not occur until 3.65 seconds after the start of fall. The choice of 20.1
such a conservative insertion rate means that there is over 1 second
after the trip point is reached before significant shutdown 20.2
reactivity is inserted into the case core. This conservatism is important 20.3 X
for hot full power accidents.

The minimum design shutdown margin available for this plant at hot 20.4
zero power (HZP) may be reached only at end-of-life in the
equilibrium cycle. This value includes an allowance for the worst 20.5
stuck rod, and adverse xenon distribution, conservative Doppler and
moderator defects, and an allowance for calculation uncertainties.
Physics calculations for this plant have shown that the effect of two 20.6
stuck RCCAs (one of which is the worst ejected rod) is to reduce the
shutdown by about an additional 1 percent Δk. Therefore, following ~~20.7~~
a reactor trip resulting from an RCCA ejection accident, the reactor
will be subcritical when the core return to HZP.

Depressurization calculations have been performed for a typical 20.8
four-loop plant, assuming the maximum possible size break (2.75-inch 20.9
diameter) located in the reactor pressure vessel head. The results 20.10
show a rapid pressure drop and a decrease in system water mass due to
the break. The safety injection system is actuated on low 20.11
pressurizer pressure within 1 minute after the break. The RCS 20.12
pressure continues to drop and reaches saturation (1,200 psi) in
about 2 to 3 minutes. Due to the large thermal inertia of primary 20.13
and secondary system, there has been no significant decrease in the
RCS temperature below no-load by this time, and the depressurization 20.14
itself has caused an increase in shutdown margin by about 0.2 percent 20.16
Δk due to the pressure coefficient. The cooldown transient could not ~~20.17~~
absorb the available shutdown margin until more than 10 minutes after
the break. The addition of borated (2,000 ppm) safety injection flow 20.18
starting 1 minute after the break is sufficient to ensure that the 20.19
core remains subcritical during the cooldown.

Reactor Protection 20.22

As discussed in Section 15.4.8.1.1, reactor protection for a rod 20.24
ejection is provided by high neutron flux trip (high and low setting)
and high rate of neutron increase trip. These protection functions 20.26
are part of the reactor trip system. No single failure of the 20.27
reactor trip system will negate the protection functions required for
the rod ejection accident, or adversely affect the consequences of
the accident.

Results 20.30

Cases are presented for both beginning-and end-of-life at zero and 20.32
full power.

1. Beginning of cycle, full power 20.36

   Control bank D was assumed to be inserted to its insertion 20.38
   limit. The worst ejected rod worth and hot channel factor 20.39
   were conservatively calculated to be 0.23 percent Δk and ~~20.41~~
   5.9, respectively. The peak hot spot clad average 20.42
   temperature was 2,081°F (1,845°F for N-1 loop operation).
   The peak hot spot fuel center temperature reached was 20.43
   4,823°F (4,159°F for N-1 loop operation).

2. Beginning of cycle, zero power 20.46

   For this condition, control bank D was assumed to be fully 20.48
   inserted and banks B and C were at their insertion limits. 20.49
   The worst ejected rod is located in control bank D and has a 20.51
   worth of 0.78 percent Δk and a hot channel factor of 11.5. ~~20.52~~
   The peak hot spot clad temperature reached 2,115°F, the fuel 20.53
   center temperature was 3,075°F.

3. End of cycle, full power 20.56

Control bank D was assumed to be inserted to its insertion 20.58
limit. The ejected rod worth and hot channel factors were 20.59
conservatively calculated to be 0.25 percent Δk and 6.4,
respectively. This resulted in a peak clad temperature of 21.2
2,101°F (1,845°F for N-1 loop operation). The peak hot spot 21.3
fuel center temperature reached melting at 4,800°F.
However, melting was restricted to less than 10 percent of 21.4
the pellet. The peak hot spot fuel center temperature for 21.5
N-1 loop operation was 4,182°F; thus, no fuel ~~meeting~~ melting was X
predicted.

4. End of cycle, zero power 21.8

The ejected rod worth and hot channel factor for this case 21.10
were obtained assuming control bank D to be fully inserted
with banks C and B at their insertion limits. The results 21.13
were 0.90 percent Δk and 20.00, respectively. The peak clad ~~21.14~~
and fuel center temperatures were 2,454 and 3,405°F. The 21.15
Doppler weighting factor for this case is significantly
higher than for the other cases due to the very large
transient hot channel factor. 21.16

A summary of the cases presented is given in Table 15.4-3. Results 21.19
for the cases with N-1 loops in operation are also presented in
Table 15.4-3. The nuclear power and hot spot fuel and clad 21.20
temperature transients for the worst cases (beginning-of-life, full
power) and also for end-of-life, full power, and zero power are 21.21
presented on Figures 15.4-26 through 15.4-29 for N-loop.
Figures 15.4-26A through 15.4-29A illustrate N-1 loop operation. 21.22

The calculated sequence of events for the worst case rod ejection 21.23
accidents, as shown on Figures 15.4-26 through 15.4-29, are presented 21.24
in Table 15.4-1. For all cases, reactor trip occurs very early in 21.25
the transient, after which the nuclear power excursion is
terminated. As discussed previously in Section 15.4.8.2.2, the 21.26
reactor will remain subcritical following reactor trip.

The ejection of an RCCA constitutes a break in the RCS, located in 21.27
the reactor pressure vessel head. The effects and consequences of 21.28
loss of coolant accidents are discussed in Section 15.6.5. Following 21.29
the RCCA ejection, the operator would follow the same emergency
instructions as for any other loss-of-coolant accident to recover 21.30
from the event.

Fission Product Release 21.33

It is assumed that fission products are released from the gaps of all 21.35
rods entering DNB. In all cases considered, less than 10 percent of 21.37
the rods entered DNB based on a detailed three dimensional THINC
analysis (WCAP-7588, Rev. 1-A). Although limited fuel melting at the 21.38
hot spot was predicted for the beginning-of-life and the full power
cases, in practice melting is not expected since the analysis 21.39

conservatively assumed that the hot spots before and after ejection were coincident. 21.41

Pressure Surge 21.44

A detailed calculation of the pressure surge for an ejection worth of 21.46
one dollar at beginning-of-life, hot full power, indicates that the 21.48
peak pressure does not exceed that which would cause stress to exceed
the faulted condition stress limits (WCAP-7588, Rev. 1-A). Since the 21.49
severity of the present analysis does not exceed the "worst case" analysis, the accident for this plant will not result in an excessive pressure rise or further damage to the RCS.

Lattice Deformations 21.52

A large temperature gradient will exist in the region of the hot 21.54
spot. Since the fuel rods are free to move in the vertical 21.55
direction, differential expansion between separate rods cannot 21.57
produce distortion. However, the temperature gradients across 21.58
individual rods may produce a differential expansion tending to bow the midpoint of the rods toward the hotter side of the rod.
Calculations have indicated that this bowing would result in a 21.59
negative reactivity effect at the hot spot since Westinghouse cores are under-moderated, and bowing will tend to increase the under-
moderation, at the hot spot. Since the 17 x 17 fuel design is also 22.1
under-moderated, the same effect would be observed. In practice, no 22.2
significant bowing is anticipated, since the structural rigidity of the core is more than sufficient to withstand the forces produced.
Boiling in the hot spot region would produce a net flow away from 22.3
that region. However, the heat from the fuel is released to the 22.4
water relatively slowly, and it is considered inconceivable that cross flow will be sufficient to produce significant lattice forces.
Even if massive and rapid boiling, sufficient to distort the lattice, 22.5
is hypothetically postulated, the large void fraction in the hot spot region would produce a reduction in the total core moderator to fuel
ratio, and a large reduction in this ratio at the hot spot. The net 22.7
effect would therefore be a negative feedback. It can be concluded 22.8
that no conceivable mechanism exists for a net positive feedback
resulting from lattice deformation. In fact, a small negative 22.9
feedback may result. The effect is conservatively ignored in the 22.10
analysis.

15.4.8.3 Conclusions 22.12

Even on a pessimistic basis, the analyses indicate that the described 22.13
fuel and clad limits are not exceeded. It is concluded that there is 22.14
no danger of sudden fuel dispersal into the coolant. Since the peak 22.16
pressure does not exceed that which would cause stresses to exceed the faulted condition stress limits, it is concluded that there is no
danger of further consequential damage to the RCS. The analyses 22.17
have demonstrated that upper limit in fission product release as a result of a number of fuel rods entering DNB amounts to 10 percent.

The RCS integrated break flow to containment following a rod ejection 22.18
accident is shown on Figure 15.4-30.

15.4.8.4 Radiological Consequences 22.20

The radiological consequences of a postulated rod ejection accident 22.21
incorporate the fuel failure modes described in Regulatory 22.22
Guide 1.77.

To evaluate the radiological consequences of a control rod ejection 22.23
accident, it is assumed that 10 percent of the fuel rods experience 22.27
clad damage, thereby releasing their respective gap activities to the
reactor coolant. The gap activity is assumed to be released 22.28
instantaneously into the containment atmosphere via the break in the
reactor vessel head. In addition, it is further postulated that 22.30
0.25 percent of the core fuel experiences melting resulting in
100 percent of the noble gases and 25 percent of iodines in the 22.31
fraction of melted fuel to be available for release from the 22.32
containment. The releases to the environment are assumed to take 22.33
place from the secondary system and from leakage of the containment 22.34
building until such time, respectively, that the secondary system
pressure decreases below relief valve actuation, and the containment 22.35
building pressure is reduced to subatmospheric conditions. Activity 22.36
released from the secondary system is derived from the technical
specification primary to secondary leakage of reactor coolant 22.37
containing activity associated with technical specification fuel
defects, releases from fuel with clad damage, and 100 percent of the 22.38
noble gases and 50 percent of the iodine contained in this fraction
of fuel assumed to have melted. Releases from the secondary side are 22.41
evaluated assuming coincident loss of offsite power. Pertinent 22.42
parameters used to describe the secondary side releases are presented
in Tables 15.4-4 and 15.4-6.

Following initiation of the rod ejection accident, approximately one 22.43
minute will elapse before actuation of the SLCRS from the SIS signal. 22.44
The SLCRS is estimated to become effective in one minute. 22.45

The result is that the activity in containment which leaks at the 22.46
design basis leak rate is assumed to be an unfiltered ground level 22.47
release for the first two minutes of the accident. Between two 22.48
minutes and one hour following the onset of the accident only the
secondary containment bypass leakage is uncollected and released 22.49
unfiltered to the environment at ground level, whereas the design
basis leakage from the primary containment is collected and filtered 22.50
before discharge to the environment.

Of the collected leakage, a fraction is discharged via the SLCRS, 22.51
HEPA, and charcoal filter system to the Unit 1 stack (Section 6.2.3). 22.52

However, for purposes of conservatism, all collected leakage is 22.53
assumed to exhaust via a release point located above the turbine 22.54
building. This assumption is made as a result of the simultaneous 22.55
operation of the charging pump ventilation supply and exhaust system 22.56
which may entrap and filter some fraction of containment leakage as

described in Section 9.4.3. This effluent is analyzed as a ground 22.57
level release.

After one hour, the containment becomes subatmospheric and all 22.58
releases from the containment cease. A summary of the releases is 22.59
presented in Table 15.4-7. The releases, together with the 22.60
atmospheric dispersion factors listed in Table 15.0-11, are used to
compute the doses to the EAB (0-2 hr) and LPZ (0-8 hr). 23.2

The radiological consequences of a postulated rod ejection accident 23.3
are analyzed (for both N-loop and N-1 loop operation) with the 23.4
information contained in Regulatory Guide 1.77 and the Standard 23.5
Review Plan 15.4.8. For the N-1 loop analysis, it is assumed that 23.6
the plant had been in N-loop operation at full power sufficiently
long to achieve equilibrium core activities and coolant 23.7
concentrations. The plant then began N-1 loop operation, shortly 23.8
after which the rod ejection accident occurred. The calculated dose 23.9
results (for both the N-loop and the N-1 loop analyses) described in
Table 15.0-8 for the rod ejection accident e presented separately 23.10
for the releases from the containment building and the releases via 23.11
the secondary system.

The radiological consequences of the postulated rod ejection accident 23.12
are well within the guidelines of 10CFR100; i.e., 75 Rem to the 23.13
thyroid and 6 Rem to the whole body.

15.4.9 References for Section 15.4 23.16

Bishop, A.A.; Sandburg, R.O.; and Tong, L.S., 1965. Forced 23.18
Convection Heat Transfer at High Pressure After the Critical Heat
Flux. ASME 65-HT-31.

Liimataninen, R.C. and Testa, F.J. 1966. Studies in TREAT of 23.21
Zircaloy-2-Clad, UO_2 Core Simulated Fuel Elements. ANL-7225, 23.22
January - June 1966, p. 177.

WCAP-3269-26, 1963. Barry, R.F. LEOPARD - A Spectrum Dependent Non- 23.25
spatial Depletion Code for the IBM-7094.

WCAP-7213-A (Proprietary) and WCAP-7758-A (Non-proprietary), 1975. 23.26
Barry, R.F. and Altomare, S. The TURTLE 24.0 Diffusion Depletion 23.28
Code.

WCAP-7588, Revision 1-A, 1975. Risher, D.H., Jr. An Evaluation of 23.31
the Rod Ejection Accident in Westinghouse Pressurized Water Reactors
Using Spatial Kinetics Methods.

WCAP-7907, 1972. Burnett, T.W.T. et al. LOFTRAN Code Description. 23.34

WCAP-7908, 1972. Hargrove, H.G. FACTRAN - A Fortran-IV Code for 23.37
Thermal Transients in a UO_2 Fuel Rod.

WCAP-7979-A (Proprietary) and WCAP-8028-A (Non-proprietary), 1975. 23.38
Risher, D. H., Jr. and Barry, R. F. TWINKLE - A Multi-Dimensional Neutron Kinetics Computer Code. 23.40

WCAP-8330, 1974. Westinghouse Anticipated Transients Without Trip Analysis. 23.42

Taxelius, T. G. (Ed) 1970. Annual Report- Spert Project, October, 1968, September, 1969. Idaho Nuclear Corporation IN-1370. 23.45 23.46

TABLE 15.4-1

TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH CAUSE REACTIVITY AND POWER DISTRIBUTION ANOMALIES

Accident	Event	N-Loop Time (Sec)	N-1 Loop Time (Sec)
Uncontrolled RCCA bank withdrawal from a subcritical or low power startup condition	Initiation of uncontrolled rod withdrawal from 10^{-9} of nominal power	0.0	
	Power range high neutron flux low setpoint reached	10.2	
	Peak nuclear power occurs	10.4	
	Rods begin to fall into core	10.7	
	Peak heat flux occurs	11.9	
	Minimum DNBR occurs	12.0	
	Peak average clad temperature occurs	12.2	
	Peak average fuel temperature occurs	12.5	
Uncontrolled RCCA bank withdrawal at power			
1. Case A	Initiation of uncontrolled RCCA withdrawal at a high reactivity insertion rate (70 pcm/sec)	0	0.0
	Power range high neutron flux high trip point reached	4.32	4.23
	Rods begin to fall into core	4.82	4.43
	Minimum DNBR occurs	5.2	5.2
2. Case B	Initiation of uncontrolled RCCA withdrawal at a small reactivity insertion rate (3 pcm/sec)	0	0.0

TABLE 15.4-1 (Cont)

Accident	Event	N-Loop Time (Sec)	N-1 Loop Time (Sec)
	Overtemperature ΔT, reactor trip signal initiated	209.68	246.54
	Rods begin to fall into core	211.68	248.54
	Minimum DNBR occurs	212.1	249.1
Startup of an inactive reactor coolant loop at an incorrect temperature and boron concentration	Initiation of pump startup	0.0	
	Power reaches high nuclear flux trip setpoint	961.1	
	Rods begin to drop	961.6	
	Minimum DNBR occurs	961.1	
	Shutdown margin is lost	3361	
CVCS malfunction that results in a decrease in the boron concentration in the reactor coolant	To be provided later		
Boron dilution event sequence to be provided later.			
	Shutdown is lost (if dilution continues after trip)	>3600	
Rod cluster control assembly ejection accident			
1. End-of-life, full power	Initiation of rod ejection	0.0	0.0
	Power range high neutron flux setpoint reached	0.035	0.05

TABLE 15.4-1 (Cont)

Accident	Event	N-Loop Time (Sec)	N-1 Loop Time (Sec)
	Peak nuclear power occurs	0.14	0.14
	Rods begin to fall into core	0.54	0.55
	Peak fuel average temperature occurs	1.92	2.00 1.96
	Peak clad temperature occurs	2.04	2.06
	Peak heat flux occurs	2.076	2.12
2. End-of-life, zero power	Initiation of rod ejection	0.0	0.0
	Power range high neutron flux low setpoint reached	0.16	0.04
	Peak nuclear power occurs	0.19	0.14
	Rods begin to fall into core	0.66	0.54
	Peak clad temperature occurs	1.26	2.04 0.06
	Peak heat flux occurs	1.29	2.12
	Peak average fuel temperature occurs	1.35	2.00

TABLE 15.4-3

PARAMETERS USED IN THE ANALYSIS OF THE ROD CLUSTER CONTROL ASSEMBLY

	N-Loop				N-1 Loop	
Time in Life	Beginning	Beginning	End	End	Beginning	End
Power level (percent)	102	0	102	0	77	77
Ejected rod worth (percent ΔK)	0.23	0.78	0.25	0.90	0.23	0.25
Delayed neutron fraction (percent)	0.55 0.55	0.55 0.44	0.44	0.44	0.55	0.44
Feedback reactivity weighting	1.3	2.07	1.0	3.6	1.3	1.3
Trip reactivity (percent ΔK)	5.0	2.0	4.0	2.0	3.0	3.0
F_q before rod ejection	2.55	--	2.55	--	2.55	2.55
F_q after rod ejection	5.90	11.5	6.4	20.0	5.90	6.4
Number of operational pumps	4	2	4	2	3	3
Max. fuel pellet average temperature (°F)	3638	2719	3677	3107	3111	3119
Max. fuel center temperature (°F)	4823	3075	4839	3405	4159	4182
Max. clad average temperature (°F)	2141	2094	2060	2430	1812	1815
Max. fuel stored energy (cal/gm)	156	111	158	129	128	128

TABLE 15.4-4

PARAMETERS USED IN ROD EJECTION ACCIDENT ANALYSIS

	Expected	Analysis Input Parameters: N-Loop	Analysis Input Parameters: N-1 Loop
1. Core thermal power (MWt)	3,411	3,636(1)	3,636(1)
2. Containment free volume (ft^3)	2.32×10^6	2.32×10^6	2.32×10^6
3. Primary coolant concentrations	Table 11.1-2	Table 15.0-10	Table 15.0-10
4. Primary to secondary leak rate (gpm)	0.009	1.0	1.0
5. Secondary coolant concentration	Tables 11.1-6 & 11.1-7	Table 15.0-10	Table 15.0-10
6. Fuel defects (%)	0.12	0.29(2)	0.29(2)
7. Failed fuel as a result of the accident (%)	0.0	10.0	10.0
8. Core and gap activity	Table 15.0-7	Table 15.0-7	Table 15.0-7
9. Quantity of fuel in the core which melts as a result of the accident (%)	0.0	0.25	0.25
10. Quantity of radionuclides from the melted fuel available for release from the containment (%)			
a. Iodine	0.0	25.0	25.0
b. Noble gases	0.0	100	100

TABLE 15.4-4 (Cont)

		Analysis Input Parameters	
	Expected	N-Loop	N-1 Loop
11. Quantity of radionuclides from the melted fuel available for release from the secondary side via primary-to-secondary leakage (%)			
a. Iodines	0.0	50.0	50.0
b. Noble gases	0.0	100	100
12. Iodine partition factor in steam generator prior to and during accident	0.01	0.01	0.01
13. Offsite power	Available	Lost	Lost
14. Steam dump from relief valves (lb)	0.0	40,604	40,604
15. Duration of dump from relief valves (sec)	0.0	125.0	125.0
16. Containment leak rate during the first hour following the occurrence of the accident (% per day)	0.9	0.9	0.9
17. Bypass leakage (fraction of containment leakage)	0.0103	0.0103	0.0103
18. Time between accident and equalization of primary and secondary pressures (sec)	140.0	140.0	140.0
19. Time estimated for SLCRS to become effective (min)	1.0	1.0	1.0
20. Duration of leakage from containment (min)	60.0	60.0	60.0
21. Iodine removal filter efficiency (%)	95.0	95.0	95.0

TABLE 15.4-4 (Cont)

		Analysis Input Parameters		
	Expected	N-Loop	N-1 Loop	
22. Steam generator contents (lb/SG)				3.23 3.24
a. Steam	8,000	8,000	7,600	3.26
b. Liquid	103,000	103,000	104,000	3.27
23. Primary coolant mass (lb)	520,000	520,000	350,000[3]	3.30

NOTES: 3.33

1. Fuel gap activities are based on reactor power of 3,636 MWt. 3.37

2. Based upon 1.0 Ci/gm I-131 dose equivalent. 3.38

3. In the N-1 loop operation analysis, the pressurizer volume has been conservatively excluded from the primary coolant. 3.40

TABLE 15.4-7

RADIOACTIVITY RELEASED TO THE ENVIRONMENT AS A RESULT OF A ROD EJECTION ACCIDENT

N-LOOP

Isotope	Containment Building Releases (Ci)	Main Steam Valve Building Releases (Ci)	Total Releases (Ci)
Kr-83m	6.27E+01[1]	2.97E+00	6.57E+01
Kr-85m	1.74E+02	7.45E+00	1.81E+02
Kr-85	4.17E+00	1.65E-01	4.34E+00
Kr-87	2.81E+02	1.44E+01	2.95E+02
Kr-88	4.53E+02	2.03E+01	4.73E+02
Kr-89	5.07E+01	2.35E+01	7.42E+01
Xe-131m	3.94E-01	1.50E-02	4.09E-01
Xe-133m	2.36E+01	9.22E-01	2.45E+01
Xe-133	9.67E+02	3.84E+01	1.01E+03
Xe-135m	9.49E+02	1.01E+01	9.59E+02
Xe-135	3.21E+02	1.01E+01	3.31E+02
Xe-137	7.99E+01	3.13E+01	1.11E+02
Xe-138	2.75E+02	3.30E+01	3.08E+02
I-131	3.32E+01	4.31E-01	3.36E+01
I-132	4.30E+01	4.02E-01	4.34E+01
I-133	7.38E+01	8.08E-01	7.46E+01
I-134	6.87E+01	6.32E-01	6.93E+01
I-135	6.65E+01	6.12E-01	6.71E+01

NOTE:

1. 6.27E+01 = 6.27×10^{1}.

TABLE 15.4-7 (Cont)

N-1 LOOP

Isotope	Containment Building Releases (Ci)	Main Steam Valve Building Releases (Ci)	Total Releases (Ci)
Kr-83m	6.30E+01[1]	6.60E-01	6.37E+01
Kr-85m	1.80E+02	1.70E+00	1.82E+02
Kr-85	4.20E+00	3.70E-02	4.24E+00
Kr-87	2.80E+02	3.20E+00	2.83E+02
Kr-88	4.70E+02	4.60E+00	4.75E+02
Kr-89	5.20E+01	4.50E+00	5.65E+01
Xe-131m	2.40E+00	3.30E-03	2.93E-01
Xe-133m	3.00E+01	2.00E-01	2.42E+01
Xe-133	1.00E+02	8.30E+00	9.50E+02 8
Xe-135m	5.30E+03	2.20E+00	9.52E+02
Xe-135	6.00E+02	2.30E+00	3.22E+02
Xe-137	8.00E+01	6.00E+00	8.60E+01
Xe-138	2.80E+02	[illegible].20E+00	2.87E+02
I-131	3.30E+01	4.90E-02	3.30E+01
I-132	4.40E+01	5.70E-02	4.41E+01
I-133	7.10E+01	1.00E-01	7.11E+01
I-134	7.10E+01	9.40E-02	7.11E+01
I-135	6.60E+01	8.20E-02	6.61E+01

NOTE:

1. 6.30E+01 = 6.30×10^{1}.

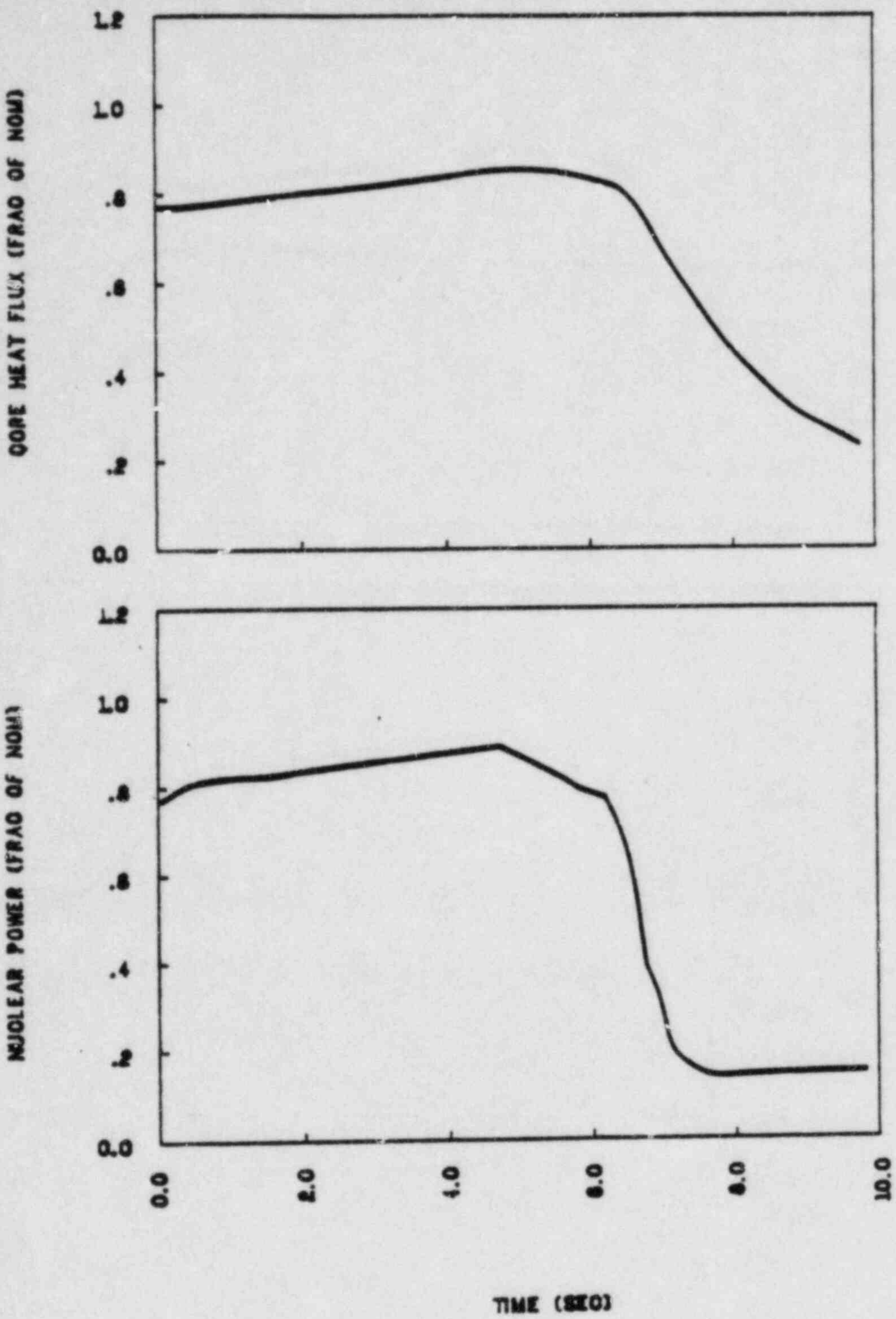


FIGURE 15.4-4a

UNCONTROLLED RCCA BANK
WITHDRAWAL FROM FULL POWER
WITH MAXIMUM REACTIVITY
FEEDBACK
(70 PCM/SEC. WITHDRAWAL RATE)
(N-1 LOOP OPERATION)

INTERACTIVE GRAPHICS
D.CHA.STOU.0751AXX.XXX.01.004.082983

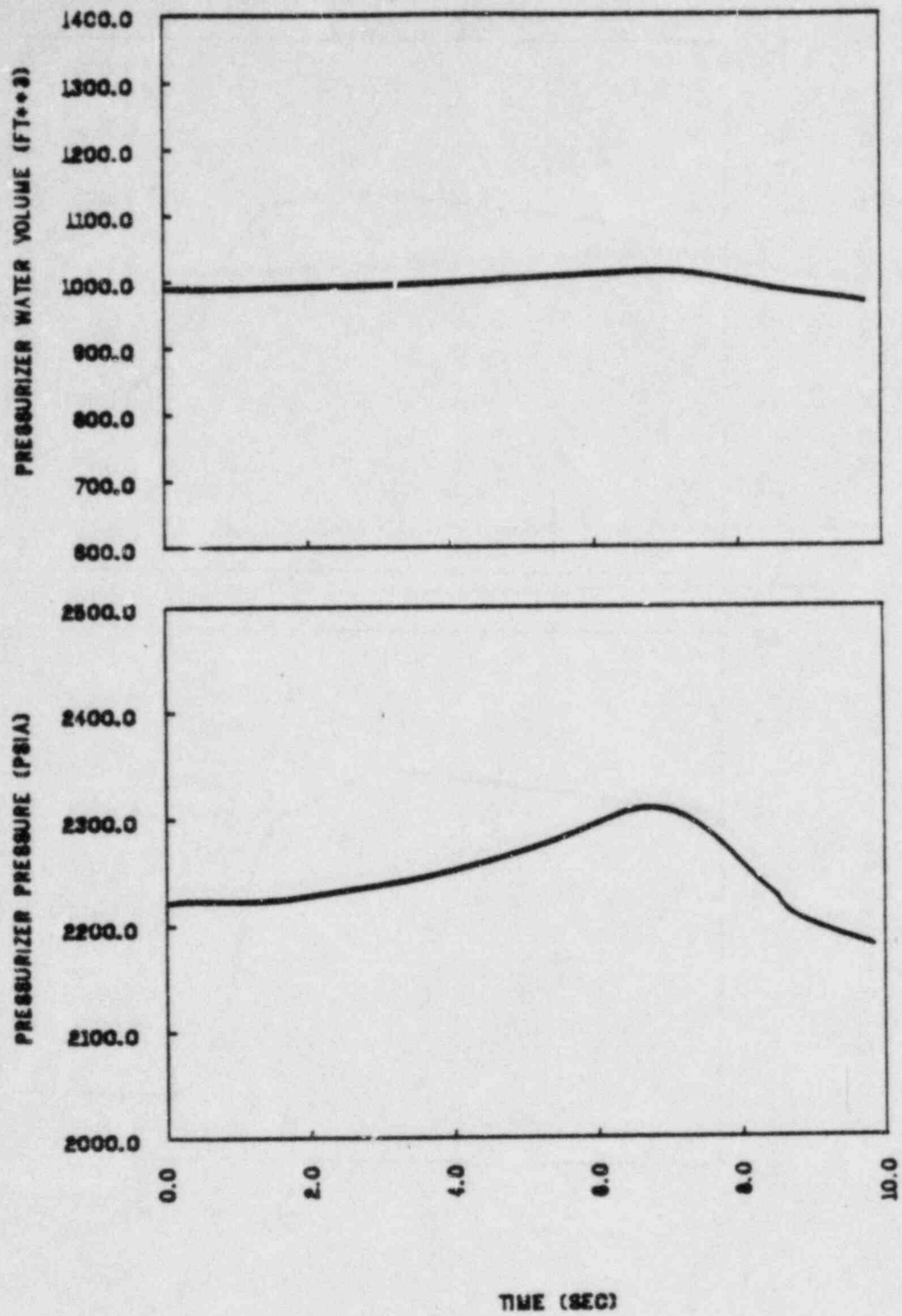


FIGURE 15.4-5a

UNCONTROLLED RCCA BANK
WITHDRAWAL FROM FULL POWER
WITH MAXIMUM REACTIVITY
FEEDBACK
(70 PCM/SEC. WITHDRAWAL RATE)
(N-1 LOOP OPERATION)

INTERACTIVE GRAPHICS
D.CHA.STOU.0751A-X..XX.01.005.082983

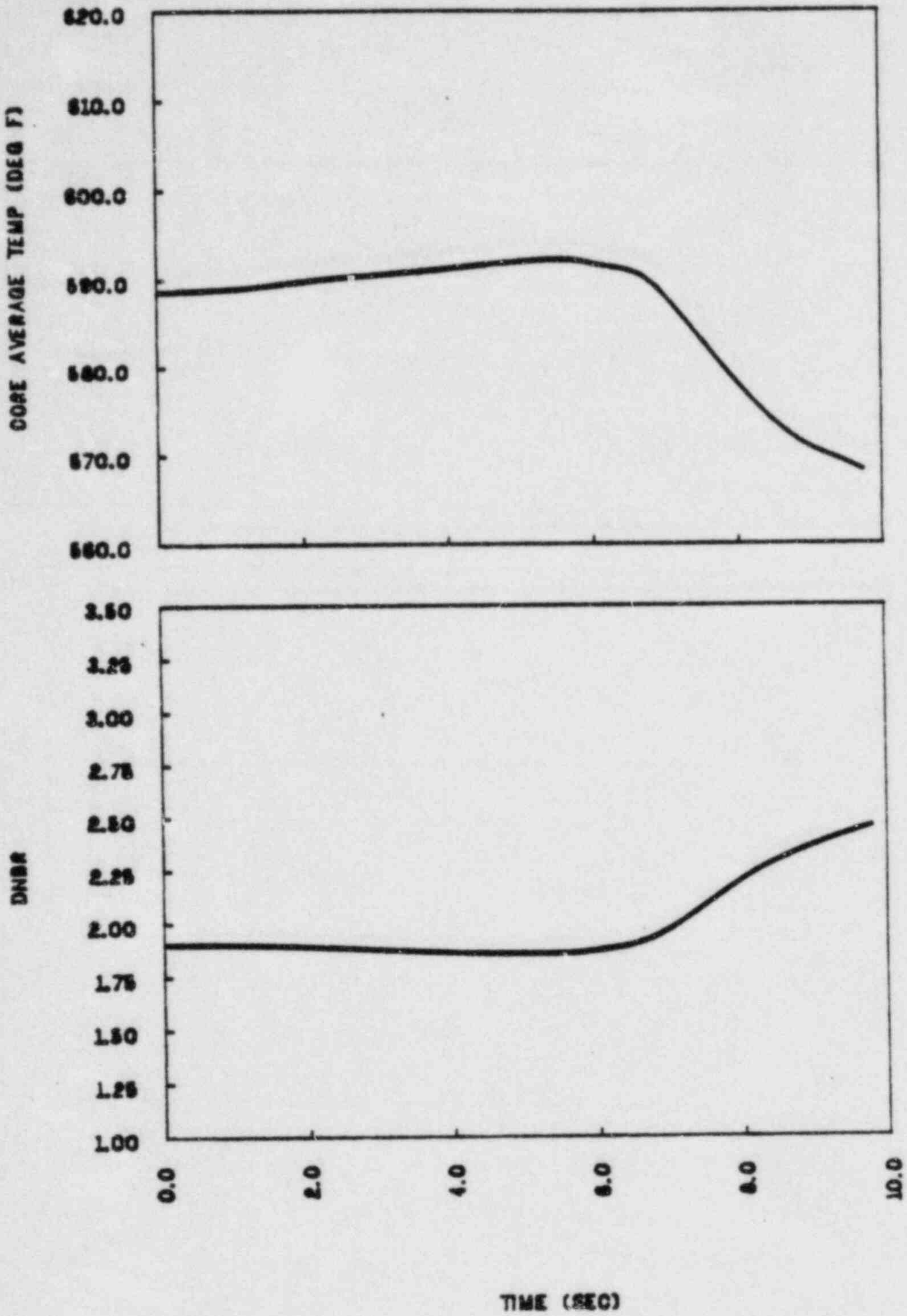


FIGURE 15.4-6a

UNCONTROLLED RCCA BANK
WITHDRAWAL FROM FULL POWER
WITH MAXIMUM REACTIVITY
FEEDBACK
(70 PCM/SEC. WITHDRAWAL RATE)
(N-1 LOOP OPERATION)

INTERACTIVE GRAPHICS
D.CHA.STOU.0751AXX.XXX.01.006.082983

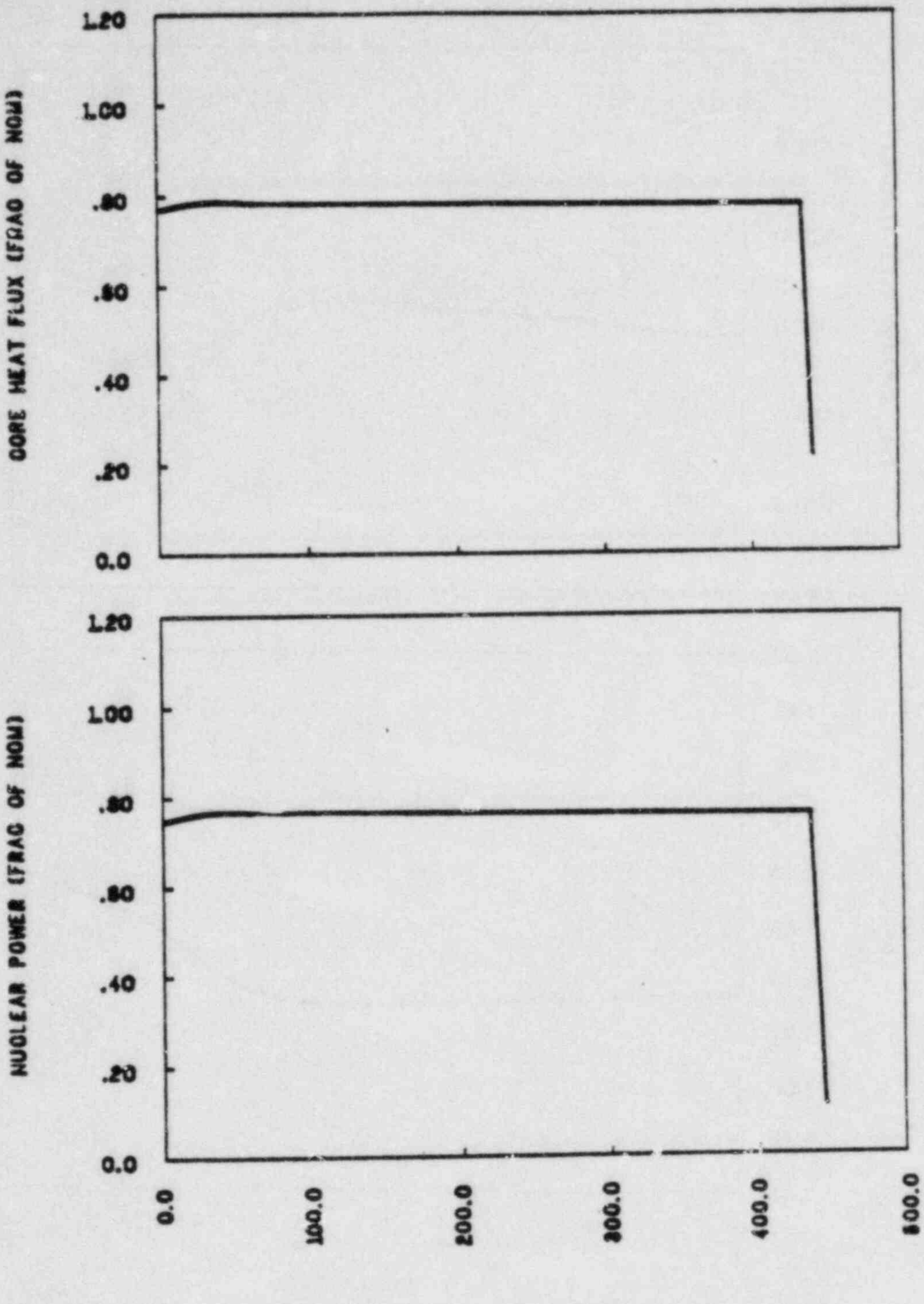


FIGURE 15.4-7a

UNCONTROLLED RCCA BANK
WITHDRAWAL FROM FULL POWER
WITH MAXIMUM REACTIVITY
FEEDBACK
(3 PCM/SEC. WITHDRAWAL RATE)
(N-1 LOOP OPERATION)

INTERACTIVE GRAPHICS
D.CHA.STOU.0751AXX.XXX.01.009.082983

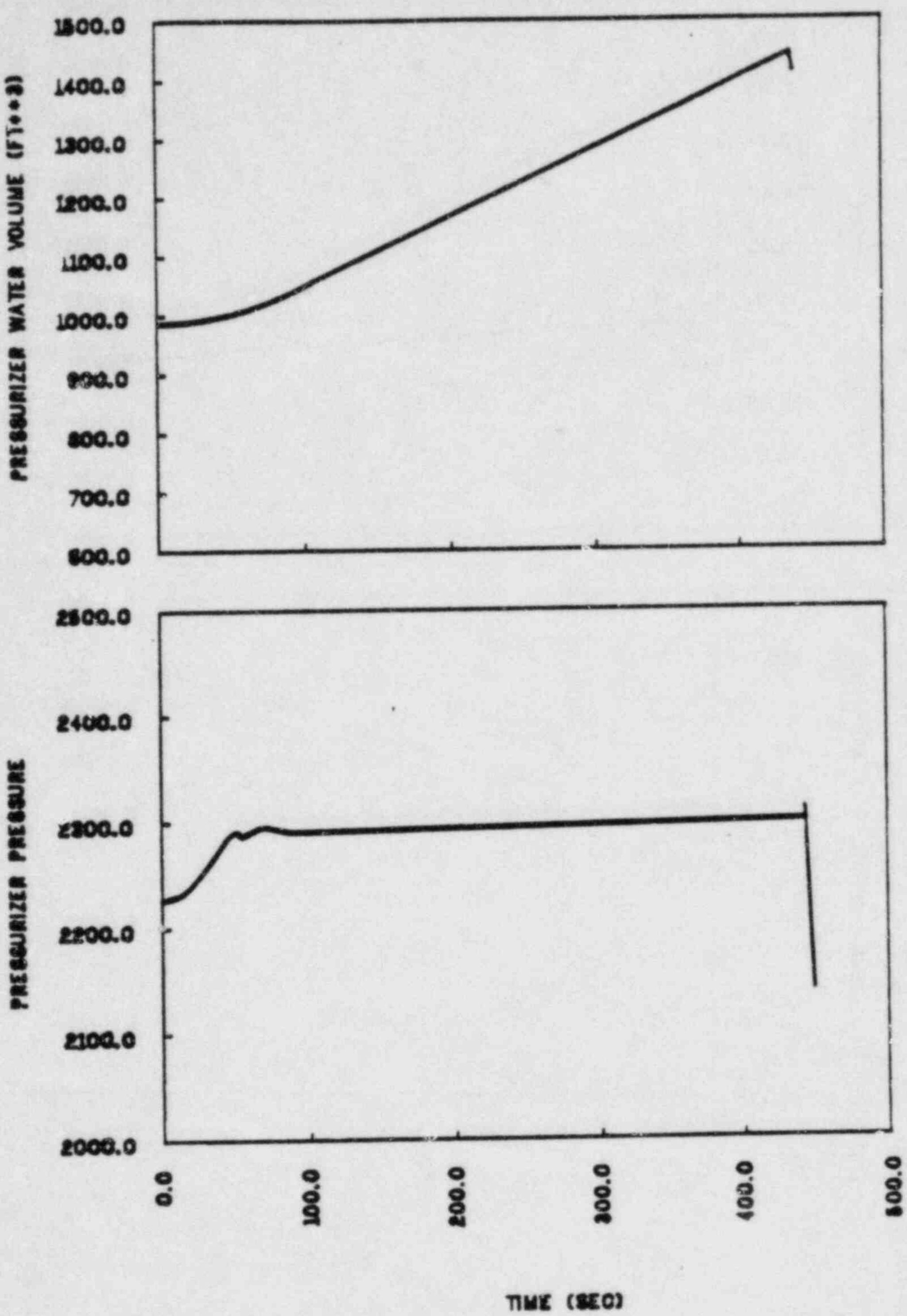


FIGURE 15.4-8a

UNCONTROLLED RCCA BANK WITHDRAWAL FROM FULL POWER WITH MAXIMUM REACTIVITY FEEDBACK
(3 PCM/SEC. WITHDRAWAL RATE)
(N-1 LOOP OPERATION)

INTERACTIVE GRAPHICS
D.CHA.STOU.0751AXX.XX.X.01.008.082983

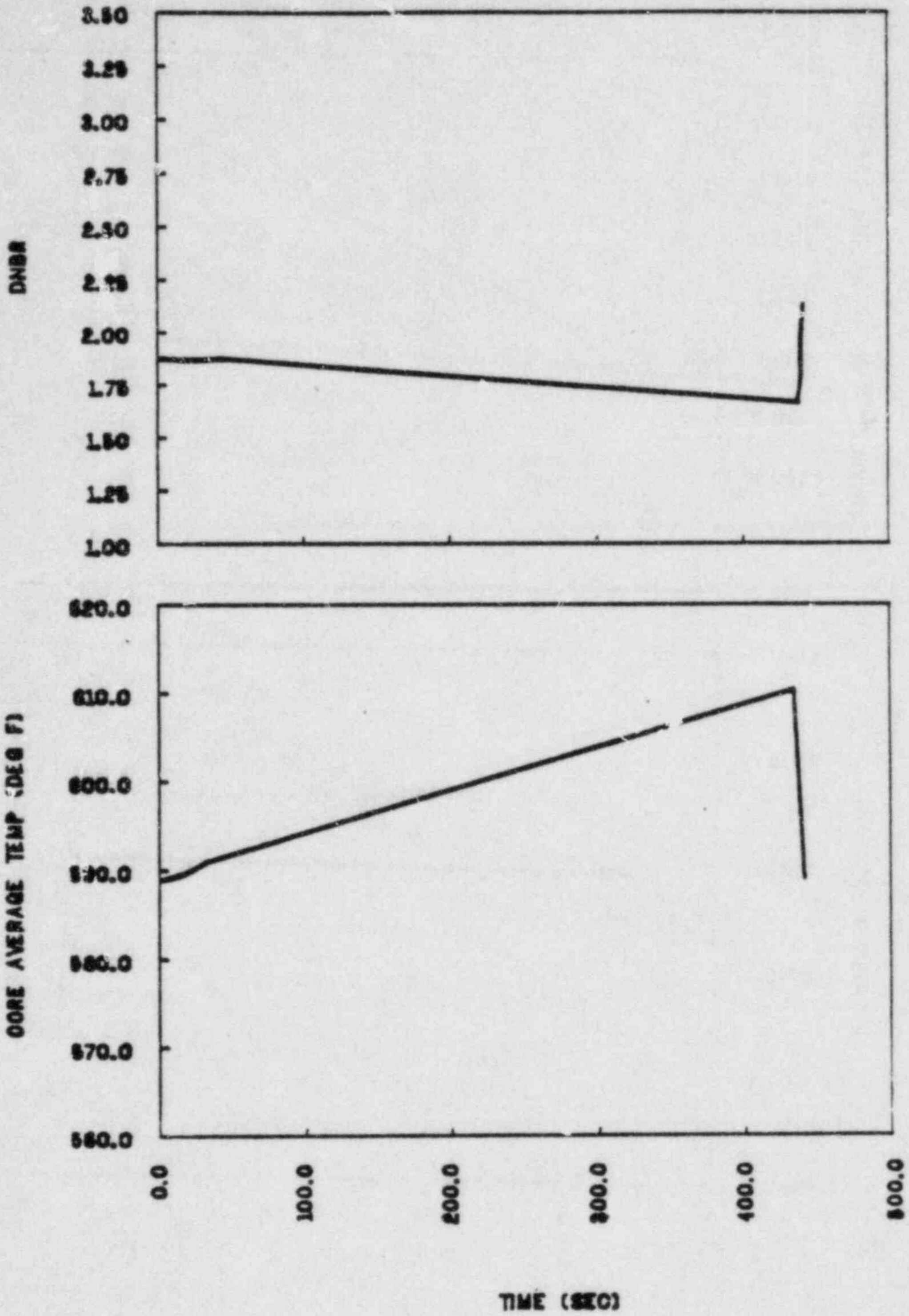


FIGURE 15.4-9a

UNCONTROLLED RCCA BANK
WITHDRAWAL FROM FULL POWER
WITH MAXIMUM REACTIVITY
FEEDBACK
(3 PCM/SEC. WITHDRAWEL RATE)
N-1 LOOP OPERATION

INTERACTIVE GRAPHICS
D.CHA.STOU.0751AXX.XXX.01.007.082983

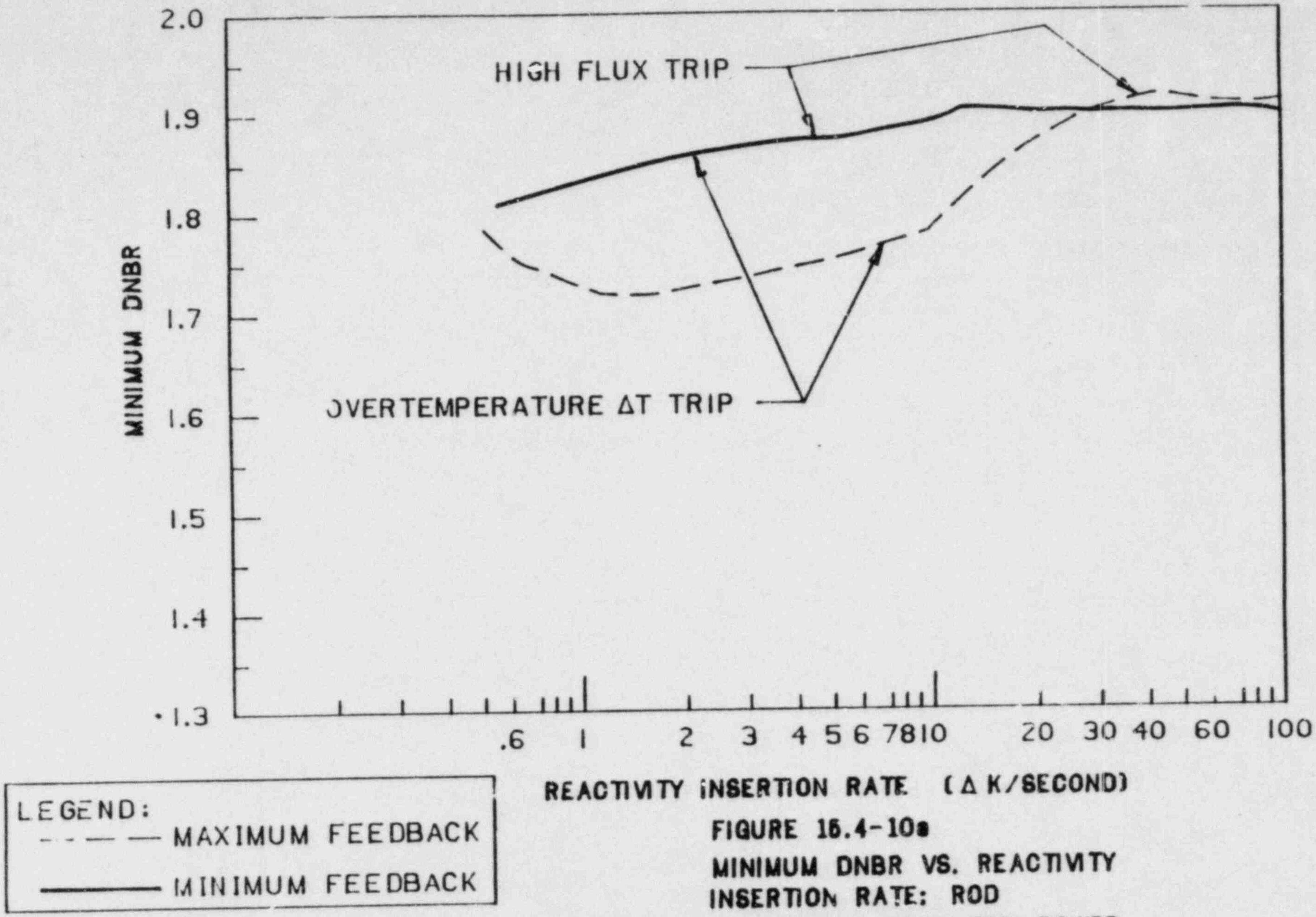


FIGURE 15.4-10a
MINIMUM DNBR VS. REACTIVITY INSERTION RATE: ROD WITHDRAWAL FROM 75% POWER N-1 LOOP OPERATION

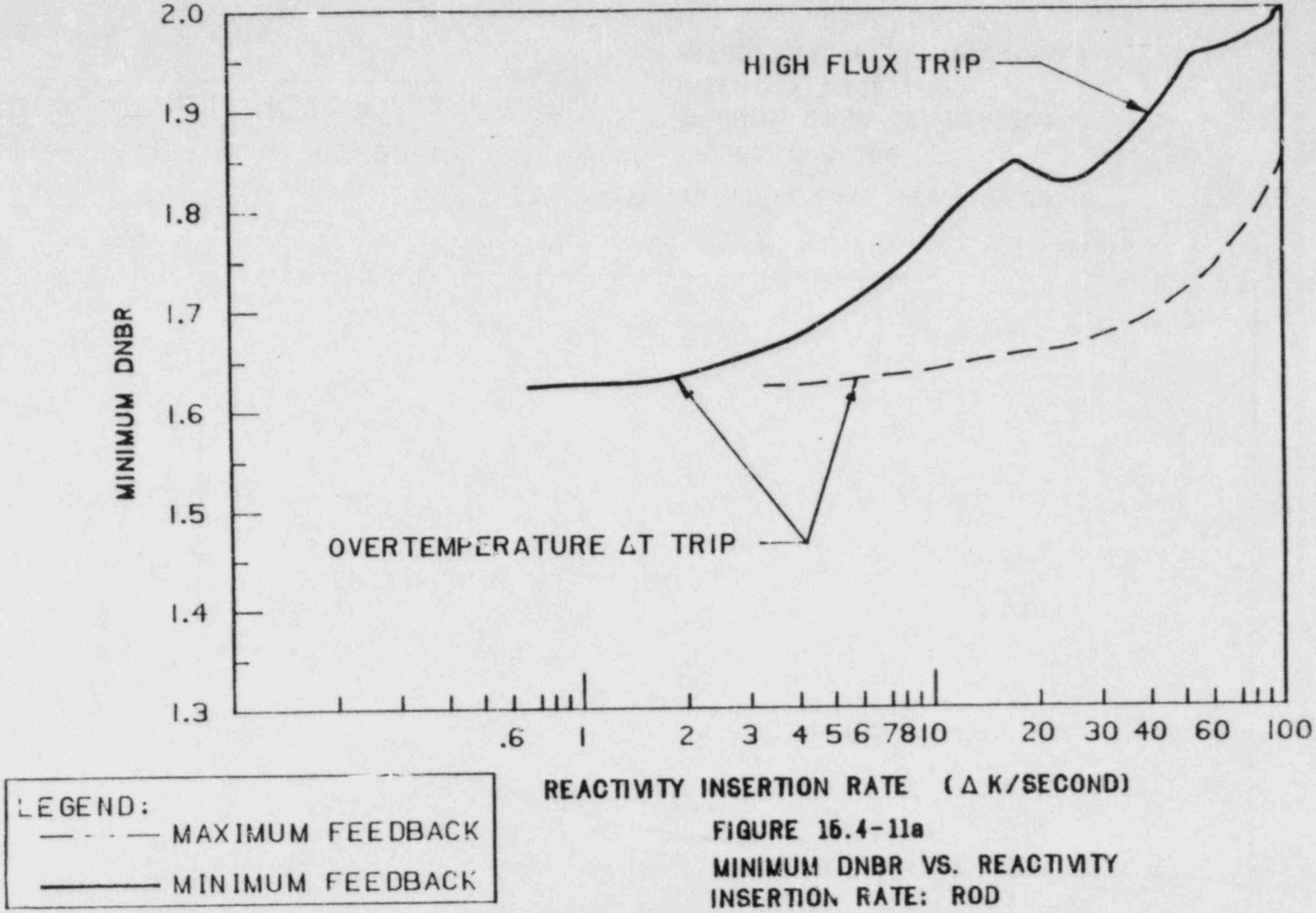


FIGURE 15.4-11a
MINIMUM DNBR VS. REACTIVITY INSERTION RATE: ROD WITHDRAWAL FROM 10% POWER N-1 LOOP OPERATION

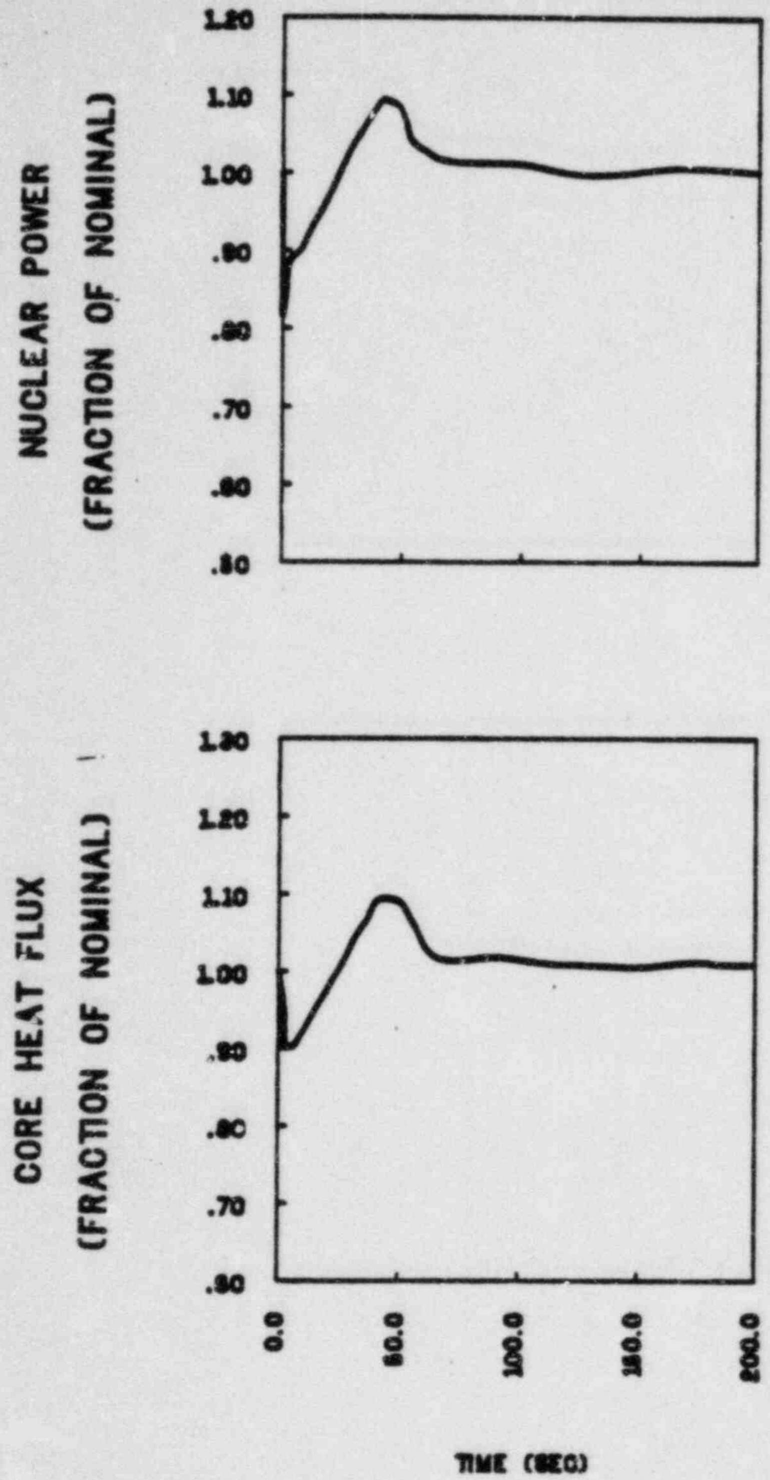


FIGURE 15.4-13

DROPPED ROD CLUSTER CONTROL
ASSEMBLY. AUTOMATIC CONTROL
N LOOP OPERATION

INTERACTIVE GRAPHICS
D.CHA.STJU.0751AXX.XXX.01.060.082983

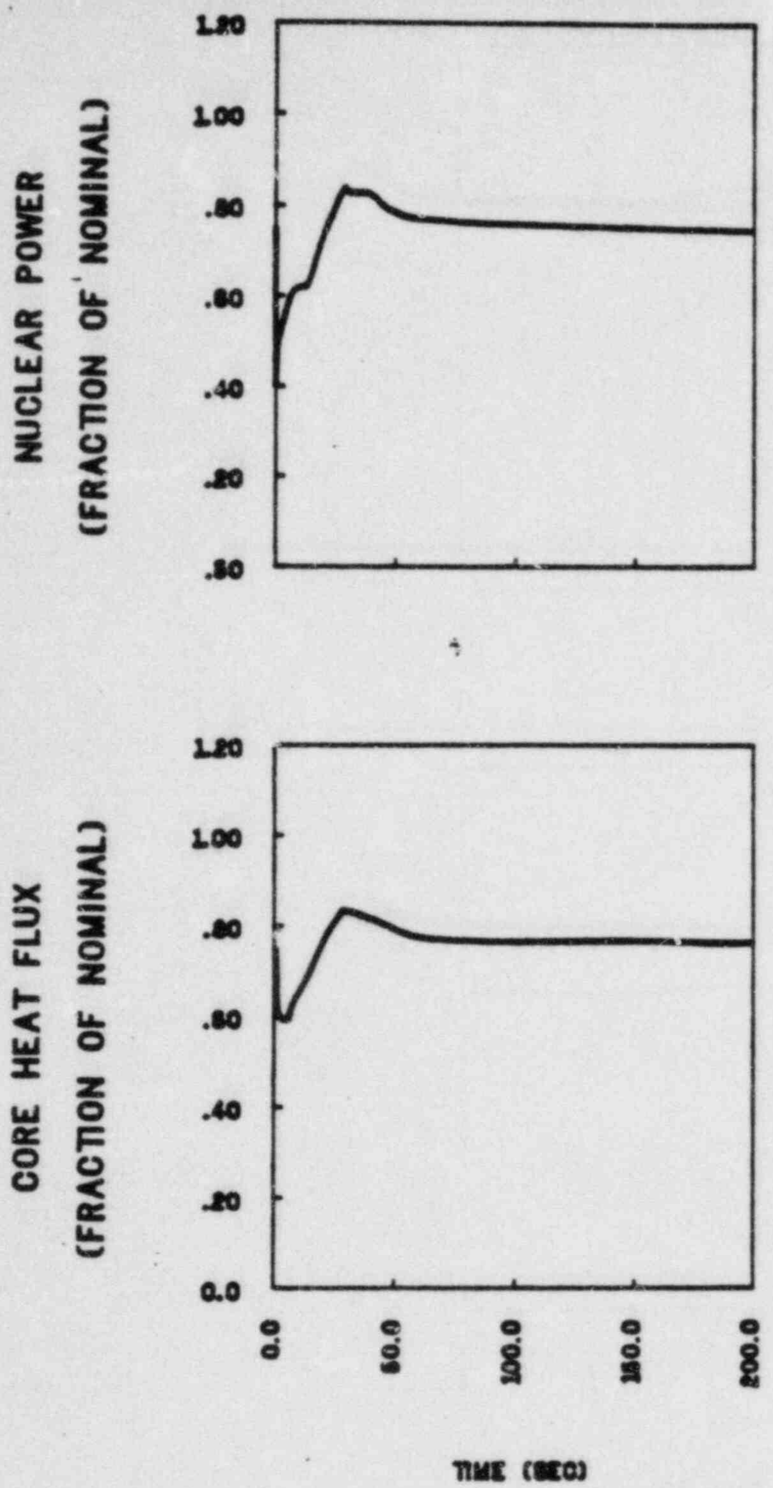


FIGURE 15.4-13a

DROPPED ROD CLUSTER CONTROL
ASSEMBLY. AUTOMATIC CONTROL
N-1 LOOP OPERATION

INTERACTIVE GRAPHICS
D.CHA.STOU.0751AXX.XXX.01.061.082983

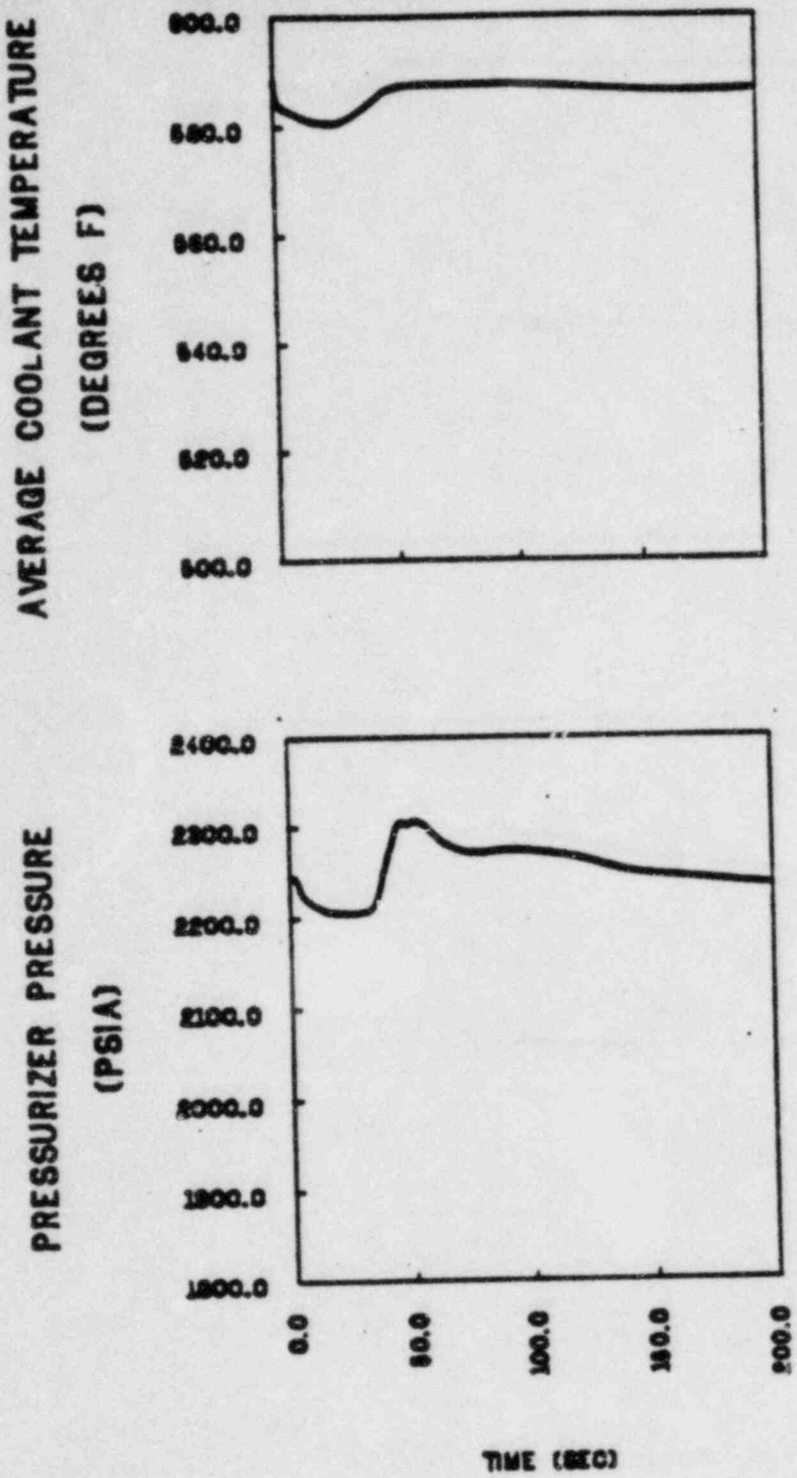


FIGURE 15.4-14

DROPPED ROD CLUSTER CONTROL
ASSEMBLY. AUTOMATIC CONTROL
N LOOP OPERATION

INTERACTIVE GRAPHICS
D.CHA.STC[illegible]XXX.XXX.01.062.082983

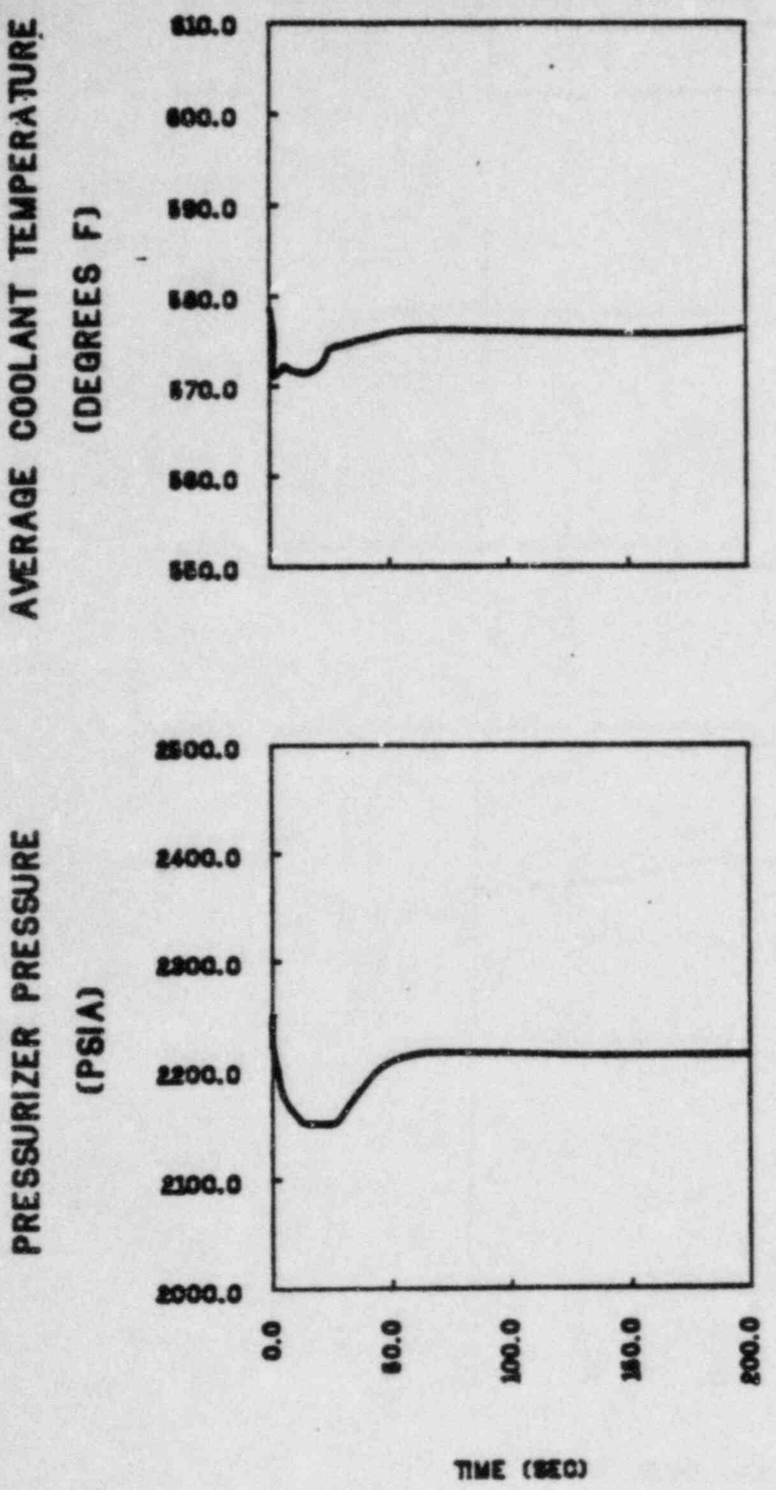


FIGURE 15.4-14a

DROPPED ROD CLUSTER CONTROL
ASSEMBLY, AUTOMATIC CONTROL
N-1 LOOP OPERATION

INTERACTIVE GRAPHICS
D.CHA.STOU.0751AXX.XXX.01.063.082983

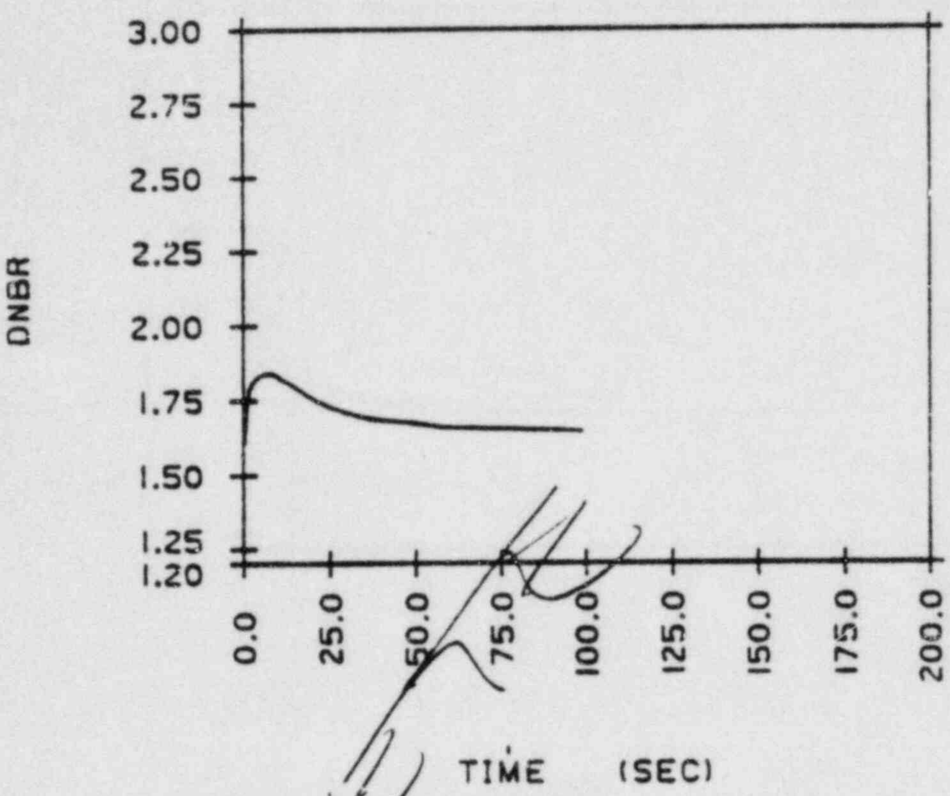


FIGURE 15.4-15
DROPPED ROD CLUSTER CONTROL ASSEMBLY, MANUAL CONTROL
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

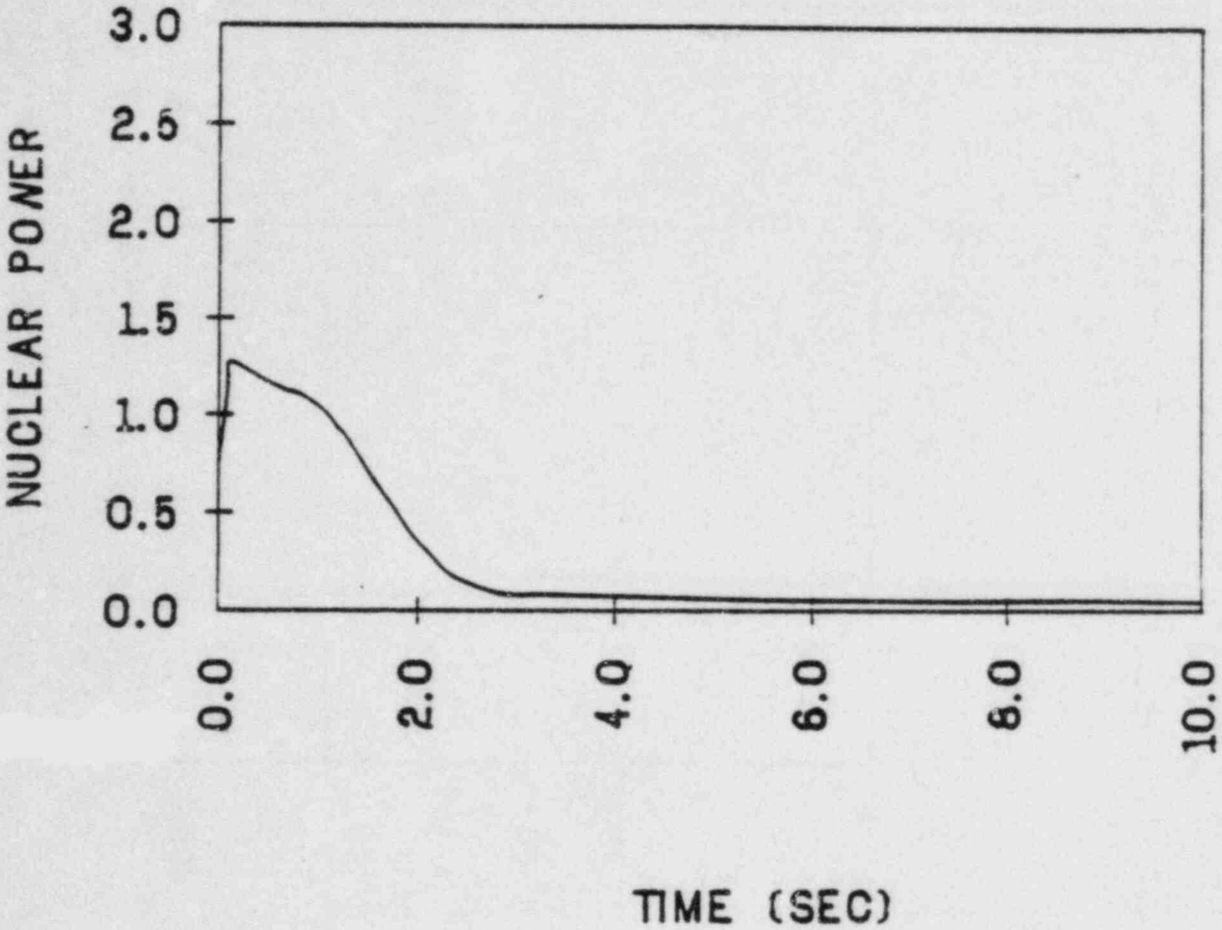


FIGURE 15.4-26A
NUCLEAR POWER TRANSIENT, BOL
HFP ROD EJECTION ACCIDENT
(N-1 LOOP OPERATION)

INTERACTIVE GRAPHICS
D.CHA.GEOR.0751AXX.XXX.01.040.083183

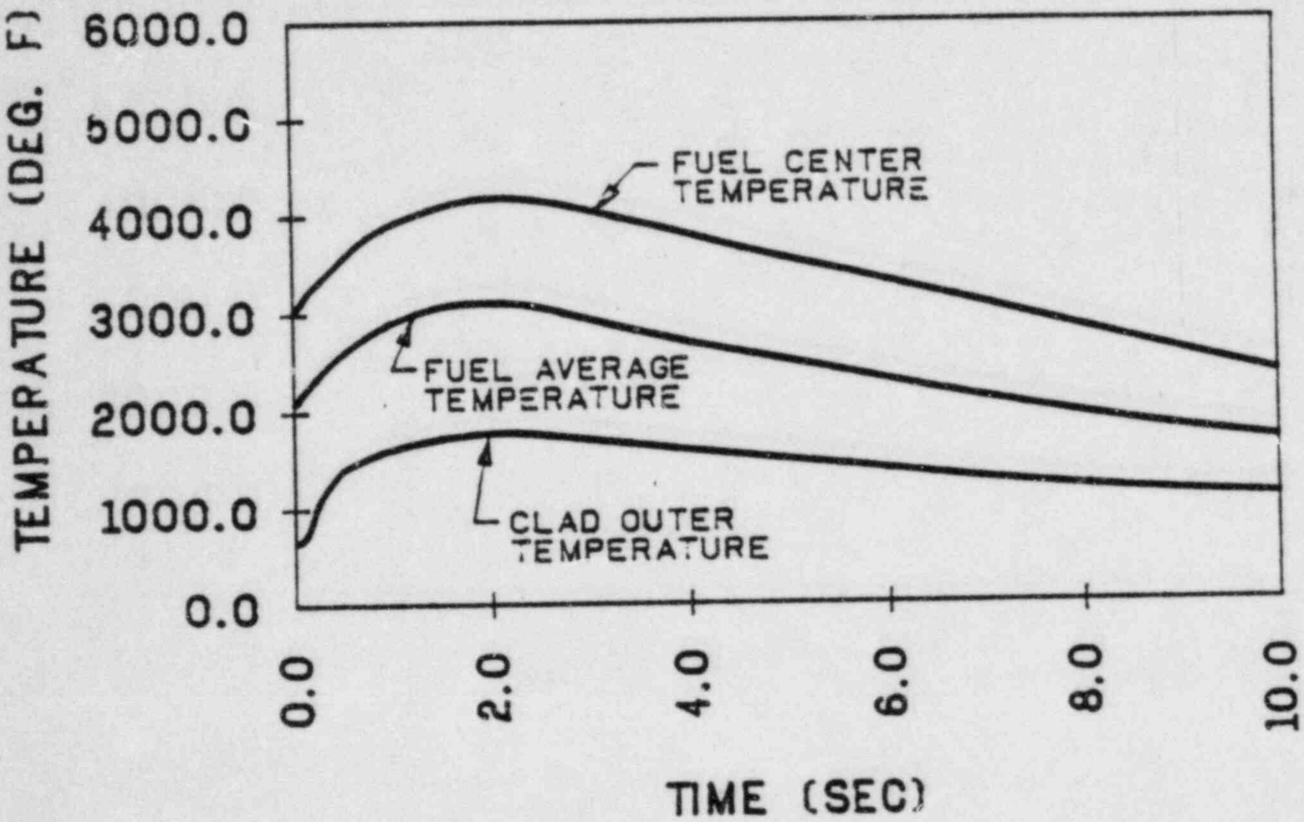


FIGURE 15.4-27A
HOT SPOT FUEL
AND CLAD TEMPERATURES VS. TIME
BOL. HFP ROD EJECTION ACCIDENT
(N-1 LOOP OPERATION)

INTERACTIVE GRAPHICS
D.CHA.MAYF.0751AXX.XXX.01.025.083083

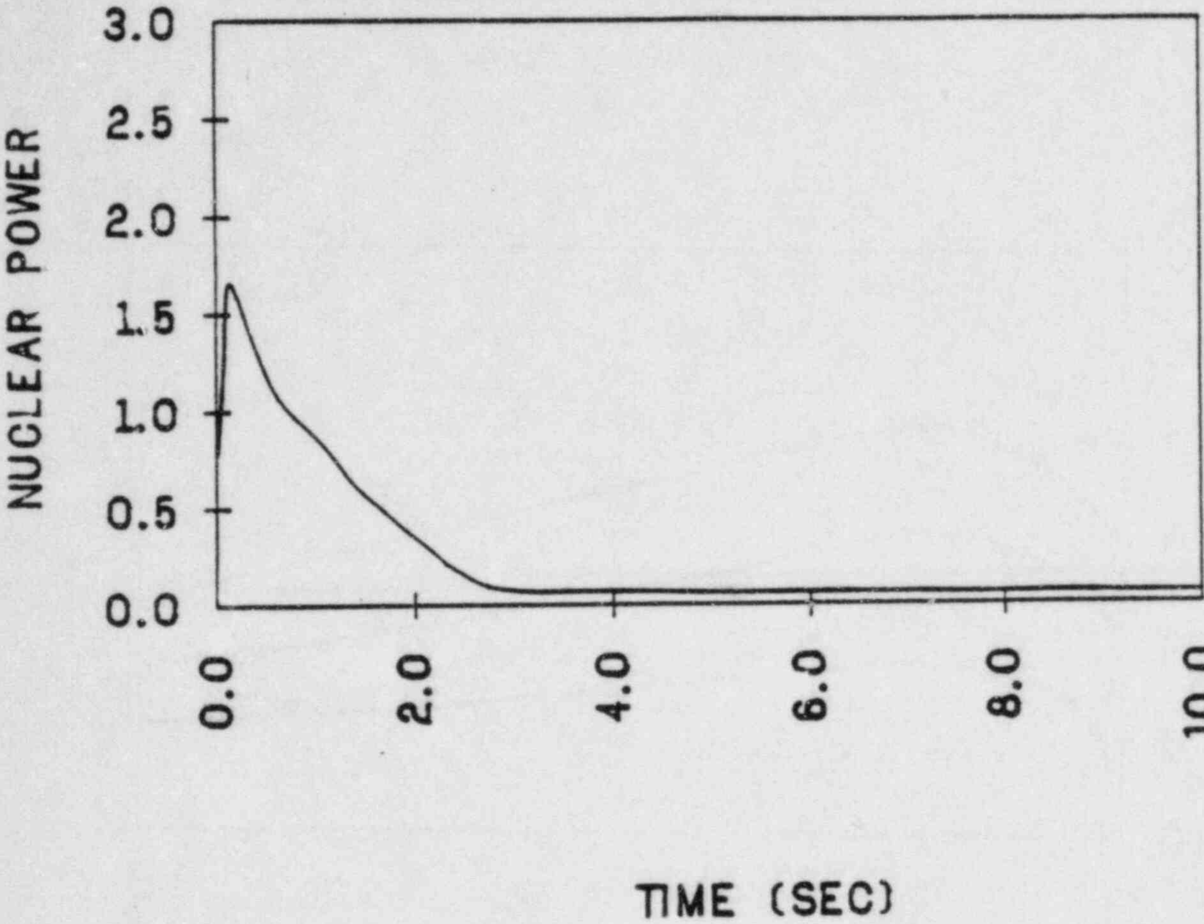


FIGURE 15.4-28A
NUCLEAR POWER TRANSIENT, EOL
HFP ROD EJECTION ACCIDENT
(N-1 LOOP OPERATION)

INTERACTIVE GRAPHICS
D.CHA.GEOR.0751AXX.XXX.01.041.093183

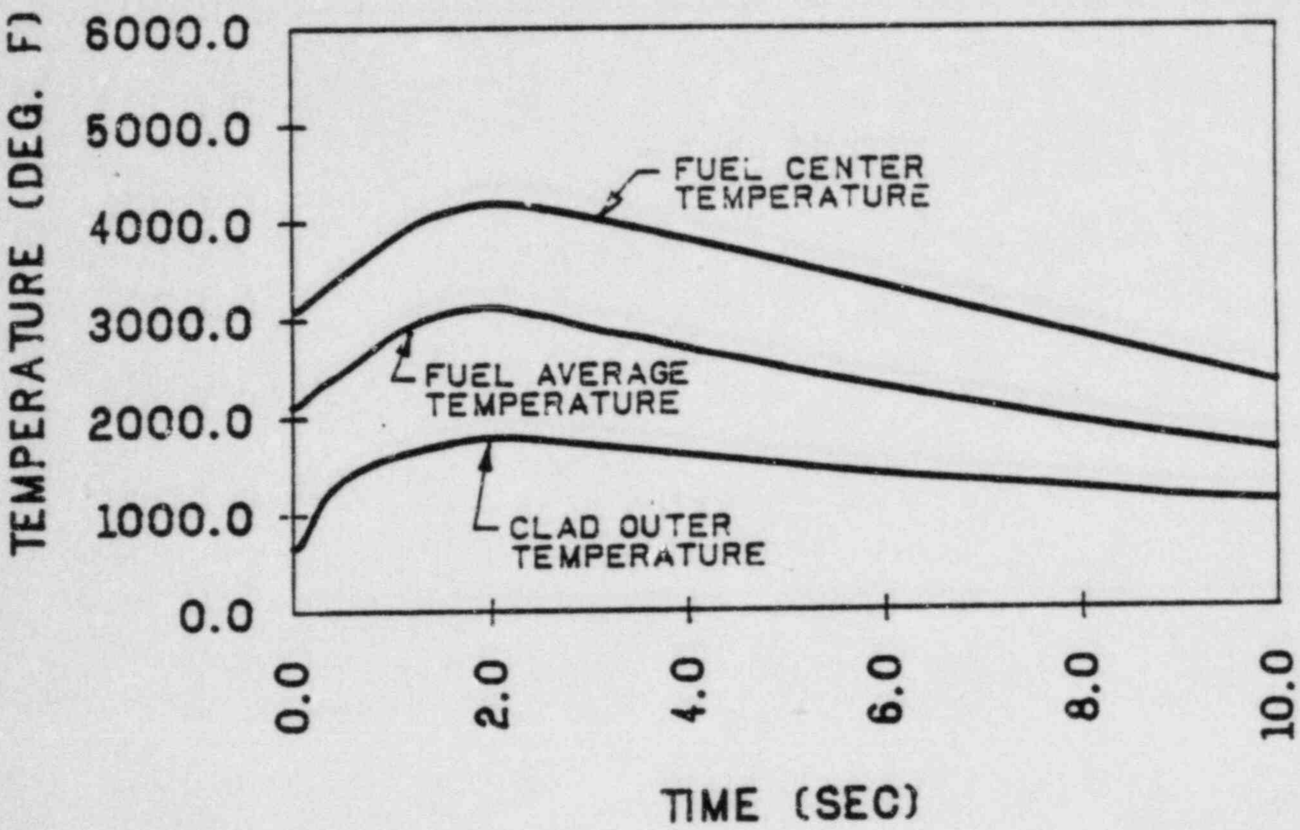


FIGURE 15.4-29A
HOT SPOT FUEL
AND CLAD TEMPERATURES VS. TIME
EOL HFP ROD EJECTION ACCIDENT
(N-1 LOOP OPERATION)

INTERACTIVE GRAPHICS
D.CHA.MAYF.0751AXX.XXX.01.026.083083

15.5 INCREASE IN REACTOR COOLANT INVENTORY 1.9

Discussion and analysis of the following events are presented in this section. 1.10

1. Inadvertent operation of the emergency core cooling system during power operation. 1.12

2. Chemical and volume control system malfunction that increases reactor coolant inventory. 1.13

3. A number of BWR transients (not applicable to Millstone 3). 1.14

These events, considered to be ANS Condition II, cause an increase in 1.16
reactor coolant inventory. Section 15.0.1 contains a discussion of 1.18
ANS classifications.

15.5.1 Inadvertent Operation of the Emergency Core Cooling System during Power Operation 1.22

15.5.1.1 Identification of Causes and Accident Description 1.25

Spurious emergency core cooling system (ECCS) operation at power 1.26
could be caused by operator error or a false electrical actuation 1.27
signal. A spurious signal may originate from any of the safety 1.29
injection actuation channels as described in Section 7.3.

Following the actuation signal, the suction of the coolant charging 1.30
pumps is diverted from the volume control tank to the RWST. The 1.32
charging/high head safety injection pumps then force concentrated
(1,950 ppm) boric acid solution from refueling water storage tank, 1.34
through the header and injection line and into the cold leg of each
loop. The safety injection pumps also start automatically but 1.35
provide no flow when the reactor coolant system (RCS) is at normal 1.36
pressure. The passive injection system and the low head system also 1.37
provide no flow at normal RCS pressure.

A safety injection system (SIS) signal normally results in a reactor 1.38
trip followed by a turbine trip. However, it cannot be assumed that 1.39
any single fault that actuates the SIS will also produce a reactor
trip. If a reactor trip is generated by the spurious SIS signal, the 1.40
operator should determine if the spurious signal was transient or 1.41
steady state in nature. The operator must also determine if the 1.42
safety injection signal should be blocked. For a spurious 1.43
occurrence, the operator would stop the safety injection and maintain
the plant in the hot shutdown condition. If the ECCS actuation 1.45
instrumentation must be repaired, future plant operation would be in
accordance with the Technical Specifications. 1.46

If the reactor protection system does not produce an immediate trip 1.47
as a result of the spurious SIS signal, the reactor experiences a 1.48
negative reactivity excursion due to the injected boron causing a
decrease in reactor power. The power mismatch causes a drop in T_{Avg} ~~1.49~~
and consequent coolant shrinkage. Pressurizer pressure and water 1.50

level drop. Load will decrease due to the effect of reduced steam 1.51
pressure on load after the turbine throttle valve is fully open. If 1.52
automatic rod control is used, these effects will be lessened until
the rods have moved out of the core. The transient is eventually 1.53
terminated by the reactor protection system low pressure trip or by
manual trip.

The time to trip is affected by initial operating conditions 1.54
including core burnup history, which affects initial boron 1.55
concentration, rate of change of boron concentration, Doppler, and
moderator coefficients.

Recovery from this second case is made in the same manner as 1.56
described for the case where the SIS signal results directly in a 1.57
reactor trip. The only difference is the lower T_{avg} and pressure ~~1.58~~
associated with the power mismatch during the transient. The time at 1.59
which reactor trip occurs is of little concern for this transient.
At lower loads, coolant contraction will be slower resulting in a 1.60
longer time to trip.

This event is classified as a Condition II incident (a fault of 2.1
moderate frequency), as defined in Section 15.0.1.

15.5.1.2 Analysis of Effects and Consequences 2.3

<u>Method of Analysis</u> 2.5

The spurious operation of the SIS is analyzed by employing the 2.7
detailed digital computer program LOFTRAN (WCAP-7907, 1972). The 2.10
code simulates the neutron kinetics, RCS, pressurizer, pressurizer
relief and safety valves, pressurizer spray, steam generator, steam 2.11
generator safety valves, and the effect of the SIS. The program 2.12
computes pertinent plant variables including temperatures, pressures,
and power level.

Because of the power and temperature reduction during the transient, 2.13
operating conditions do not approach the core limits. Analysis of 2.15
several cases has shown that the results are relatively independent
of time to trip.

A typical transient is presented representing minimum reactivity 2.16
feedback. Results with maximum reactivity feedback are similar 2.17
except that the transient is slower. For calculational simplicity, 2.18
zero injection line purge volume was assumed in this analysis, thus
the boration transient begins immediately when the appropriate valves 2.19
are opened. Plant characteristics and initial conditions are further 2.20
discussed in Section 15.0.3.

The assumptions are as follows. 2.21

1. Initial operating conditions 2.24

The initial reactor power and RCS temperatures are assumed 2.26
at their maximum values consistent with the steady state
normal power operation including allowance for calibration 2.27
and instrument errors (for N-loop or N-1 loop operation). X

2. Moderator and Doppler coefficients of reactivity 2.31

A least negative moderator temperature coefficient was used. 2.33
A low (absolute value) Doppler power coefficient was 2.34
assumed.

3. Reactor control 2.38

The reactor was assumed to be in manual control. 2.40

4. Pressurizer heaters 2.44

Pressurizer heaters were assumed to be inoperable in order 2.46
to increase the rate of pressure drop.

5. Boron injection 2.50

At time zero, two charging/high head safety injection pumps 2.52
inject 1,950 ppm borated water into the cold leg of each 2.53
loop.

6. Turbine load 2.57

Turbine load was assumed constant until the governor drives 2.59
the throttle wide open, then turbine load drops as steam 2.60
pressure drops.

7. Reactor trip 3.4

Reactor trip was initiated by low pressurizer pressure. 3.6

Plant systems and equipment which are available to mitigate the 3.9
effects of the accident are discussed in Section 15.0.8 and listed in 3.10
Table 15.0-6. No single active failure in any of these systems or 3.11
equipment will adversely affect the consequences of the accident.

Results 3.14

Figures 15.5-1 through 15.5-3 (for four loops in operation) and 3.16
Figures 15.5-1A through 15.5-3A (for three loops in operation) show 3.17
the transient response to inadvertent operation of ECCS during power 3.18
operation. Neutron flux starts decreasing immediately due to boron 3.20
injection but steam flow does not decrease until later in the
transient when the turbine throttle valve goes wide open. 3.22
~~[illegible] operation, the turbine throttle [illegible] does not completely [illegible]~~ X
The mismatch between load and nuclear 3.23
power causes T_{Avg}, pressurizer water level, and pressurizer pressure

to drop. When the low pressure trip setpoint is reached, the reactor trips and control rods start moving into the core. Departure from nucleate boiling ratio (DNBR) increases throughout the transient.

The calculated sequence of events is shown in Table 15.5-1. After reactor trip, pressure and temperature slowly rise, since the turbine is tripped and the reactor is producing some power due to delayed neutron fissions and decay heat. Recovery from this accident is discussed in Section 15.5.1.1.

15.5.1.3 Conclusions

Results of the analysis show that the spurious safety injection without immediate reactor trip presents no hazard to the integrity of the RCS.

If the reactor does not trip immediately, the low pressure reactor trip will be actuated. This trips the turbine and prevents excess cooldown, thereby, expediting recovery from the incident.

15.5.1.4 Radiological Consequences

There are only minimal radiological consequences associated with inadvertent ECCS operation. The reactor trip causes a turbine trip and heat is removed from the secondary system through the steam generator power relief valves or safety valves. Since no fuel damage is postulated to occur from this transient, the radiological consequences associated with atmospheric steam release from this event are less severe than the steam line break event analyzed in Section 15.1.5.

15.5.2 Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory

An increase in reactor coolant inventory, which results from the addition of cold, unborated water to the RCS, is analyzed in Section 15.4.6, Chemical and Volume Control System Malfunction That Results in a Decrease in Boron Concentration in the Reactor Coolant. An increase in reactor coolant inventory, which results from the injection of borated water into the RCS, is analyzed in Section 15.5.1, Inadvertent Operation of the Emergency Core Cooling System during Power Operation.

15.5.3 A Number of BWR Transients

This section is not applicable to Millstone 3.

15.5.4 Reference for Section 15.5

WCAP-7907, 1972. Burnett, T.W.T. et al. LOFTRAN Code Description.

TABLE 15.5-1

TIME SEQUENCE OF EVENTS FOR INCREASE IN
REACTOR COOLANT INVENTORY EVENTS

Accident	Event	N-Loop Time (sec)	N-1 Loop Time (sec)
Inadvertent actuation of the ECCS during power operating	Spurious safety injection signal generated; two charging pumps begin injecting borated water	0	0
	Turbine throttle valve wide open, load begins to drop with steam pressure	38.5	~~38.5~~ 73.5
	Low pressurizer pressure reactor trip setpoint reached	84.1	84.4
	Control rod motion begins	86.1	86.4

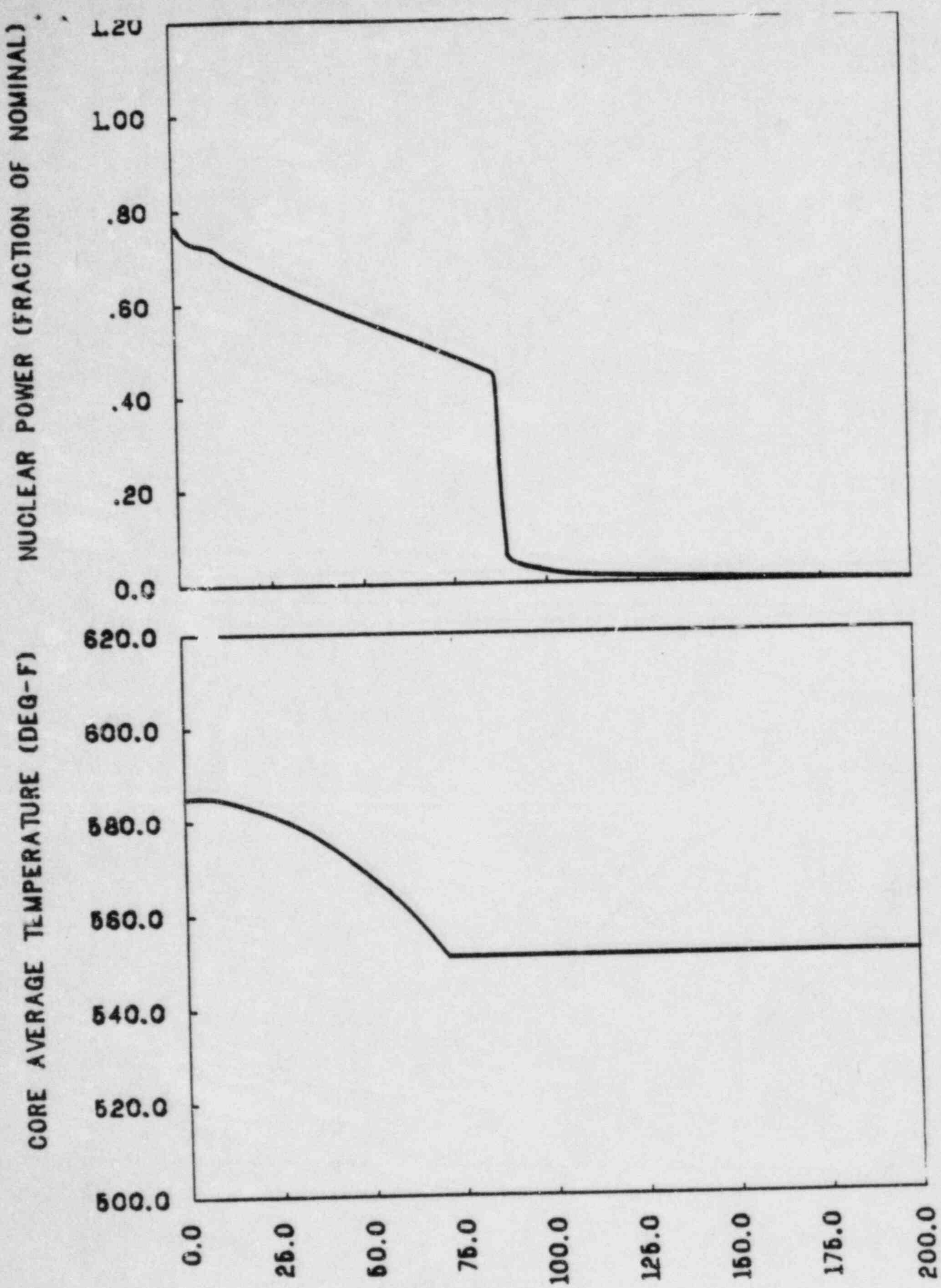


FIGURE 15.5-1a

INADVERTANT ACTUATION OF
ECCS DURING POWER OPERATION
N-1 LOOP OPERATION

INTERACTIVE GRAPHICS
D.CHA.STOU.0751AXX.XXX.01.019.082983

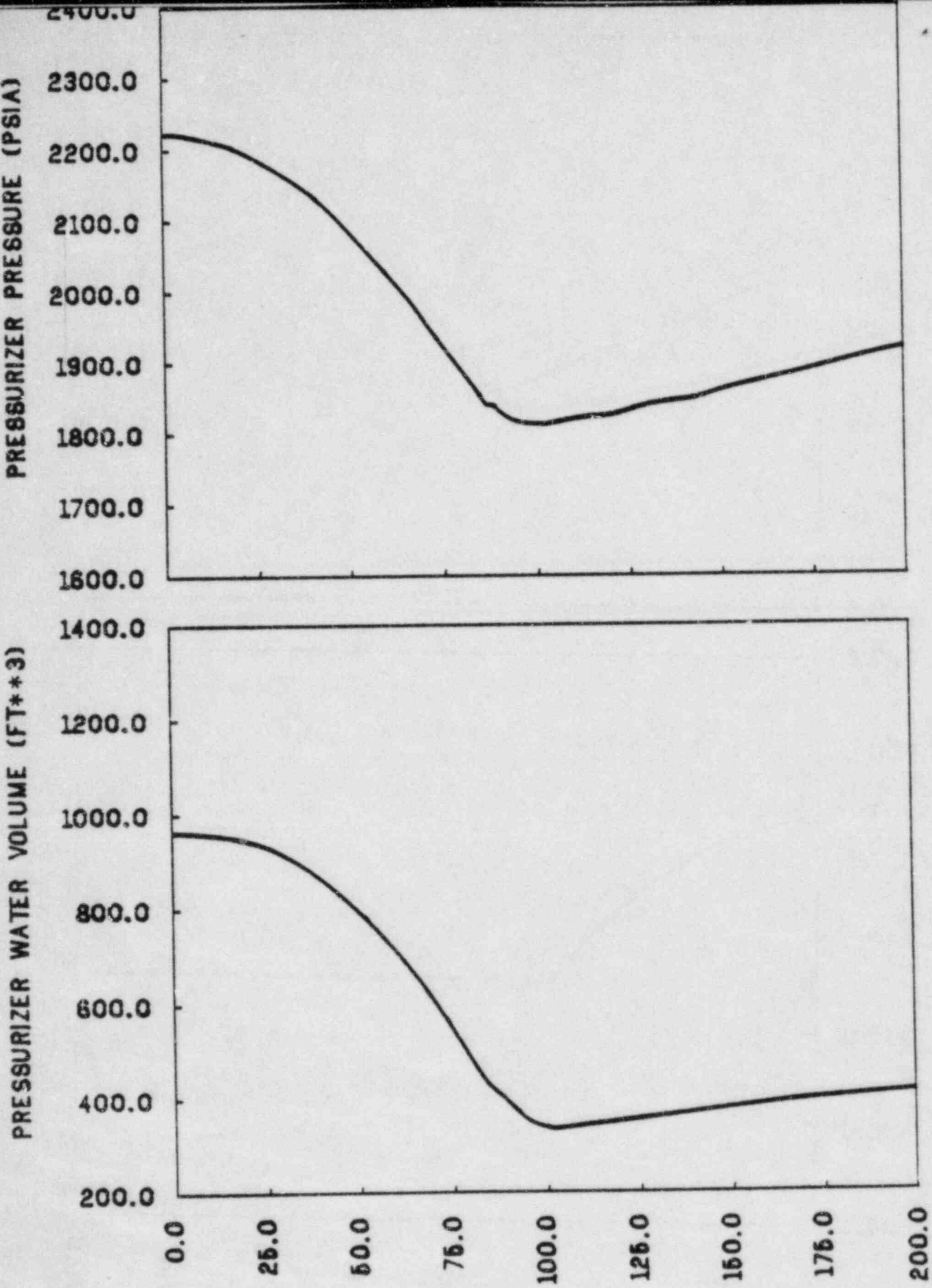


FIGURE 15.5-2a

INADVERTANT OPERATION OF
ECCS DURING POWER OPERATION
N-1 LOOP OPERATION

INTERACTIVE GRAPHICS
D.CHA.STOU.075:AXX.XXX.01.018.082983

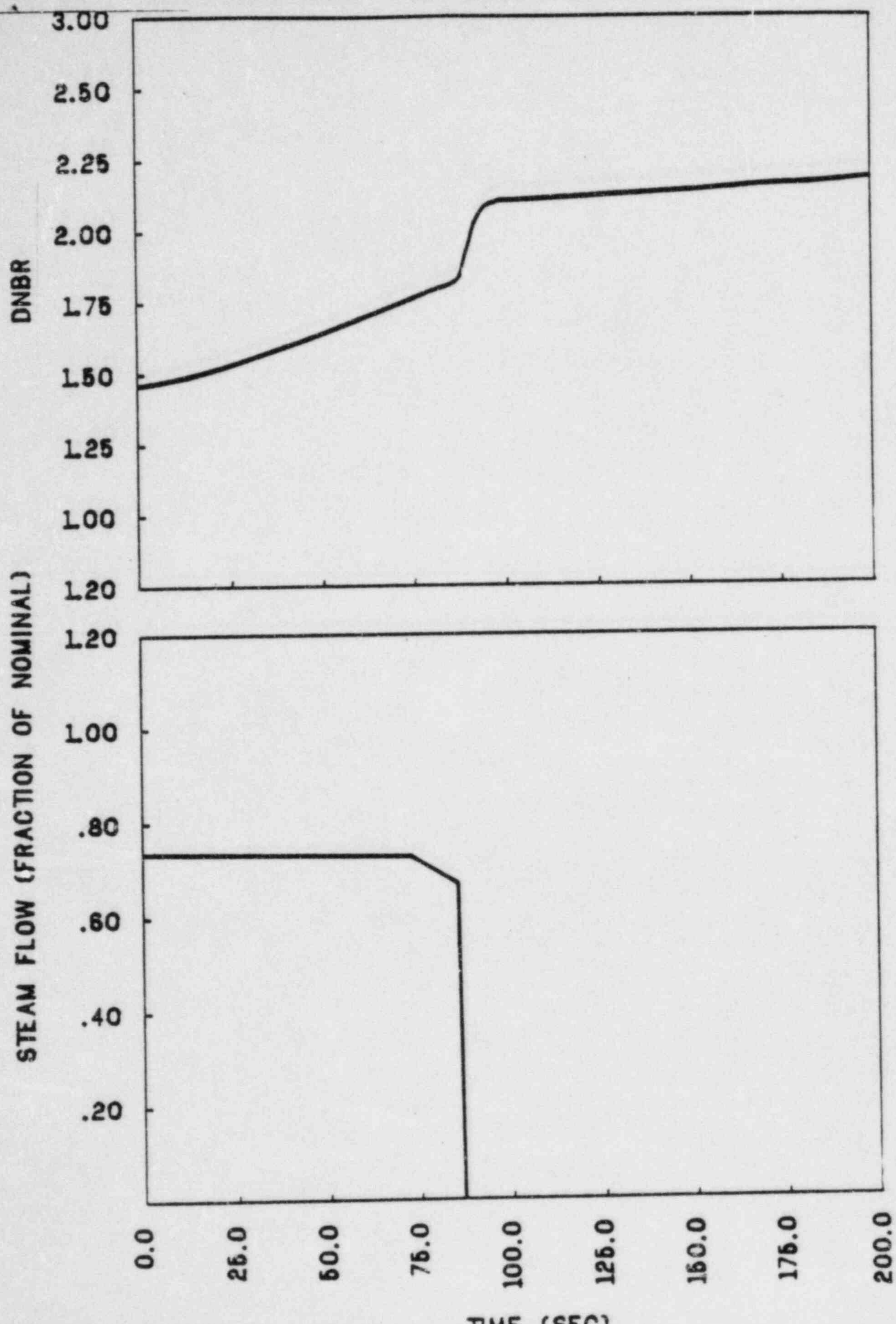


FIGURE 15.5-3a

INADVERTANT OPERATION OF
ECCS DURING POWER OPERATION
N-1 LOOP OPERATION

INTERACTIVE GRAPHICS
D.CHA.STOU.0751AXX.XXX.01.017.082983

15.6 DECREASE IN REACTOR COOLANT INVENTORY 1.9

Events which result in a decrease in reactor coolant inventory as 1.10
discussed in this section are as follows.

1. Inadvertent opening of a pressurizer safety or relief valve. 1.12

2. Break in instrument line or other lines from reactor coolant 1.13
pressure boundary (RCPB) that penetrate containment.

3. Steam generator tube failure. 1.14

4. Spectrum of boiling water reactor (BWR) steam system piping 1.15
failures outside of containment (not applicable to 1.16
Millstone 3).

5. Loss-of-coolant accident (LOCA) resulting from a spectrum of 1.17
postulated piping breaks within the RCPB.

6. A number of BWR transients (not applicable to Millstone 3). 1.18

15.6.1 Inadvertent Opening of a Pressurizer Safety or Relief Valve 1.21

15.6.1.1 Identification of Causes and Accident Description 1.22

An accidental depressurization of the reactor coolant system (RCS) 1.23
could occur as a result of an inadvertent opening of a pressurizer 1.24
relief or safety valve. Since a safety valve is sized to relieve 1.26
approximately twice the steam flow rate of a relief valve, and will
therefore allow a much more rapid depressurization upon opening, the 1.27
most severe core conditions resulting from an accidental
depressurization of the RCS are associated with an inadvertent 1.28
opening of a pressurizer safety valve. Initially, the event results 1.29
in a rapidly decreasing RCS pressure until this pressure reaches a
value corresponding to the hot leg saturation pressure. At this 1.31
time, the pressure decrease is slowed considerably. The pressure 1.32
continues to decrease throughout the transient. The effect of the 1.33
pressure decrease would be to decrease power via the moderator
density feedback, but the RCS (if in the automatic mode) functions to 1.34
maintain the power essentially constant throughout the initial stage 1.35
of the transient. The average coolant temperature decreases slowly, 1.36
but the pressurizer level increases until reactor trip.

The reactor may be tripped by the following reactor protection system 1.38
signals:

1. Overtemperature ΔT 1.40

2. Pressurizer low pressure 1.41

An inadvertent opening of a pressurizer safety valve is classified as 1.44
an American Nuclear Society (ANS) Condition II event, a fault of 1.45
moderate frequency. Section 15.0.1 discusses Condition II events. 1.46

15.6.1.2 Analysis of Effects and Consequences 1.48

Method of Analysis 1.50

The accidental depressurization transient is analyzed by employing 1.52
the detailed digital computer code LOFTRAN (WCAP-7907, 1972). The 1.55
code simulates the neutron kinetics, RCS, pressurizer, pressurizer
relief and safety valves, pressurizer spray, steam generator, and 1.56
steam generator safety valves. The code computes pertinent plant 1.57
variables, including temperatures, pressures, and power level.

Initial operating conditions are assumed at values consistent with 1.58
steady state N-loop and N-1 loop operations. Plant characteristics 1.59
and initial conditions are discussed in Section 15.0.3. To give 1.60
conservative results in calculating the departure from nucleate
boiling ratio (DNBR) during the transient, the following assumptions 2.1
are made.

1. Initial conditions of maximum core power and reactor coolant 2.3
temperatures and minimum reactor coolant pressure resulting 2.4
in the minimum initial margin to DNB (Section 15.0.3).

2. A least negative moderator coefficient of reactivity is 2.5
assumed. The spatial affect of void due to local or 2.6
subcooled boiling is not considered in the analysis with
respect to reactivity feedback or core power shape. 2.7

3. A large (absolute value) Doppler coefficient of reactivity 2.8
such that the resultant amount of positive feedback is
conservatively high in order to retard any power decrease 2.9
due to moderator reactivity feedback.

4. Cases are analyzed considering four loops in operation and 2.10
three loops in operation.

Plant systems and equipment which are necessary to mitigate the 2.12
effects of RCS depressurization caused by an inadvertent safety valve 2.13
opening are discussed in Section 15.0.8 and listed in Table 15.0-6.

Normal RCS are not required to function. The rod control system is 2.15
assumed to be in the automatic mode to hold the core at full power
longer and thus delay the trip. This is a worst case assumption; if 2.17
the reactor were in manual control, a trip could occur earlier on
overtemperature ΔT or low pressurizer pressure. The reactor 2.19
protector system functions to trip the reactor on the appropriate
signal. No single active failure will prevent the reactor protection 2.20
system from functioning properly.

Results 2.23

The system response to an inadvertent opening of a pressurizer safety 2.25
valve is shown on Figures 15.6-1 and 15.6-2. Figure 15.6-1 2.28
illustrates the nuclear power transient following the
depressurization. Nuclear power is maintained at the initial value 2.29

until reactor trip occurs on overtemperature ΔT. The pressure decay ~~2.30~~
transient and average temperature transient following the accident
are given on Figure 15.6-2. Pressure drops more rapidly while core 2.31
heat generation is reduced via the trip, and then slows once
saturation temperature is reached in the hot leg. The departure from 2.33
nucleate boiling ratio (DNBR) decreases initially, but increases 2.34
rapidly following the trip (Figure 15.6-1). The DNBR remains above 2.35
1.30 throughout the transient. The DNBR design basis is described in 2.36
4.4 Section [illegible]. Figures 15.6-1A and 15.6-2A show the transient for the 2.37
case with two reactor coolant loops in service (N-1 loop operation). 2.38

The calculated sequence of events for the inadvertent opening of a 2.39
pressurizer safety valve incident is shown in Table 15.6-1. 2.40

15.6.1.3 Conclusions 2.42

The results of the analysis show that the pressurizer low pressure 2.43
and the overtemperature ΔT reactor protection system signals provide 2.~~45~~
adequate protection against the RCS depressurization event. The DNBR 2.46
remains above 1.30 throughout the transient; thus, the DNB design
basis as described in Section 4.4 is met. The radiological 2.47
consequences of this event are not limiting.

15.6.1.4 Radiological Consequences 2.49

An inadvertent opening of a pressurizer safety or relief valve 2.50
releases primary coolant to the pressurizer relief tank; however, 2.53
even assuming a direct release to the containment atmosphere, the
radiological consequences of this event would be substantially less 2.54
than that of a LOCA (Section 15.6.5) because less primary coolant is
released and the activity is lower as fuel damage is not predicted as 2.55
a result of this event.

15.6.2 Failure of Small Lines Carrying Primary Coolant Outside Containment 2.59

There are no instrument lines connected to the RCS that penetrate the 3.2
containment. There are, however, the sample lines from the hot legs 3.4
of reactor coolant loops 2 and 3 from the steam and liquid space of 3.5
the pressurizer, and from the chemical and volume control system
(CHS) letdown and excess letdown lines penetrating the containment. 3.6
The sample lines are provided with normally closed isolation valves 3.7
on both sides of the containment wall. The CHS letdown and excess 3.8
letdown lines are provided with normally open containment isolation
valves on both sides of the containment wall. In all cases the 3.10
containment isolation valves are designed in accordance with the
requirements of General Design Criterion 55. 3.11

The most severe pipe rupture with regard to radioactivity release 3.12
during normal plant operation occurs in the CHS. This would be a 3.13
complete severance of the 3-inch letdown line just outside
containment, but between the outboard letdown isolation valve and 3.14
letdown heat exchanger (Figure 9.3-8), at rated power condition. The 3.16
occurrence of a complete severance of the letdown line would result

in a loss of reactor coolant at the rate of approximately 152 gpm 3.17
(referenced at a density of 62 lb/ft^3) which would not cause
engineered safety features system actuation. 3.18 450.4

Area radiation and leakage detection instrumentation provide the 3.19
means for detection of a letdown line rupture. Frequent operation of 3.20
the CHS reactor makeup control system and other CHS instrumentation
will also aid the operator in diagnosing a letdown line rupture. The 3.22 450.4
time required for the operator to identify the accident and manually
isolate the rupture is expected to be within 30 minutes of the 3.23
rupture. Once the rupture is identified, the operator would isolate 3.24
the letdown line rupture by closing the letdown orifice isolation 3.25
valves followed by closing the pressurizer low level isolation 3.26
valves. Alternatively, the operator would close the letdown line 3.27
containment isolation valves to isolate the rupture. All valves are 3.28
provided with control switches at the main control board. There are 3.29
no single failures that would prevent isolation of the letdown line
rupture.

Radiological Consequences 3.32 450.4

The plant is assumed to be operating at the technical specification 3.34
primary coolant activity and primary-to-secondary leakage through all 3.37
four steam generators.

The complete severance of the letdown line results in a LOCA at the 3.38
rate of approximately 152 gpm which may not result in the activation 3.39
of the engineered safety features (ESF) systems for the duration of 3.40 450.4
the release. This implies that the supplementary leak collection and 3.41
release system (SLCRS) and auxiliary building filters are not in 3.42
operation and the releases to the environment from the severed line
are assumed to be at ground level. The time needed to identify and 3.43
isolate the rupture is conservatively assumed to be 30 minutes. 450.4

The fraction of iodine release to the environment is derived from a 3.44
calculated fraction of approximately 0.40 of the primary coolant 3.45
flashing during pipe leakage. This is based on a direct release of 3.46 450.4
primary coolant at primary coolant temperature, which conservatively
bounds potential accident sequences. 3.47

Due to transients in the core at the time of the accident, it is 3.48
assumed that an iodine spike occurs concurrently with the letdown 3.49
line rupture.

The radiological consequences of the postulated small line break are 3.50
reported in Table 15.0-8. The assumptions used to perform this 3.51
evaluation are summarized in Table 15.6-2. Technical specification 3.52 450.4
column values have been used. The releases listed in Table 15.6-3 3.53
together with the atmoshperic dispersion values listed in
Table 15.0-11 are used to compute the doses to the EAB (0-2 hr). The 3.55
resulting doses to the exclusion area boundary (EAB) are a small
fraction of the 10CFR100 guidelines; i.e., 30 Rem to the thyroid and 3.56
2.5 Rem to the whole body.

15.6.3 Steam Generator Tube Failure 3.58

15.6.3.1 Identification of Causes and Accident Description 3.59

The accident examined is the complete severance of a single steam 3.60
generator tube. This event is considered an ANS Condition IV event, 4.2
a limiting fault (Section 15.0.1). The accident is assumed to take 4.3
place at full power with the reactor coolant contaminated with
fission products corresponding to the continuous operation with a 4.4
limited amount of defective fuel rods. The accident leads to an 4.5
increase in contamination of the secondary system due to leakage of
radioactive coolant from the RCS. In the event of a coincident loss 4.7
of offsite power, or failure of the condenser steam dump system,
discharge of activity to the atmosphere takes place via the steam 4.8
generator safety and/or power operated relief valves.

Complete severance of the steam generator tube rupture is considered 4.9
a somewhat conservative assumption since the Inconel-600 tube 4.10
material is highly ductile. The more probable mode of tube failure 4.11
would be one or more minor leaks of undetermined origin. Activity in 4.12
the steam and power conversion system is subject to continual
surveillance and an accumulation of minor leaks which exceed the 4.13
limits established in the Technical Specifications is not permitted
during unit operation.

The operator is expected to determine that a steam generator tube 4.14
rupture has occurred, and to identify and isolate the faulty steam 4.15
generator on a restricted time scale to minimize contamination of the
secondary system and ensure termination of radioactive release to the 4.16
atmosphere from the faulty unit. The recovery procedure can be 4.17
carried out on a time scale which ensures that break flow to the
secondary system is terminated before water level in the affected 4.18
steam generator rises into the main steam pipe. Sufficient 4.19
indications and controls are provided to enable the operator to carry
out these functions satisfactorily.

Consideration of the indications provided at the control board, 4.20
together with the magnitude of the break flow, leads to the 4.21
conclusion that the accident diagnostics and isolation procedure can
be completed within 30 minutes of initiation for the design basis 4.22
event.

If normal operation of the various plant control systems is assumed, 4.23
the following sequence of events is initiated by a tube rupture. 4.24

1. Pressurizer low pressure and low level alarms are actuated 4.26
and charging pump flow increases in an attempt to maintain 4.27
pressurizer level. On the secondary side, steam 4.28
flow/feedwater flow mismatch occurs as feedwater flow to the
affected steam generator is reduced as a result of primary 4.29
coolant break flow to that unit.

2. Decrease in RCS pressure (Figure 15.6-3A) due to continued 4.30
loss of reactor coolant inventory leads to a reactor trip 4.31

signal on low pressurizer pressure or overtemperature T.
Resultant plant cooldown (Figure 15.6-3B) following reactor 4.32
trip leads to a rapid decrease in pressurizer level
(Figure 15.6-3E), and a safety injection signal, initiated 4.33
by low pressurizer pressure, follows soon after reactor
trip. The safety injection signal automatically terminates 4.35
normal feedwater supply and initiates auxiliary feedwater
addition. 4.36

3. The reactor trip automatically trips the turbine and if 4.37
offsite power is available the steam dump valves open,
permitting steam dump to the condenser. In the event of a 4.39
coincident station blackout (loss of offsite power), as
assumed in the transients presented in this section, the 4.40
steam dump valves automatically close to protect the
condenser. The steam generator pressure (Figure 15.6-3C) 4.41
rapidly increases, resulting in steam discharge to the
atmosphere through the steam generator safety and/or power 4.42
operated relief valves. The steam flow is presented as a 4.43
function of time on Figure 15.6-3F. The flow is constant 4.44
initially until reactor trip, followed by turbine trip,
which results in a large decrease in flow and a rapid 4.45
increase in steam pressure to the safety valve setpoint.

4. Following the safety injection signal, the continued action 4.46
of the auxiliary feedwater supply and borated safety
injection flow provide a heat sink which absorbs the decay 4.47
heat. 4.48

5. Safety injection flow results in increasing pressurizer 4.49
water volume (Figure 15.6-3E).

6. At this point, steam generator blowdown is manually 4.50
isolated. In addition to the indication mentioned above, 4.51
the condenser air ejector monitor and the steam generator
blowdown liquid monitor may alarm, indicating a sharp 4.52
increase in radioactivity in the secondary system.

15.6.3.2 Analysis of Effects and Consequences 4.55

Method of Analysis 4.57

Mass and energy balance calculations are performed using LOFTRAN 4.59
(WCAP-7907, 1972) to conservatively determine primary to secondary 5.2
mass release and to conservatively determine the amount of steam
vented from each of the steam generators during the initial 30-minute 5.3
period following the tube rupture.

In estimating the mass transfer from the RCS through the broken tube, 5.4
the following assumptions are made:

1. Reactor trip occurs automatically as a result of low 5.6
pressurizer pressure or overtemperature ΔT. Loss of offsite ~~5.7~~
power occurs at reactor trip.

2. Following the initiation of the safety injection signal, two high head safety injection pumps and two charging pumps are actuated and are assumed in the analyses to continue to deliver flow until 30 minutes after accident initiation. 5.9 5.10 5.11

3. After reactor trip the break flow reaches equilibrium when incoming safety injection flow is balanced by outgoing break flow as shown on Figure 15.6-3. Break flow is assumed to persist for 30 minutes beyond initiation of the accident. 5.12 5.13 5.14

4. The steam generators are controlled at the safety valve setting rather than at the power operated relief valve setting. 5.15 5.16

5. The operator is assumed to throttle the auxiliary feedwater flow to attempt control of the steam generator level. 5.17

6. The operator identifies the accident type and terminates break flow to the faulted steam generator within 30 minutes of accident initiation. 5.18 5.19

The above assumptions, suitably conservative for the design basis tube rupture, are made to maximize doses and do not explicitly model operator actions for recovery. 5.21 5.22

Prior to reactor trip, steam is dumped to the condenser from both the faulted and nonfaulted steam generators. After the condenser is lost, following loss of offsite power at reactor trip, steam from all steam generators is released to the atmosphere. 5.23 5.24 5.25

Following isolation of the faulted steam generator, it is assumed that steam dump from the nonfaulted steam generator is used to reduce the RCS temperature to 50°F below no-load T_{avg} (557°F). From 2 to 8 hours, steam is assumed to be dumped from the non-faulted steam generators to reduce the RCS temperature and pressure to residual heat removal systems (RHRS) conditions. The faulted steam generator is depressurized to the RHRS cut-in pressure via steam release from the faulted steam generator pilot operated relief valve. After 8 hours, further plant cooldown is carried out with the RHRS. The 0.5-to 2-hour and 2-to 8-hour steam releases from and feedwater flows to the steam generators required to remove decay heat, heat due to an operating reactor coolant pump, and stored fluid energy in the RCS and steam generators are determined based on these assumptions. 5.26 5.27 5.28 5.29 5.30 5.31 5.33 5.34 5.36 5.37

<u>Key Recovery Sequence</u> 5.40

The recovery sequence to be followed consists of the following major operator actions: 5.42

1. identification of the faulted steam generator; 5.44

2. isolation of the faulted steam generator; 5.45

3. cooldown of the RCS fluid to approximately 50°F below no 5.47
load temperature to assure subcooling in the RCS non-faulted 5.48
loops at the faulted steam generator pressur.;

4. controlled depressurization of the RCS to a value equal to 5.49
the faulted steam generator pressure; and

5. subsequent termination of safety injection flow. 5.50

Results 5.53

The sequence of events for steam generator tube rupture is presented 5.55
in Table 15.6-4. These events are the normal plant response to 5.56
normal plant setpoints. Loss of offsite power is assumed to occur at 5.57
reactor trip.

The previously discussed assumptions lead to an estimate of 104,500 5.59
pounds for the total amount of reactor coolants transferred to the 5.60
secondary side of the faulted steam generator as a result of a tube
rupture accident. The steam releases to the condenser and atmosphere 6.1
from both the faulted and non-faulted steam generators are given in
Table 15.6-5. The total feedwater flows to all steam generators are 6.3
also listed in Tables 15.6-5 and 15.6-6.

The following is a list of figures of pertinent time-dependent 6.5
parameters:

Figure 15.6-3A - Core Pressure 6.7
Figure 15.6-3B - Reactor Coolant System Temperature 6.8
Figure 15.6-3C - Steam Generator Pressure (Faulted Steam 6.9
Generator) 6.10
Figure 15.6-3D - Steam Generator Temperature (Faulted Steam 6.11
Generator) 6.12
Figure 15.6-3E - Pressurizer Water Volume 6.13
Figure 15.6-3F - Steam Generator Flow (Faulted Steam Generator) 6.14
Figure 15.6-3G - Feedwater Flow to Faulted Steam Generator 6.15
Figure 15.6-3H - Faulted Steam Generator Safety/Relief Valve Flow 6.16
Rate 6.17
Figure 15.6-3I - Faulted Steam Generator Break Flow Rate 6.18
Figure 15.6-3J - Steam Generator Mass 6.19
Figure 15.6-3K - Faulted Steam Generator Liquid Volume 6.20

The DNB calculations performed with LOFTRAN (WCAP-7907, 1972) 6.23
indicate that DNB limits are met.

15.6.3.3 Radiological Consequences 6.25

It is postulated that if a steam generator tube rupture occurs, the 6.26
fraction of reactor coolant system activity resulting from long-term 6.27
operation with minor fuel defects is released to the secondary side
of the steam generator. The release to the environment from the 6.30
defective steam generator continues until this steam generator is
isolated 30 minutes after the rupture occurs. Offsite power is 6.33
conservatively assumed to be lost following the tube rupture,

resulting in the condenser being unavailable for steam dump. The 6.35
nondefective steam generators continue to release activity through
the steam relief valves until the plant is brought to a cold shutdown 6.36
8 hours after the tube rupture. The analysis considers the effect of 6.37
the highest worth control rod stuck out of the core. The calculation 6.38 440.61
indicates that the DNB limits are met, thereby precluding any release
from additional fuel failure. The activities released are based on 6.39
technical specification primary to secondary leak rate of 1 gallon
per minute. It is also assumed the reactor coolant and secondary 6.41
coolant are at equilibrium technical specification activity
concentrations. A partition factor of 0.01 is assumed for iodines 6.43
passing between the water and steam phases in both the defective and
nondefective steam generators. 6.44

The assumptions used to calculate the doses to the EAB and the LPZ 6.45
from a steam generator tube rupture are summarized in Table 15.6-5. 6.46
Two cases are analyzed to ascertain the results of increased reactor 6.47
coolant iodine concentration resulting from operating transients: 6.48

1. the iodine activity in the primary coolant is increased due 6.50
   to a preaccident iodine spike, and

2. an iodine spike occurs concurrently with the steam generator 6.51
   tube rupture.

The calculated activities released to the environment as a result of 6.53
a steam generator tube rupture based on the parameters described in 6.54
Tables 15.6-5 and 15.6-6 are shown in Table 15.6-7. The tabulated 6.55
release values are for the two cases examined:

1. an assumed preaccident iodine spike condition in the reactor 6.57
   coolant, and

2. an accident initiated concurrent iodine spike. 6.58

The releases together with the atmospheric dispersion factors listed 6.60
in Table 15.0-11 are used to compute the doses presented in 7.1
Table 15.0-8 for the EAB (0-2 hr) and the LPZ (0-5 hr).

The whole body and thyroid doses calculated for the postulated 7.2
accident assuming a preaccident iodine spike in the reactor coolant 7.3
is less than the dose guideline values described in 10CFR100, i.e.,
300 Rem to the thyroid and 25 Rem to the whole body. 7.4

For the assumed condition of a concurrent iodine spike in combination 7.5
with equilibrium iodine concentrations at full power, the analysis of 7.6
the postulated accident resulted in dose values less than a small
fraction of 10CFR100; i.e., 30 Rem to the thyroid and 2.5 Rem whole 7.7
body.

15.6.4 Spectrum of BWR Steam System Piping Failures Outside of Containment 7.11

Not applicable to Millstone 3. 7.12

15.6.5 Loss-of-Coolant Accidents Resulting from a Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary 7.18 7.19

15.6.5.1 Identification of Causes and Frequency Classification 7.20

A LOCA is the result of a pipe rupture of the RCPB (Section 5.2). 7.23
For the analyses reported here, a major pipe break (large break) is 7.25
defined as a rupture with a total cross-sectional area equal to or 7.26
greater than 1.0 square feet (ft^2). This event is considered an ANS 7.27
Condition IV event, a limiting fault, in that it is not expected to
occur during the lifetime of the plant but is postulated as a 7.28
conservative design basis (Section 15.0.1). The Millstone 3 nuclear 7.29
steam supply system (NSS) contains loop isolation valves and
therefore can be operated with one loop out of service. For the 7.31
analyses presented here, a postulated major pipe break (large break)
could occur in one of the active loops or in the reactor vessel side 7.32
of the isolated (inactive) loop while in this (N-1) loop
configuration.

A minor pipe break (small break), as considered here, is defined as a 7.33
rupture of the reactor coolant pressure boundary with a total cross- 7.35
sectional area less than 1.0 ft^2 in which the normally operating
charging system flow is not sufficient to sustain pressurizer level 7.36
and pressure. This is considered a Condition III event, in that it 7.37
is an infrequent fault which may occur during the life of the plant. 7.38

The Acceptance criteria for the LOCA are described in 10CFR50.46 7.40
(10CFR50.46 and Appendix K of 10CFR50 1974) as follows. 7.41

1. The calculated peak fuel element clad temperature is below 7.43
the requirement of 2,200°F.

2. The amount of fuel element cladding that reacts chemically 7.44
with water or steam does not exceed 1 percent of the total 7.45
amount of Zircaloy in the reactor.

3. The clad temperature transient is terminated at a time when 7.47
the core geometry is still amenable to cooling. The 7.48
localized cladding oxidation limit of 17 percent is not
exceeded during or after quenching.

4. The core remains amenable to cooling during and after the 7.49
break.

5. The core temperature is reduced and decay heat is removed 7.50
for an extended period of time, as required by the long- 7.51
lived radioactivity remaining in the core.

These criteria were established to provide significant margin in 7.53
emergency core cooling system (ECCS) performance following a LOCA. 7.54
WASH-1400 (1975) presents a recent study in regard to the probability 7.55
of occurrence of RCS pipe ruptures. Input parameters used in the 7.56
ECCS analysis are described in Table 15.6-8.

In all cases, small breaks (less than 1.0 square foot) yield results 7.57
with more margin to the Acceptance Criteria limits than large breaks. 7.60

15.6.5.2 Sequence of Events and Systems Operations 8.2

Should a major break occur, depressurization of the RCS results in a 8.3
pressure decrease in the pressurizer. The reactor trip signal 8.5
subsequently occurs when the pressurizer low pressure trip setpoint 8.6
is reached. A safety injection signal is generated when the 8.7
appropriate setpoint is reached. These countermeasures will limit 8.9
the consequences of the accident in two ways.

1. Reactor trip and borated water injection complement void 8.11
formation in causing rapid reduction of power to a residual
level corresponding to fission product decay heat. However, 8.13
no credit is taken in the LOCA analysis for boron content on
the injection water. In addition, the insertion of control 8.14
rods to shut down the reactor is neglected in the large
break analysis.

2. Injection of borated water provides for heat transfer from 8.15
the core and prevents excessive clad temperatures.

Description of Large Break LOCA Transient 8.18

The sequence of events following a large break LOCA are presented in 8.20
Table 15.6-1 and on Figure 15.6-4.

Before the break occurs, the unit is in an equilibrium condition, 8.22
i.e., the heat generated in the core is being removed via the 8.23
secondary system. During blowdown, heat from fission product decay, 8.24
hot internals and the vessel continues to be transferred to the
reactor coolant. At the beginning of the blowdown phase, the entire 8.26
RCS contains subcooled liquid which transfers heat from the core by
forced convection with some fully developed nucleate boiling. 8.27
Thereafter, the core heat transfer is based on local conditions with 8.28
transition boiling and forced convection to steam as the major heat 8.29
transfer mechanisms.

The heat transfer between the RCS and the secondary system may be in 8.30
either direction depending on the relative temperatures. In the case 8.32
of continued heat addition to the secondary, secondary system
pressure increases and the main steam safety valves may actuate to 8.33
limit the pressure. Makeup water to the secondary side is 8.34
automatically provided by the auxiliary feedwater system. The safety 8.36
injection signal actuates a feedwater isolation signal which isolates
normal feedwater flow by closing the main feedwater isolation valves, 8.37
and also initiates auxiliary feedwater flow by starting the auxiliary
feedwater pumps. The secondary flow aids in the reduction of RCS 8.38
pressure.

When the RCS depressurizes to 615 psia, the accumulators begin to 8.39
inject borated water into the reactor coolant loops. Since the loss 8.41
of offsite power is assumed, the reactor coolant pumps are assumed to

trip at the inception of the accident. The effects of pump coastdown 8.43
are included in the blowdown analysis.

1280

The blowdown phase of the transient ends when the RCS pressure 8.45
(initially assumed at [redacted] psia) falls to a value approaching that 8.46
of the containment atmosphere. Prior to or at the end of the 8.47
blowdown, the mechanisms that are responsible for the bypassing of
emergency core cooling water injected into the RCS are calculated not 8.48
to be effective. At this time (end-of-bypass) refill of the reactor 8.49
vessel lower plenum begins. Refill is complete when emergency core 8.50
cooling water has filled the lower plenum of the reactor vessel which
is bounded by the bottom of the fuel rods (bottom of core recovery 8.51
time).

The reflood phase of the transient is defined as the time period 8.52
lasting from the end-of-refill until the reactor vessel has been 8.53
filled with water to the extent that the core temperature rise has
been terminated. From the later stage of blowdown and then the 8.54
beginning-of-reflood, the safety injection accumulator tanks rapidly
discharge borated cooling water into the RCS, contributing to the 8.55
filling of the reactor vessel downcomer. The downcomer water 8.56
elevation head provides the driving force required for the reflooding
of the reactor core. The low head and high head safety injection 8.57
pumps aid in the filling of the downcomer and subsequently supply
water to maintain a full downcomer and complete the reflooding 8.58
process.

Continued operation of the ECCS pumps supplies water during long-term 8.59
cooling. Core temperatures have been reduced to long-term steady 8.60
state levels associated with dissipation of residual heat generation. 9.1
After the water level of the refueling water storage tank reaches a 9.2
minimum allowable value, coolant for long-term cooling of the core is 9.3
obtained by switching to the cold recirculation phase of operation in
which spilled borated water is drawn from the engineered safety 9.5
features containment sumps by the containment recirculation pumps and 9.6
returned to the RCS cold legs. The containment recirculation system 9.7
also continues to operate in the spray mode to further reduce
containment pressure. Approximately 24 hours after initiation of the 9.9
LOCA, the ECCS is realigned to supply water to the RCS hot legs in
order to control the boric acid concentration in the reactor vessel. 9.10

<u>Description of Small Break LOCA Transient</u> 9.13

As contrasted with the large break, the blowdown phase of the small 9.15
break occurs over a longer time period. Thus, for the small break 9.17
LOCA there are only three characteristic stages, i.e., a gradual
blowdown in which the decrease in water level is checked, core 9.18
recovery, and long-term recirculation.

15.6.5.3 Core and System Performance 9.21

1. Mathematical Model 9.23

The requirements of an acceptable ECCS evaluation model are 9.25
presented in Appendix K of 10CFR50 (1974).

a. Large Break LOCA Evaluation Model 9.28

The analysis of a large break LOCA transient is divided 9.30
into three phases: 1) blowdown, 2) refill, and
3) reflood. There are three distinct transients 9.33
analyzed in each phase, including the thermal-hydraulic
transient in the RCS, the pressure and temperature 9.35
transient within the containment, and the fuel and clad
temperature transient of the hottest fuel rod in the 9.36
core. Based on these considerations, a system of 9.37
interrelated computer codes has been developed for the
analysis of the LOCA.

A description of the various aspects of the LOCA 9.38
analysis methodology is given by WCAP-8339 (1974).
This document describes the major phenomena modeled, 9.39
the interfaces among the computer codes, and the
features of the codes which ensure compliance with the 9.40
Acceptance Criteria. The SATAN-VI, WREFLOOD, COCO, and 9.41
LOCTA-IV codes, which are used in the LOCA analysis,
are described in detail in WCAP-8301 and WCAP-8305 9.42
(1974) WCAP-8170 and WCAP-8171 (1974); WCAP-8327 and
WCAP-8326 (1974); and WCAP-8302 and WCAP-8306 (1974). 9.43
Code modifications are specified in WCAP-8471 and 9.44
WCAP-8472 (1975), WCAP-8622 and WCAP-8623 (1975).
These codes assess the core heat transfer geometry and 9.45
determine if the core remains amenable to cooling
throughout and subsequent to the blowdown, refill, and 9.46
reflood phases of the LOCA. The SATAN-VI computer code 9.47
analyses the thermal-hydraulic transient in the RCS
during blowdown and the WREFLOOD computer code 9.49
calculates this transient during the refill and reflood
phases to the accident. The COCO computer code 9.50
calculates the containment pressure transient during
all three phases of the LOCA analysis. Similarly, the 9.52
LOCTA-IV computer computes the thermal transient of the
hottest fuel rod during the three phases. 9.53

SATAN-VI calculates the RCS pressure, enthalpy, 9.54
density, and the mass and energy flow rates in the RCS,
as well as steam generator energy transfer between the 9.55
primary and secondary systems as a function of time
during the blowdown phase of the LOCA. SATAN-VI also 9.57
calculates the accumulator water mass and internal
pressure and the pipe break mass and energy flow rates
that are assumed to be vented to the containment during 9.58
blowdown. At the end of the blowdown phase, these data 9.59

are transferred to the WREFLOOD code. Also, at the 9.60
end-of-blowdown, the mass and energy release rates
during blowdown are transferred to the COCO code for
use in the determination of the containment pressure 10.1
response during this first phase of the LOCA.
Additional SATAN-VI output data from the end-of- 10.2
blowdown, including the core inlet flow rate and
enthalpy, the core pressure, and the core power decay 10.3
transient, are input to the LOCTA-IV code.

With input from the SATAN-VI code, WREFLOOD uses a 10.4
system thermal-hydraulic model to determine the core
flooding rate (i.e., the rate at which coolant enters 10.5
the bottom of the core), the coolant pressure and
temperature, and the quench front height during the 10.6
reflood phase of the LOCA. WREFLOOD also calculates 10.7
the mass and energy flow addition to the containment
through the break. Since the mass flow rate to the 10.8
containment depends upon the core flooding rate and the
local core pressure, which is a function of the 10.9
containment back-pressure, the WREFLOOD and COCO codes
are interactively linked. WREFLOOD is also linked to 10.10
the LOCTA-IV code in that thermal-hydraulic parameters
from WREFLOOD are used by LOCTA-TV in its calculation 10.11
of the fuel temperature. LOCTA-IV is used throughout 10.12
the analysis of he LOCA transient to calculate the fuel
clad temperature and metal-water reaction of the 10.13
hottest rod in the core.

The large break analysis was performed with the 10.14
December 1981 version of the Evaluation Model, which
includes modifications delineated in WCAP-9220-P-A and 10.15
WCAP-9221-P-A (1981).

The (N-1) large break analysis was performed with the 10.16
December 1981 version of the ECCS Evaluation Model as
described above and incorporated those changes 10.17
necessary to model the loop out of service. These 10.19
changes are delineated in WCAP-8904-A.

The analysis in this section was performed with the 10.20
upper head fluid temperature equal to the RCS cold leg 10.21
fluid temperature, achieved by increasing the upper
head cooling flow (WCAP-7907, 1972).

b. Small break LOCA Evaluation Model 10.24

The WFLASH program used in the analysis of the small 10.26
break LOCA is an extension of the FLASH-4 code
(WARD-TM-84, 1969) developed at the Westinghouse Bettis 10.28
Atomic Power Laboratory. The WFLASH program permits a 10.29
detailed spatial representation of the RCS.

The RCS is nodalized into volumes interconnected by 10.31
flowpaths. The broken loop is modeled explicitly with 10.32
the intact loops lumped into a second loop. The 10.33
transient behavior of the system is determined from the
governing conservation equations of mass energy and
momentum applied throughout the system. A detailed 10.35
description of WFLASH is given in WCAP-8200 and WCAP-
8261 (1974).

The use of WFLASH in the analysis involves, among other 10.36
things, the representation of the reactor core as a
heated control volume with the associated bubble rise 10.37
model to permit a transient mixture height calculation.
The multinode capability of the program enables an 10.38
explicit and detailed spatial representation of various
system components. In particular, it enables a proper 10.40
calculation of the behavior of the loop seal during a 10.41
loss-of-coolant transient.

Clad thermal analyses are performed with the LOCTA-IV 10.42
code (WCAP-8301 and WCAP-8305, 1974), which uses the
RCS pressure, fuel rod power history, steam flow past 10.44
the uncovered part of the core and mixture height
history from the WFLASH hydraulic calculations as 10.45
input.

Figure 15.6-7 presents the hot rod power shape utilized 10.46
to perform the small break analysis presented here.
This power shape was chosen because it provides an 10.47
appropriate distribution of power versus core height,
and also because local power is maximized in the upper 10.48
regions of the reactor core (10 feet to 12 feet). This 10.49
power shape is skewed to the top of the core with the
peak local power occurring at the 10.0-foot core
elevation. This is limiting for the small break 10.51
analysis because of the core uncovery process for small
breaks. As the core uncovers, the cladding in the 10.52
upper elevation of the core heats up and is sensitive
to the local power at that elevation. The cladding 10.54
temperatures in the lower elevation of the core, below
the two-phase mixture height, remains low. The peak 10.55
clad temperature occurs above 10 feet.

Schematic representations of the computer code 10.56
interfaces are given on Figures 15.6-5 and 15.6-6.

The small break analysis was performed with the 10.57
October 1975 version of the Westinghouse ECCS Small
Break Evaluation Model (WCAP-8301 and WCAP-8305, 1974; 10.58
WCAP-8200 and WCAP-8261, 1974; WCAP-8970 and WCAP-8976,
1977; and Eicheldinger 1978). 10.60

2. Input Parameters and Initial Conditions 11.3

Table 15.6-9 lists important input parameters and initial 11.5
conditions used in the analysis of both N-Loop and N-1 loop 11.6
configurations.

The analysis presented in this section was performed with a 11.8
reactor vessel upper head temperature equal to the RCS cold 11.9
leg temperature. The effect of using the cold leg 11.10
temperature in the reactor vessel upper head is described in
WCAP-7907, 1972. In addition, the analysis in this section 11.11
utilized the upflow barrel-baffle methodology described in
WCAP-9168 and WCAP-9169, 1977. 11.12

The bases used to select the numerical values that are input 11.13
parameters to the analysis have been conservatively 11.14
determined from extensive sensitivity studies (WCAP-8341 and
WCAP-8342, 1974; WCAP-8340 and WCAP-8356,1974; WCAP-8586-P-A 11.15
and WCAP-8566-A, 1975). In addition, the requirements of 11.16
Appendix K regarding specific model features were met by
selecting models which provide a significant overall 11.17
conservatism in the analysis. The assumptions made pertain 11.18
to the conditions of the reactor and associated safety
system equipment at the time that the LOCA occurs and 11.20
include such items as the core peaking factors, the
containment pressure, and the performance of the ECCS. 11.21
Decay heat generated throughout the transient is also 11.22
conservatively calculated.

3. Results 11.25

a. Large Break Results 11.27

Based on the results of the LOCA sensitivity studies 11.29
(WCAP-8341 and WCAP-8342 1974; WCAP-8340 and WCAP-8356,
1974; WCAP-8586-P-A and WCAP-8566-A, 1975), the 11.30
limiting large break was found to be the double ended
cold leg guillotine (DECLG). Therefore, only the DECLG 11.33
beak is considered in the large break ECCS performance
analysis. Calculations were performed for a range of 11.34
Moody break discharge coefficients. The results of 11.35
these calculations are summarized in Tables 15.6-1 and
15.6-11.

In accordance with the methodology presented in the 11.36
Letter from E.P. Rahe of W to R.L. Tedesco of the NRC,
the worst break for this plant (C_D=0.6) was repeated, ~~11.38~~ D
assuming no single failure (maximum safeguards). The 11.39
most limiting of the minimum or maximum safeguards
results for the limiting break is taken as the limiting
result for the plant. 11.40

Based on LOCA sensitivity studies for the N-1 loop 11.41
configuration (WCAP-8904, 1979), the limiting N-1 large

break was again found to be the double ended cold leg 11.42
guillotine (DECLG), occurring in one of the active
loops. Calculations were performed for a range of 11.43
Moody discharge coefficients. For the worst break 11.44
analyzed, an additional calculation was performed with
the break occurring in the inactive loop to bound the 11.45
results. The results of these calculations are 11.46
summarized in Tables 15.6-1 and 15.6-11.

The mass and energy release data for the break 11.47
resulting in the highest calculated peak clad
temperature are presented in Section 6.2.1.5 for both 11.49
N-loop and N-1 loop configurations.

Figures 15.6-8 through 15.6-30 present the parameters 11.50
of principal interest for the N-loop large break ECCS
analyses. Figures 15.6-44 through 15.6-67 present the 11.51
parameters of principal interest for the N-1 loop large
break ECCS analysis. For all cases analyzed, transients 11.63
of the following parameters are presented:

- hot spot clad temperature; 11.55
- coolant pressure in the reactor core; 11.56
- water level in the core and downcomer during reflood; 11.57
- core reflooding rate; and 11.58
- thermal power during blowdown. 11.59

The minimum containment pressure transient resulting 12.1
from a LOCA is presented in Section 6.2.1.5. For the 12.2
limiting break analyzed, the following additional
transient parameters are presented:

- core flow during blowdown (inlet and outlet); 12.4
- core heat transfer coefficients; 12.5
- hot spot fluid temperature; 12.6
- mass released to containment during blowdown; 12.7
- energy released to containment during blowdown; 12.8
- fluid quality in the hot assembly during blowdown; 12.9
- mass velocity during blowdown~~s~~; 12.10
- accumulator water flow rate during blowdown; and 12.11

- pumped safety injection water flow rate during 12.12
  reflood.

The maximum clad temperature calculated for the N-loop 12.14
large break is 2,132°F, which is less than the
acceptance criteria limit of 2,200°F. The maximum clad 12.16
temperature calculated for the N-1 active loop large
break is 1,878°F, which is also less than the
acceptance criteria limit of 2,200°F. The maximum 12.18
local metal-water reaction for the N-loop is
6.47 percent, and is 8.9 percent for the N-1 loop
analysis which are well below the embrittlement limit 12.19
of 17 percent as required by 10CFR50.46. The total 12.20
core metal-water reaction is less than 0.3 percent for
all breaks, (both N-loop and N-1 loop) as compared with 12.21
the 1 percent criterion of 10CFR50.46. The clad 12.22
temperature transient is terminated at a time when the
core geometry is still amendable to cooling. As a 12.23
result, the core temperature will continue to drop and
the ability to remove decay heat generated in the fuel 12.24
for an extended period of time will be provided.

b. Small Break Results 12.27

As noted previously, the calculated peak clad 12.29
temperature resulting from a small break LOCA is less
than that calculated for a large break. Based on the 12.32
results of the LOCA sensitivity studies (WCAP-8341 and
WCAP-8342, 1974) the limiting small break was found to 12.33
be less than a 10-inch diameter rupture of the RCS cold
leg. Therefore, a range of small break analyses are 12.34
presented, which establishes the limiting break size.
The results of these analyses are summarized in 12.35
Tables 15.6-1 and 15.6-11.

Figure 15.6-7 and Figures 15.6-31 through 15.6-43 12.36
present the principal parameters of interest for the
small break ECCS analyses. For all cases analyzed the 12.40
following transient parameters are presented:

- RCS pressure; 12.42
- core mixture height; and 12.43
- hot spot clad temperature. 12.44

For the limiting break analyzed, the following 12.46
additional transient parameters are presented:

- RCS pressure; 12.48
- core mixture height; and 12.49
- hot spot clad temperature. 12.50

For the limiting break analyzed, the following 12.52
additional transient parameters are presented:

- core steam flow rate; 12.54
- core heat transfer coefficient; 12.55
- hot spot fluid temperature. 12.56

The maximum calculated peak clad temperature for small 12.58
breaks analyzed is 1,483°F. These results are well 12.59
below all Acceptance Criteria limits of 10CFR50.46 and
in all cases are not limiting when compared to the 12.60
results presented for large breaks.

15.6.5.4 Radiological Consequences 13.3

A LOCA would increase the pressure in the primary containment 13.4
building resulting in containment isolation and initiation of the 13.5
ECCS and the containment spray systems. A safety injection system 13.7
(SIS) signal automatically starts operation of the supplementary leak
collection and release system (SLCRS). This system maintains a 13.9
negative pressure within the secondary containment enclosure,
auxiliary building, engineered safety features (ESF) building, 13.10
hydrogen recombiner building, and the main steam valve building 13.11
during accident conditions. The nuclide inventory assumed to be 13.12
initially available for release from within the containment building
consists of 100 percent of the core noble gases and 25 percent of the 13.13
halogens, as described in Regulatory Guide 1.4.

The SIS signal also initiates control room isolation and 13.14
pressurization with bottled air for a period of 1 hours. There is a 13.15
1-minute delay from control room isolation to initiation of the
bottled air system. The control room remains pressurized for that 13.16
1-minute due to ventilation system operation prior to isolation.
Therefore, the control room is effectively pressurized for a period 13.17
of 61 minutes. After 1 hour, leakage from the containment building 13.18
terminates and the control room ventilation system begins operation, 13.19
taking filtered air into the control room. During the period of 13.20
pressurization, 10 cfm of unfiltered inleakage is assumed.

<u>Release Pathways</u> 13.23

The release pathways to the environment subsequent to a loss-of- 13.25
coolant DBA are leakages from the containment building and ESF 13.26
systems, which are collected and processed, and leakage from the 13.28
containment building which is assumed to bypass SLCRS.

<u>Containment Leakage Pathway</u> 13.31

The containment is assumed to leak at the design leak rate for 1-hour 13.33
after the accident. After 1 hour, the ESF systems return the 13.35
containment to subatmospheric conditions, preventing further leakage
from the containment building. 13.37

shielding serves to protect the operators from direct radiation due to the passing cloud of radioactive effluent assumed to have leaked from the containment structure and from the ESF system. The control building walls also provide shielding protection for radiation emanating from buildings located onsite which may contain significant quantities of radioactivity.

A SIS signal from Millstone 3 initiates control room isolation, and after a 60 second delay, pressurization with bottled air for a period of 1 hour. After 1 hour, leakage from the containment building terminates and the control room emergency ventilation system begins operation, taking filtered air into the control room via HEPA and charcoal filters. The calculated whole body dose, beta skin dose, and thyroid dose are presented in Table 15.6-13 and are below the General Design Criterion 19 limits.

Normally, outside air is provided to the control room by the air intake duct, located on the roof of the control building. The intake duct, which is equipped with redundant radiation monitors, will automatically isolate the control room on a high alarm, and after a 60 second delay, the control room is pressurized with bottled air for a period of 1 hour. Approximately 3.7 seconds of continued unfiltered intake is assumed to account for damper actuation and control room isolation subsequent to a signal from either radiation monitor. After 61 minutes, the control room ventilation system will intake outside and recirculation air into the control room via HEPA and charcoal filters.

Since other operating reactors are located on the site, an assessment was made of the habitability of the Millstone 3 control room subsequent to an assumed DBA at either Millstone 1 or 2. All other DBAs (e.g., small line breaks) were considered and found to be bounded by the LOCA. For the assumed DBA at Millstone Unit 2, a low wind speed condition and a high wind speed condition have been analyzed. As described in the Millstone Unit 2 FSAR, Question 6.15.2, Amendment 27, it has been assumed that the high wind speed condition exists for 36 hours after the LOCA and 10 percent of the activity in the enclosure building bypasses the enclosure building filtration system resulting in a ground level release to the environment.

Displacement of the enclosure building atmosphere with outside air would begin at wind speed above 25 mph. However, for conservatism, it is assumed that this displacement would begin with a 23 mph wind. The releases from a postulated LOCA at Millstone 1 are shown in Tables 15.6-14 through 15.6-17, and from a postulated LOCA at Millstone 2 in Tables 15.6-18 through 15.6-21. The calculated Millstone 3 control room whole body dose and thyroid dose from Millstone 1 releases and from Millstone 2 releases are presented in Table 15.6-13. The tabulated control room doses also include the calculated beta skin dose.

Technical Support Center Habitability 14.55

The potential radiation doses to a person occupying the technical 14.57
support center (TSC) have been evaluated for the following design 14.59
basis accidents:

1. Unit 3 LOCA, and 15.1
2. Unit 2 LOCA (high wind speed condition). 15.2

Due to the proximity of the two structures and the ventilation 15.5
intakes (see Figure 1.2-2), meteorological parameters used for the 15.6
control room habitability analysis are applicable to the TSC.

The results of the control room habitability analysis in 15.7
Table 15.6-13, show that for the three units on site, a postulated 15.8
Unit 2 LOCA under high wind speed conditions is the limiting event 15.9
for the Unit 3 control room. Therefore, in order to verify that the 15.10
Unit 3 LOCA is limiting for the TSC, the Unit 2 LOCA high speed wind
case has been analyzed in addition to the Unit 3 LOCA. The dose 15.13
results in Table 15.6-22 confirm that the Unit 3 LOCA is limiting.

The TSC is designed for continuous operation for the duration of the 15.14
accident (i.e., 30 days). The building roof and walls provide 15.15
adequate shielding to protect the occupants against direct radiation
from the external radioactive cloud and from the containment during 15.16
the postulated LOCA. Double vestibule doors are provided at the 15.17
building entrance to eliminate inleakage due to personnel
ingress/egress.

The TSC ventilation system is described in Section 9.4.13. A safety 15.19
injection signal from Unit 3 initiates the CBI signal which in turns
initiates isolation of the TSC. In the event of a LOCA at either 15.20
Unit 1 or Unit 2, the redundant Unit 3 control room intake duct
monitors will initiate a TSC isolation signal upon a high 15.21
radioactivity alarm. This closes the damper in the TSC ventilation 15.22
intake and isolates the building from outside air within 3.7 seconds,
which accounts for the monitor response and the damper closure times. 15.23
Prior to isolation, unfiltered intake at the normal operating rate of 15.24
100 cfm is assumed. Upon isolation, the TSC remains isolated for 30 15.25
minutes with no ventilation intake and 2,000 cfm filtered
recirculation. During this period, a conservative unfiltered 15.27
infiltration rate of 50 cfm is assumed. The actual infiltration rate 15.28
is expected to be much less, since the TSC is predominately a below
ground structure with 2-foot thick concrete walls and a 1-foot thick 15.29
concrete roof. There are a minimum number of penetrations into the 15.31
building; these penetrations are sealed to minimize infiltration of
outside air. In addition, there is an upper floor atop the TSC, 15.33
which further reduces the infiltration to the TSC.

After 30 minutes, there is a 100 cfm filtered ventilation intake of 15.34
the outside air with 1,900 cfm filtered recirculation for the 15.36
duration of the accident. During this period, the TSC is pressurized 15.37
with the intake air and no infiltration is assumed.

The data and assumptions used in the TSC dose evaluation are given in Table 15.6-21. Meteorological parameters (χ/Q values) are given in Table 15.0-11. Releases due to a postulated Unit 3 LOCA are shown in Tables 15.6-19 and 15.6-20. Releases from a Unit 2 LOCA, high wind speed case are shown in Tables 15.6-18 through 15.6-18C. 471.13

The 30-day integrated thyroid dose, whole body gamma dose, and beta skin dose for an individual occupying the TSC following the DBA are presented in Table 15.6-22.

Dose Computation

The radiological dose consequences resulting from a postulated LOCA at Millstone 3 are reported in Table 15.0-8. Assumptions used to perform the evaluation are summarized in Table 15.6-9. The inventory of noble gases and halogens in the containment building atmosphere available for release are presented in Table 15.6-10. Nuclide releases to the environment from containment building leakage and ESF system leakage are shown in Tables 15.6-19 and 15.6-20, respectively. These releases together with atmospheric dispersion factors listed in Table 15.0-11 are used to compute the doses to the EAB (0-2 hr) and LPZ (0-30 day) reported in Table 15.0-8. The calculations show that the thyroid and whole body doses are within the appropriate exposure guideline values specified in 10CFR100. The dose methodology used to determine the results of the hypothetical DBAs are described in Appendix 15A. 471.13

Radiological consequences of a LOCA at Millstone 3 during N-1 loop operation are bounded by the consequences of a LOCA during N-loop operation.

15.6.5.5 Conclusions

Analysis shows that the acceptance criteria described in Section 15.6.5.1 are met and that the radiological consequences are with 10CFR100 guidelines. Control room doses are within the limits of General Design Criterion 19.

15.6.6 BWR Transients

Not applicable to Millstone 3.

15.6.7 References for Section 15.6

Anderson, T. M. 1979. Westinghouse Electric Corporation. Letter to Stolz, J., Nuclear Regulatory Commission (NRC), Letter Number NS-TMA-2030.

Eicheldinger, C. 1976. (Westinghouse) Letter to Vassallo, D.B., (NRC).

Eicheldinger, C. 1978. (Westinghouse). Letter to Stolz, J.F., (NRC), Letter Number NS-CE-1672.

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

WARD-TM-84, 1969. Porsching, T.A.; Murphy, J.H.; Redfield, J.A.; and Davis, V.C. FLASH-4, A Fully Implicit FORTRAN-IV Program for the Digital Simulation of Transients in a Reactor Plant. WARD-TM-84, Bettis Atomic Power Laboratory.

WASH-1400, NUREG-75/104, 1975. Reactor Safety Study - An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants.

WCAP-7907, 1972. Burnett, F.W.F. et al. LOFTRAN Code Description.

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

WCAP-8200, Revision 2 (Proprietary) and WCAP-8261, Revision 1 (Non-proprietary), 1974. Esposito, V.J.; Kesavan, K.; and Maul, B.A. WFLASH, A FORTRAN-IV Computer Program for Simulation of Transients in a Multi-Loop PWR.

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

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

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

WCAP-8339, 1974. Bordelon, F.M.; Massie, H.W.; and Zordan, T.A. Westinghouse ECCS Evaluation Mode - Summary.

WCAP-8340 (Proprietary) and WCAP-8356 (Non-proprietary), 1974. Salvatori, R. Westinghouse ECCS - Plant Sensitivity Studies.

WCAP-8341 (Proprietary) and WCAP-8342 (Non-proprietary), 1974. Westinghouse ECCS Evaluation Model Sensitivity Studies.

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

WCAP-8586-P-A (Proprietary) and WCAP-8566-A (Non-proprietary), 1975. Johnson, W.J.; Massie, H.OW.; and Thompson, C.M. Westinghouse ECCS-Four Loop Plant (17 x 17) Sensitivity Studies.

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

WCAP-8904-A (Non-proprietary), 1979. Kemper, R.M. Westinghouse Emergency Core Cooling System Evaluation Model for (N-1) Loop Operation of plant with Loop Isolation Valves. 17.8 17.9

WCAP-8970 (Proprietary) and WCAP-8971 (Non-proprietary), 1977. Skwarek, R.J.; Johnson, W.J.; and Meyer, P.E. Westinghouse Emergency Core Cooling System Small Break October 1975 Model. 17.10 17.12

WCAP-9168 (Proprietary) and WCAP-9169 (Non-proprietary), 1977. Johnson, W.J. and Thompson, C.M. Westinghouse Emergency Core Cooling System Evaluation Model - Modified October 1975 Version. 17.13 17.15

WCAP-9220-P-A (Proprietary Version) and WCAP-9221-P-A, (Non-proprietary Version), Revison 1, 1981. Eicheldinger, C. Westinghouse ECCS Evaluation Mode, 19781 Version. 17.16 17.17 17.18

letter from E.P. Rahe of Westinghouse Electric Corporation to Robert L. Tedesco of the Nuclear Regulatory Commission, letter Number NS-ER-2538, December 1981.

TABLE 15.6-1

TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH CAUSE A DECREASE IN REACTOR COOLANT INVENTORY

Accident	Event	N-Loop Time (sec)	N-1 Loop Time (sec)
Inadvertent opening of a pressurizer safety valve	Safety valve fully opens	0.0	0.0
	Overtemperature ΔT reactor trip setpoint reached	20.0	32.5
	Rods begin to drop	22.0	34.5
	Minimum DNBR occurs	20.6	35.2
Large break LOCA			
1. DECLG $C_D = 0.8$	Start	0.0	
(Minimum SI)	Reactor trip signal	0.59	
	Safety injection signal	2.13	
	Accumulator injection begins	12.9	
	End-of-bypass	26.96	
	End-of-blowdown	27.18	
	Pump injection begins	27.13	
	Bottom of core recovery	39.46	
	Accumulator empty	51.66	
2 DECLG $C_D = 0.6$	Start	0.0	
(Minimum SI)	Reactor trip signal	0.6	
	Safety injection signal	2.5	
	Accumulator injection begins	15.5	
	Pump injection begins	27.50	
	End-of-bypass	31.68	

TABLE 15.6-1 (Cont)

Accident	Event	N-Loop Time (sec)	N-1 Loop Time (sec)
	End-of-blowdown	31.68	
	Bottom of core recovery	45.29	
	Accumulator empty	55.34	
2.A. DECLG C_D = 0.6	Start	0.0	
(Maximum SI)	Reactor trip signal	0.6	
	Safety injection signal	2.5	
	Accumulator injection begins	15.5	
	Pump injection begins	27.5	
	End-of-bypass	31.68	
	End-of-blowdown	31.68	
	Bottom of core recovery	44.58	
	Accumulator empty	56.64	
3. DECLG C_D = 0.4	Start	0.0	
(Minimum SI)	Reactor trip signal	0.61	
	Safety injection signal	3.21	
	Accumulator injection begins	20.1	
	End-of-bypass	39.50	
	End-of-blowdown	39.50	
	Pump injection begins	28.21	
	Bottom of core recovery	54.03	
	Accumulator empty	61.84	

TABLE 15.6-1 (Cont)

Accident	Event	N-Loop Time (sec)	N-1 Loop Time (sec)	
Large break LOCA (N-1)				3.15
4. DECLG C_D = 0.4	Start		0.0	~~3.17~~
Active Loop Break	Reactor trip signal		0.552	3.19
	Safety injection signal		3.79	3.22
	Accumulator injection begins		18.5	3.25 3.26
	End-of-bypass		35.488	3.29
	End-of-blowdown		35.531	3.31
	Pump injection begins		28.79	3.33
	Bottom of core recovery		50.751	3.35
	Accumulator empty		68.78	3.37
5. DECLG C_D = 0.6	Start		0.0	~~3.40~~
Active Loop Break	Reactor trip signal		0.542	3.42
	Safety injection signal		2.89	3.45
	Accumulator injection begins		14.1	3.48 3.49
	Pump injection begins		27.89	3.52
	End-of-bypass		28.656	3.54
	End-of-blowdown		28.702	3.56
	Bottom of core recovery		43.416	3.58
	Accumulator empty		63.98	3.60
6. DECLG C_D = 0.4	Start		0.0	~~4.3~~
Inactive Loop Break	Reactor trip signal		0.577	4.5
	Safety injection signal		6.11	4.8

TABLE 15.6-1 (Cont)

Accident	Event	N-Loop Time (sec)	N-1 Loop Time (sec)
	Accumulator injection begins		30.4
	End-of-bypass		52.776
	End-of-blowdown		52.907
	Pump injection begins		31.11
	Bottom of core recovery		69.069
	Accumulator empty		83.40
Small break LOCA			
1. 3-inch	Start	0.0	
	Reactor trip signal	29.7	
	Top of core uncovered	754	
	Accumulator injection begins	1886	
	Peak clad temperature occurs	1515	
	Top of core covered	2054	
2. 4-inch	Start	0.0	
	Reactor trip signal	19.2	
	Top of core uncovered	366	
	Accumulator injection begins	892	
	Peak clad temperature occurs	868	
	Top of core covered	457	

TABLE 15.6-1 (Cont)

Accident	Event	N-Loop Time (sec)	N-1 Loop Time (sec)	
3. 6-inch	Start	0.0		5.3
	Reactor trip signal	11.7		5.5
	Top of core uncovered	132		5.7
	Accumulator injection begins	331		5.10 5.11
	Peak clad temperature occurs	336.4		5.14 5.15
	Top of core covered	372		5.17

IMAGE EVALUATION
TEST TARGET (MT-3)

150mm

6"

IMAGE EVALUATION
TEST TARGET (MT-3)

150mm

6"

TABLE 15.6-8

INPUT PARAMETERS USED IN THE ECCS ANALYSIS

	N-Loop	N-1 Loop
Licensed core power (MWt)[1]	3,411	2,217
Peak linear power, includes 102 percent factor (kW/ft)	12.88	9.385
Total peaking factor (F_Q^T)	2.32	2.60
Power shape		
Large break	Chopped cosine	Chopped cosine
Small break	See Fig. 15.6-7	-
Fuel assembly array	17 x 17	17 x 17
Accumulator water volume, nominal (ft^3/accumulator)	900	1,000
Accumulator tank volume, nominal (ft^3/accumulator)	1,350	1,350
Accumulator gas pressure, minimum (psia)	615	600
Safety injection pumped flow	See Figures 15.6-20 and 15.6-43	See Figure 15.6-57
Containment parameters	See Section 6.2	See Section 6.2
Initial loop flow (lb/sec)	9,817	10,420
Vessel inlet temperature (°F)	554.7	548.87
Vessel outlet temperature (°F)	616.5	608.15
Reactor coolant pressure (psia)	2,280	2,280
Steam pressure (psia)	975	925.11
Steam generator tube plugging level (percent)	0	0

NOTE:

1. Two percent is added to this power to account for calorimetric error.

TABLE 15.6-11

LARGE AND SMALL BREAK LOCA RESULTS FUEL CLADDING DATA

Large Break LOCA Results Fuel Cladding Data

Results	DECLG			
	C_D=0.4 (min SI)	C_D=0.6 (min SI)	C_D=0.6 (max SI)	C_D=0.8 (min SI)
Peak clad temp (°F)	1881.0	2100.0	2132.0	1936.
Peak clad location (ft)	7.5	7.5	7.5	7.5
Local Zr/H_2O Rxn (max) (%)	2.84	6.47	6.43	3.69
Local Zr/H_2O location (ft)	7.5	7.5	7.5	7.5
Total Zr/H_2O Rxn (%)	<0.3	<0.3	<0.3	<0.3
Hot rod burst time (sec)	92.06	67.8	69.2	70.0
Hot rod burst location (ft)	7.0	6.0	6.0	6.5

Calculation			
NSSS power MWt 102 of	3,411		
Peak linear power, kW/ft	12.88		
Peaking factor (at license rating)	2.32		
Accumulator water volume (ft^3)	900/accumulator		
Fuel Region - cycle analyzed		Cycle	Region
Unit 3		1	All

TABLE 15.6-11 (Cont)

Small Break LOCA Results Fuel Cladding Data 1.58

Results	3-Inch	4-Inch	6-Inch	
				2.1
Peak clad temperature (°F)	1,356	1,483	1,277	2.3
Peak clad temperature location (ft)	12.0	11.50	11.0	2.5 2.6
Local Zr/H_2O reaction, maximum (%)	0.21	0.3585	0.10	2.9 2.10
Local Zr/H_2O location (ft)	12.0	12.0	11.0	2.12
Total Zr/H_2O reaction (%)	<0.3	<0.3	<0.3	2.14
Hot rod burst time (sec)	N/A	N/A	N/A	2.16
Hot rod burst location (ft)	N/A	N/A	N/A	2.18

TABLE 15.6-11 (Cont)

Large Break (N-1) Loop LOCA Results Fuel Cladding Data

	DECLG		
	Active Loop		Inactive Loop
Results	C_D=0.4	C_D=0.6	C_D=0.4
Peak clad temp (°F)	1878.0	1791.0	1591.0
Peak clad location (ft)	9.0	9.25	9.0
Local Zr/H_2O Rxn (max) (%)	3.14	2.41	0.73
Local Zr/H_2O location (ft)	9.25	9.5	9.25
Total Zr/H_2O Rxn (%)	<0.3	<0.3	<0.3
Hot rod burst time (sec)	97.04	106.73	-
Hot rod burst location (ft)	6.25	6.75	-

Calculation

NSSS power MWt 102 of	2,217		
Peak linear power, kW/ft	9.385		
Peaking factor (at license rating)	2.60		
Accumulator water volume (ft^3)	1000.0/accumulator		
Fuel Region - cycle analyzed		Cycle	Region
Unit 3		1	All

[illegible]

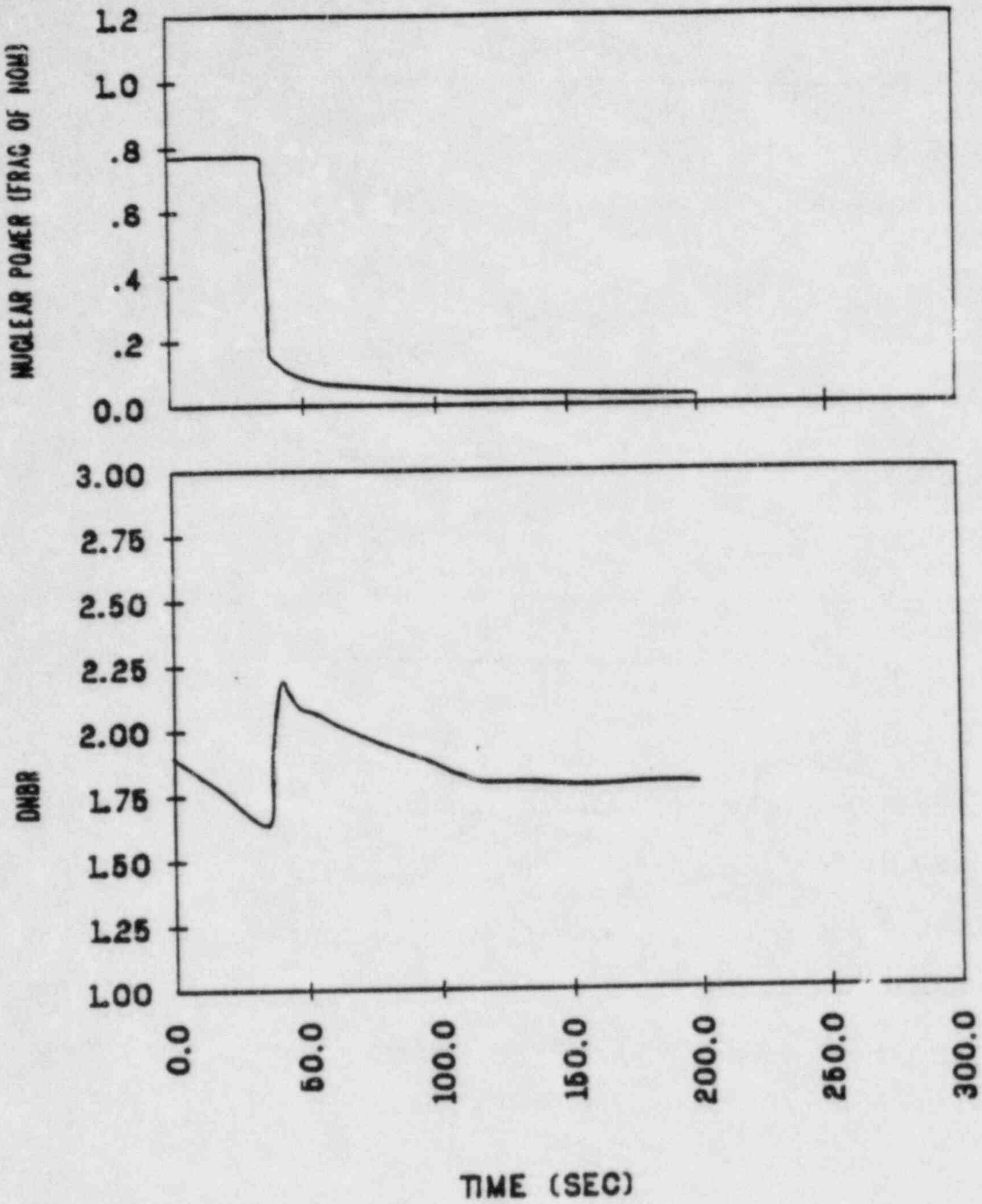


FIGURE 15.6-1A
INADVERTENT OPENING OF A PRESSURIZER SAFETY VALVE
N-1 /LOOP OPERATION

INTERACTIVE GRAPHICS
D.CHA.GEOR.0751AXX.XXX.01.053.083183

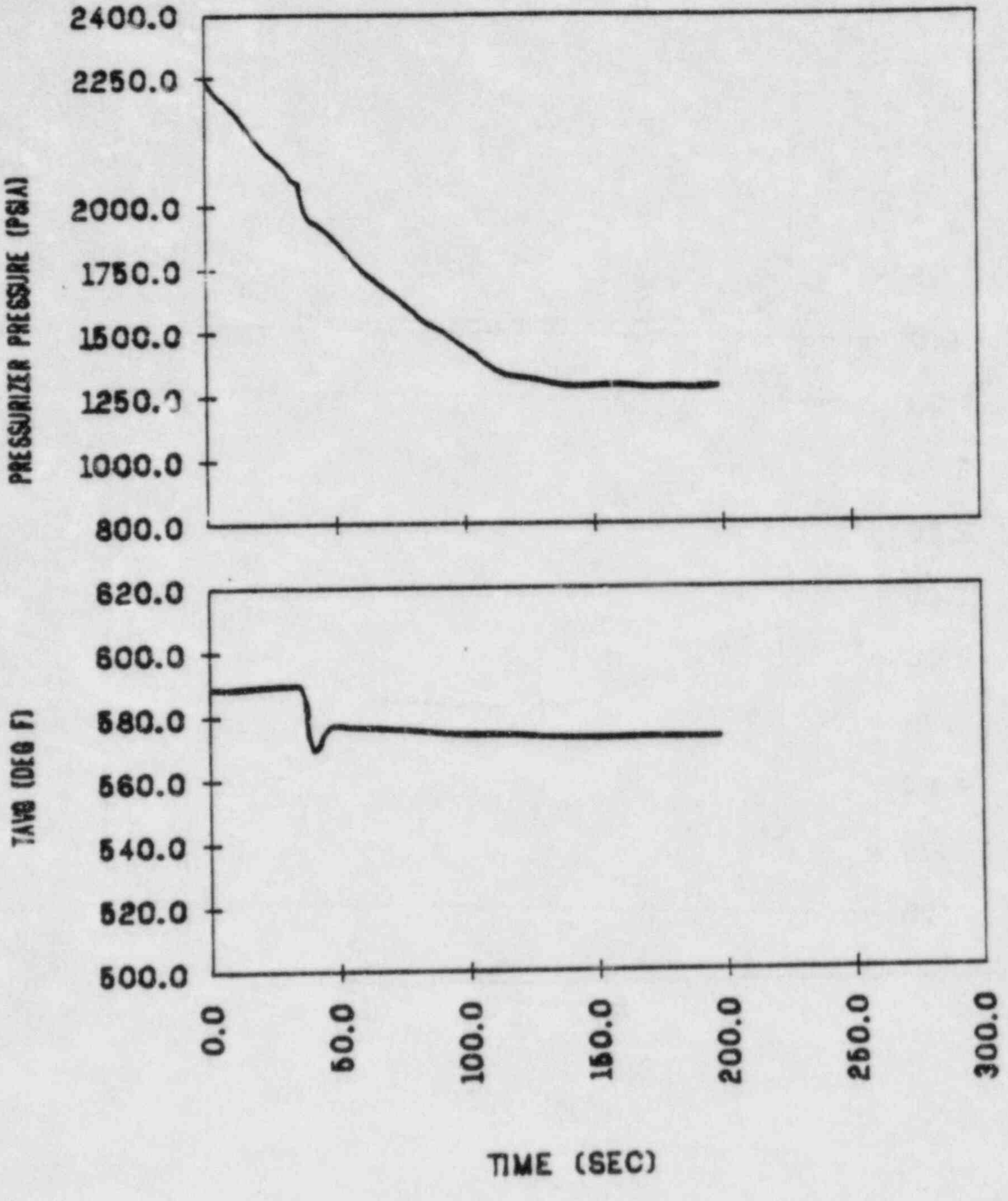


FIGURE 15.6-2A
INADVERTENT OPENING OF A
PRESSURIZER SAFETY VALVE
N-1 /LOOP OPERATION

INTERACTIVE GRAPHICS
D.CHA.GE OR.0751AXX.XXX.01.051.083183

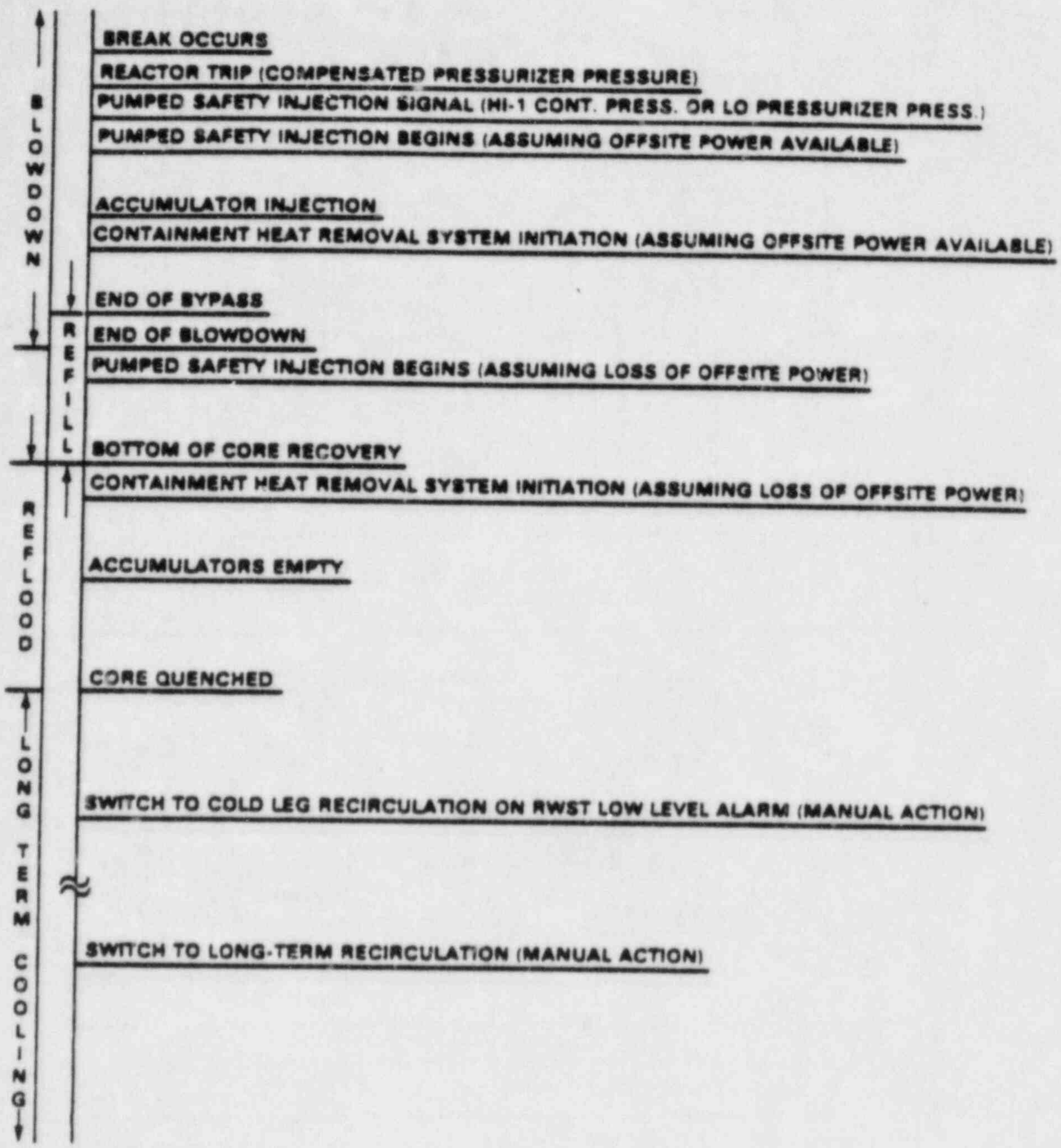


FIGURE 15 6-4
SEQUENCE OF EVENTS FOR
LARGE BREAK LOCA ANALYSIS
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

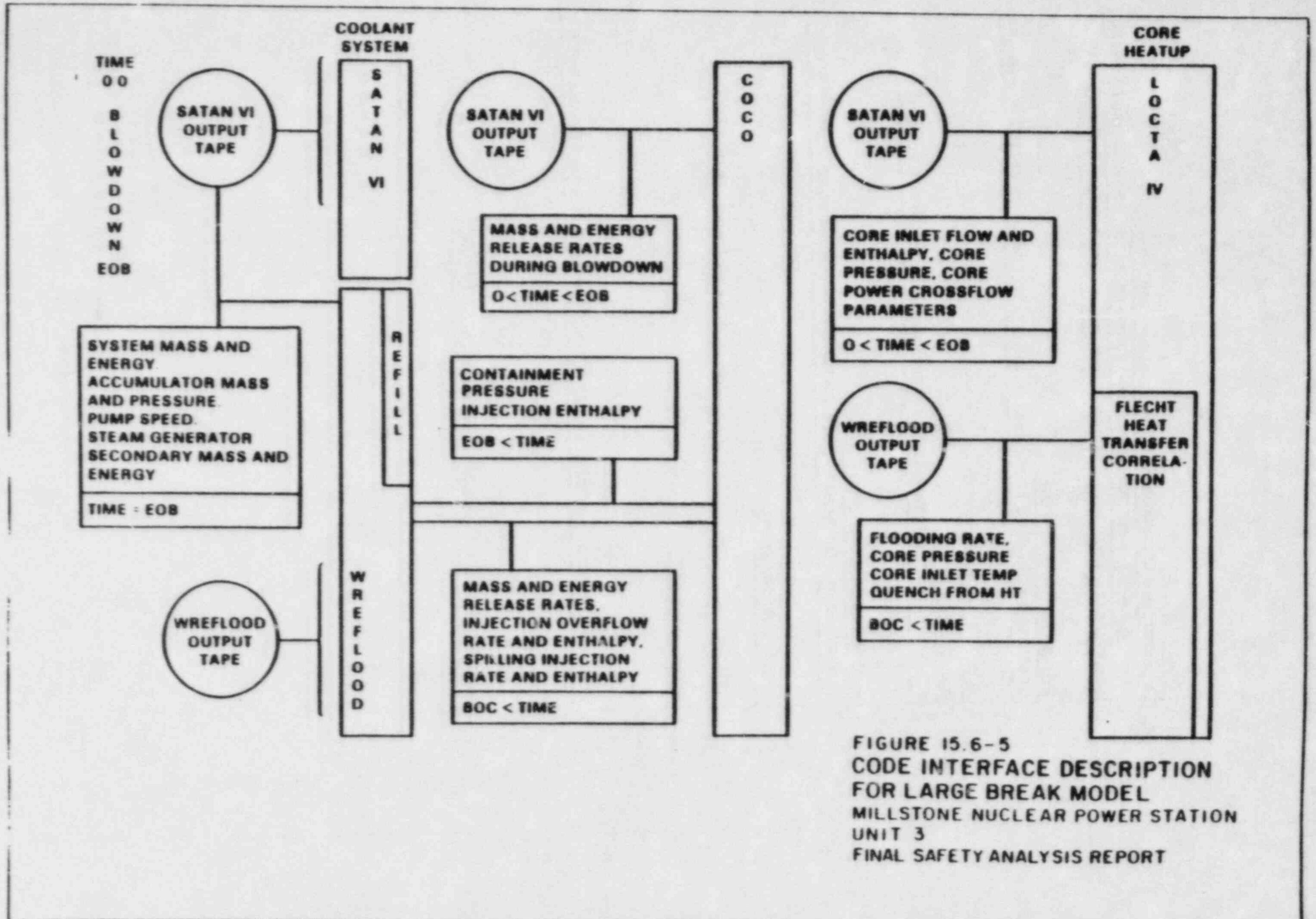


FIGURE 15.6-5
CODE INTERFACE DESCRIPTION FOR LARGE BREAK MODEL
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

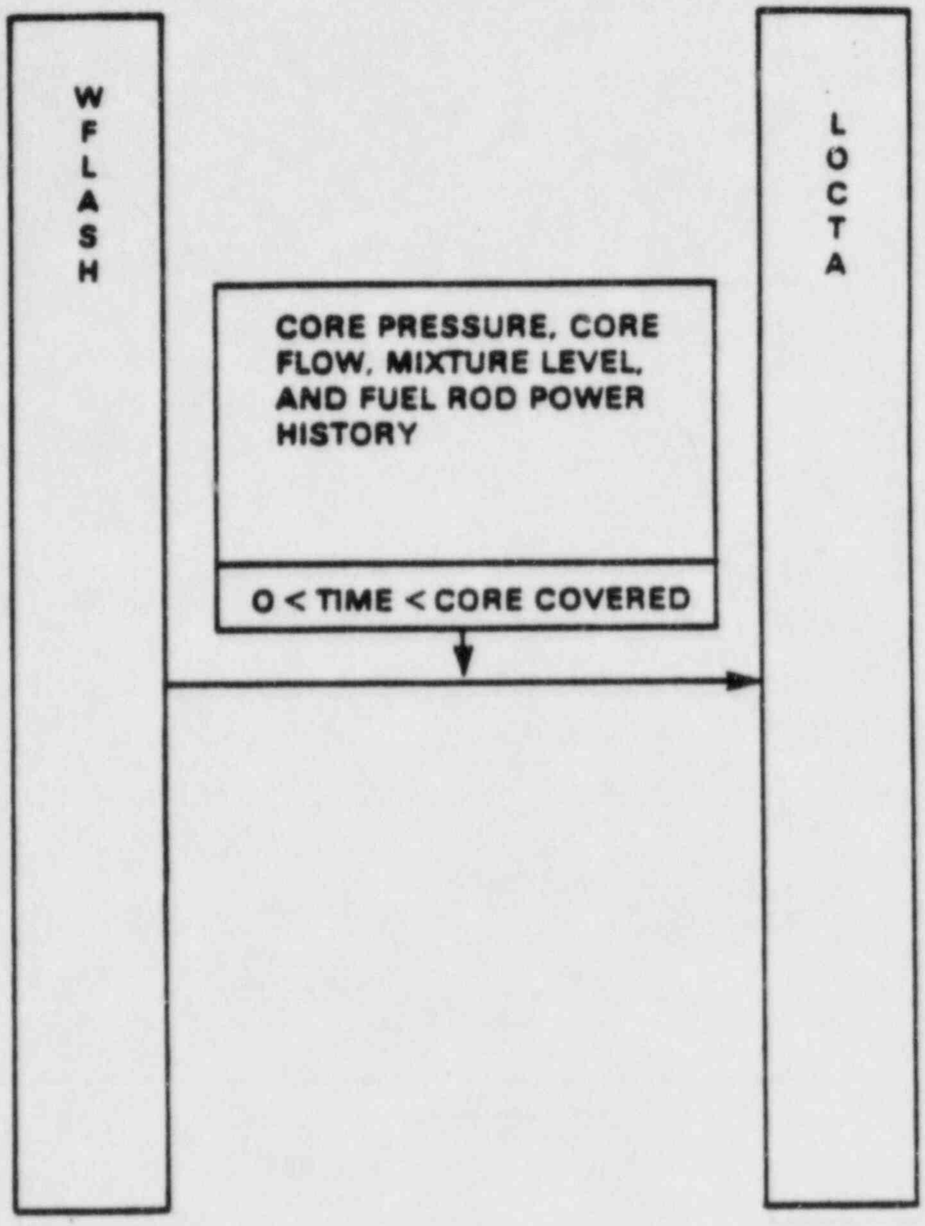


FIGURE 15.6-6
CODE INTERFACE DESCRIPTION FOR SMALL BREAK MODEL
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

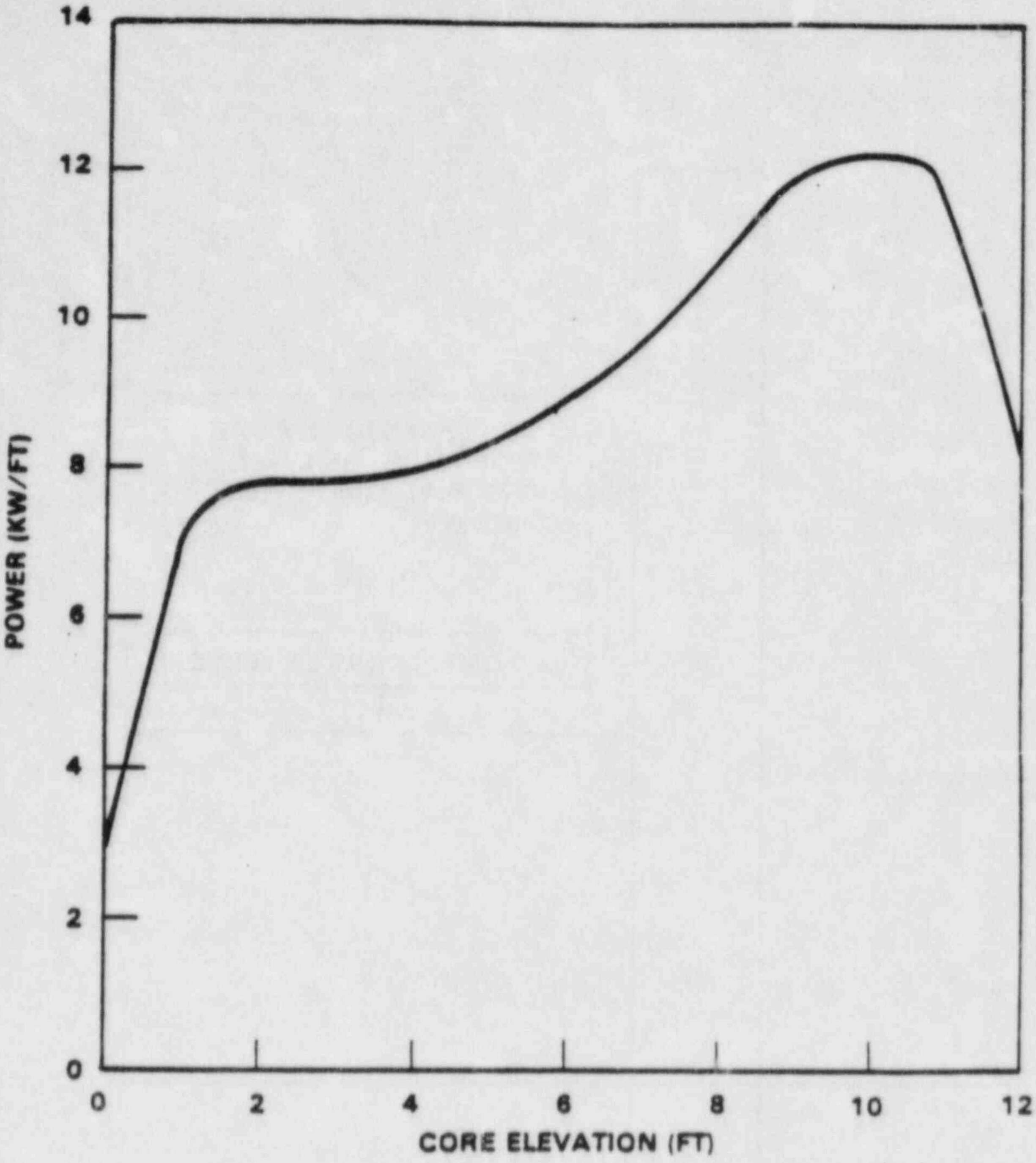


FIGURE 15.6-7
SMALL BREAK POWER DISTRIBUTION
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

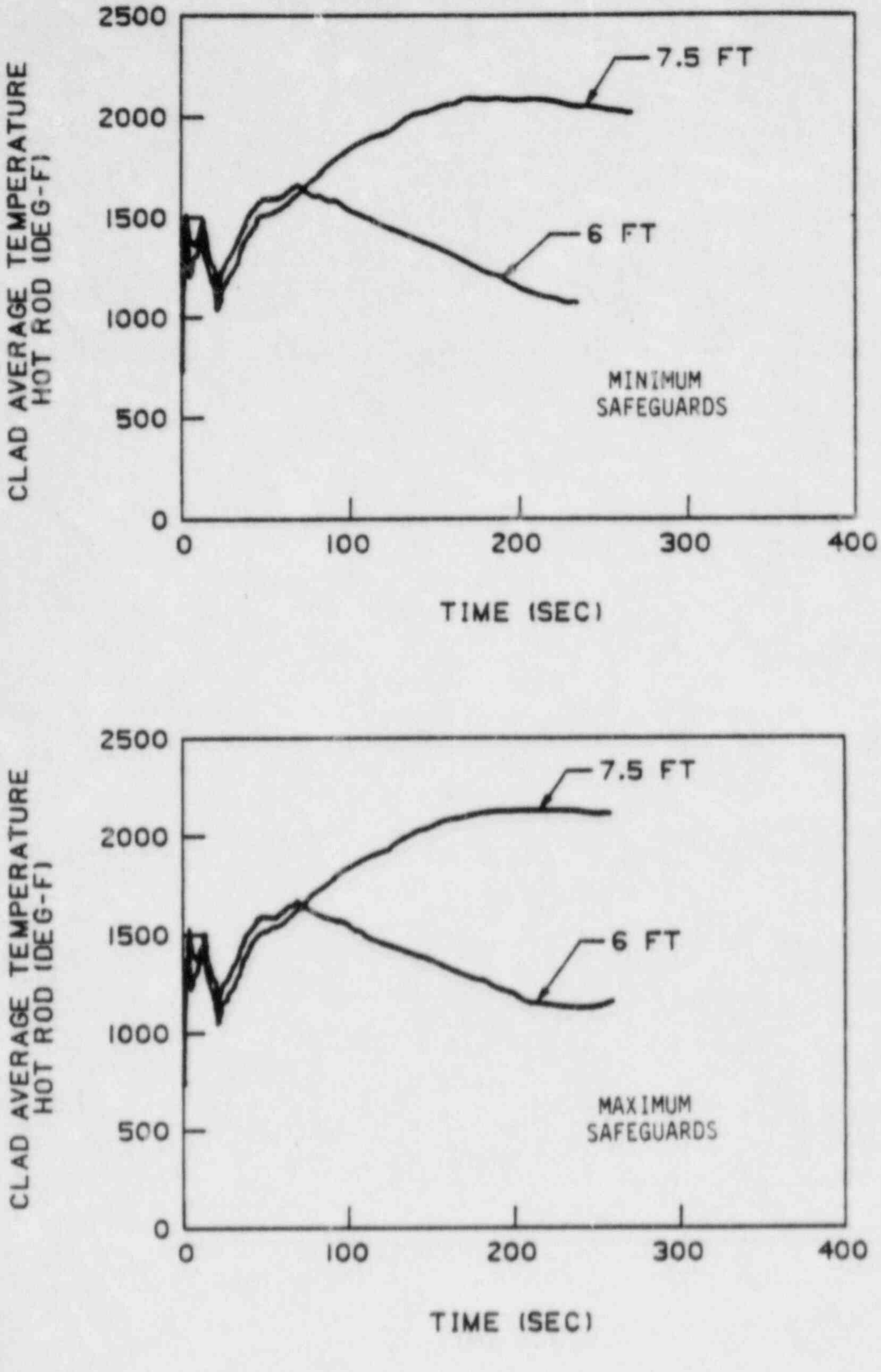


FIGURE 15.6-8
PEAK CLAD TEMPERATURE – DECLG (C_D = 0.6)
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

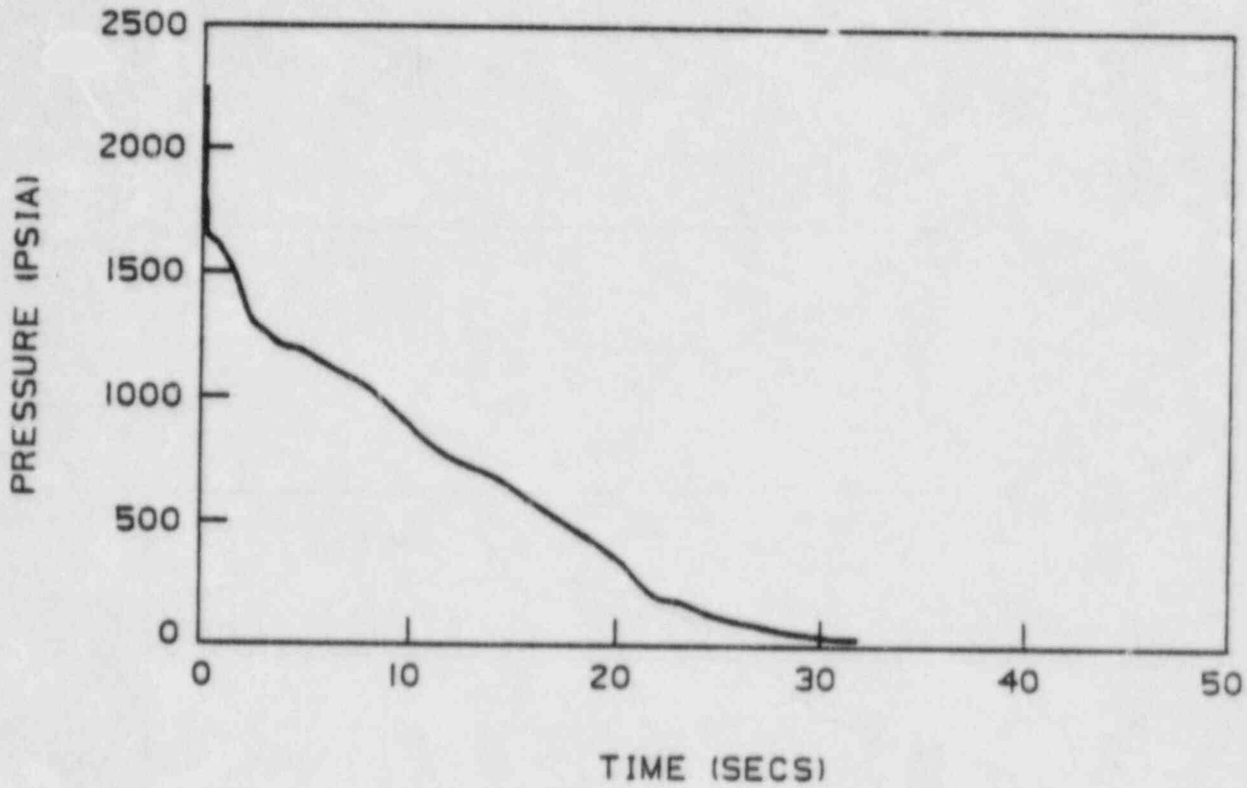
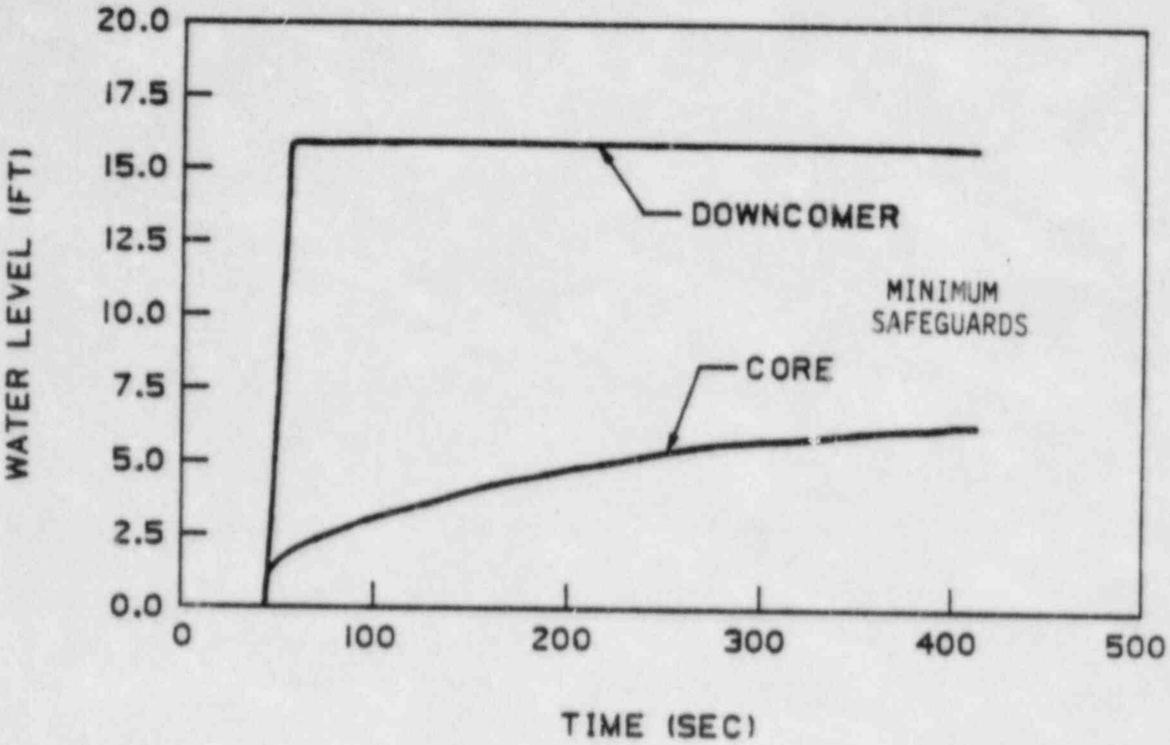


FIGURE 15.6-8a
CORE PRESSURE
DECLG (C_D=0.6)
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT



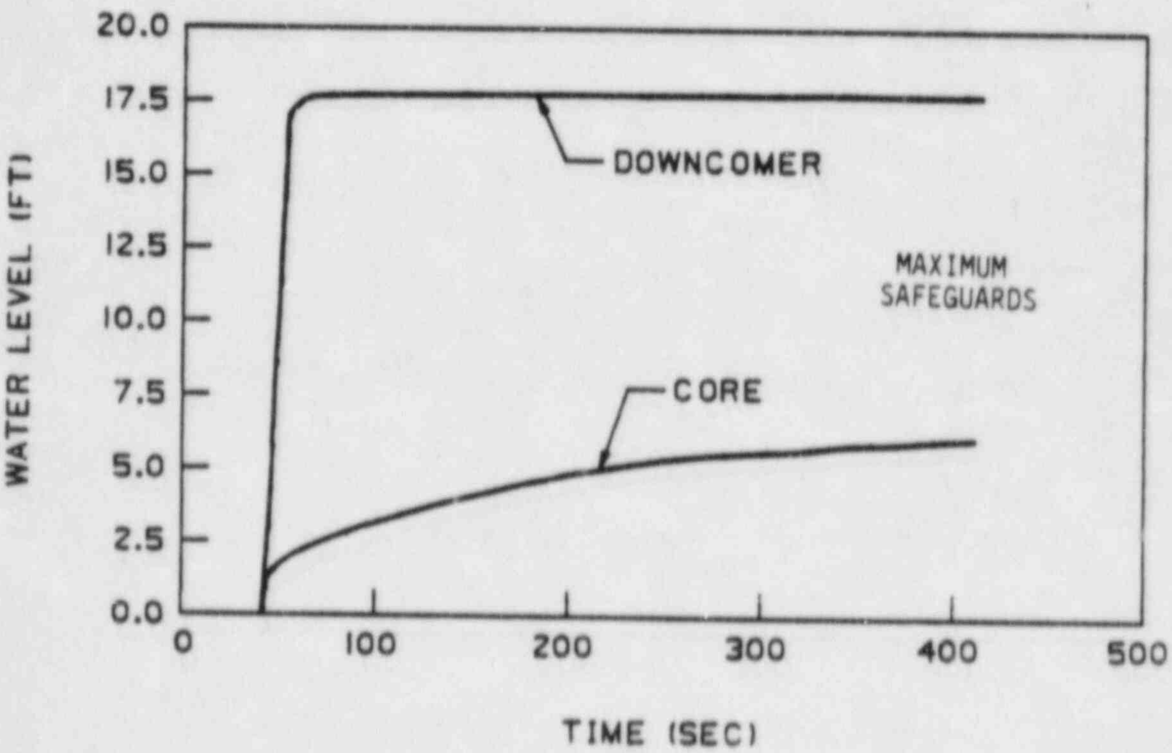
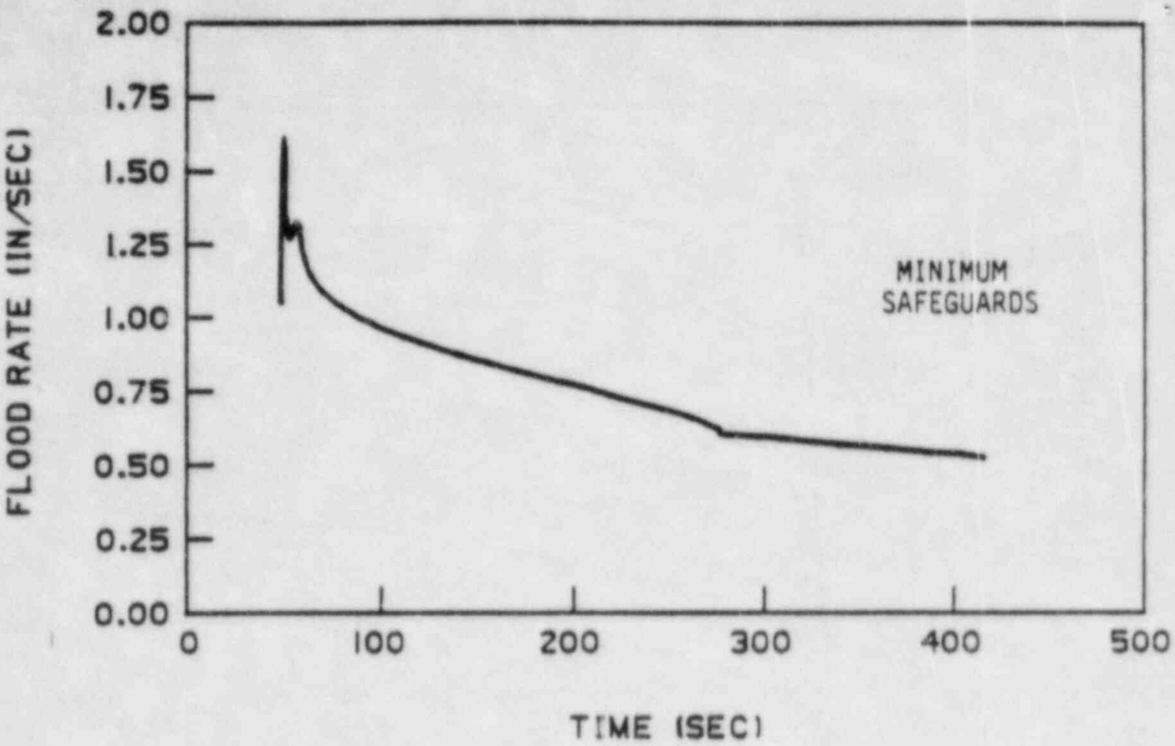


FIGURE 15.6-9
DOWNCOMER AND CORE WATER LEVELS
DURING REFLOOD-DECLG (C_D=0.6)
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT



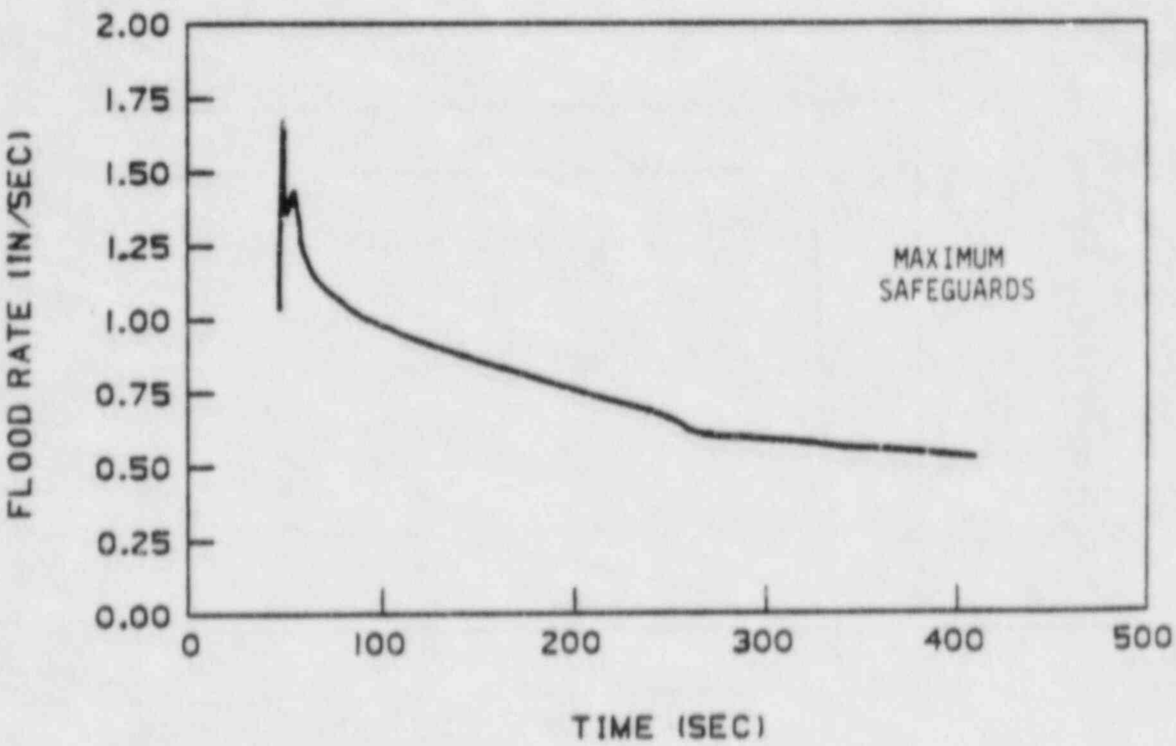


FIGURE 15.6-10
CORE INLET VELOCITY DURING
REFLOOD-DECLG (C_D=0.6)
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

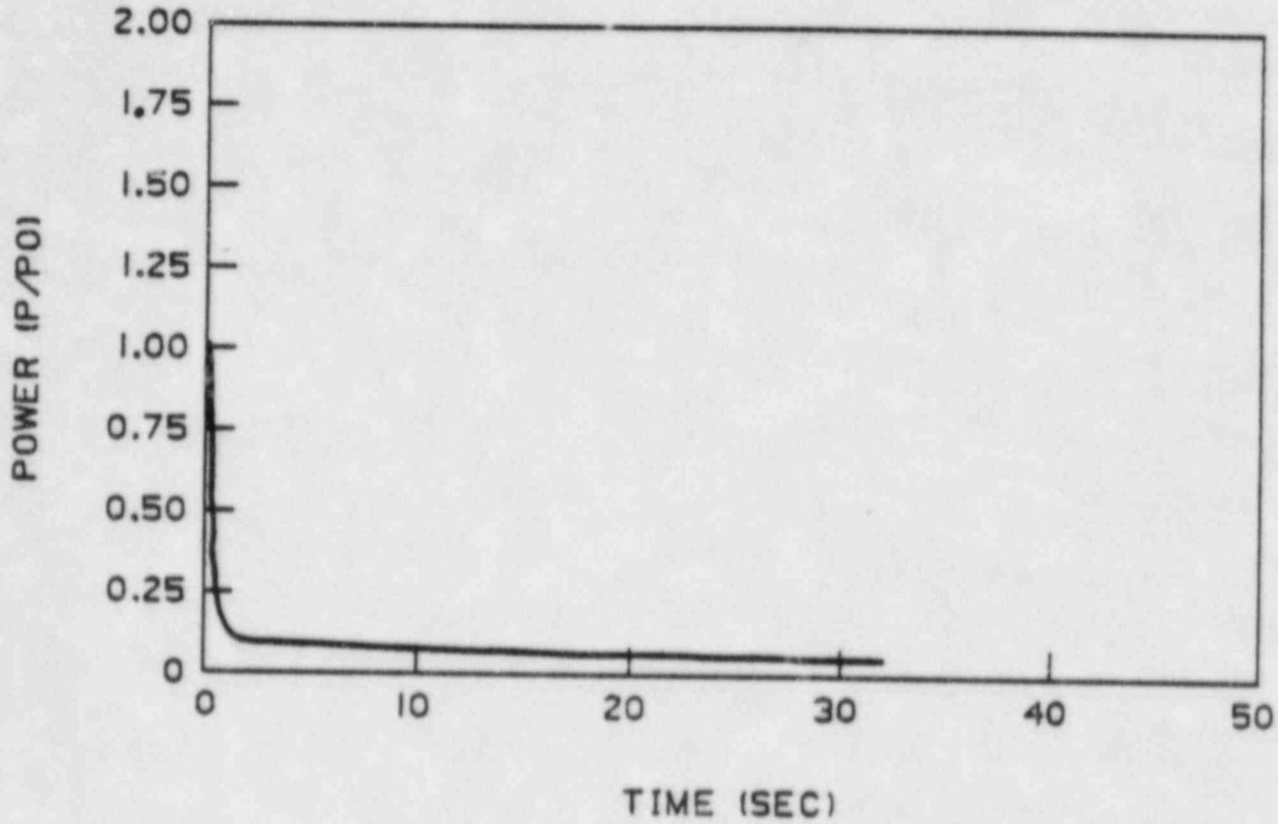


FIGURE 15.6-11
CORE POWER TRANSIENT
DECLG (C_D=0.6)

MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

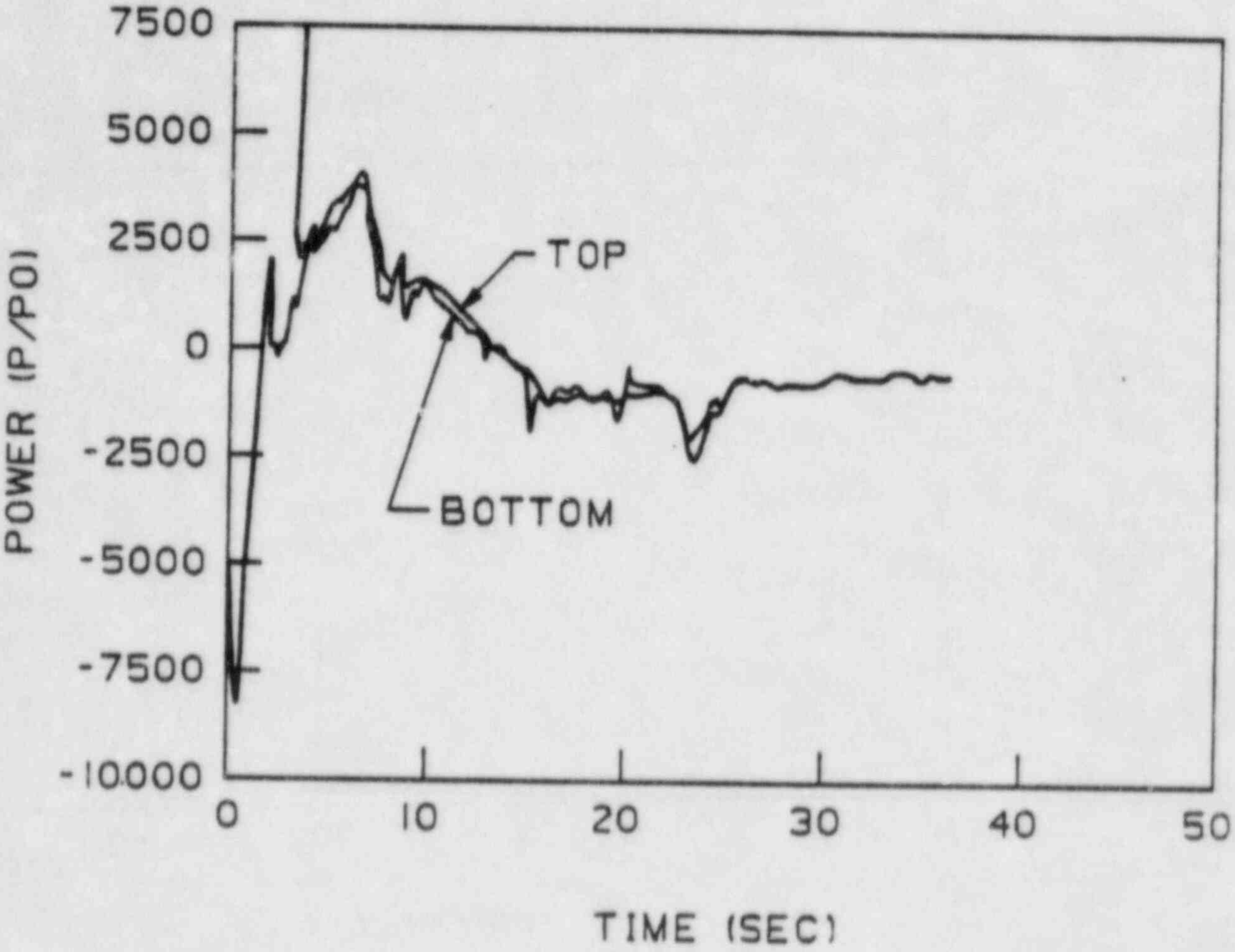


FIGURE 15.6-12
CORE FLOW TOP AND BOTTOM
DECLG (C_D=0.6)
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

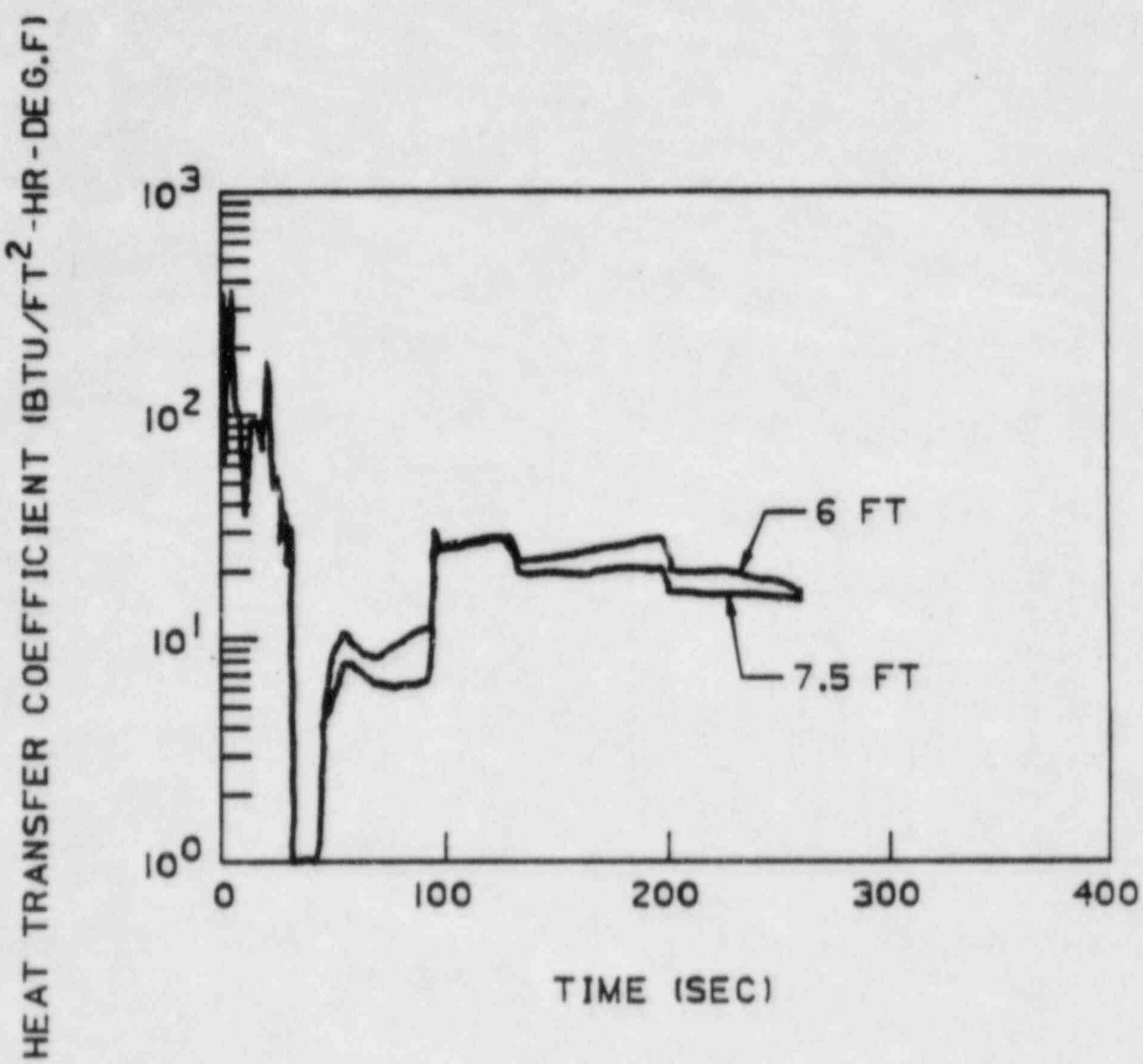


FIGURE 15.6-13
CORE HEAT TRANSFER
COEFFICIENT - DECLG (C_D = 0.6)
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

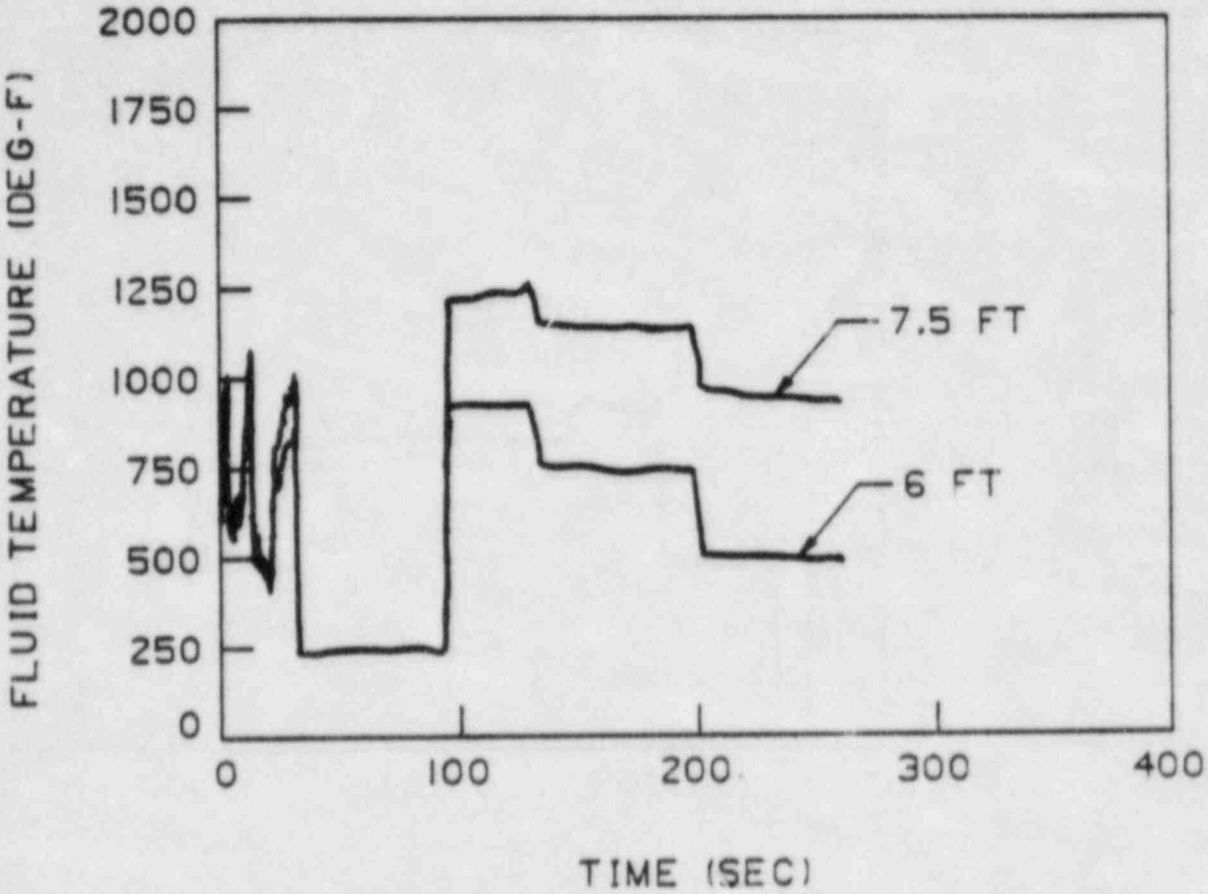


FIGURE 15.6-14
FLUID TEMPERATURE
DECLG (C_D = 0.6)
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

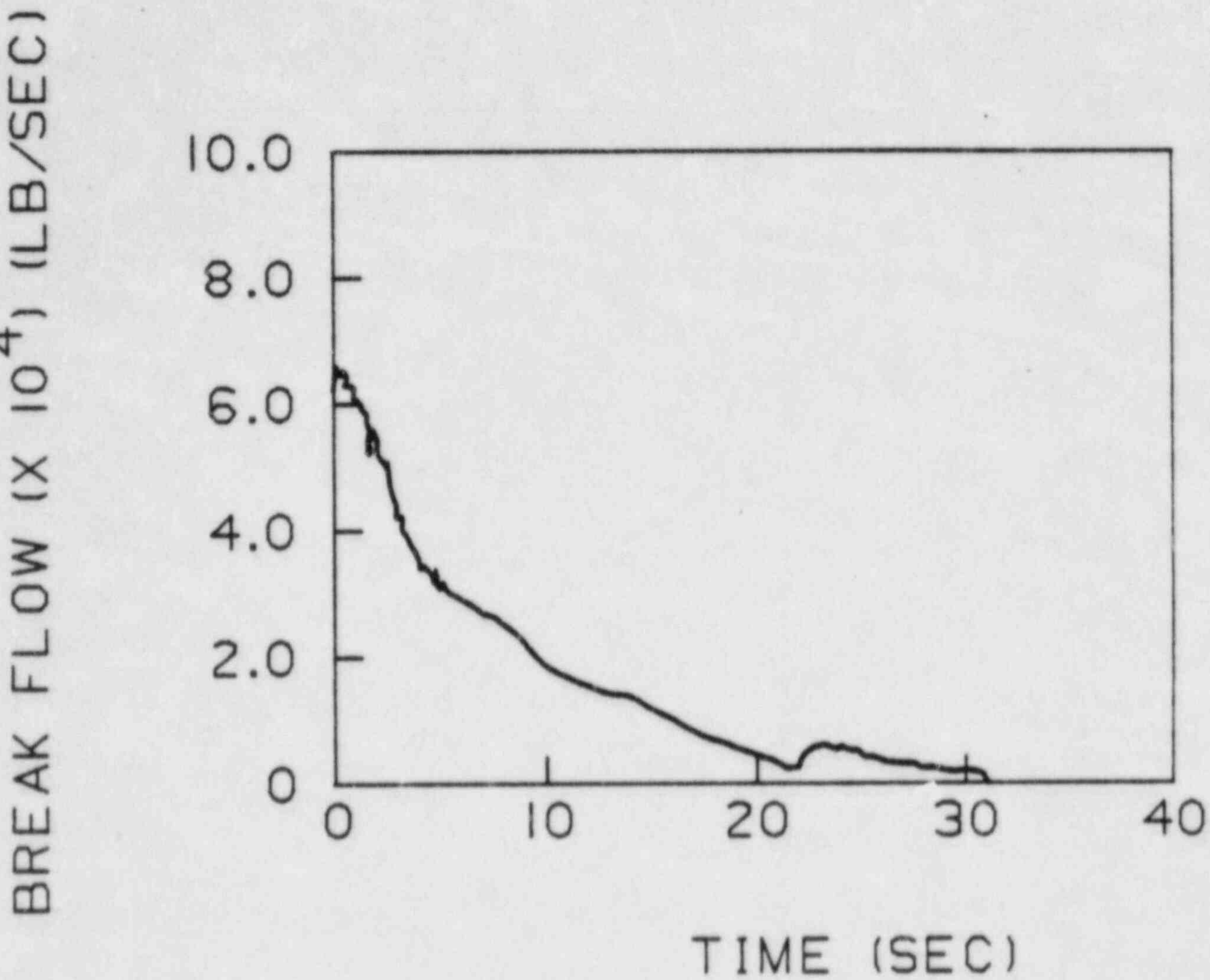


FIGURE 15.6-15
BREAK MASS FLOW RATE
DECLG(C_D=0.6)
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

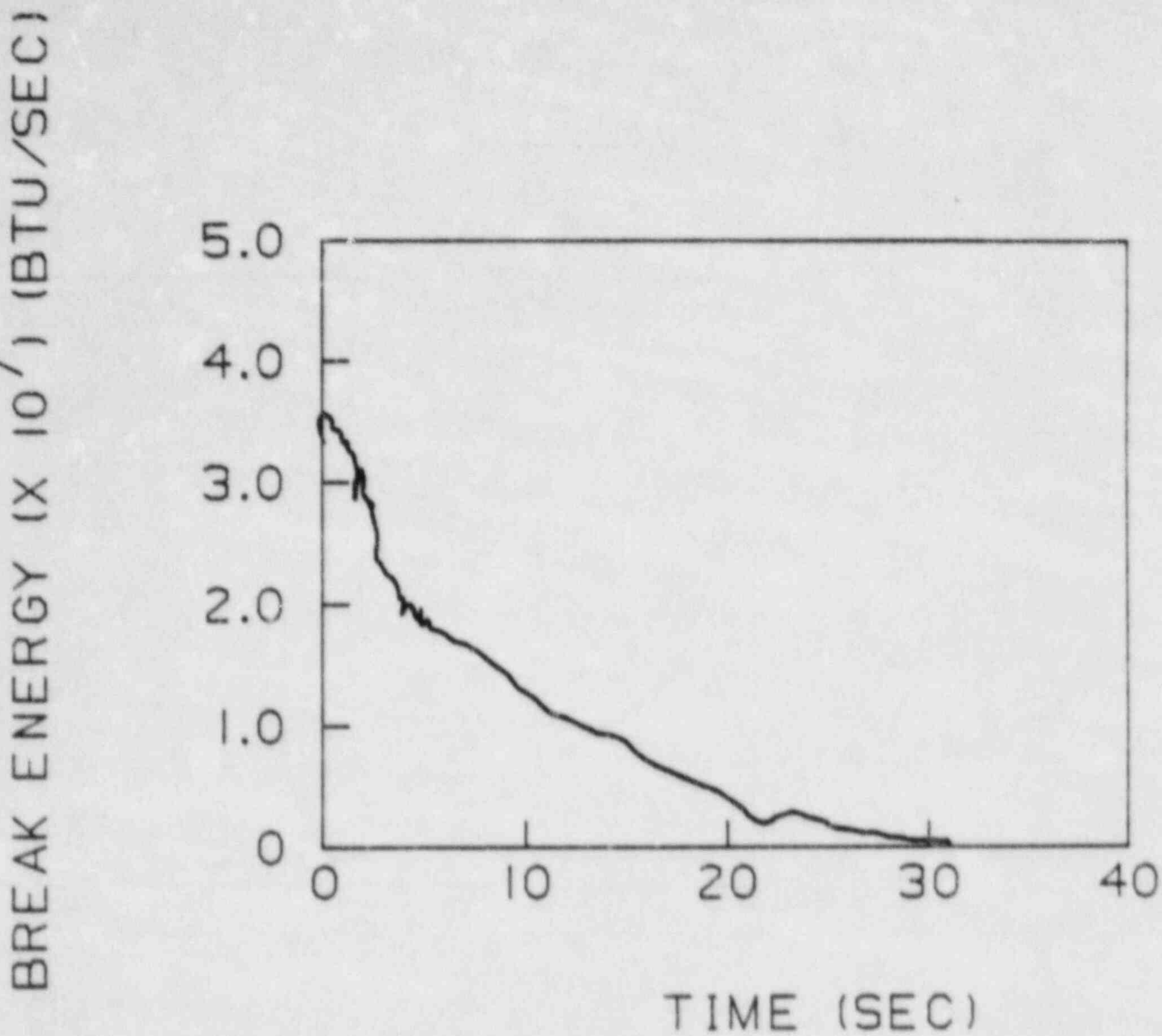


FIGURE 15.6-16
BREAK ENERGY RELEASE RATE
DECLG (C_D = 0.6)
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

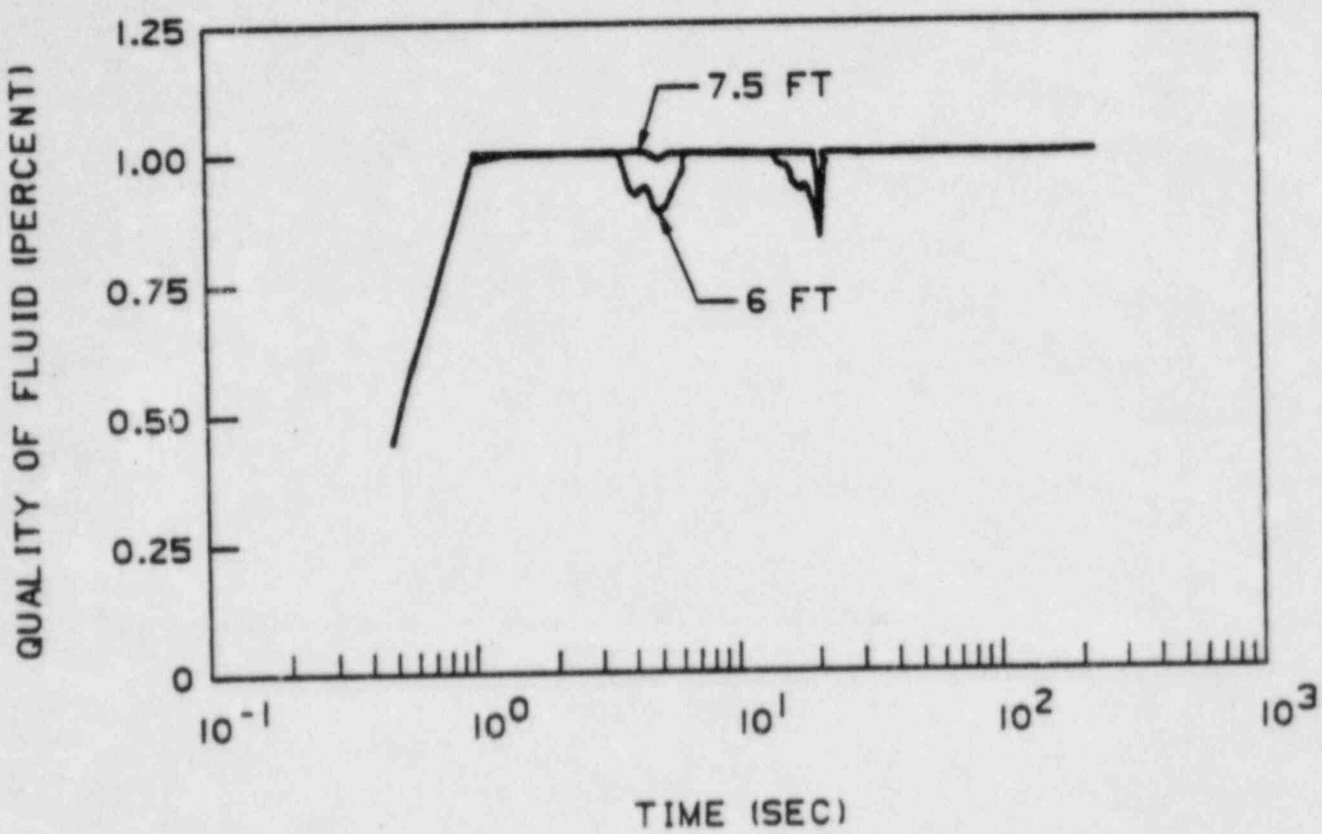


FIGURE 15.6-17
FLUID QUALITY-DECLG (C_D = 0.6)
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

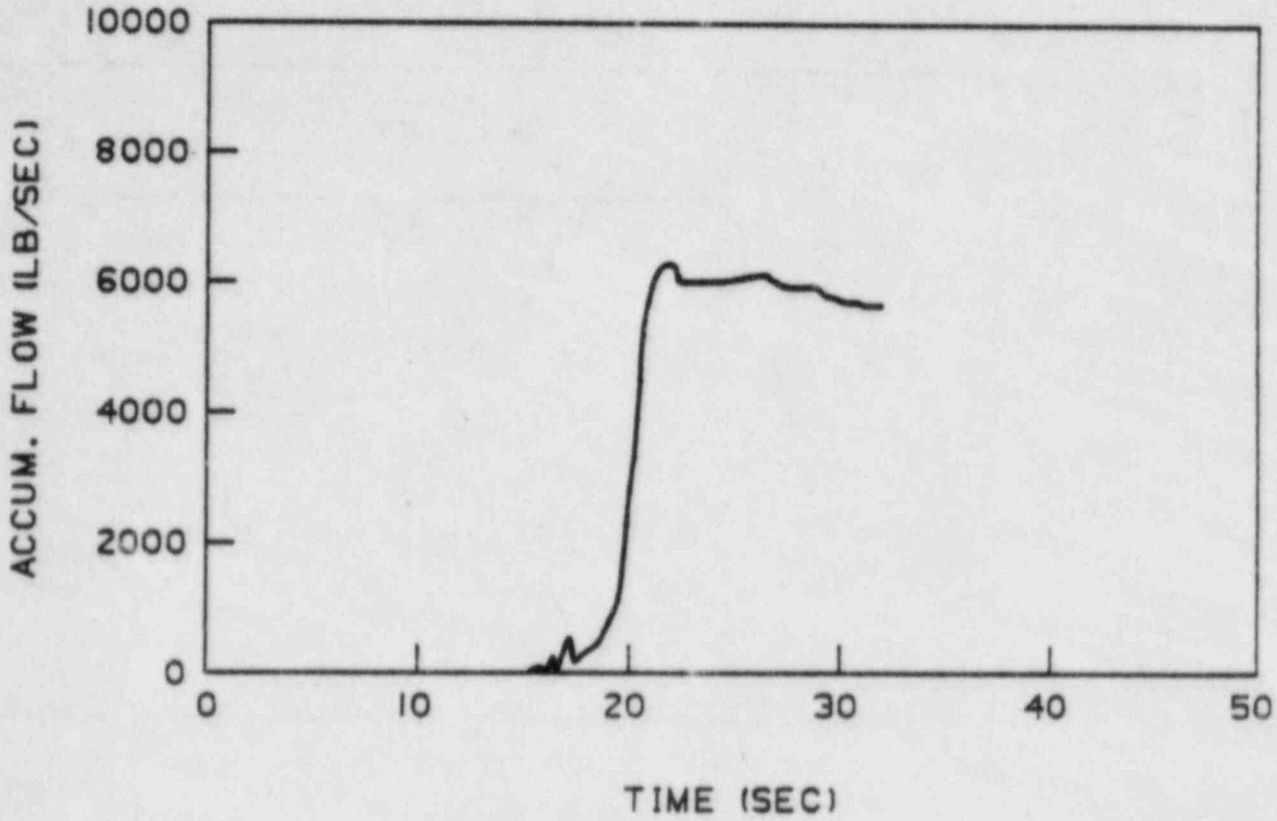


FIGURE 15.6-18
ACCUMULATOR FLOW DURING BLOWDOWN-DECLG (C_D=0.6)
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

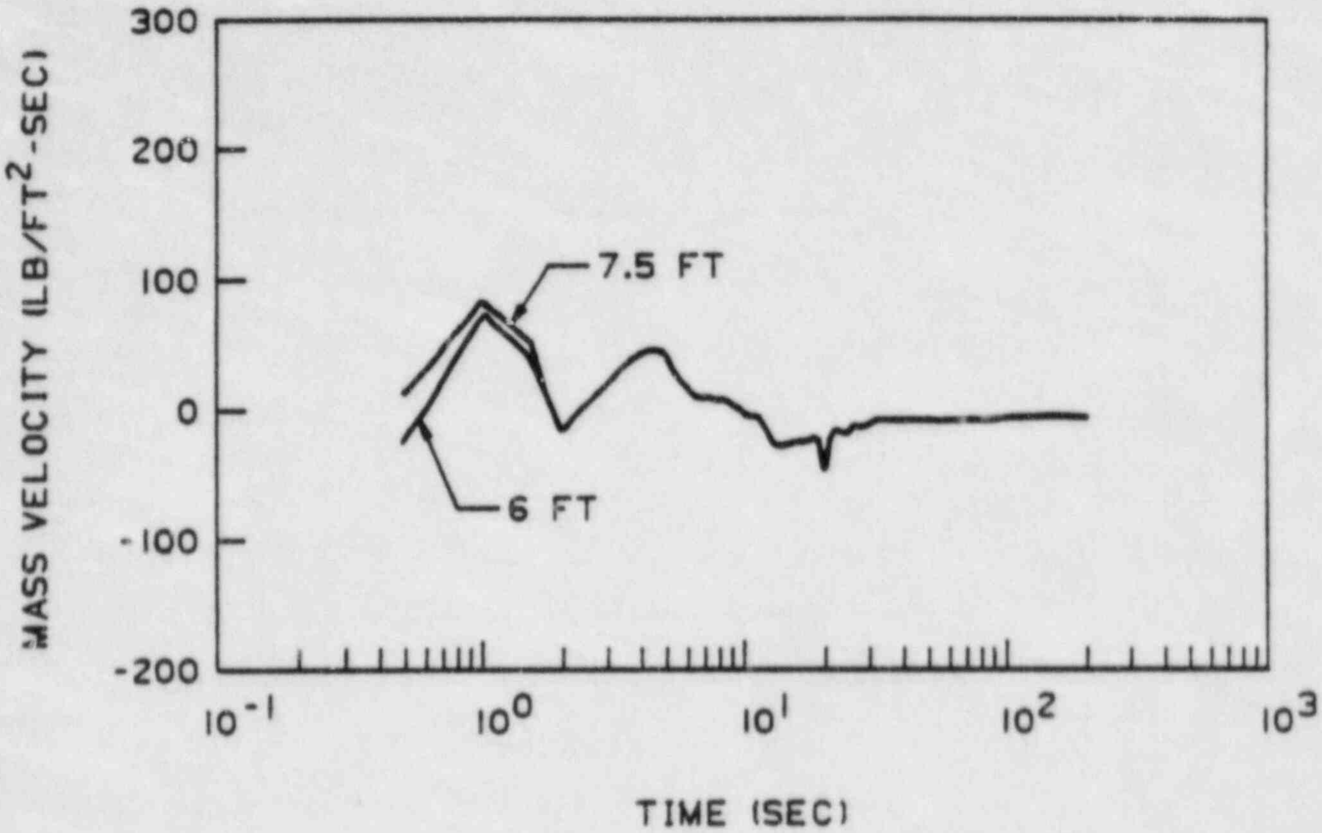


FIGURE 15.6-19
MASS VELOCITY-DECLG (C_D= 0.6)
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

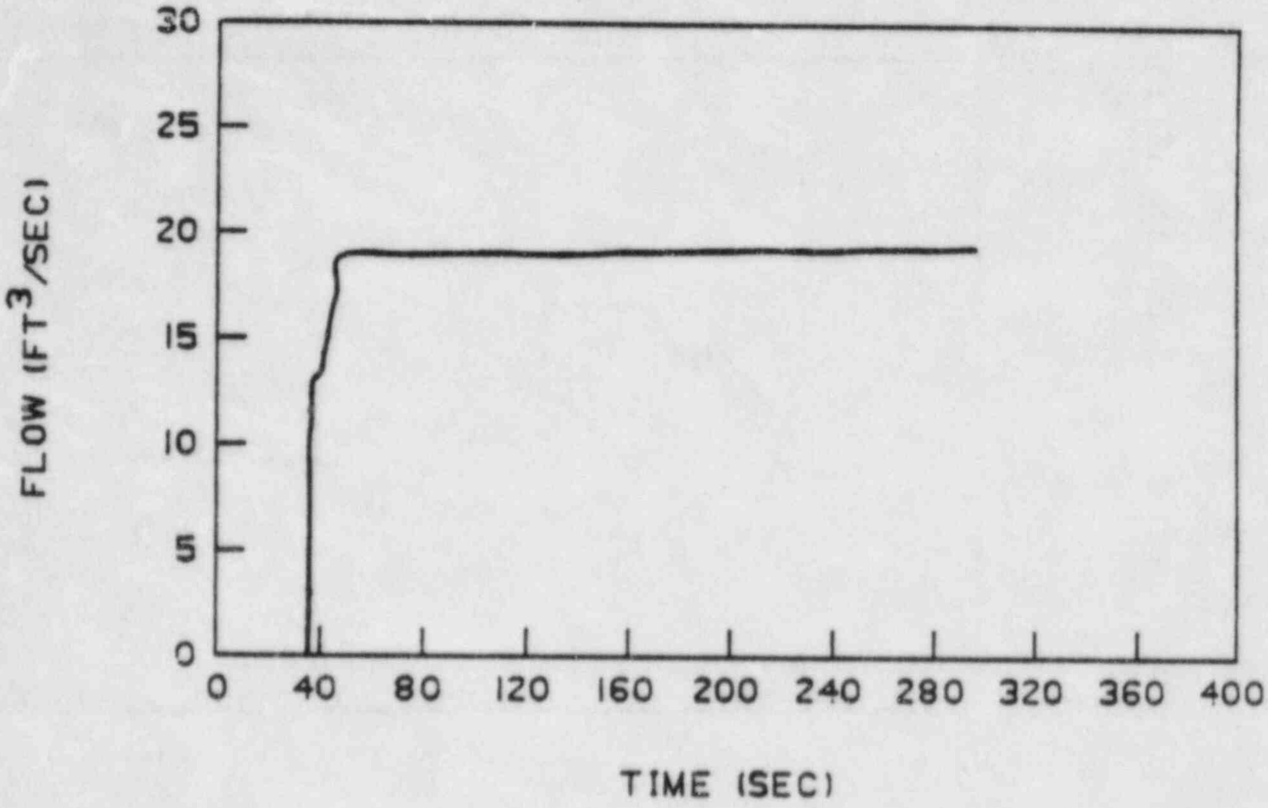


FIGURE 15.6-20

PUMPED ECCS FLOW DURING REFLOOD-DECLG (C_D=0.6)

MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

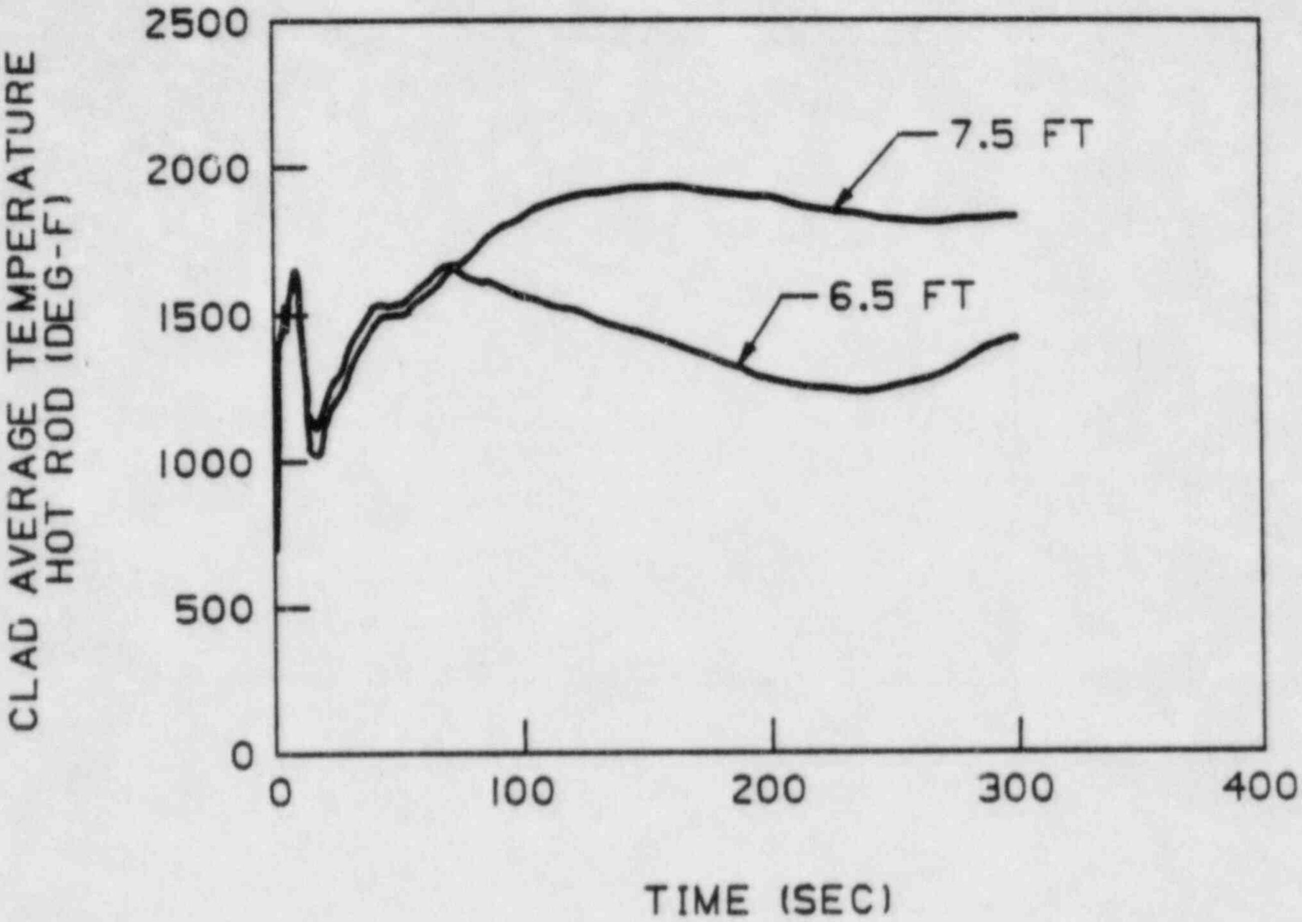


FIGURE 15.6-21
PEAK CLAD TEMPERATURE
DECLG (C_D=0.8)

MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

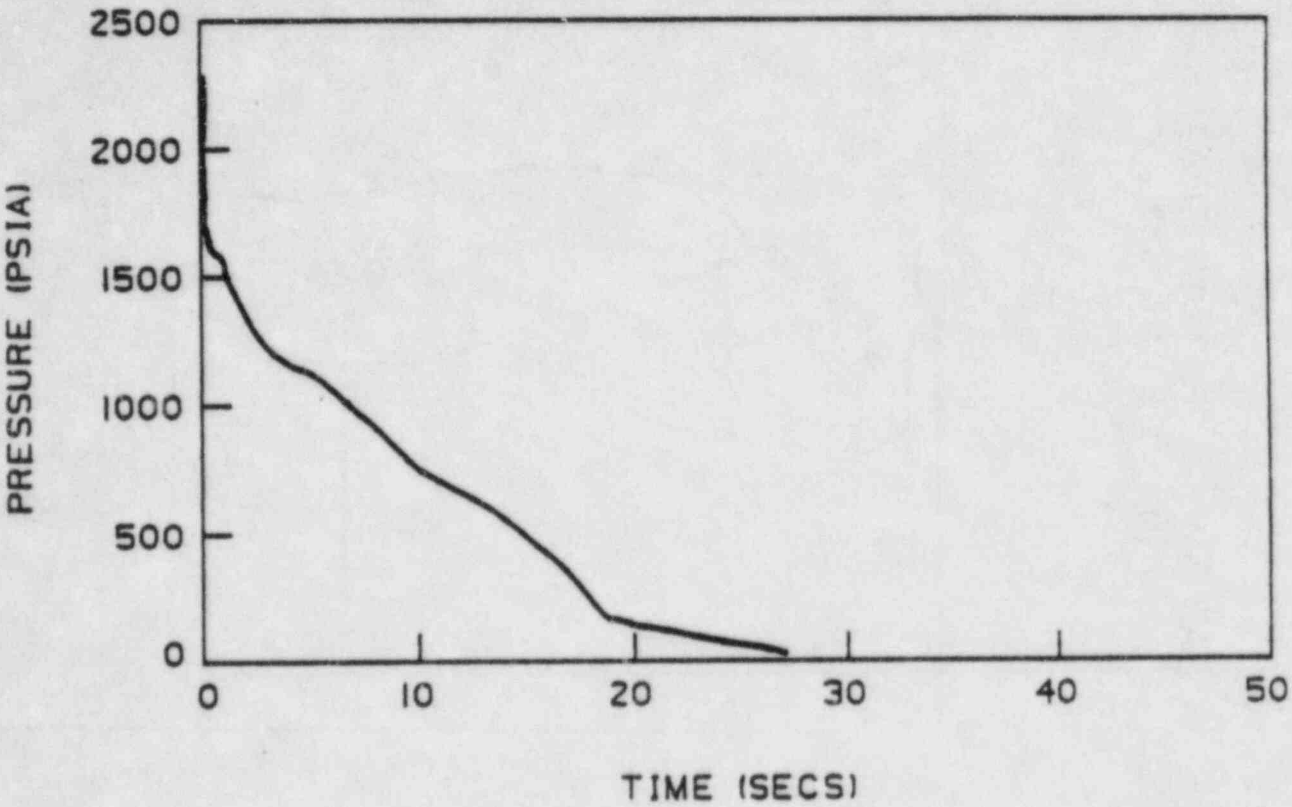


FIGURE 15.6-22
CORE PRESSURE —
DECLG (C_D = 0.8)
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

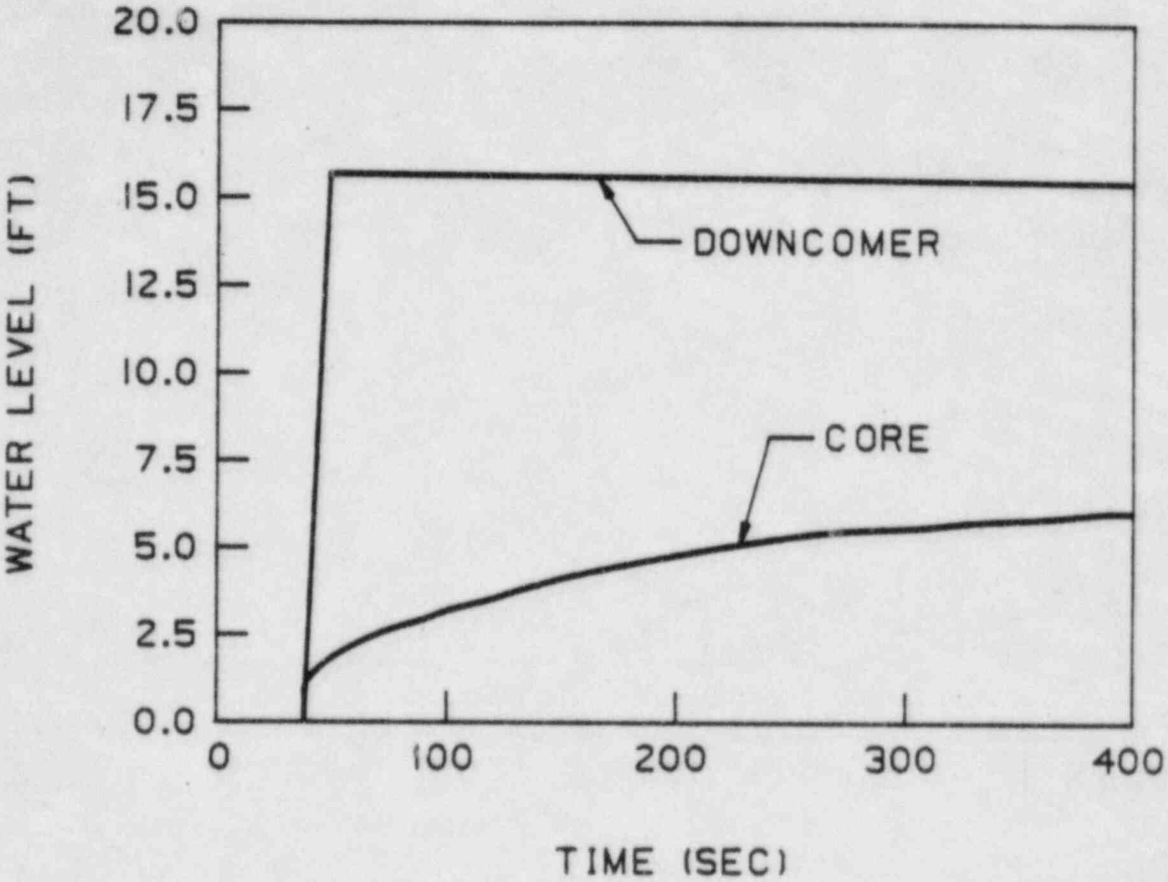


FIGURE 15.6-23
DOWNCOMER AND CORE WATER LEVELS
DURING REFLOOD-DECLG (C_D=0.8
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

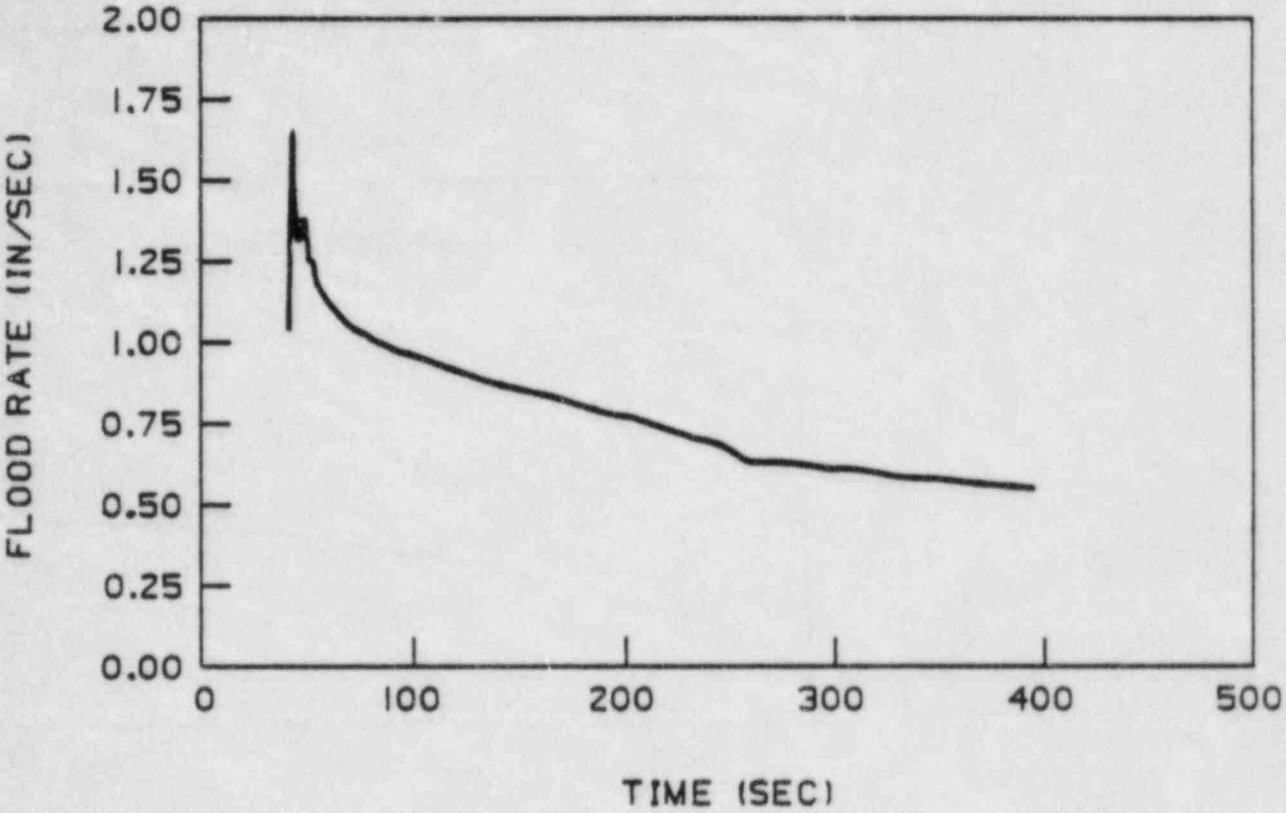


FIGURE 15.6-24
CORE INLET VELOCITY
DURING REFLOOD-DECLG (C_D=0.8)
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

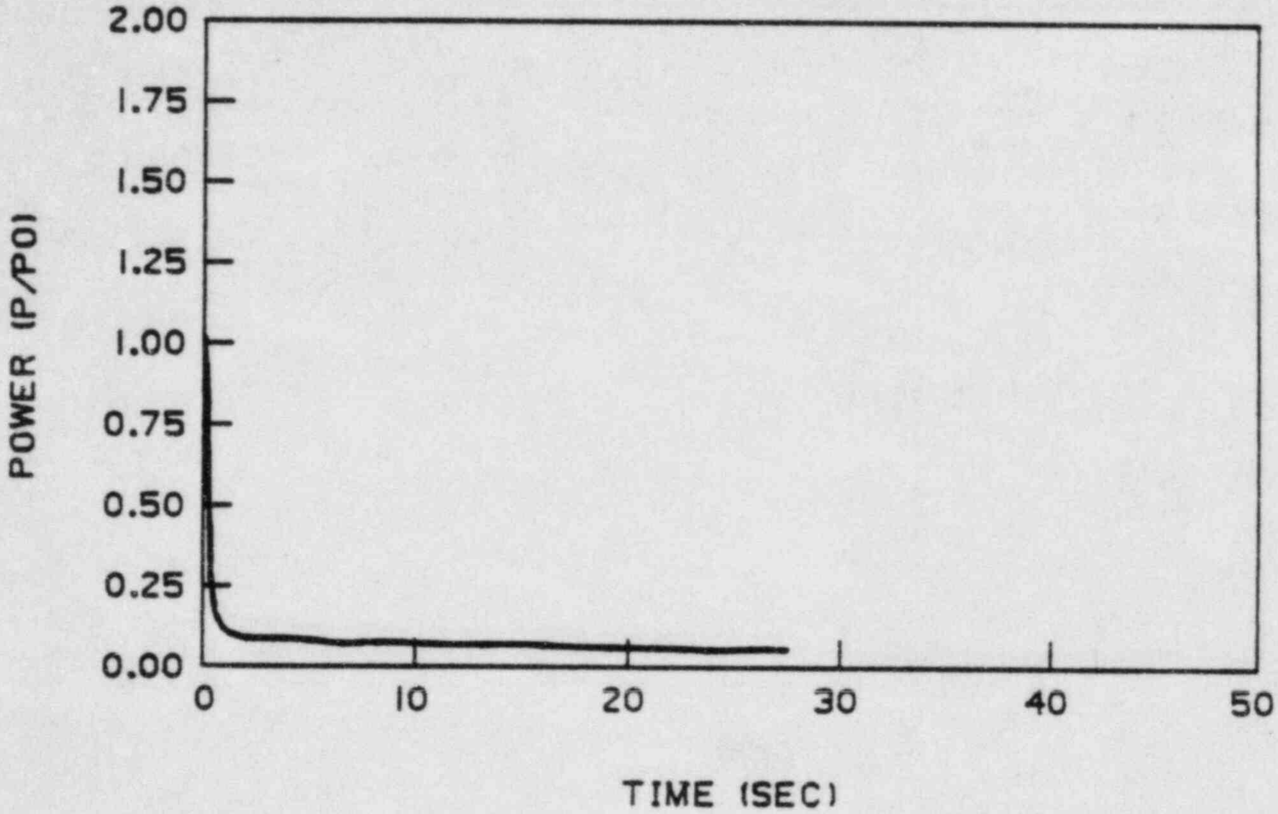


FIGURE 15.6-25
CORE POWER TRANSIENT
DECLG (C_D=0.8)
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

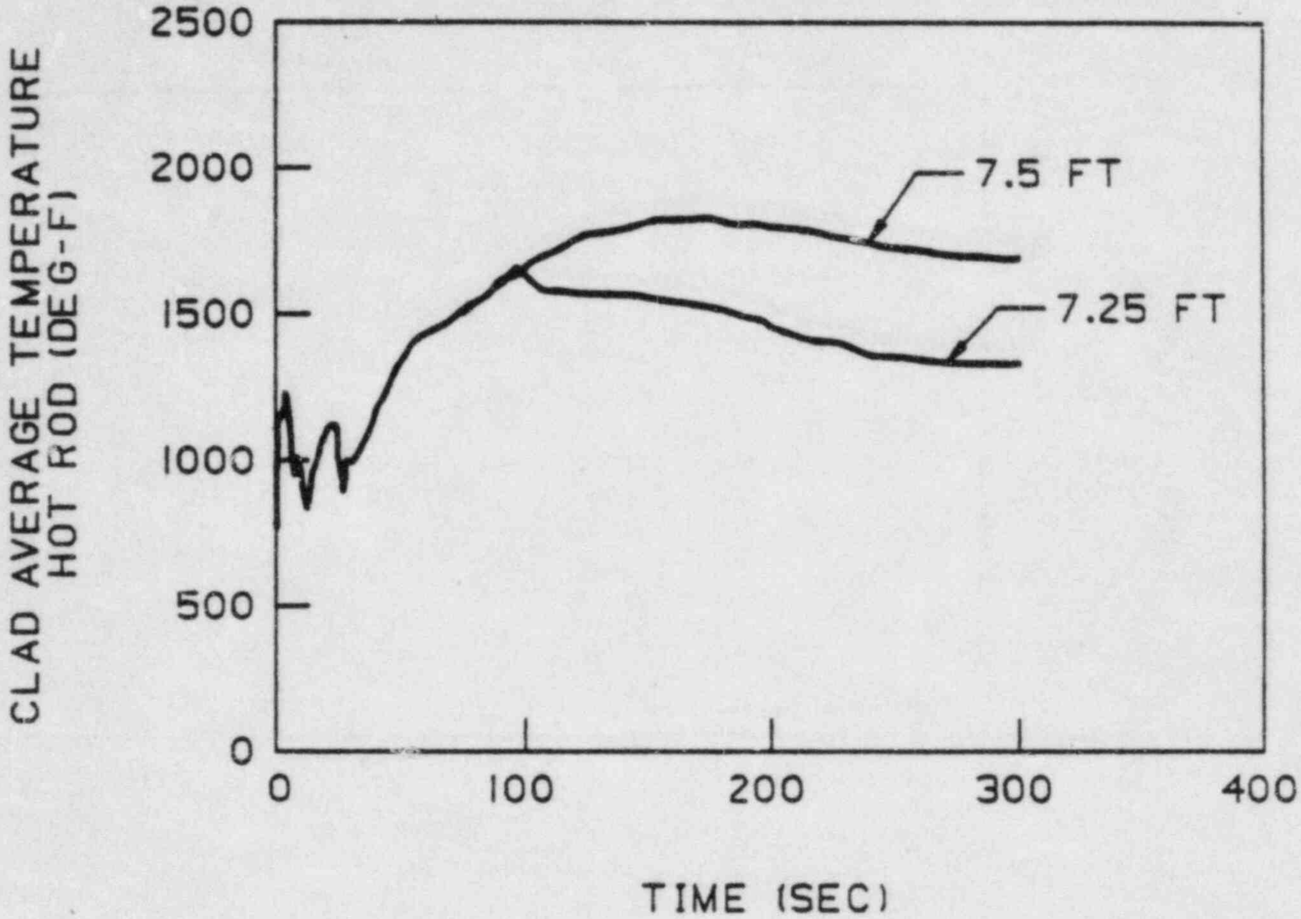


FIGURE 15.6-26
PEAK CLAD TEMPERATURE
DECLG (C_D=0.4)
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

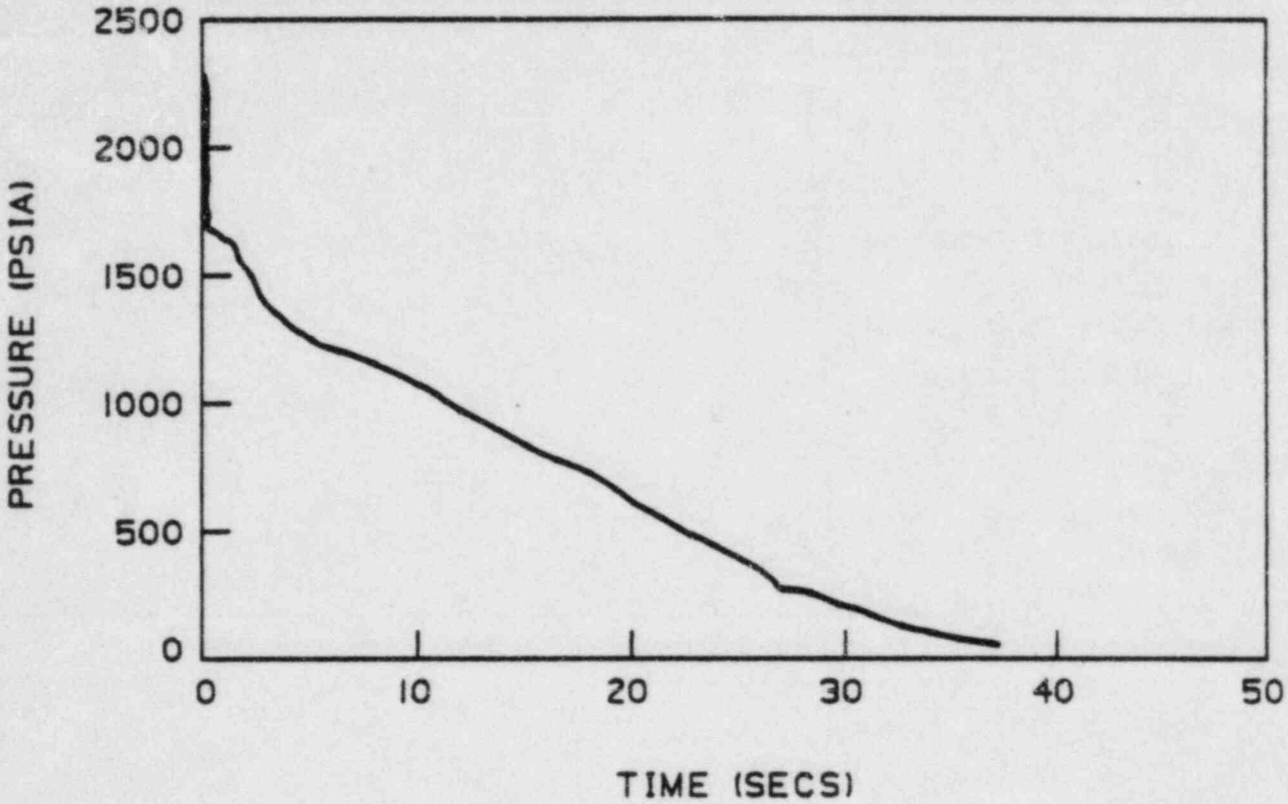


FIGURE 15.6-27
CORE PRESSURE
DECLG (C_D = 0.4)
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORTS

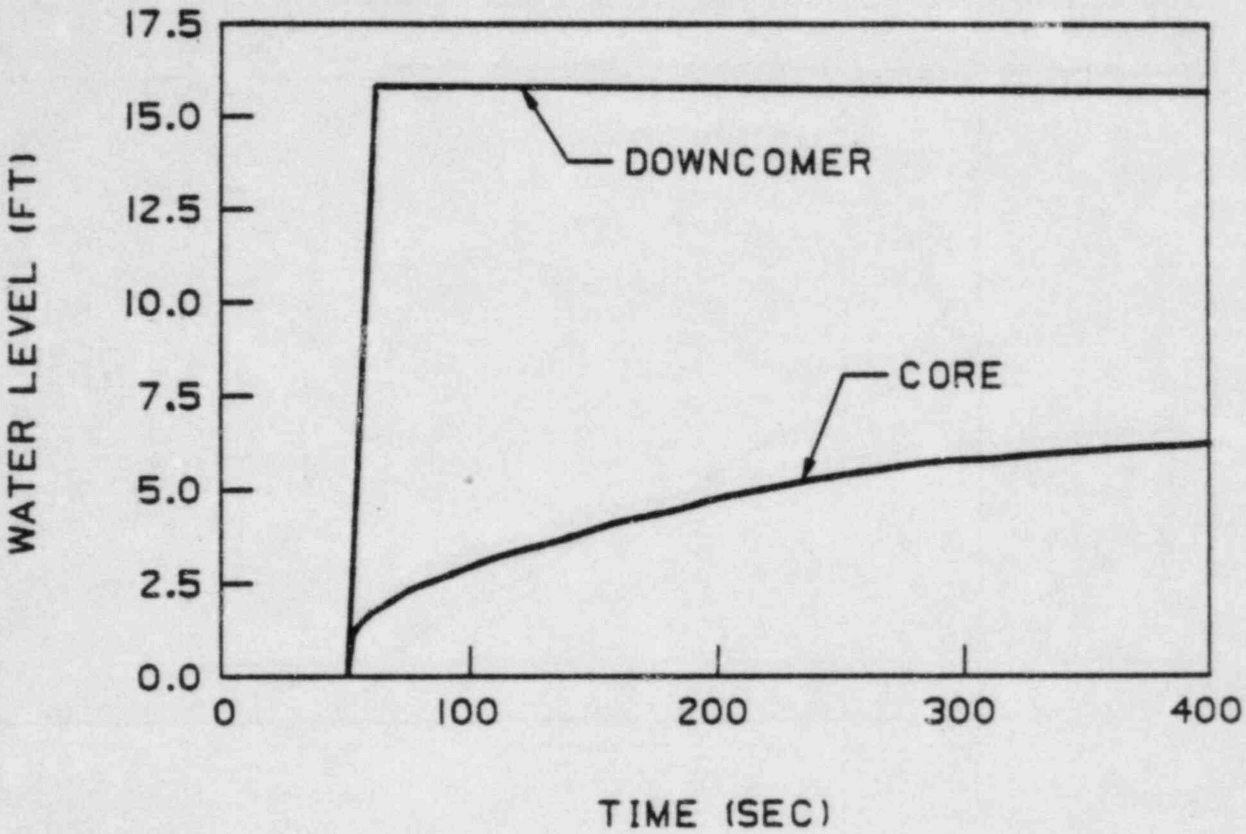


FIGURE 15.6.28
DOWNCOMER AND CORE WATER LEVELS
DURING REFLOOD-DECLG(C_D=O4)
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

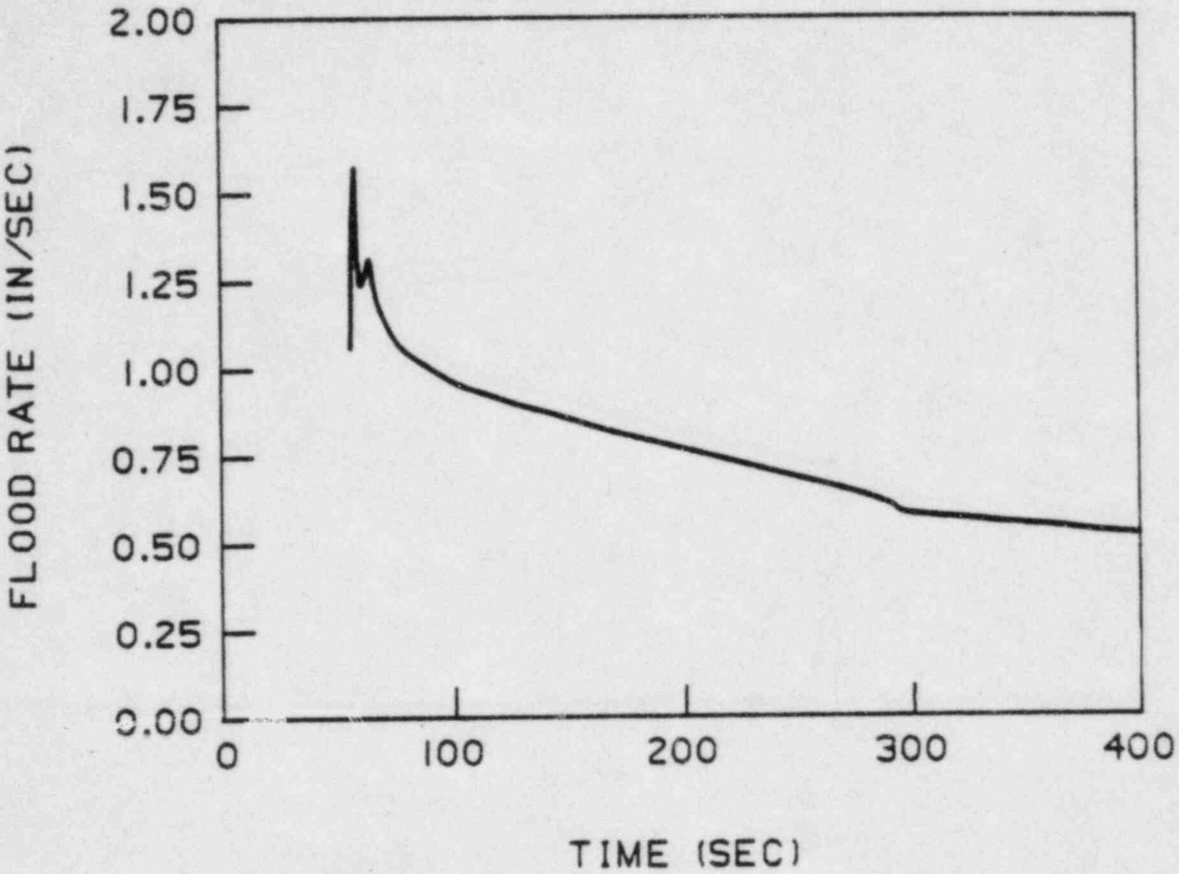


FIGURE 15.6-29
CORE INLET VELOCITY
DURING REFLOOD - DECLG (C_D=0.4)
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

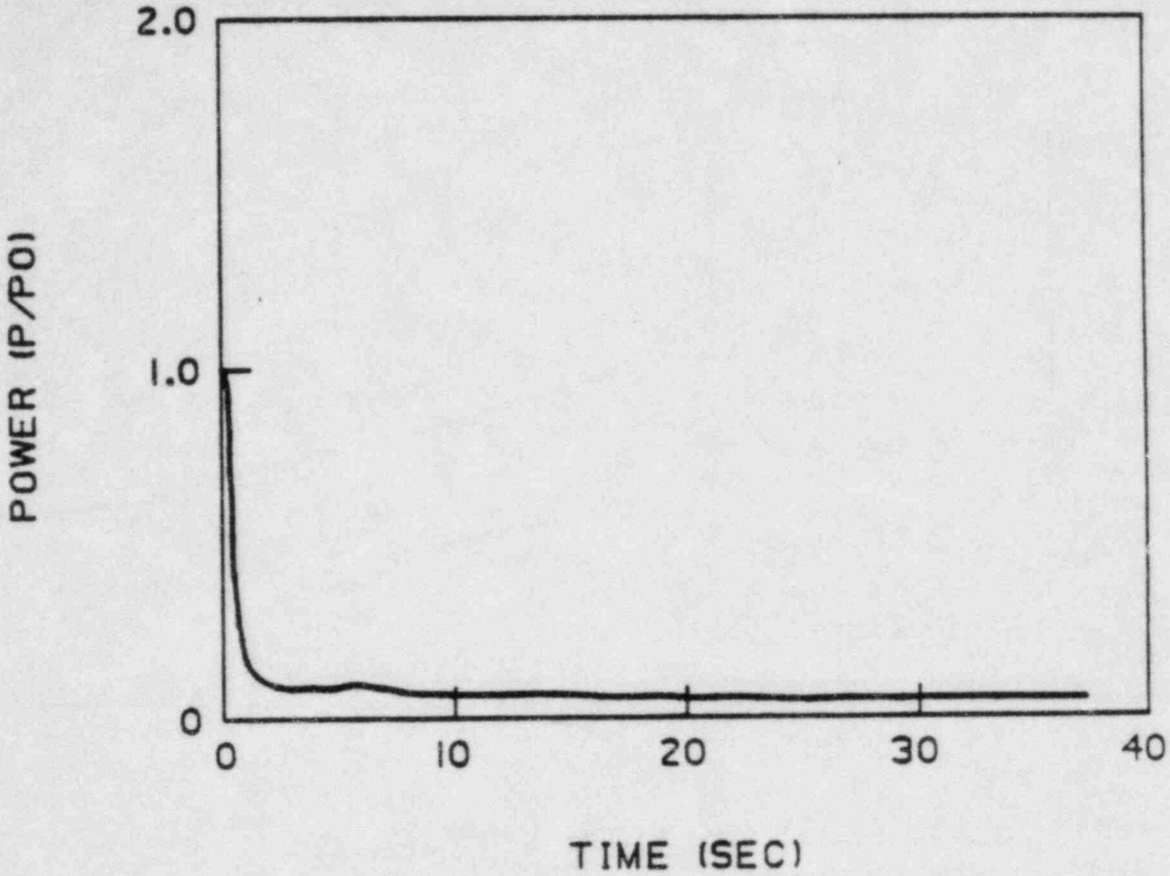


FIGURE 15.6-30
CORE POWER TRANSIENT—
DECLG (C_D=0.4)
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

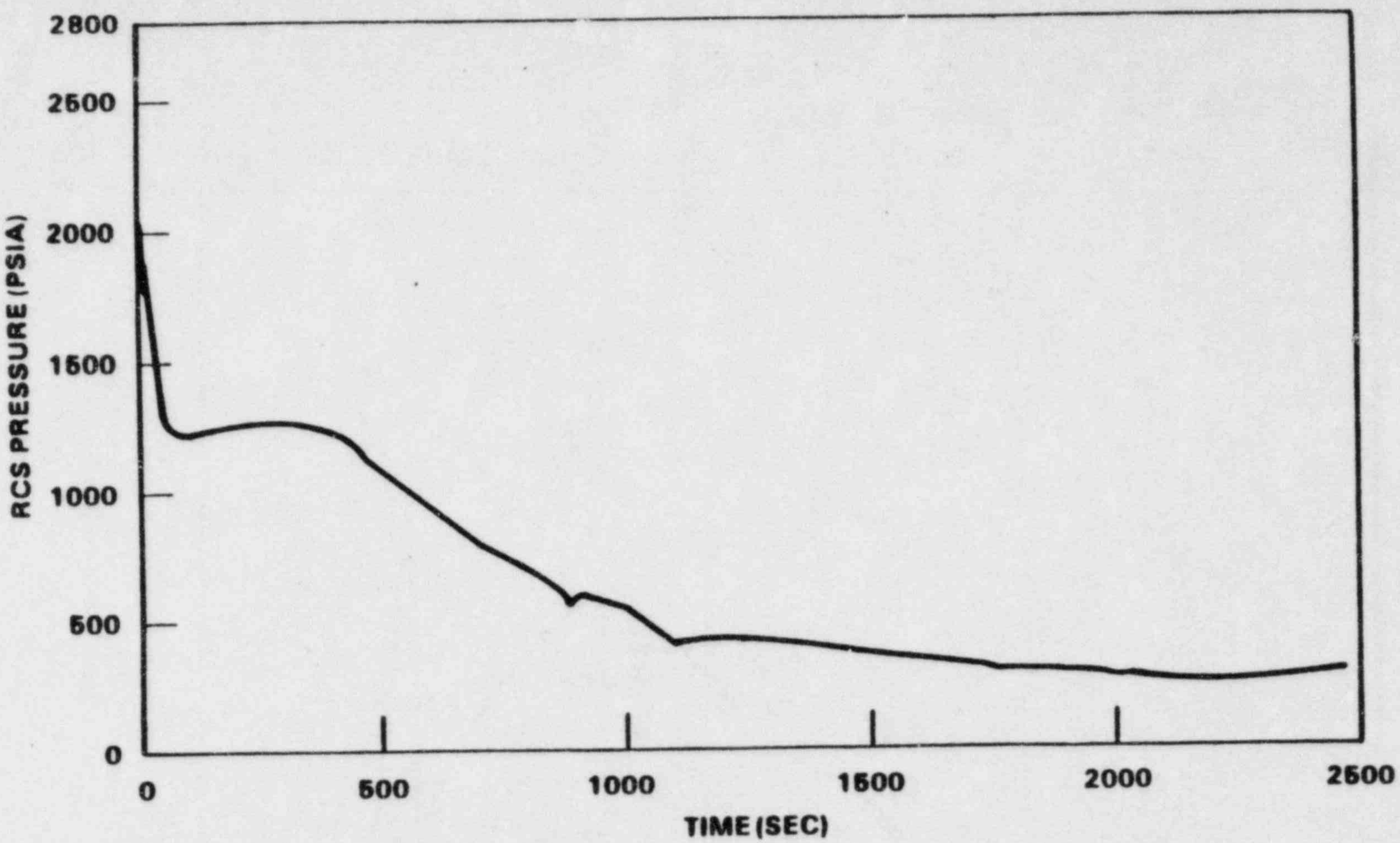


FIGURE 15.6-31
REACTOR COOLANT SYSTEM
DEPRESSURIZATION TRANSIENT
(4 INCH BREAK)
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

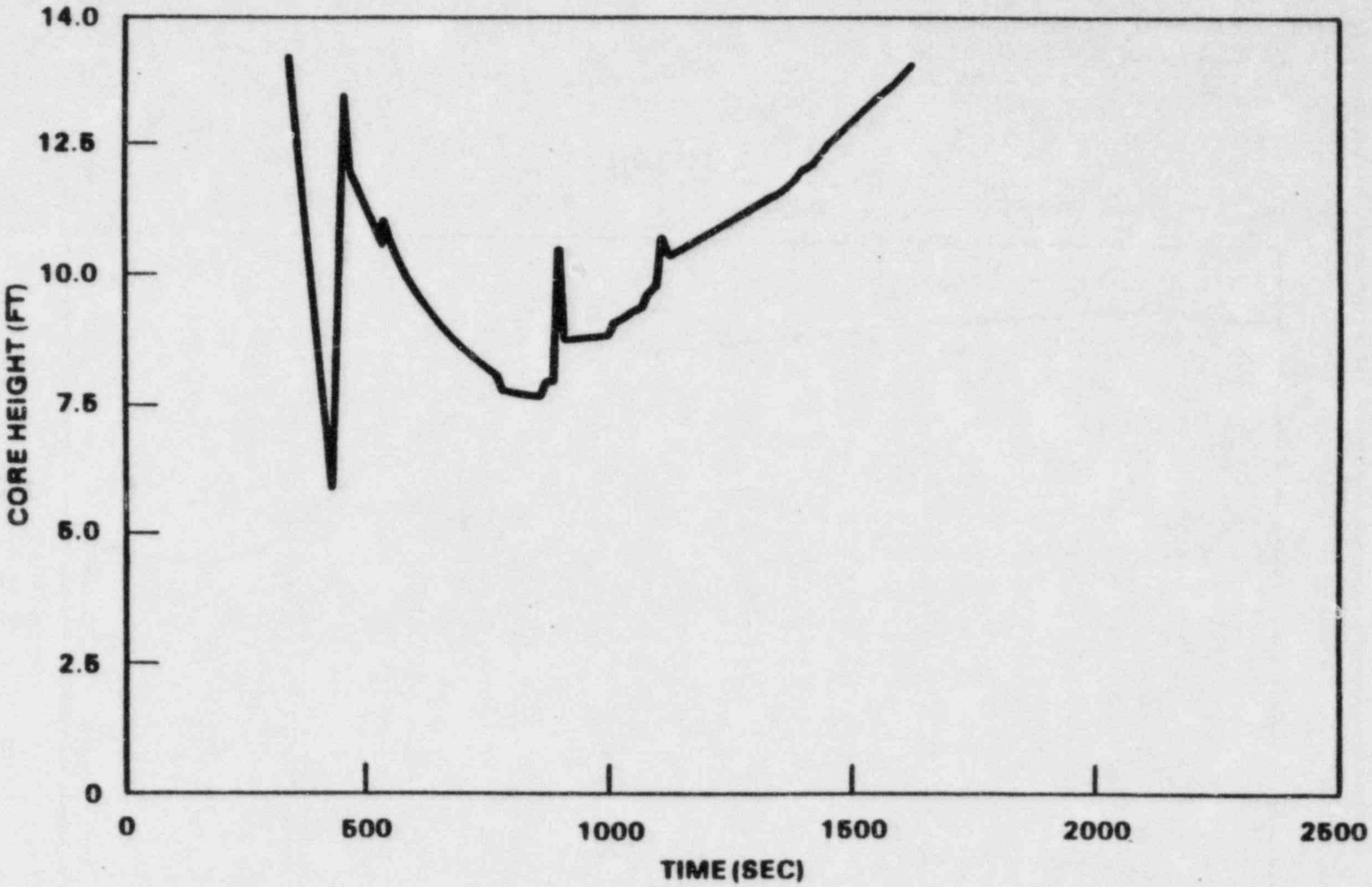


FIGURE 15.6-32
CORE MIXTURE
(4 INCH BREAK)
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

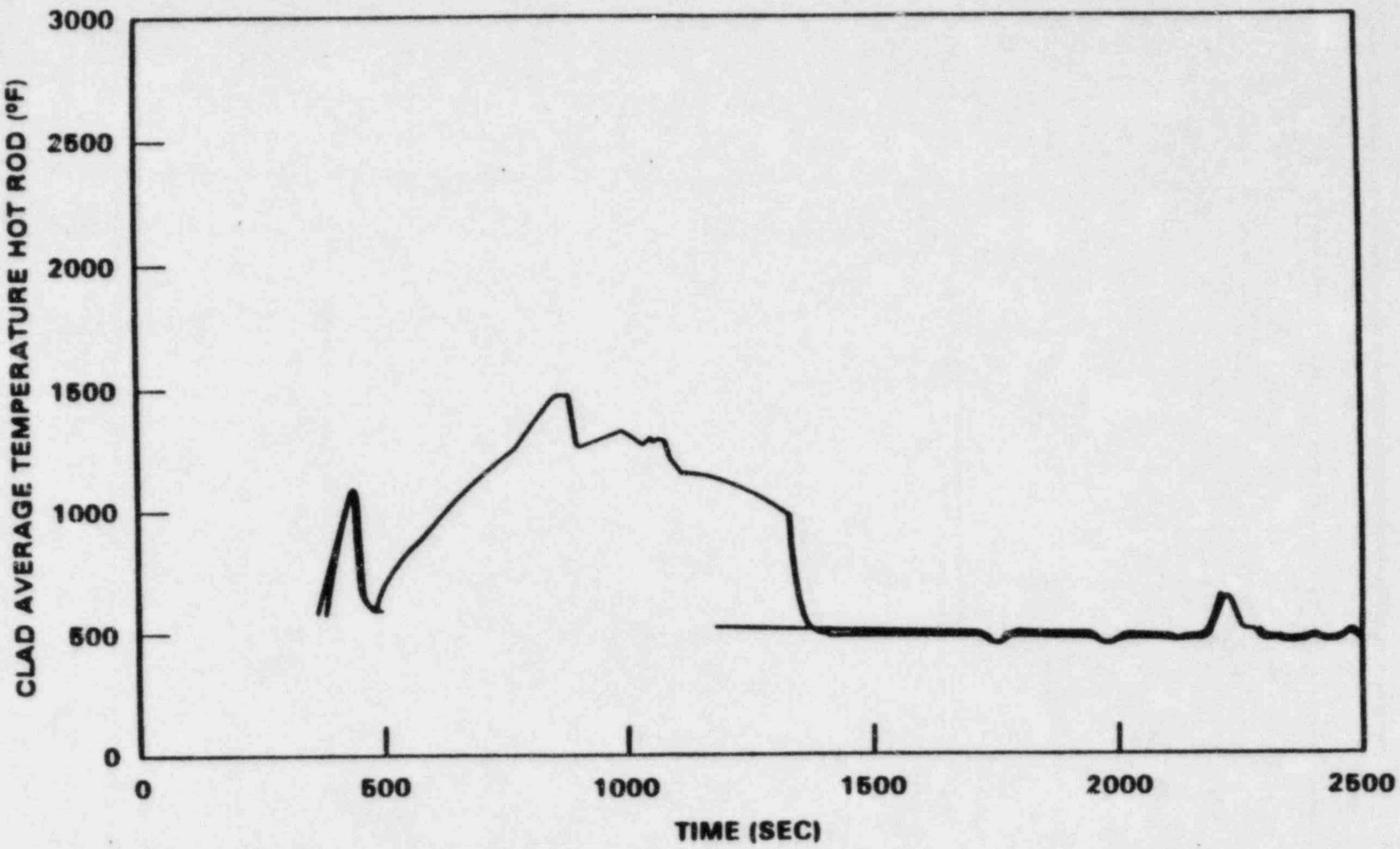


FIGURE 15.6-33
CLAD TEMPERATURE TRANSIENT
(4 INCH BREAK)
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

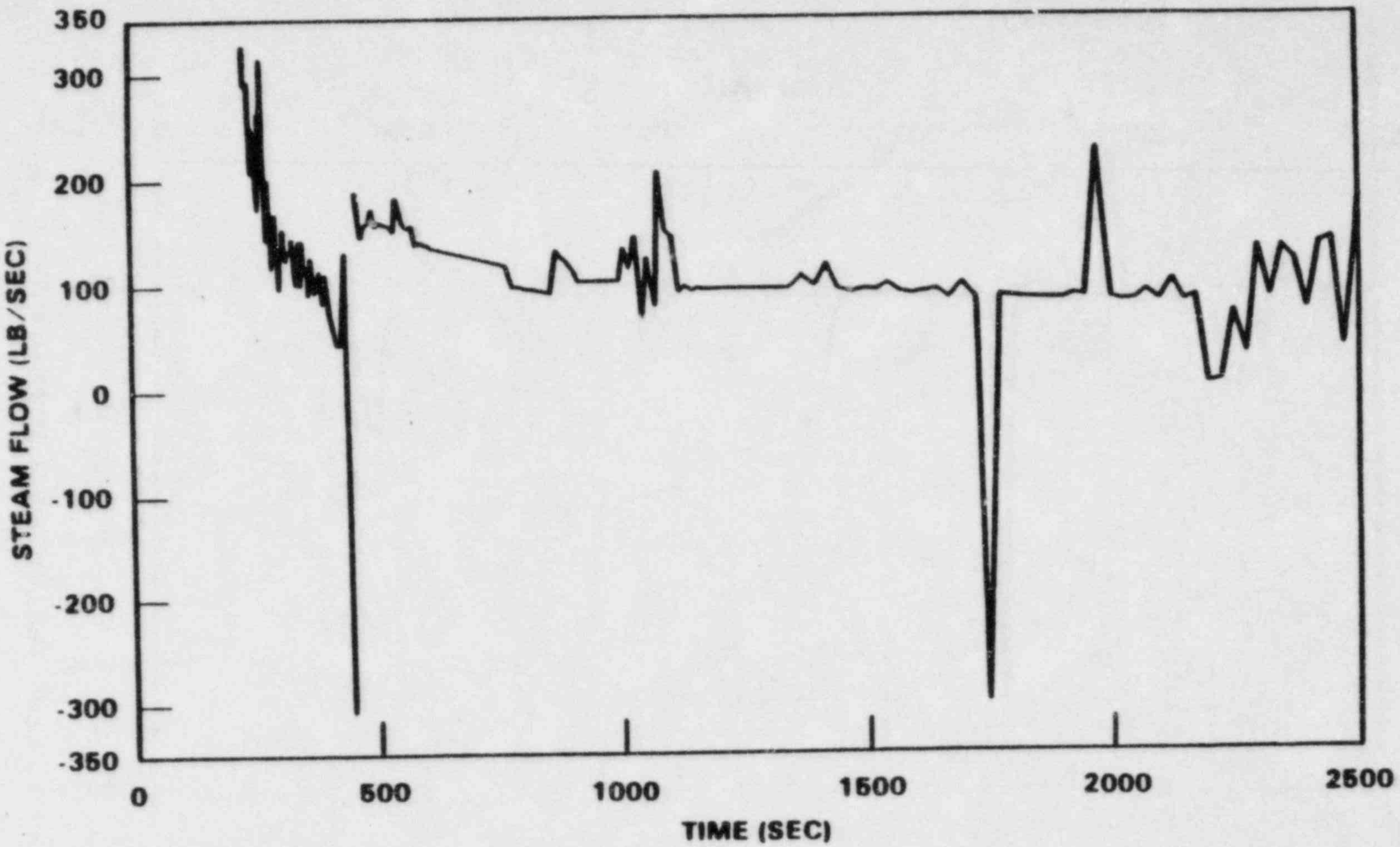


FIGURE 15.6-34
STEAM FLOW
(4 INCH BREAK)
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

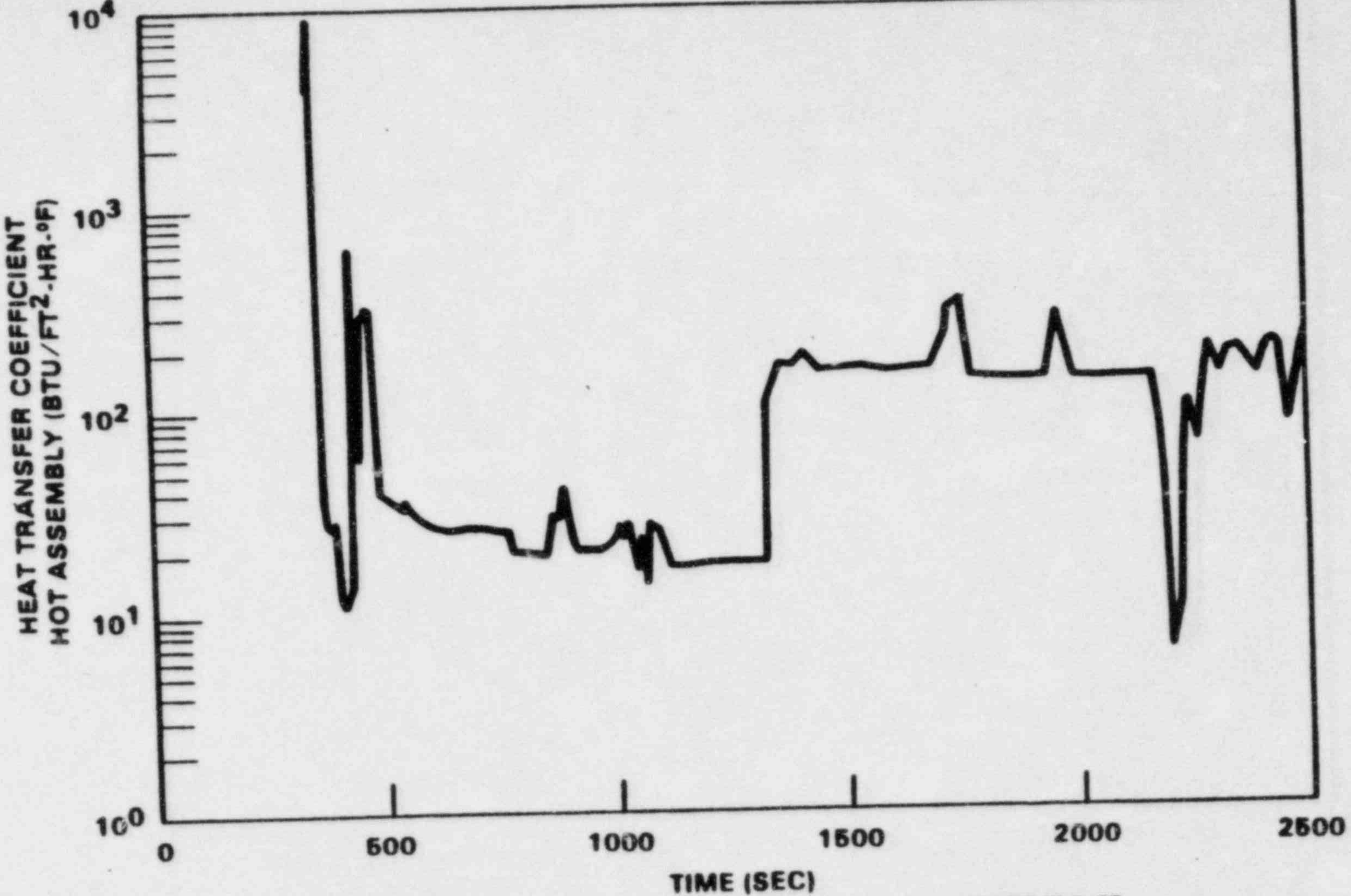


FIGURE 15.6-35
ROD FILM HEAT TRANSFER
COEFFICIENT - 4 INCH BREAK
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

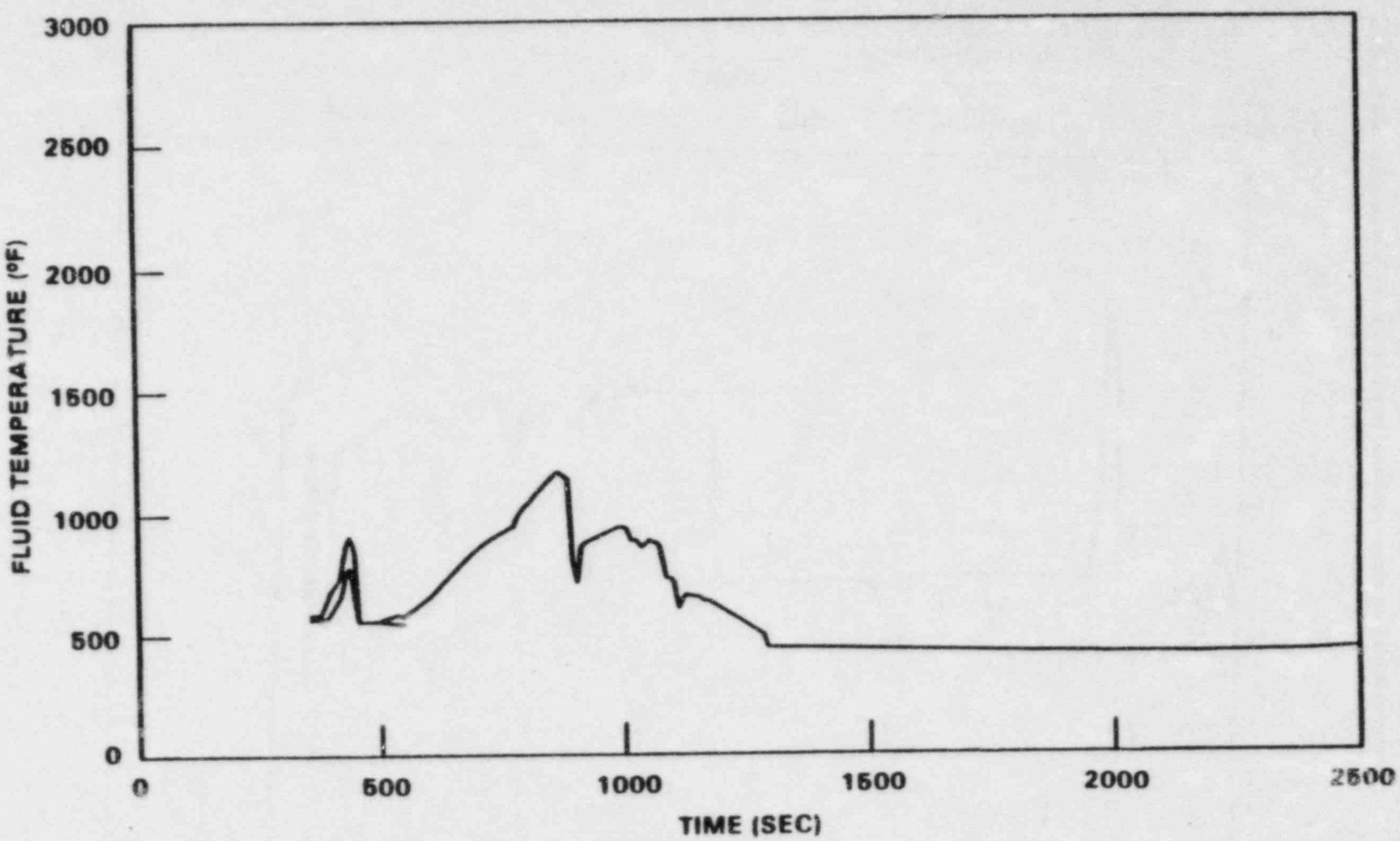


FIGURE 15.6-36
HOT SPOT FLUID TEMPERATURE
(4 INCH BREAK)
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

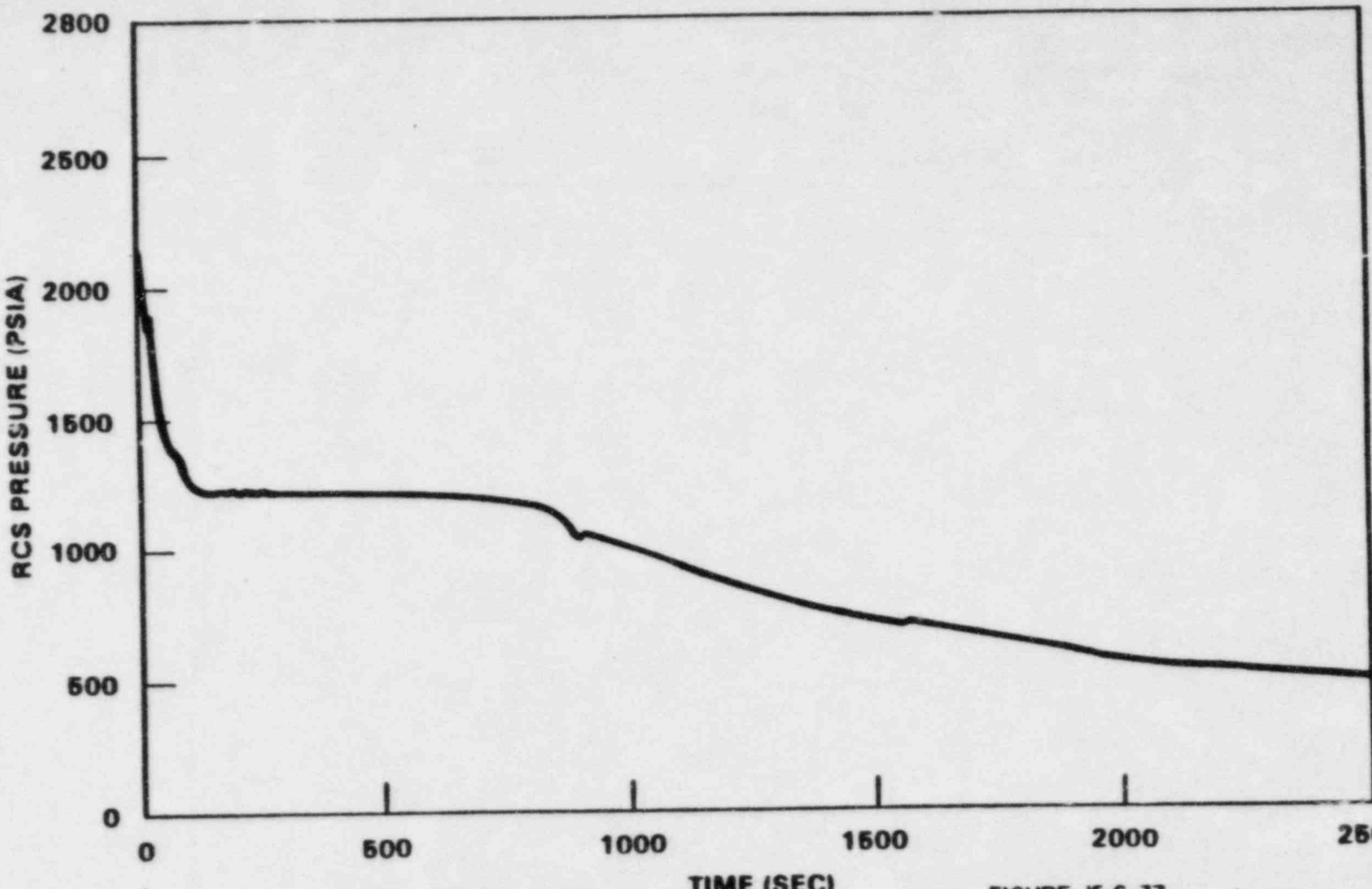


FIGURE 15.6-37
REACTOR COOLANT SYSTEM
DEPRESSURIZATION TRANSIENT
(3 INCH BREAK)

MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

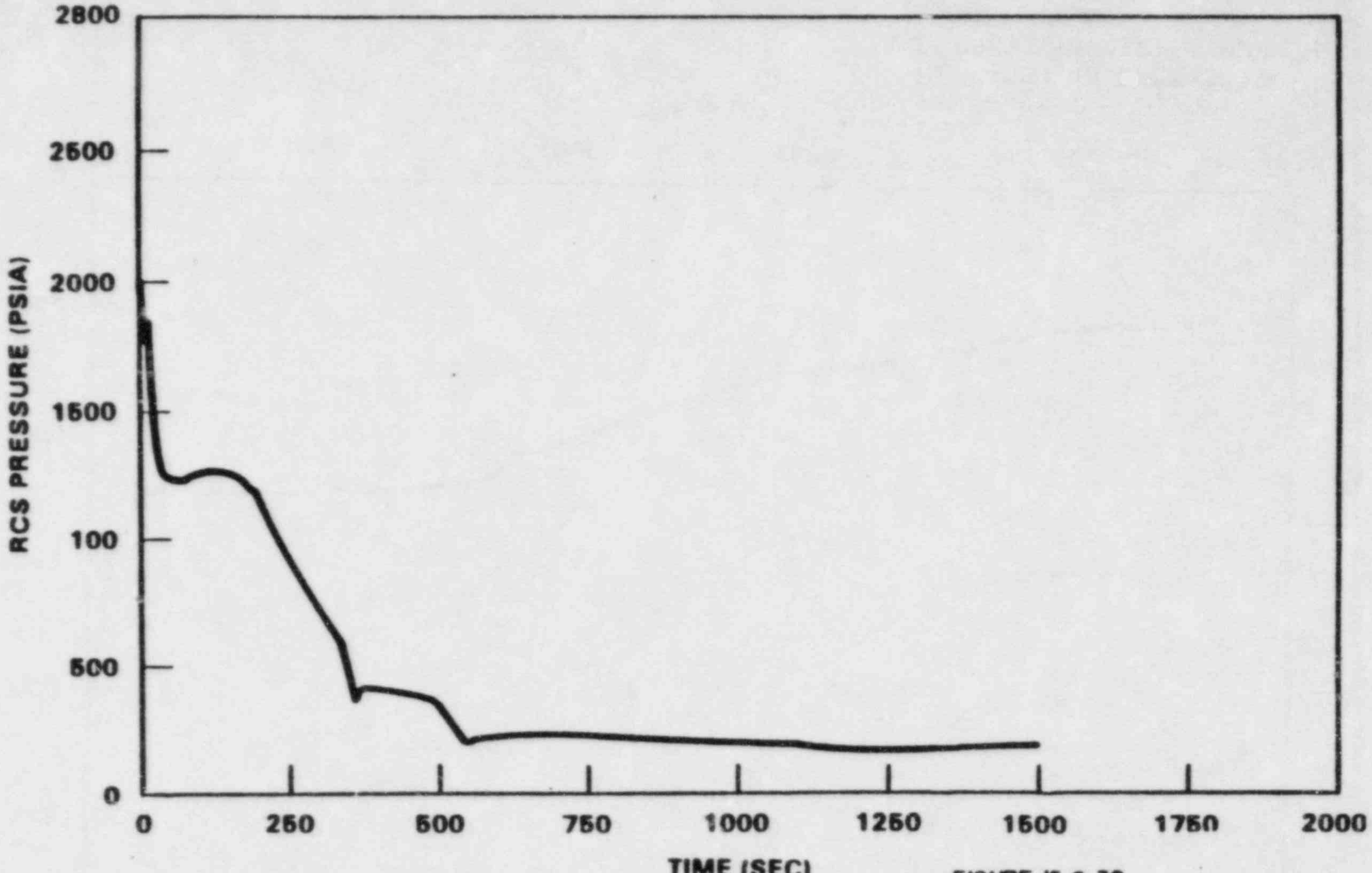


FIGURE 15.6-38
REACTOR COOLANT SYSTEM
DEPRESSURIZATION TRANSIENT
(6 INCH BREAK)
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

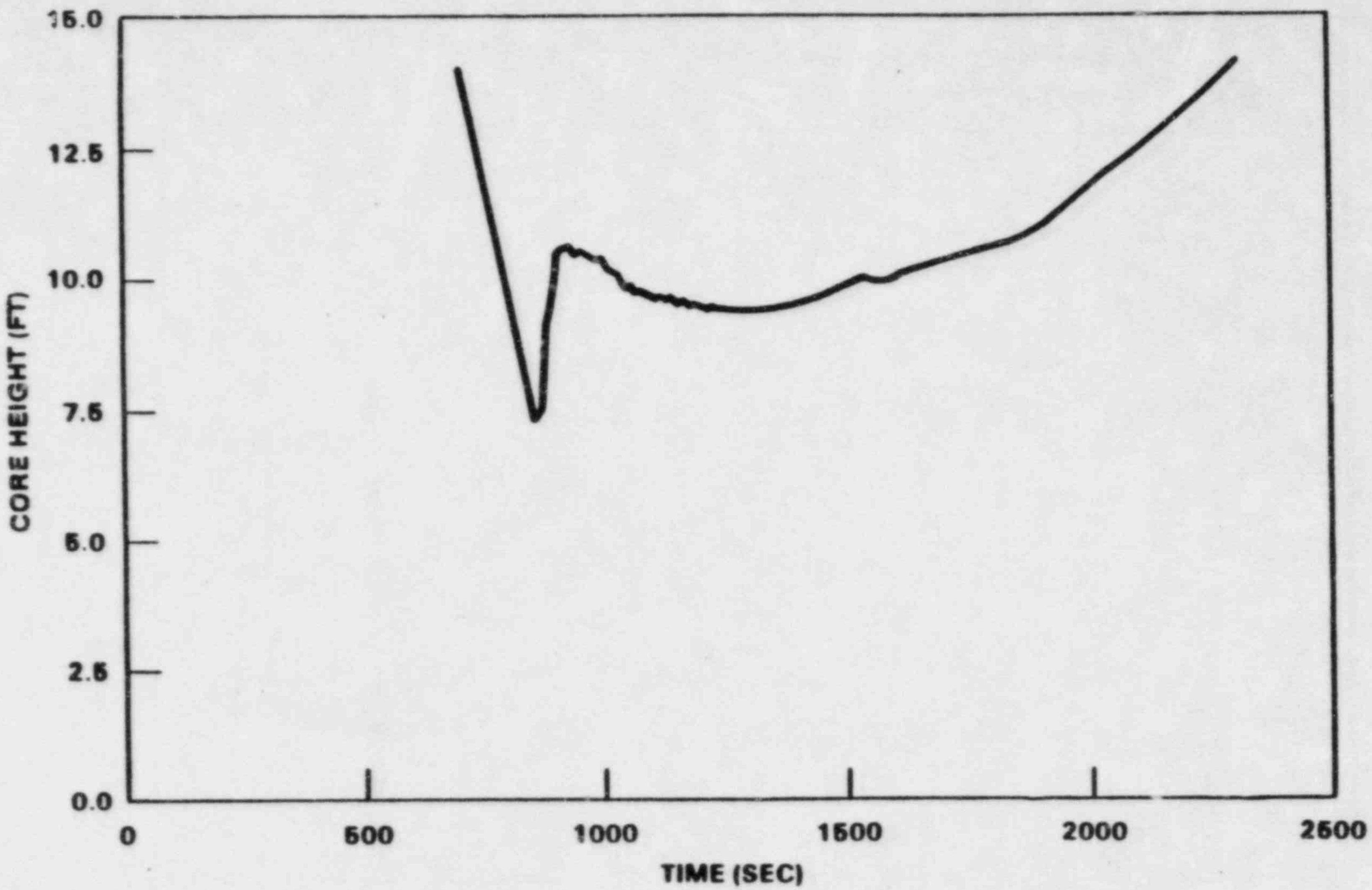


FIGURE 15.6-39
CORE MIXTURE HEIGHT
(3 INCH BREAK)
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

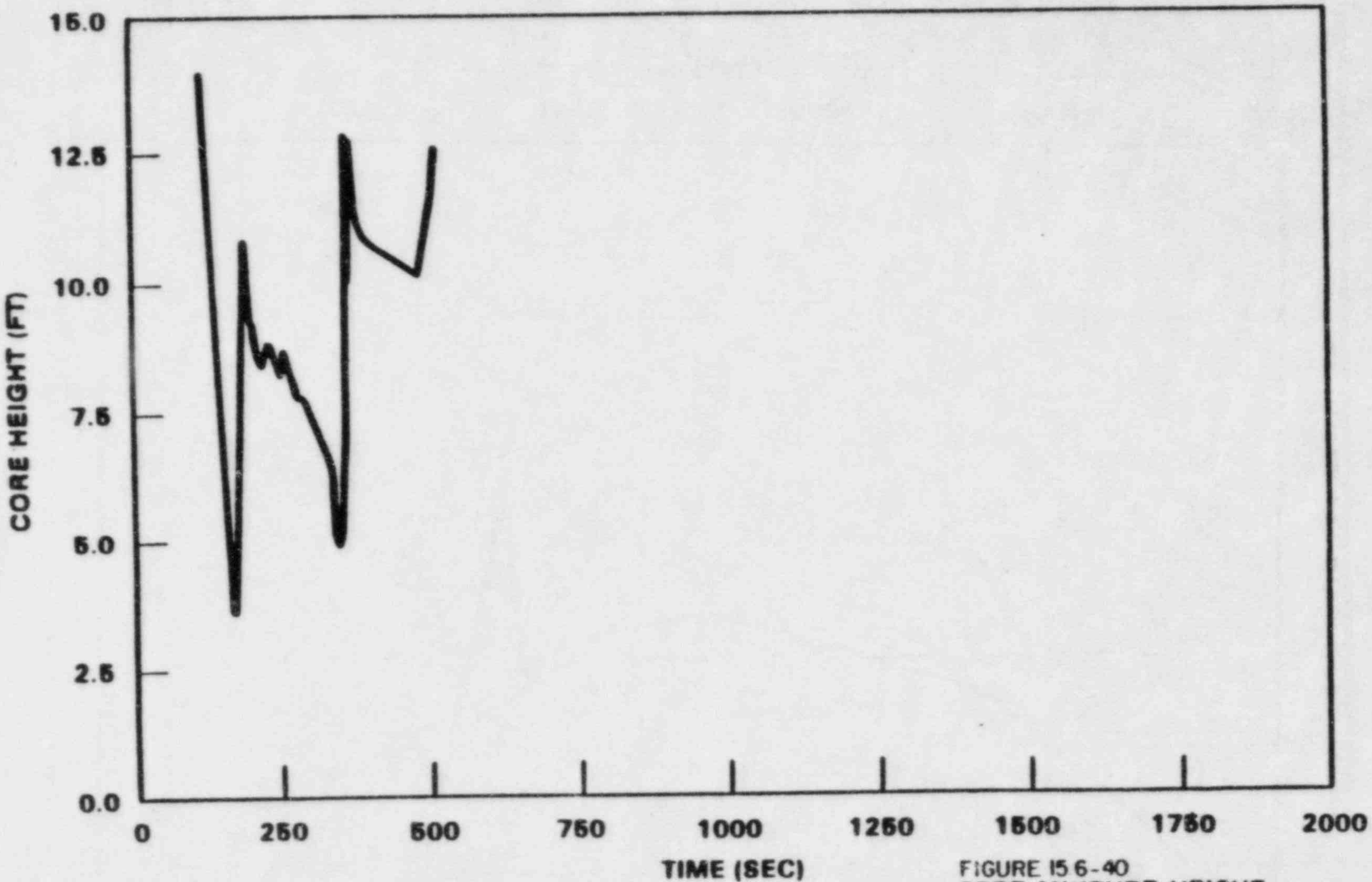


FIGURE 15.6-40
CORE MIXTURE HEIGHT
(6 INCH BREAK)

MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

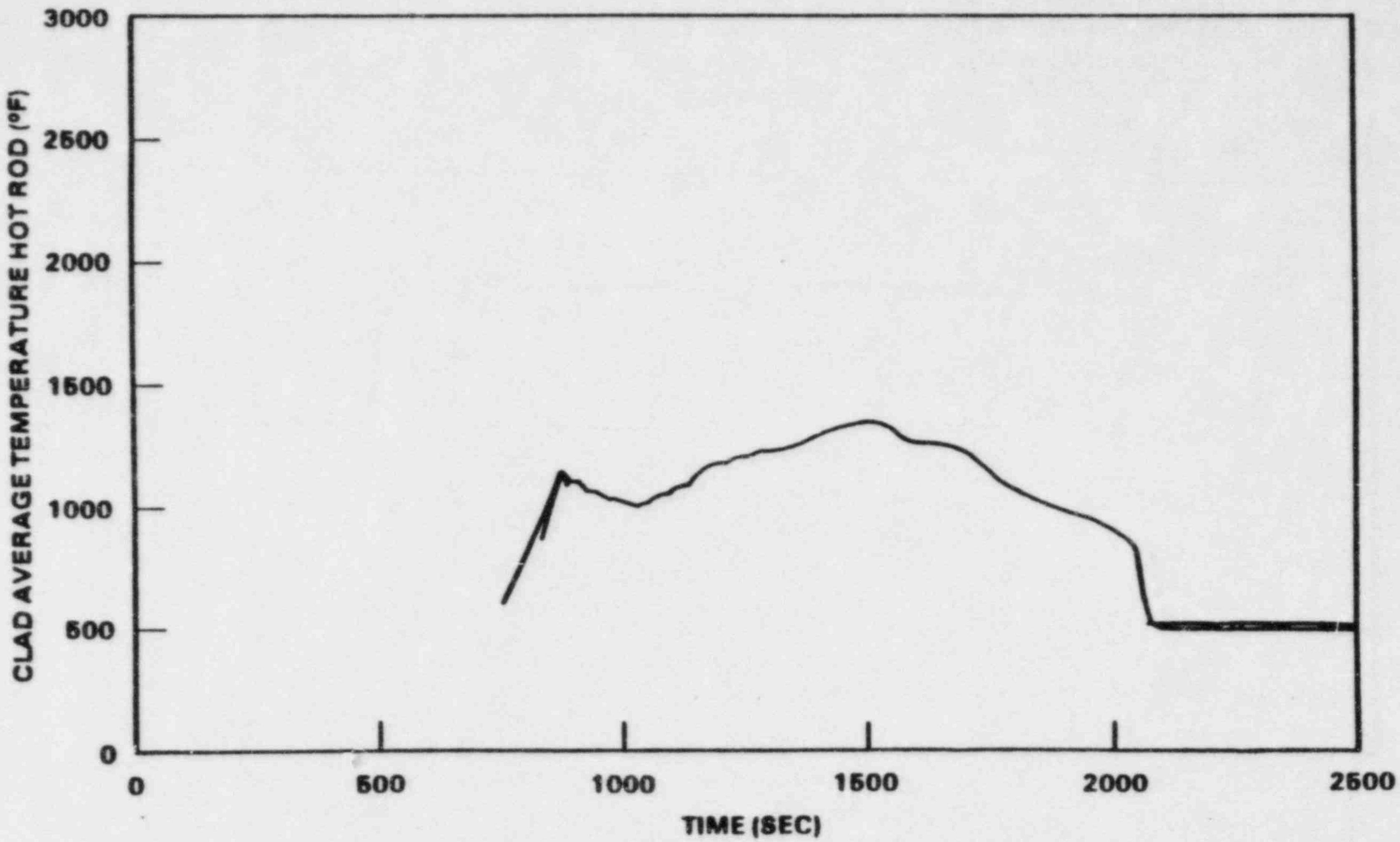


FIGURE 15.6-41
CLAD TEMPERATURE TRANSIENT
(3 INCH BREAK)
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

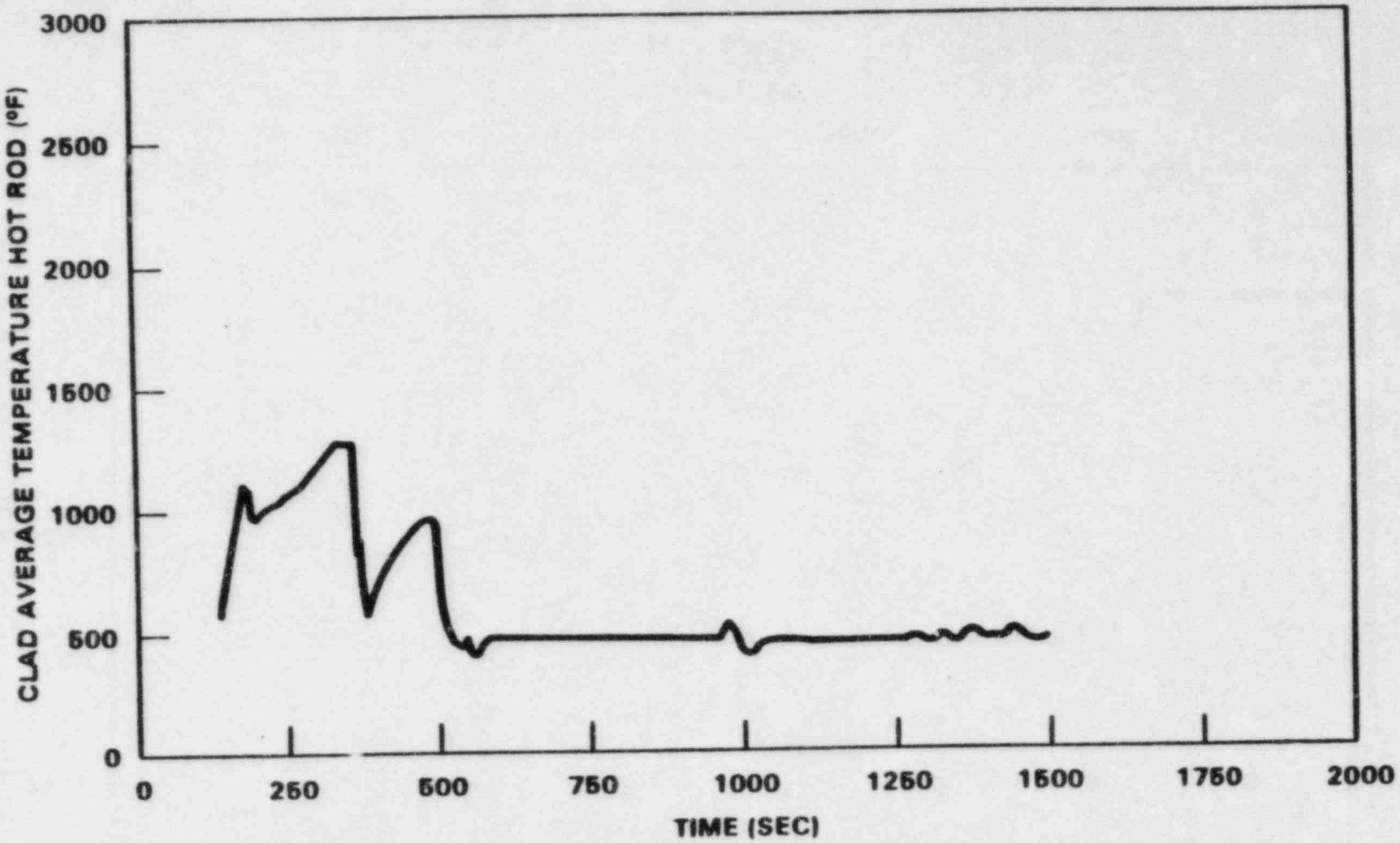


FIGURE 15.6-42
CLAD TEMPERATURE TRANSIENT
(6 INCH BREAK)
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

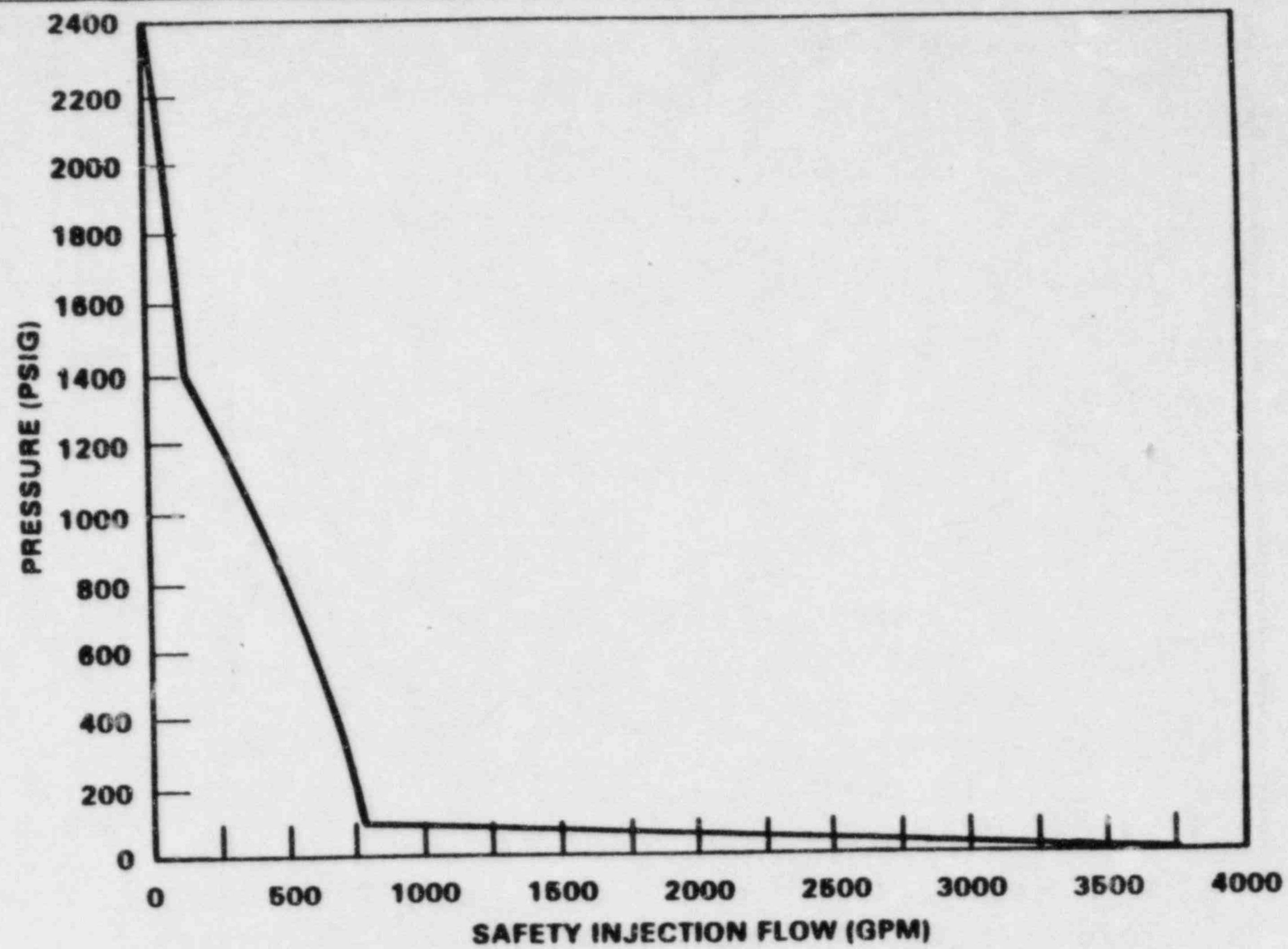


FIGURE 15.6-43
SAFETY INJECTION FLOWRATE
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

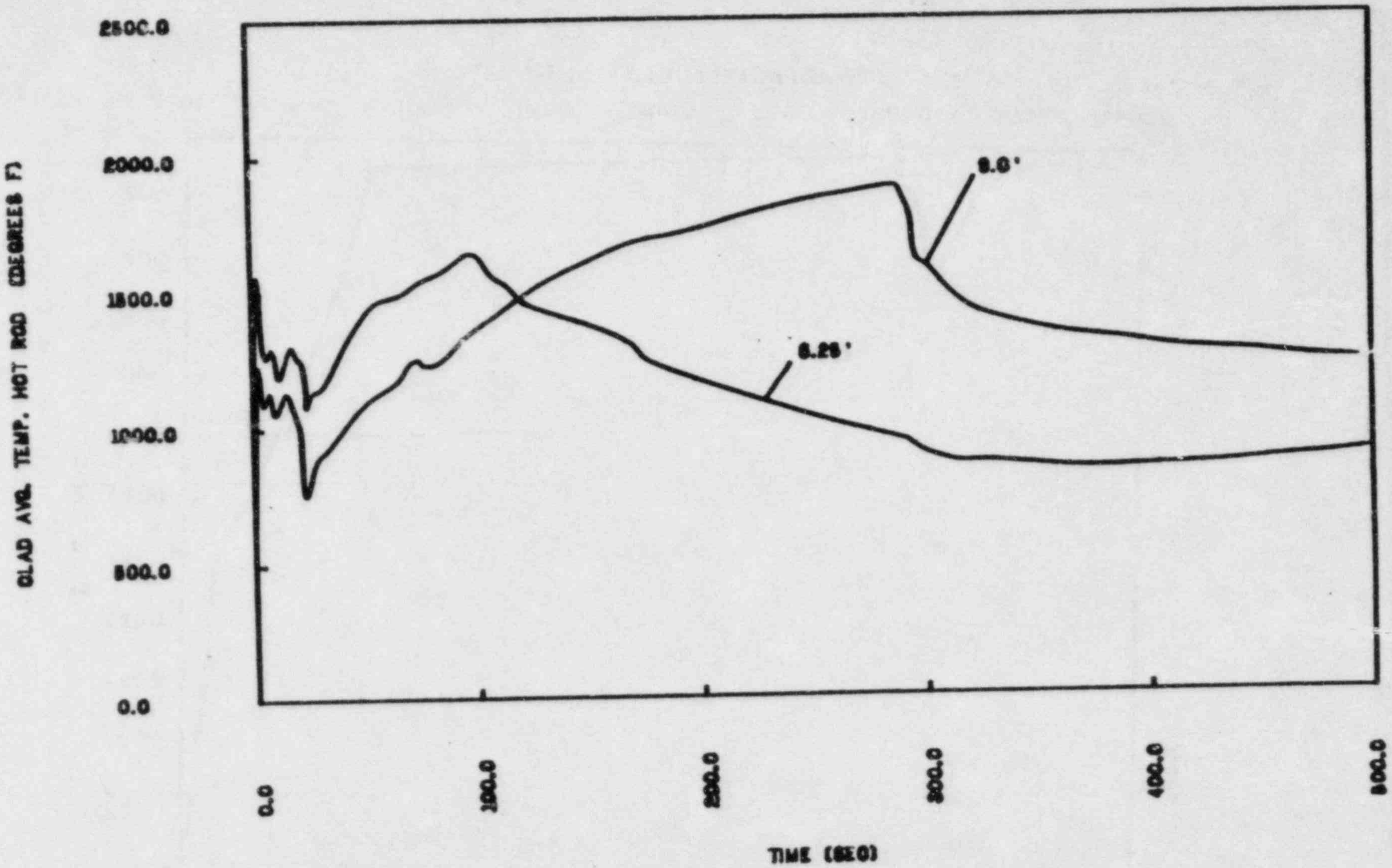


FIGURE 15.6-44
HOT SPOT CLAD TEMPERATURE (Active Loop Break C_D = 0.4)
N-1 LOOP OPERATION

INTERACTIVE GRAPHICS

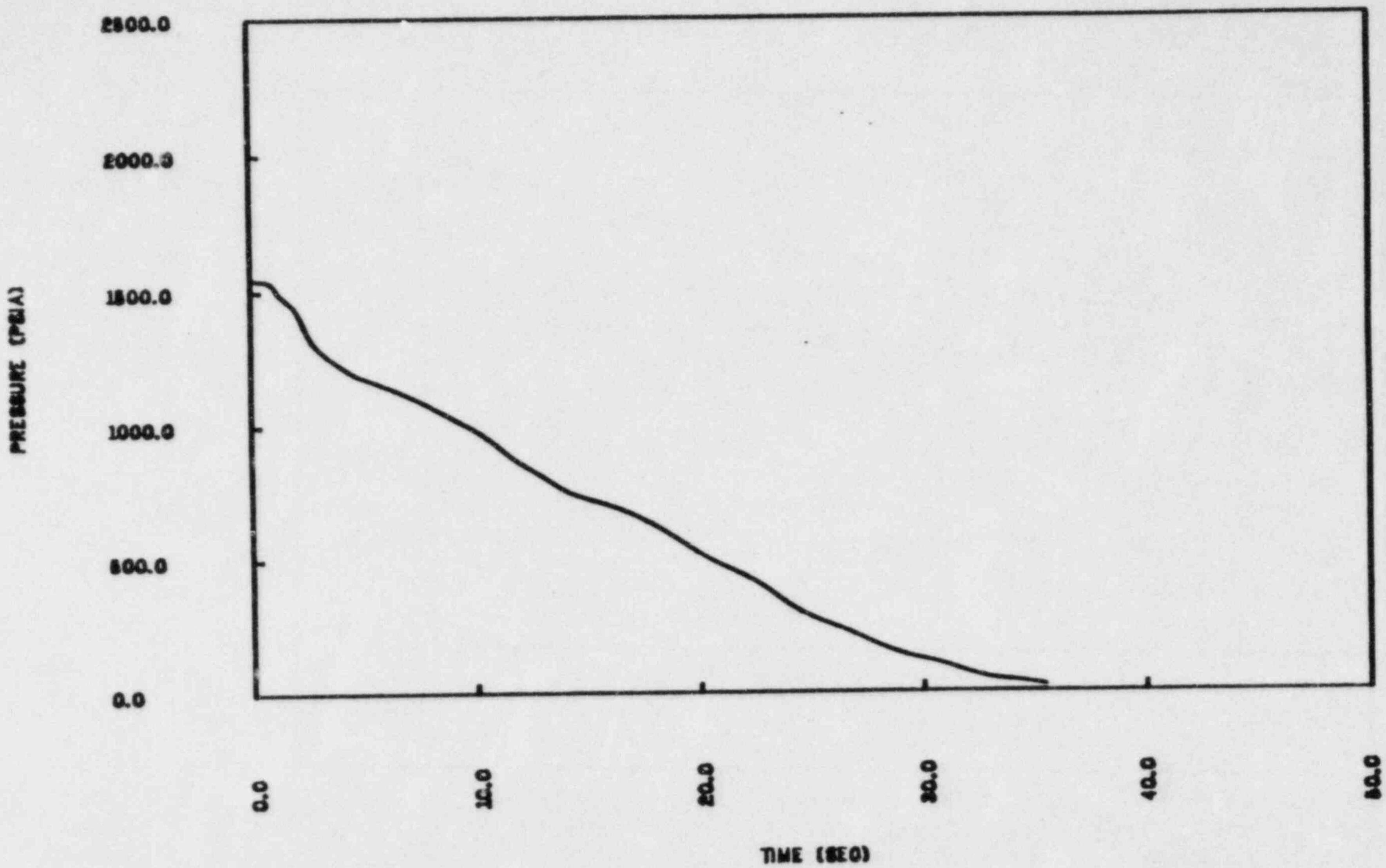
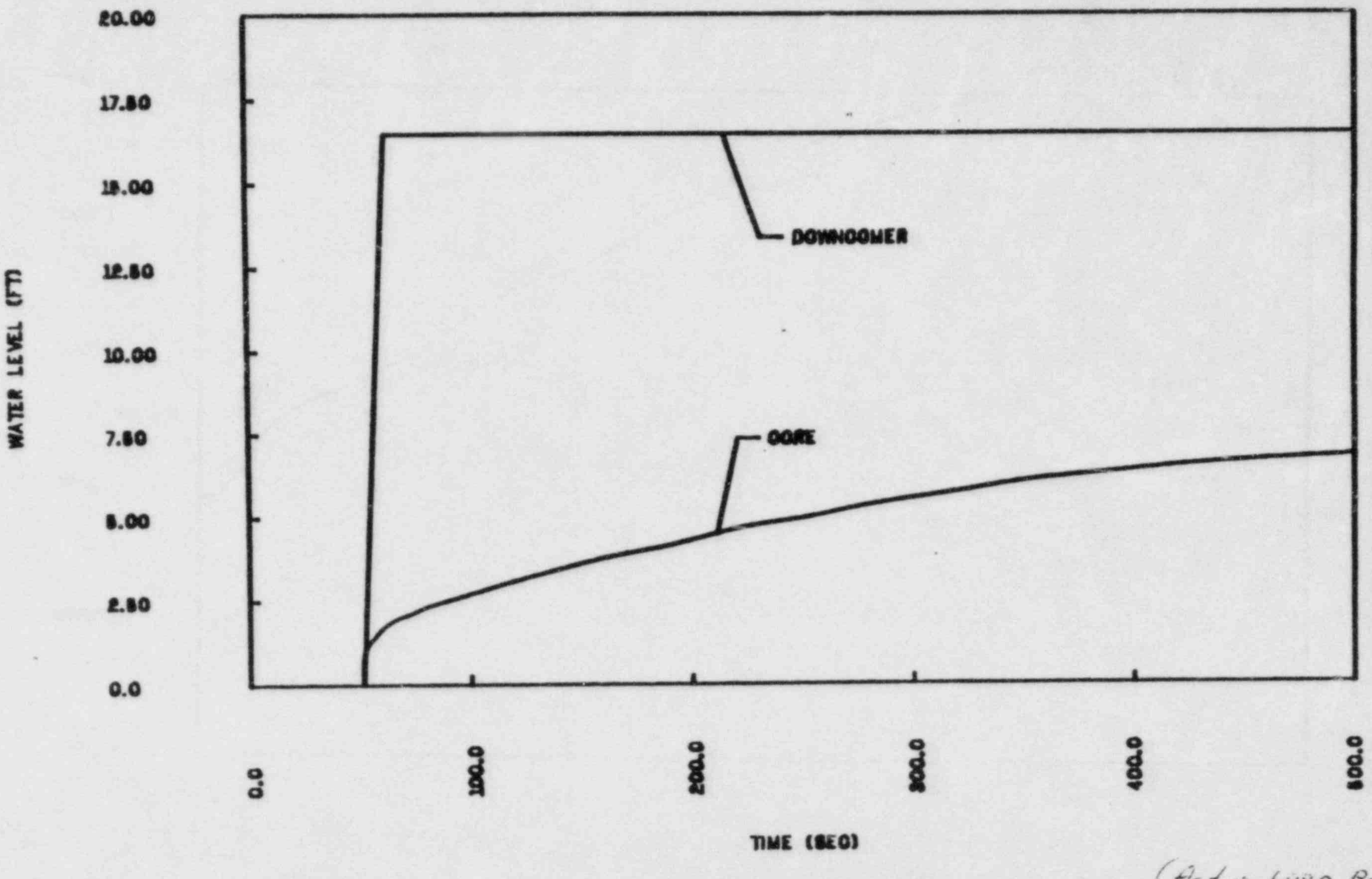


FIGURE 15.6-45
COOLANT PRESSURE IN THE REACTOR CORE (Active Loop Break CD=0.7
N-1 LOOP OPERATION



(Active Loop Break CD=

FIGURE 15.6-46
WATER LEVEL IN THE CORE AND DOWNCOMER DURING REFLOOD ^
N-1 LOOP OPERATION

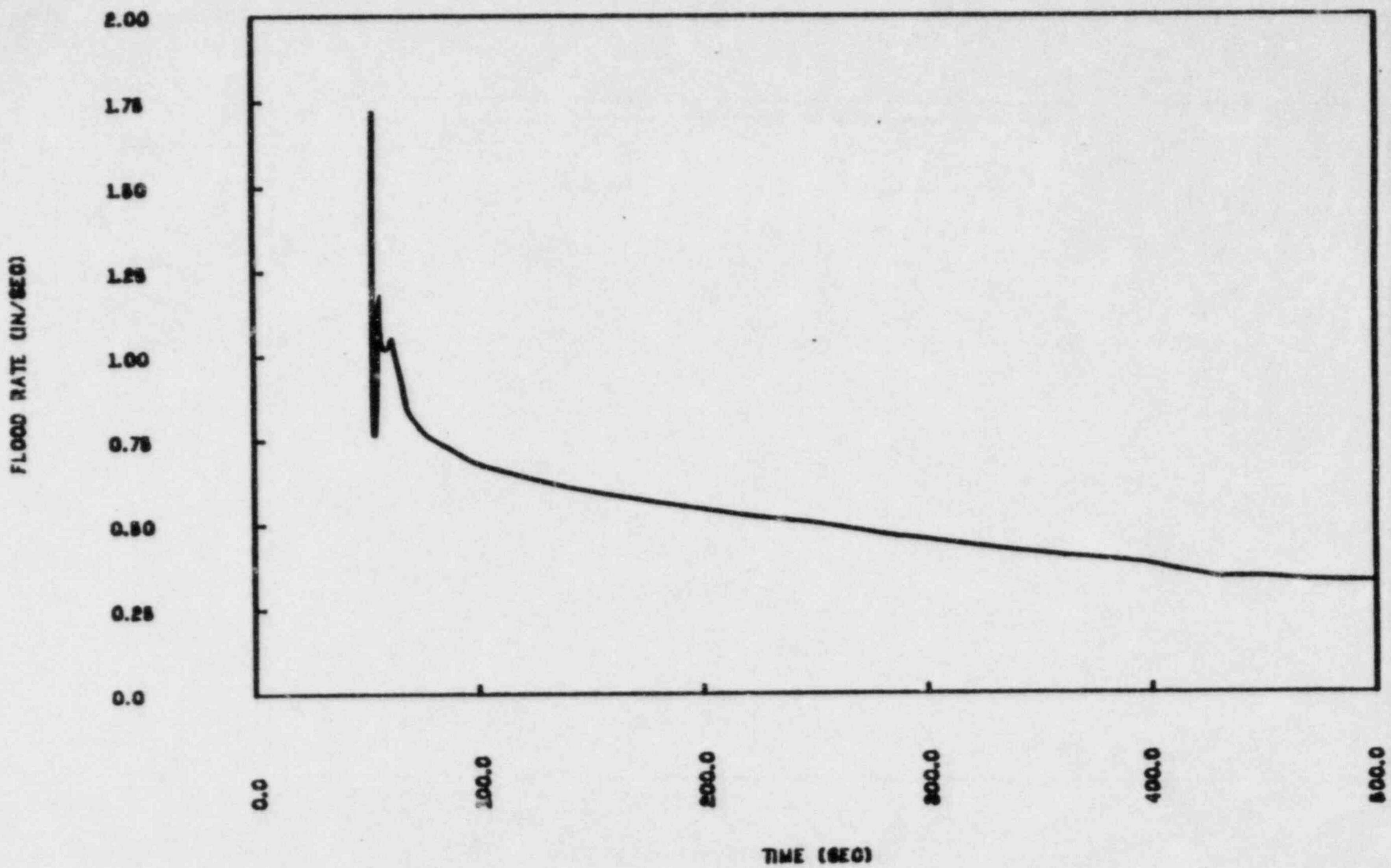


FIGURE 15.6-47
CORE REFLOODING RATE (Active Loop Break CD=0.4)
N-1 LOOP OPERATION

INTERACTIVE GRAPHICS

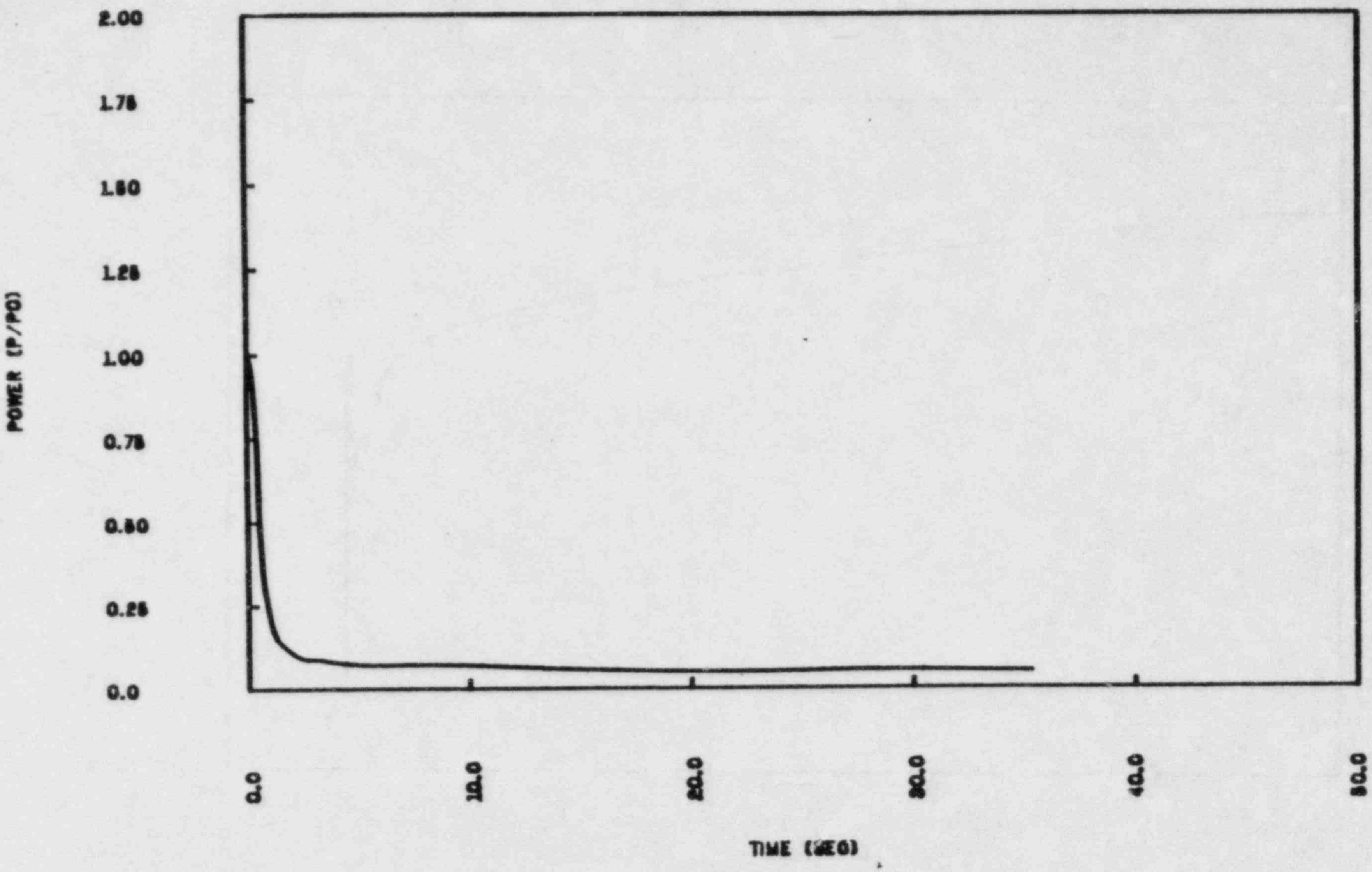


FIGURE 15.8-48
THERMAL POWER DURING BLOWDOWN (Active Loop Break CD=0.4)
N-1 LOOP OPERATION

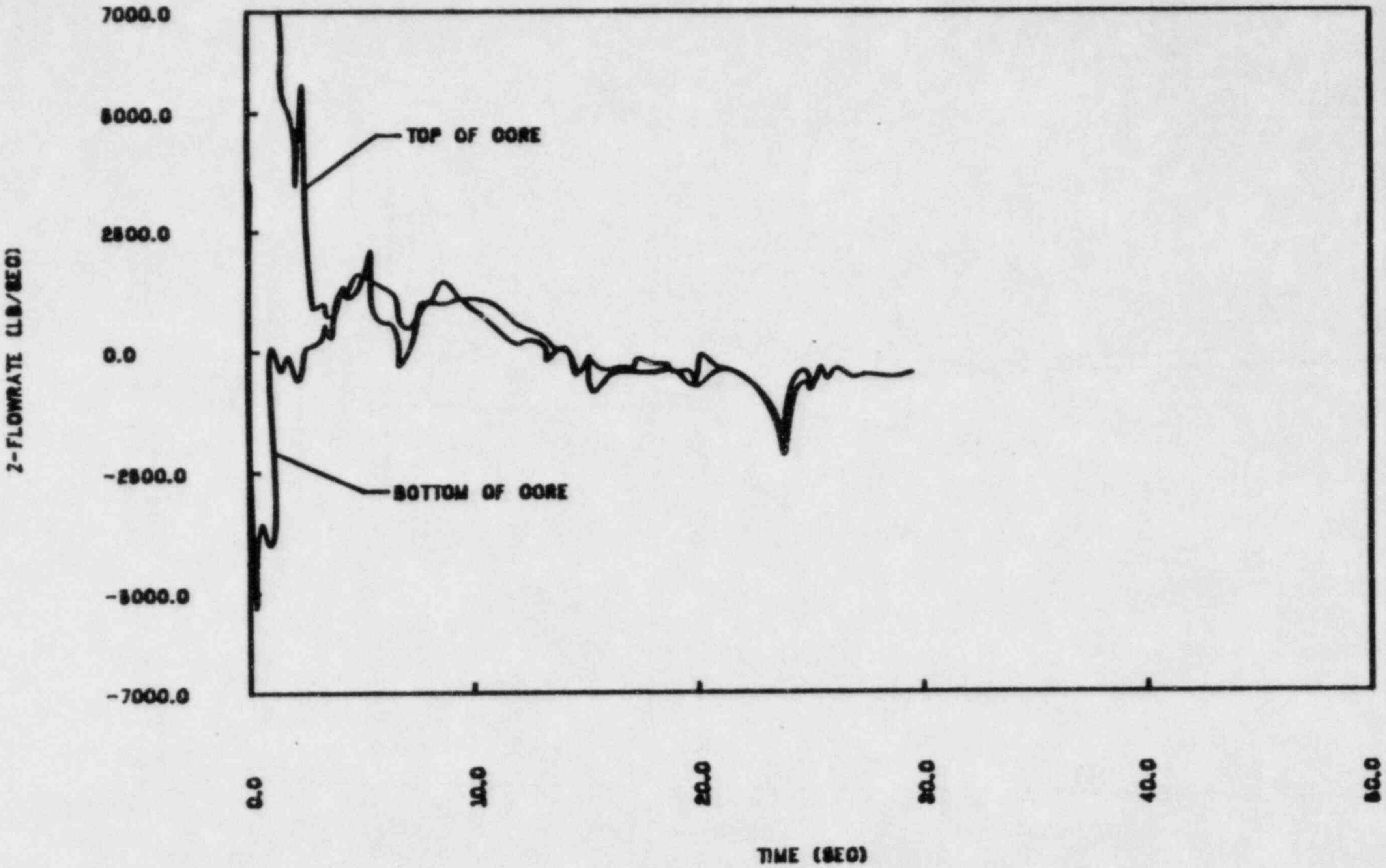


FIGURE 15.6-49
CORE FLOW DURING BLOWDOWN (INLET AND OUTLET) (Active Loop Break CD= 0.4
N-1 LOOP OPERATION

INTERACTIVE GRAPHICS

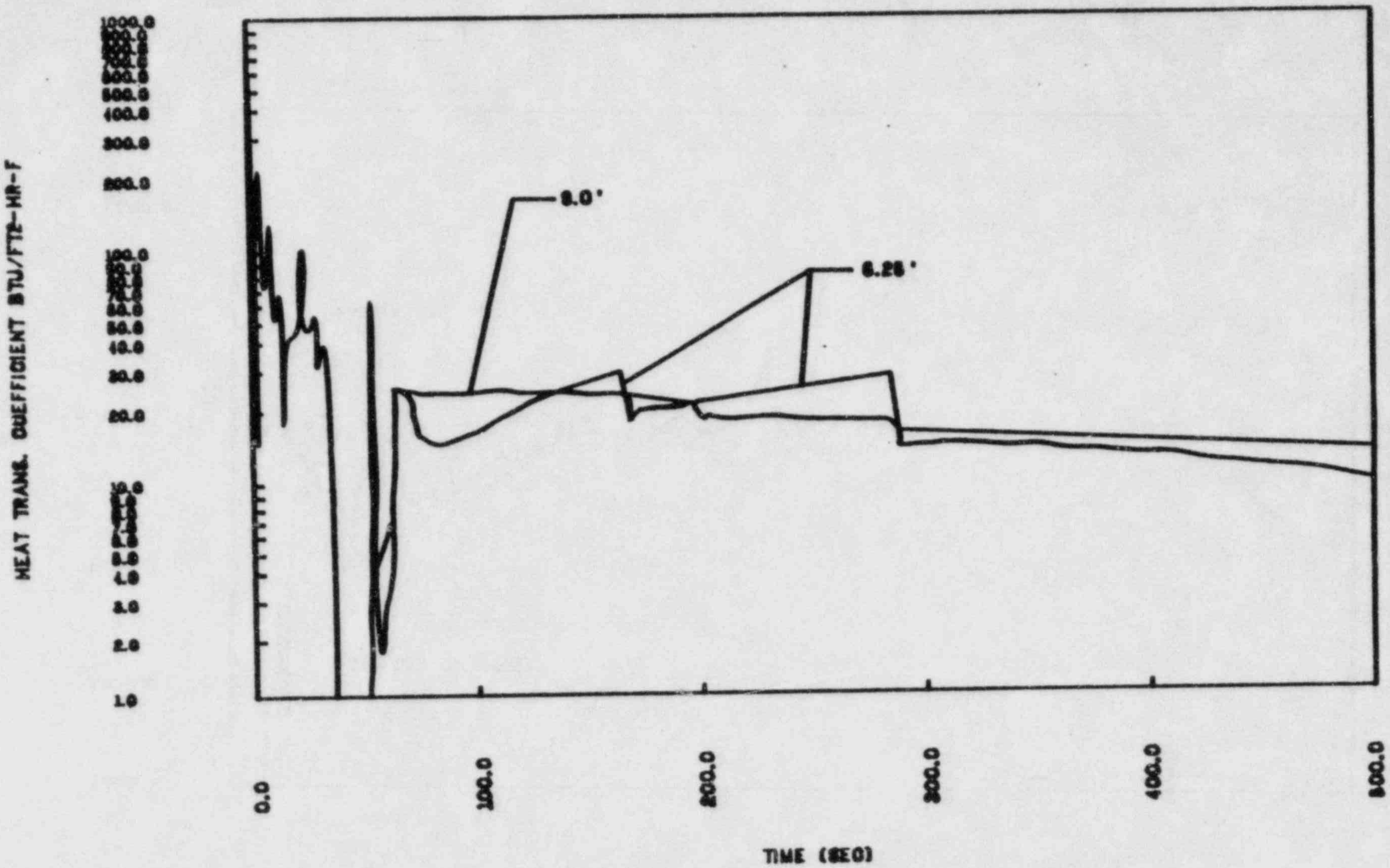


FIGURE 15.6-50
CORE HEAT TRANSFER COEFFICIENTS (Active Loop Break CD=0.4)
N-1 LOOP OPERATION

INTERACTIVE GRAPHICS

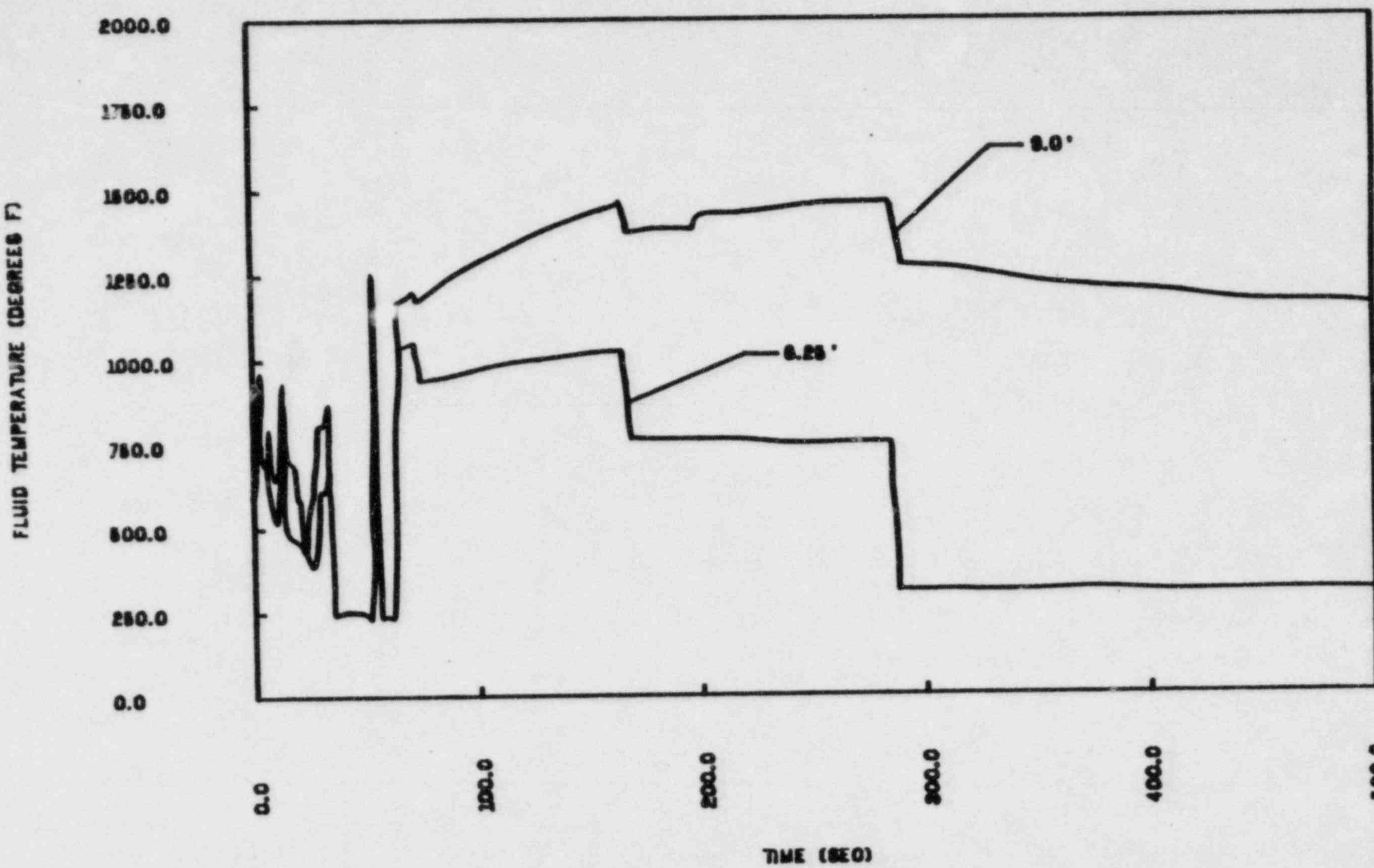


FIGURE 15.6-51
HOT SPOT FLUID TEMPERATURE (Active Loop Break CD=0.7)
N-1 LOOP OPERATION

INTERACTIVE GRAPHICS
D CHA TE SO 0751AXX.XXX.01.101.0829

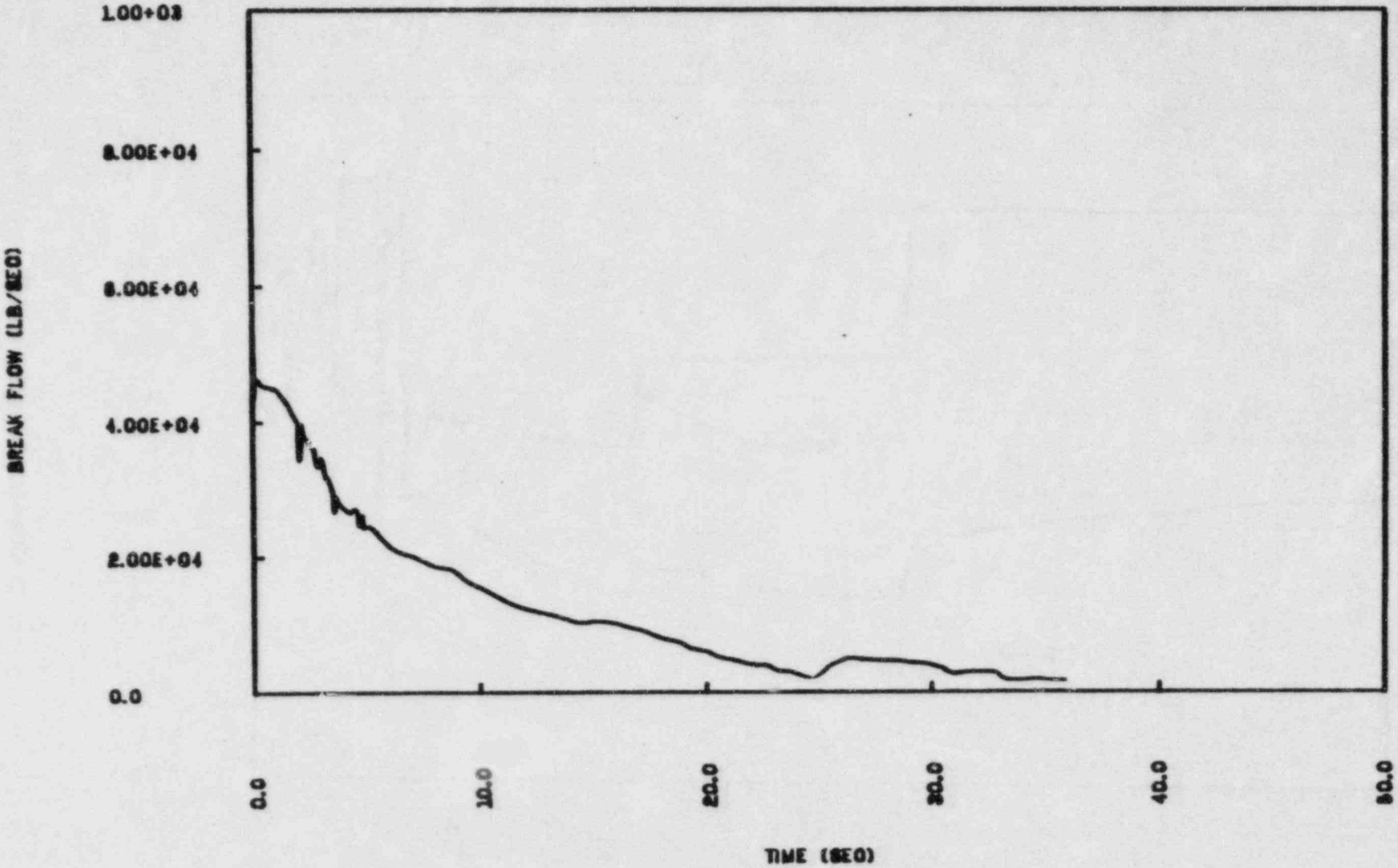
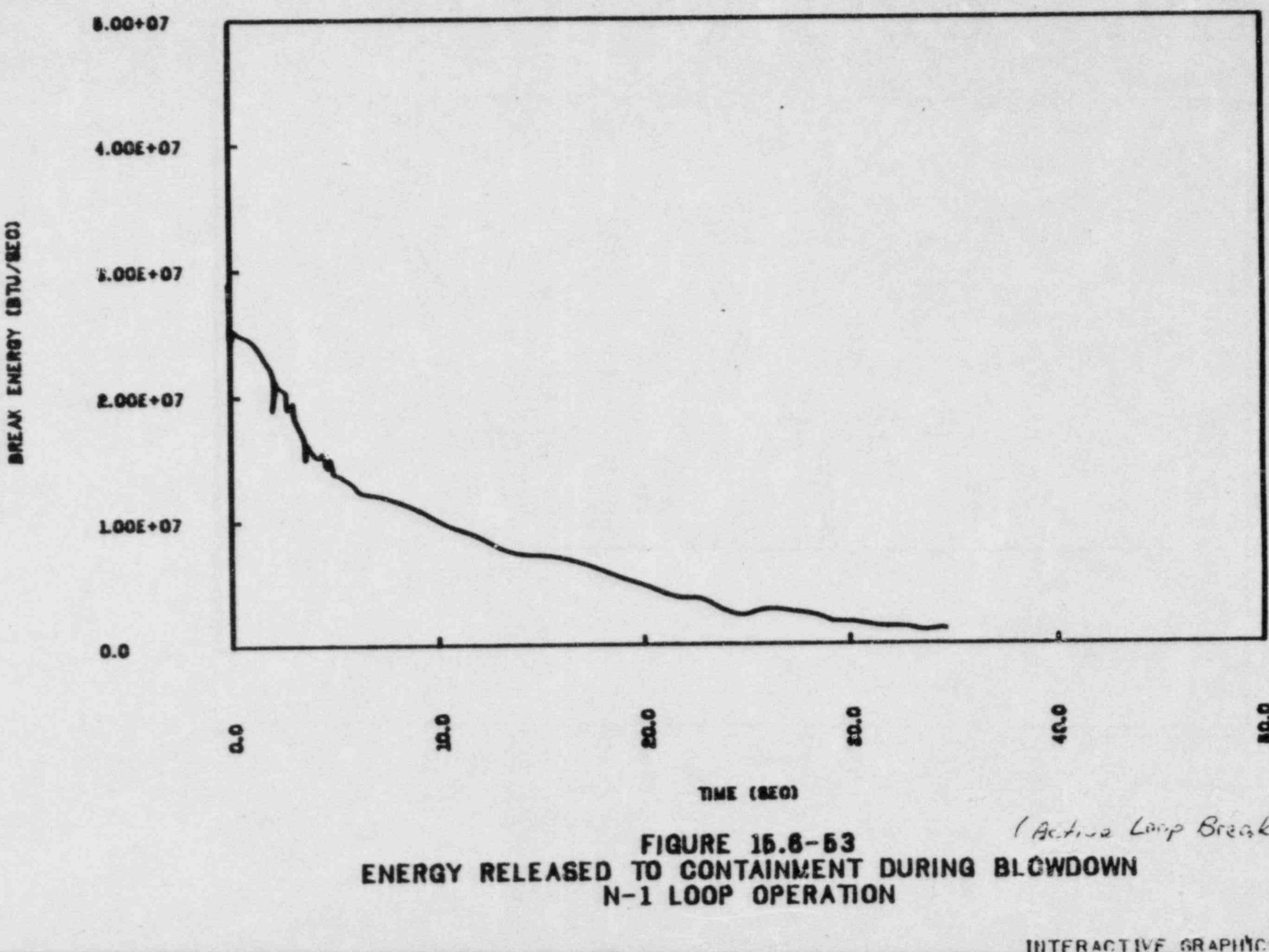


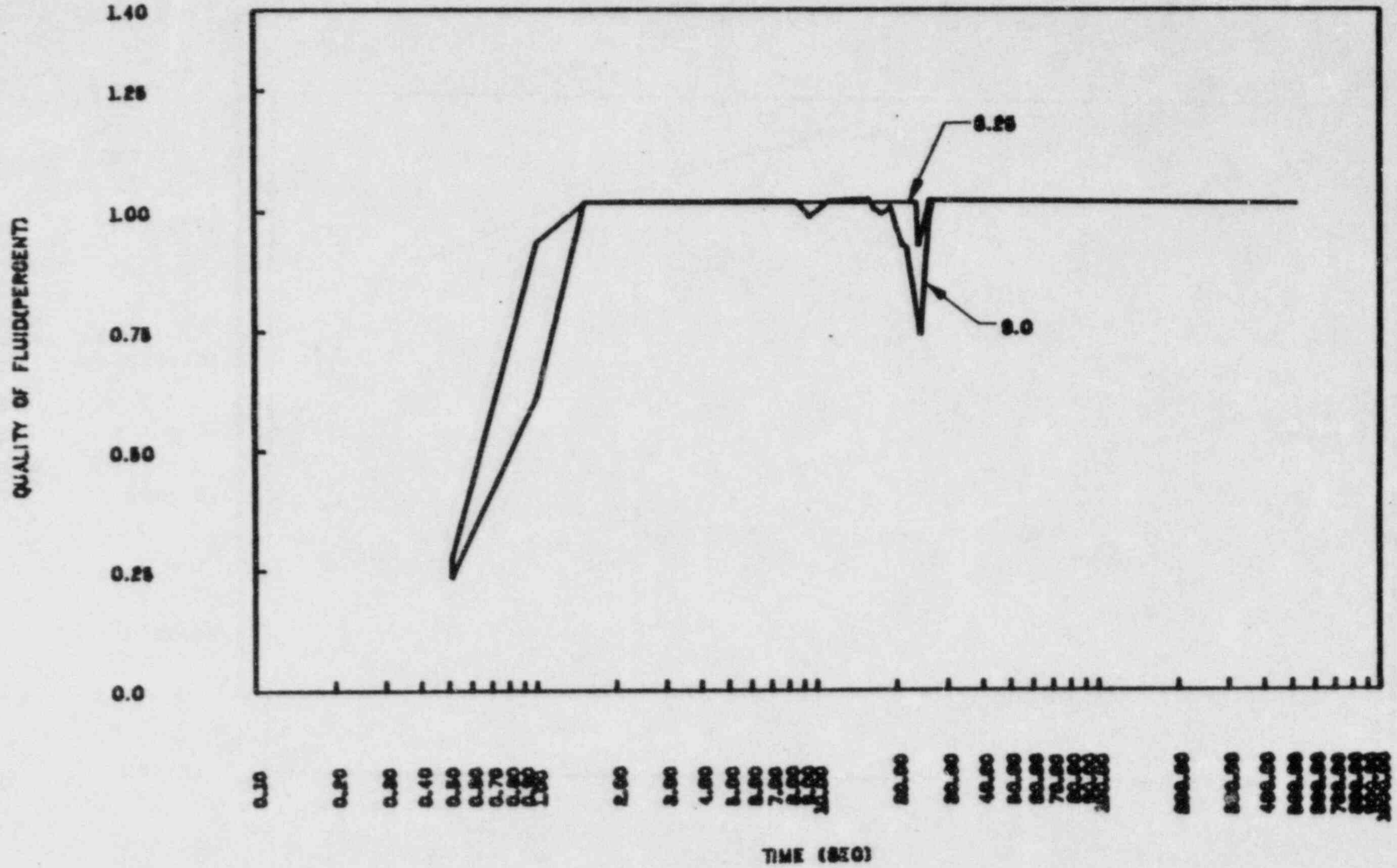
FIGURE 15.6-52
MASS RELEASED TO CONTAINMENT DURING BLOWDOWN (Active Loop Break CD=0.4
N-1 LOOP OPERATION



(Active Loop Break CD=0.4

FIGURE 15.6-53
ENERGY RELEASED TO CONTAINMENT DURING BLOWDOWN
N-1 LOOP OPERATION

INTERACTIVE GRAPHICS



(Active Loop Break CD=0.5

FIGURE 15.6-54
FLUID QUALITY IN THE HOT ASSEMBLY DURING BLOWDOWN
N-1 LOOP OPERATION

INTERACTIVE GRAPHICS

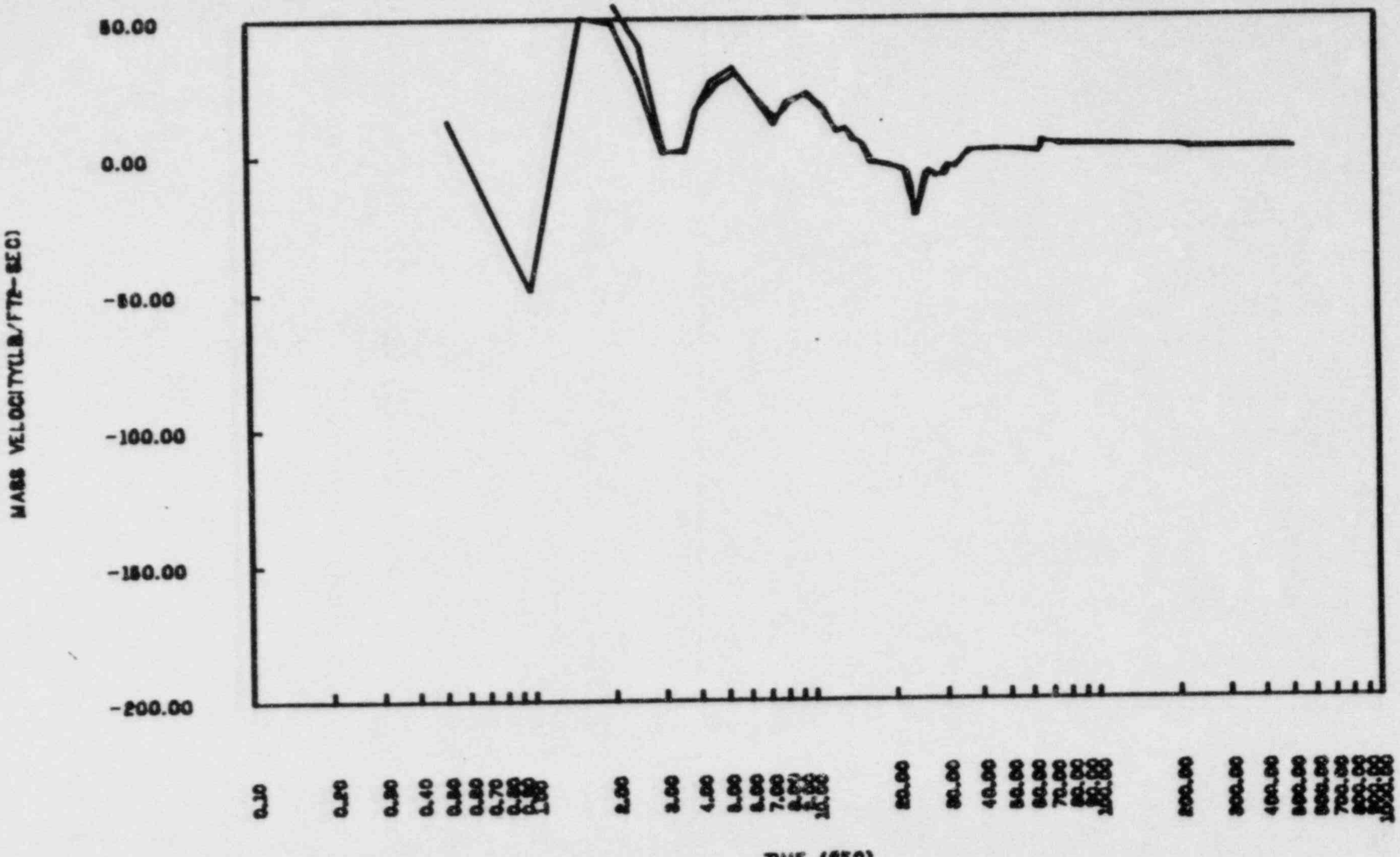
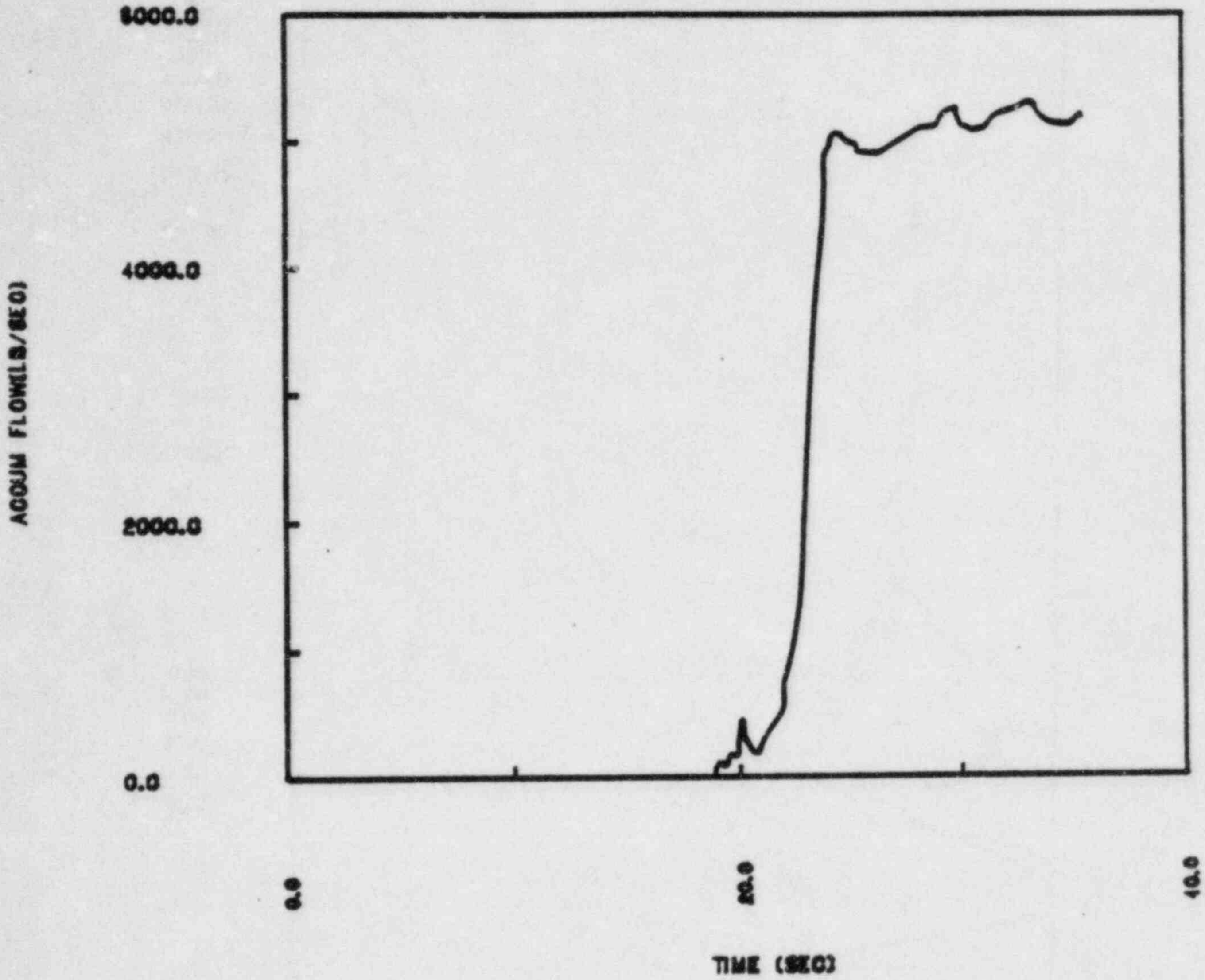
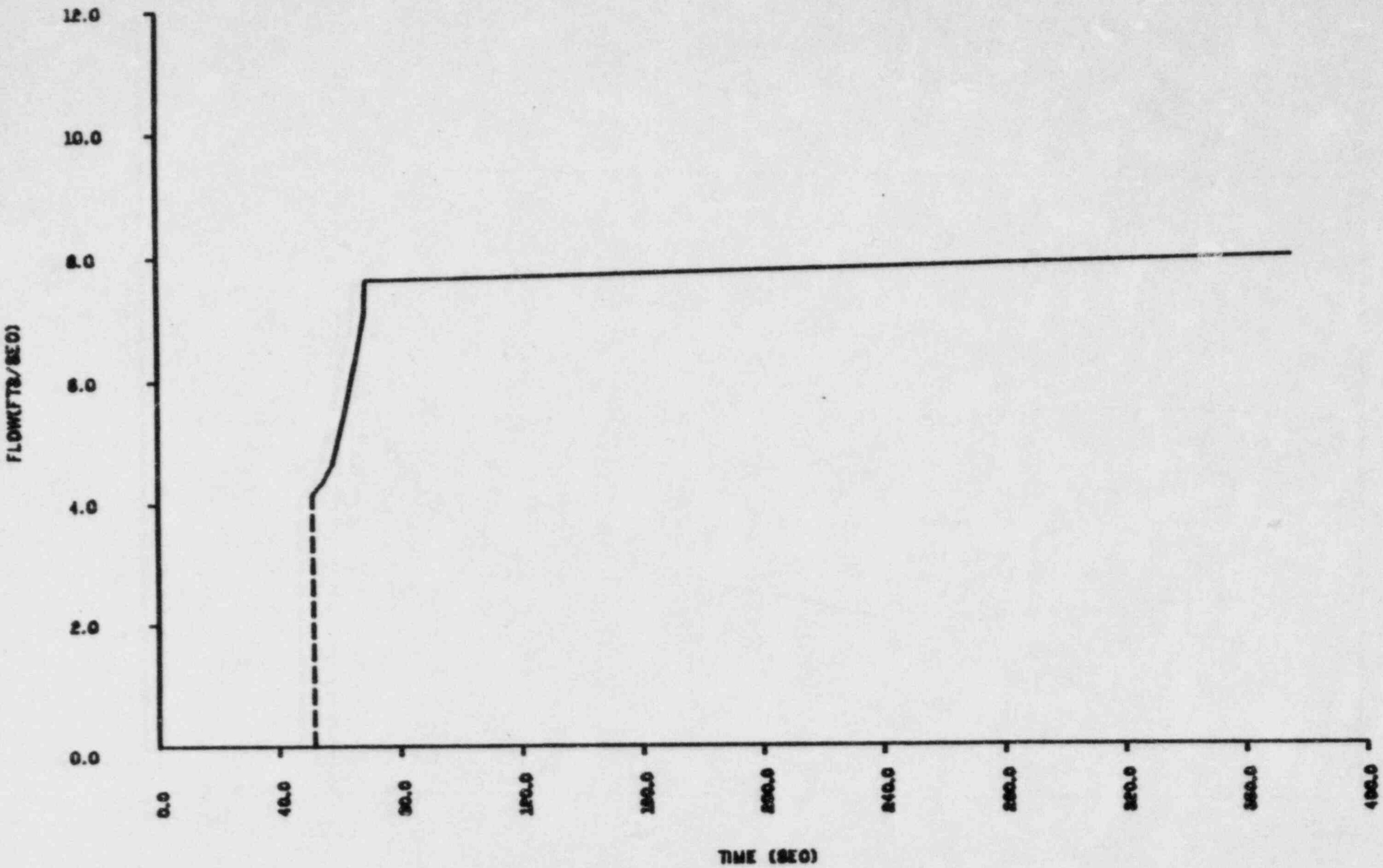


FIGURE 15.6-55
MASS VELOCITY DURING BLOWDOWN (Active Loop Break CD=0.4
N-1 LOOP OPERATION

INTERACTIVE GRAPHICS



DECLG (ACTIVE LOOP BREAK CD=0.4)
FIGURE 15.6-56
ACCUMULATOR WATER FLOW RATE DURING BLOWDOWN
N-1 LOOP OPERATION



PUMPED ECCS FLOW (REFLOOD)
DECLG (ACTIVE LOOP BREAK CD=0.4)
FIGURE 15.6-57
PUMPED SAFETY INJECTION WATER FLOW RATE DURING REFLOOD
N-1 LOOP OPERATION

INTERACTIVE GRAPHICS

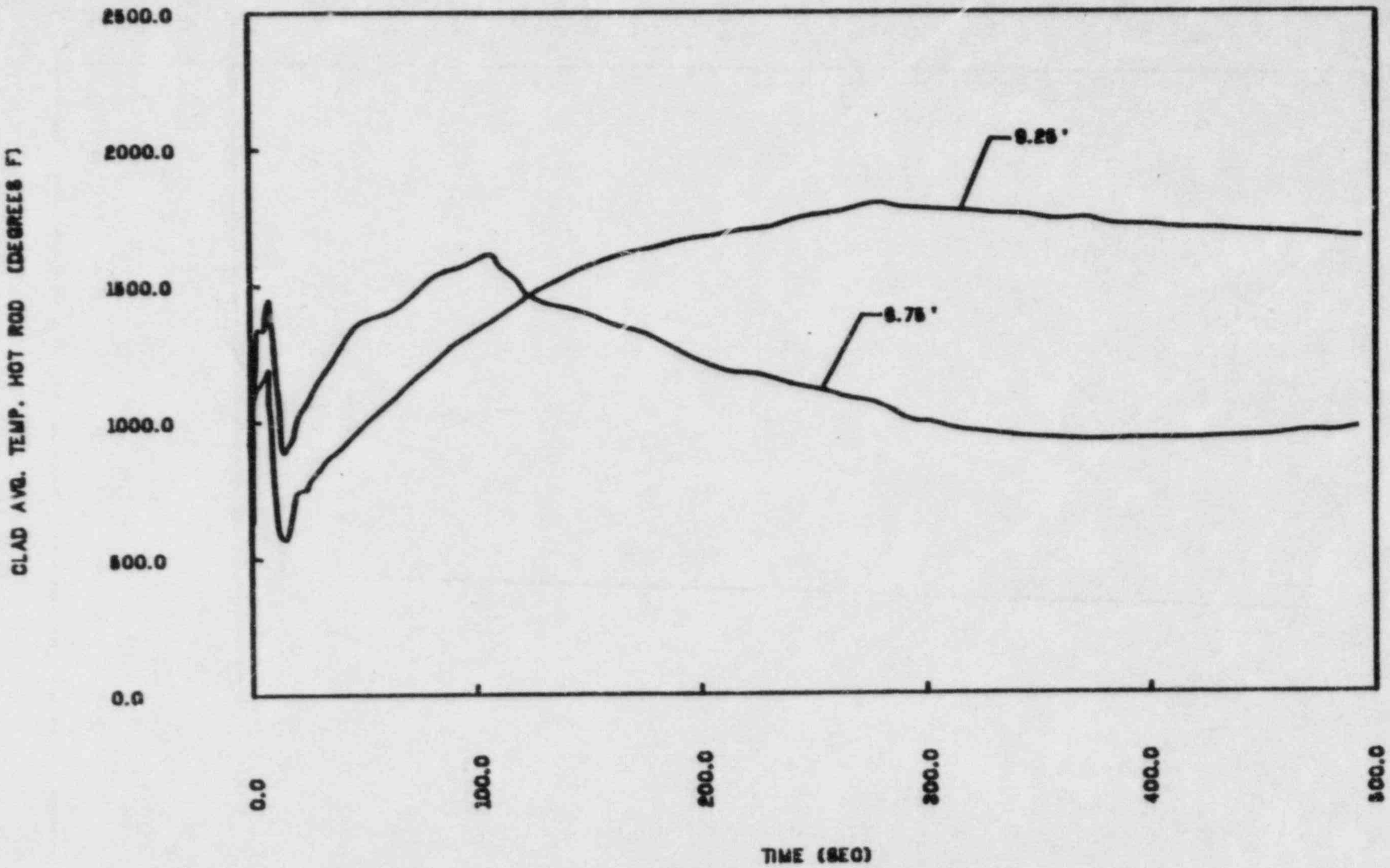


FIGURE 15.6-58
HOT SPOT CLAD TEMPERATURE (Active Loop Break $C_D = 0.6$)
N-1 LOOP OPERATION

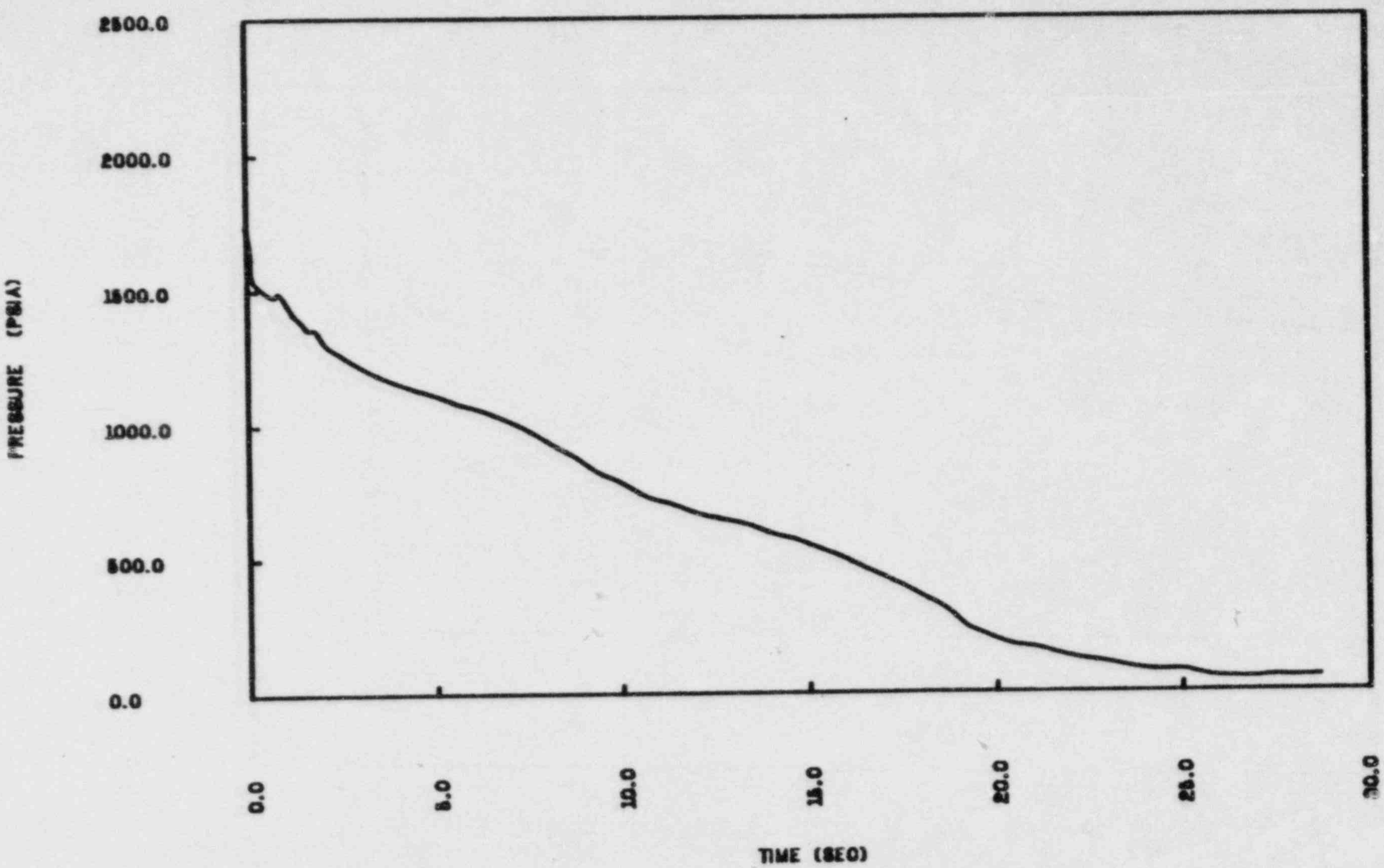
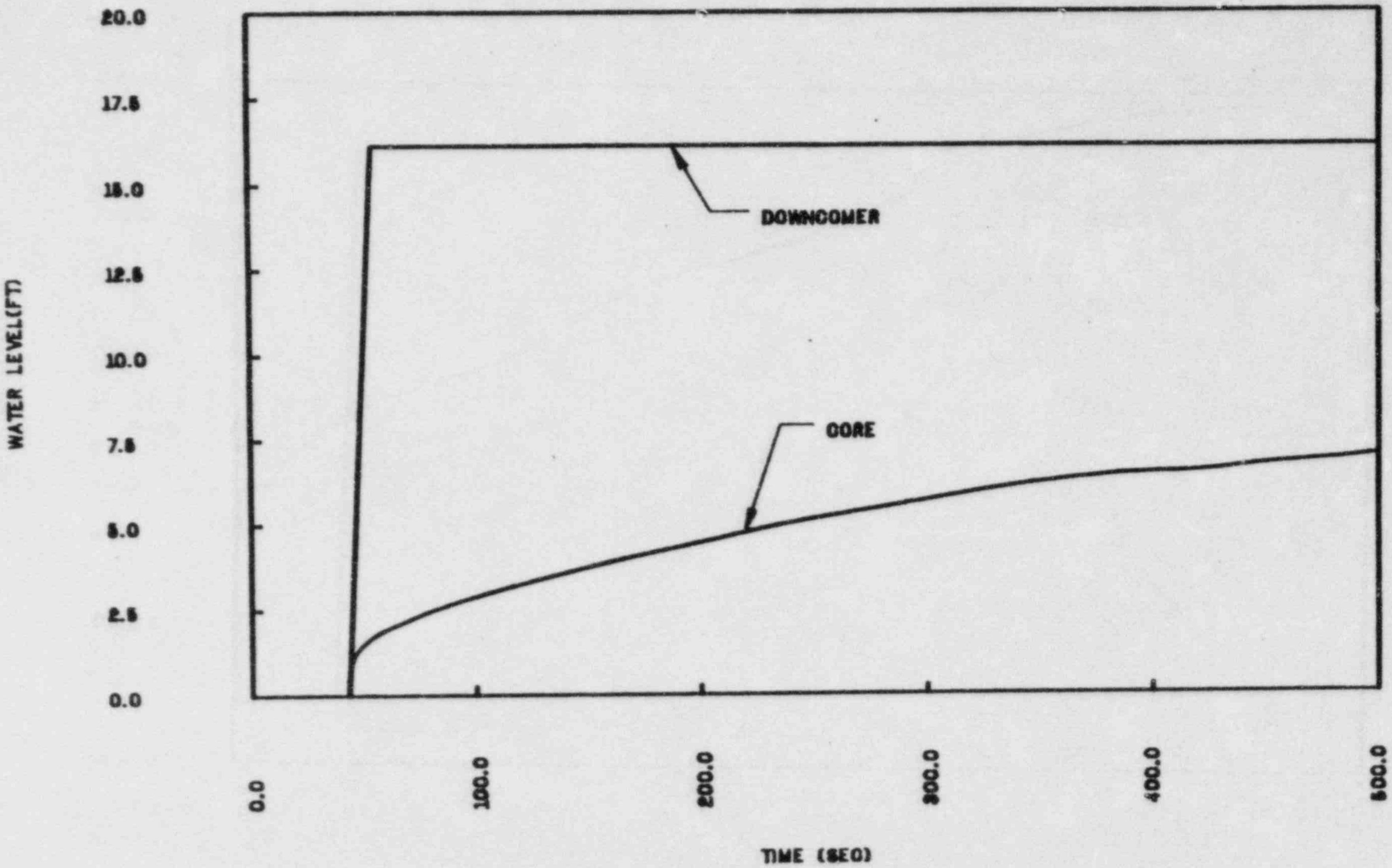


FIGURE 15.6-59
COOLANT PRESSURE IN THE REACTOR CORE (Active Loop Break CD=0
N-1 LOOP OPERATION

INTERACTIVE GRAPHICS



(Active Loop Break CD=0.6)

FIGURE 15.6-60
WATER LEVEL IN THE CORE AND DOWNCOMER DURING REFLOOD
N-1 LOOP OPERATION

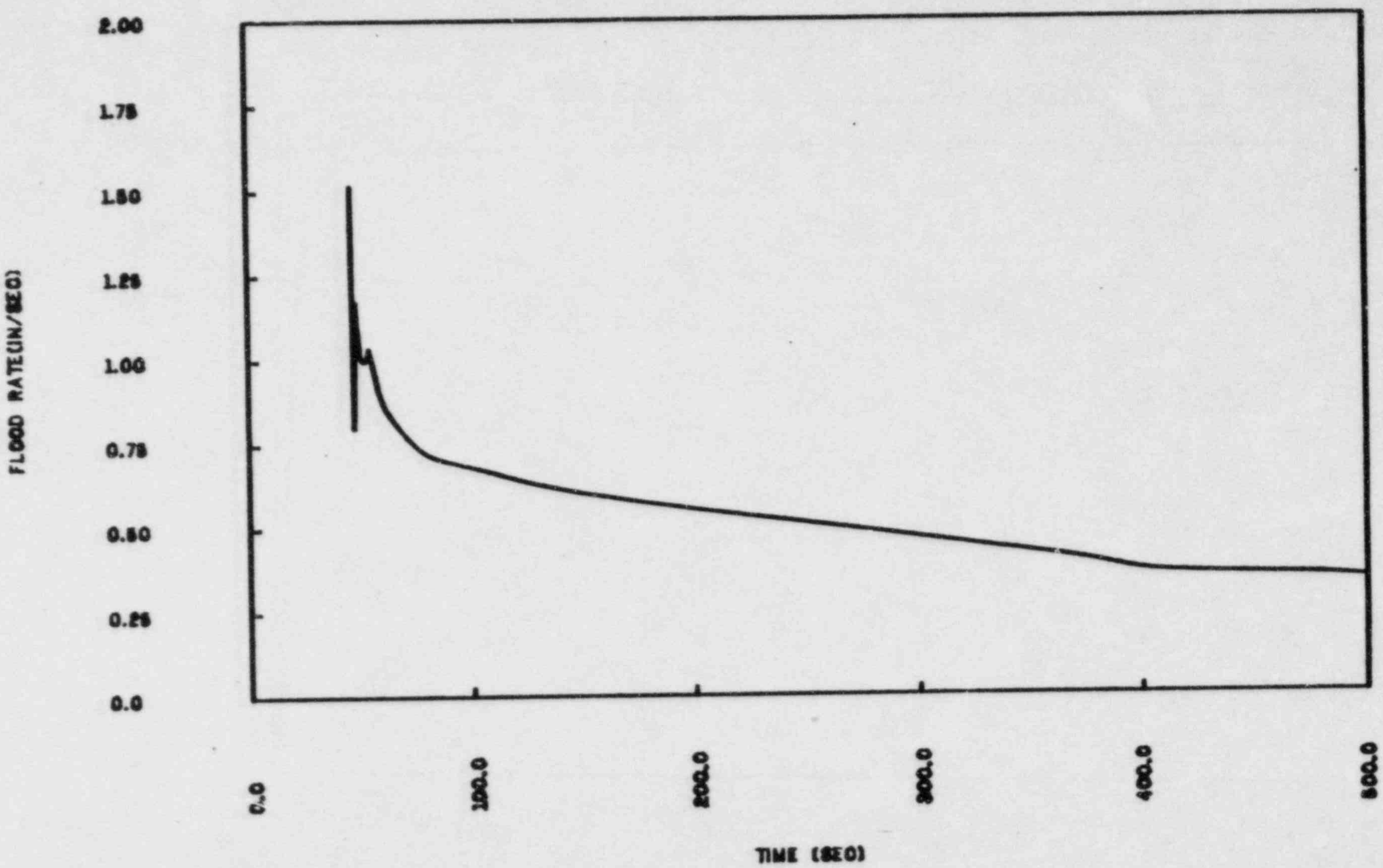


FIGURE 15.6-61
CORE REFLOODING RATE (Active Loop Break CD=0.6)
N-1 LOOP OPERATION

INTERACTIVE GRAPHICS

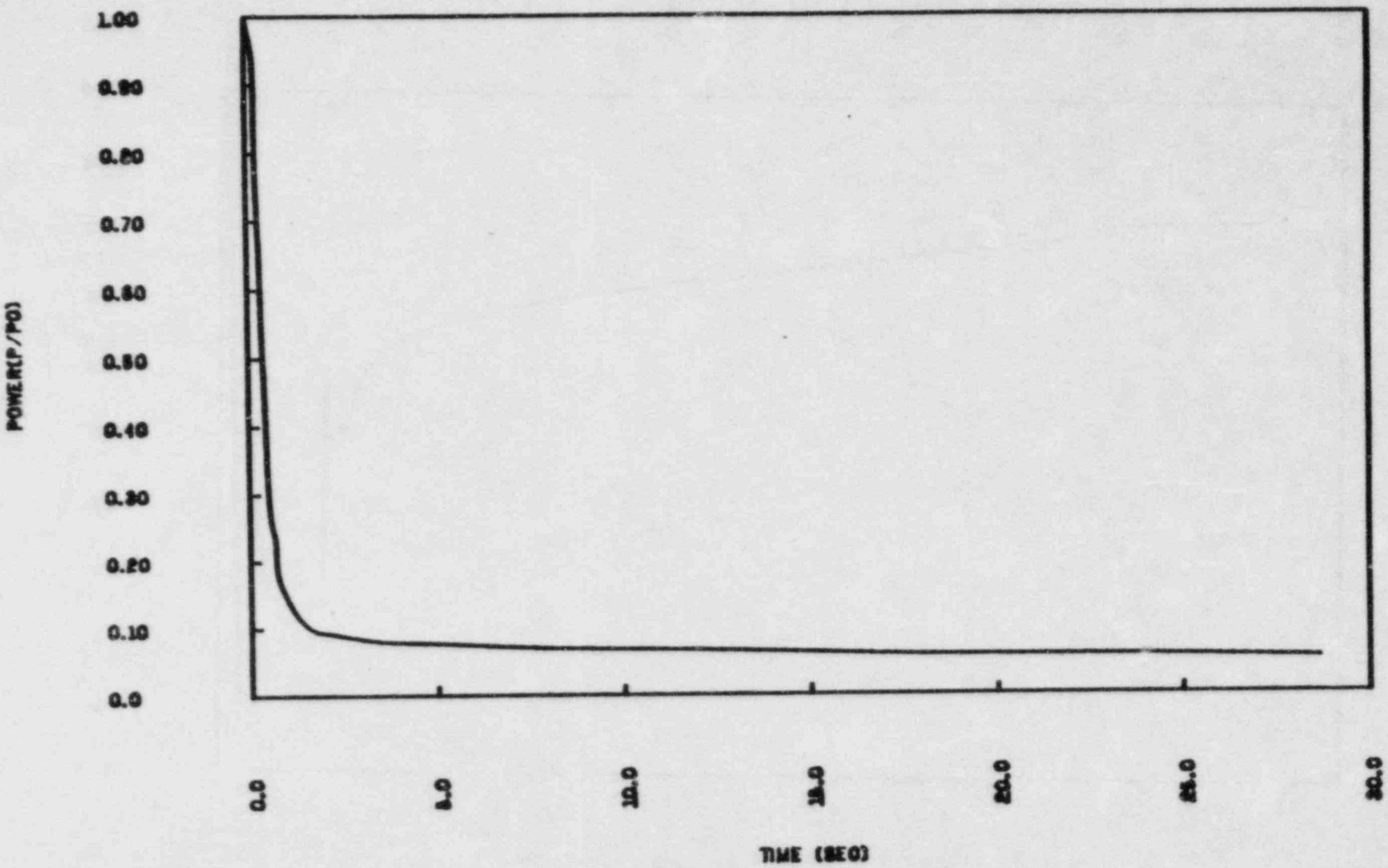


FIGURE 15.6-82
THERMAL POWER DURING BLOWDOWN (Active Loop Break CD=0.6
N-1 LOOP OPERATION

INTERACTIVE GRAPHICS

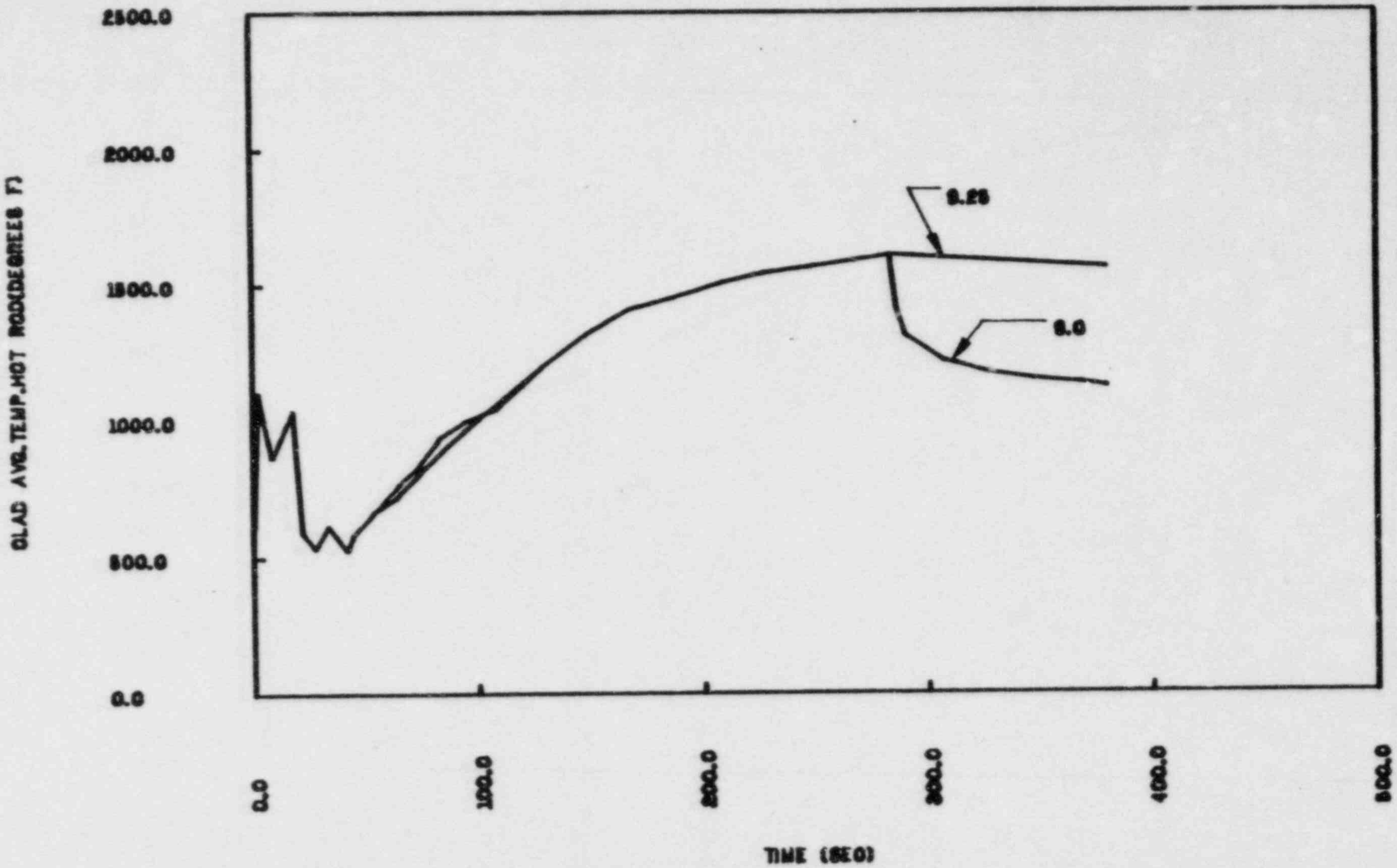


FIGURE 15.6-63
HOT SPOT CLAD TEMPERATURE (Inactive Loop Break CD= 0.7)
N-1 LOOP OPERATION

INTERACTIVE GRAPHICS

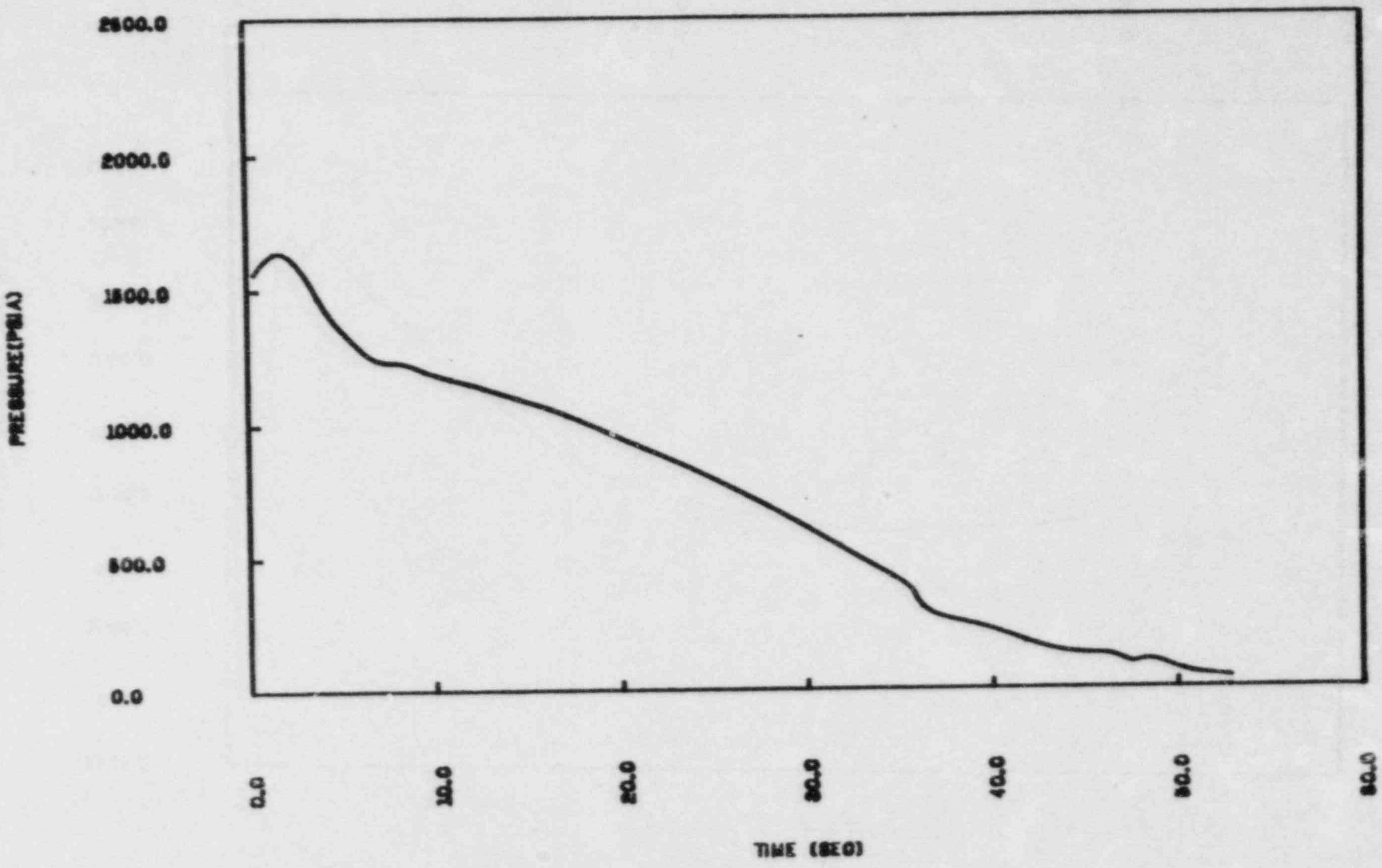
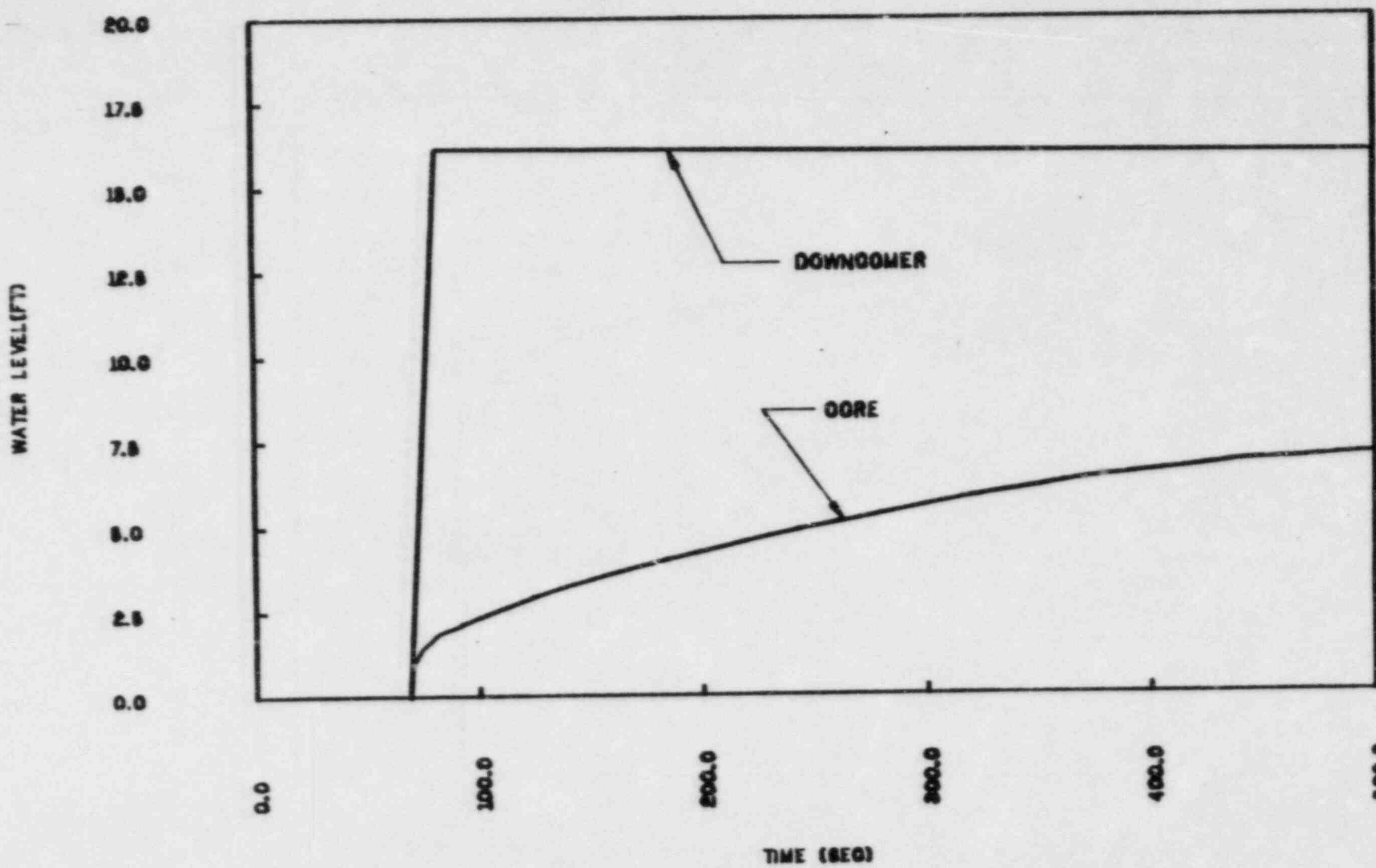


FIGURE 15.6-64
COOLANT PRESSURE IN THE REACTOR CORE (Inactive Loop Break CD=0.
N-1 LOOP OPERATION



(Inactive Loop Break CD=0.

FIGURE 15.6-65
WATER LEVEL IN THE CORE AND DOWNCOMER DURING REFLOOD
N-1 LOOP OPERATION

INTERACTIVE GRAPHICS

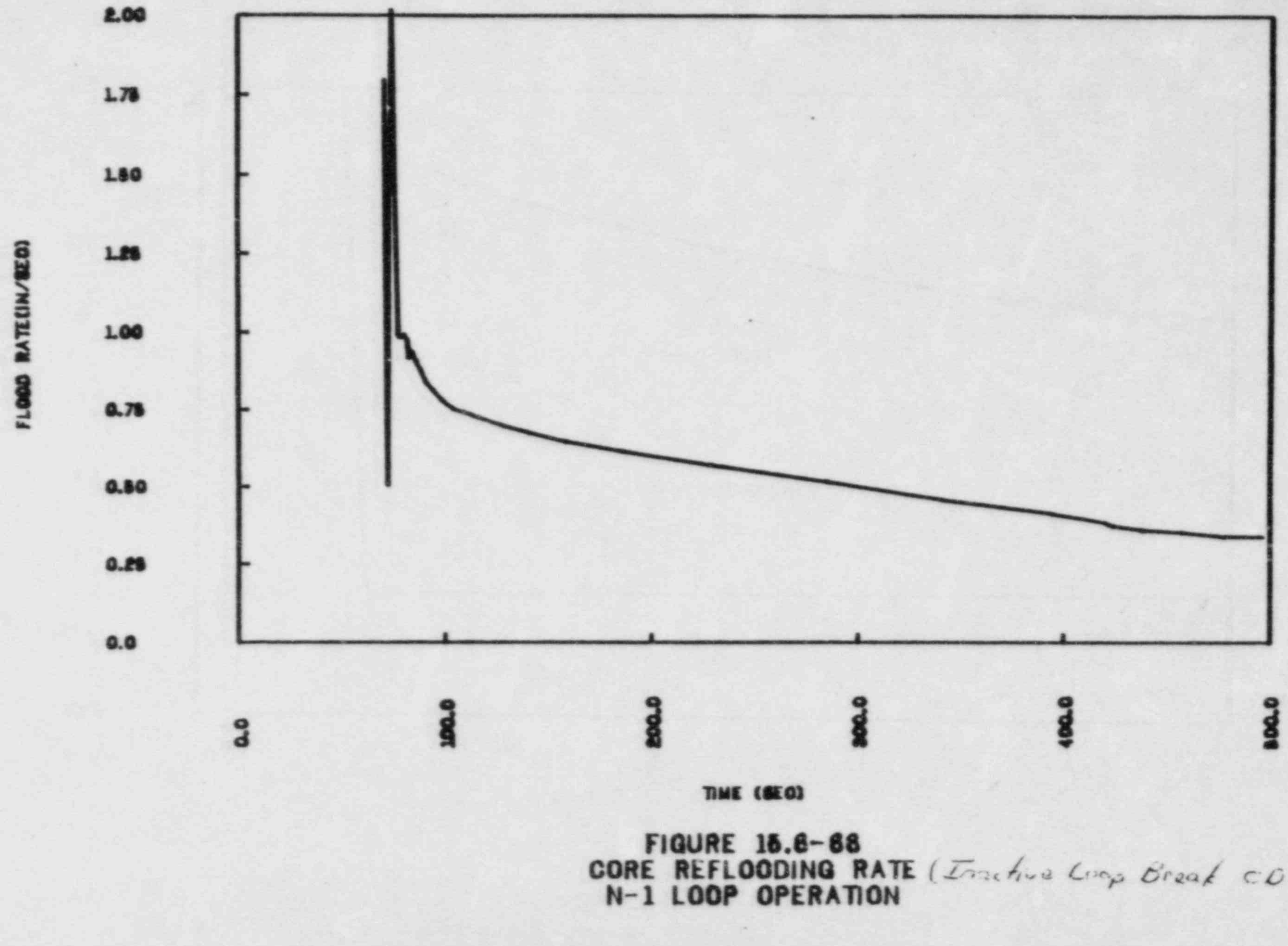


FIGURE 15.6-68
CORE REFLOODING RATE (Inactive Loop Break CD=0.4).
N-1 LOOP OPERATION

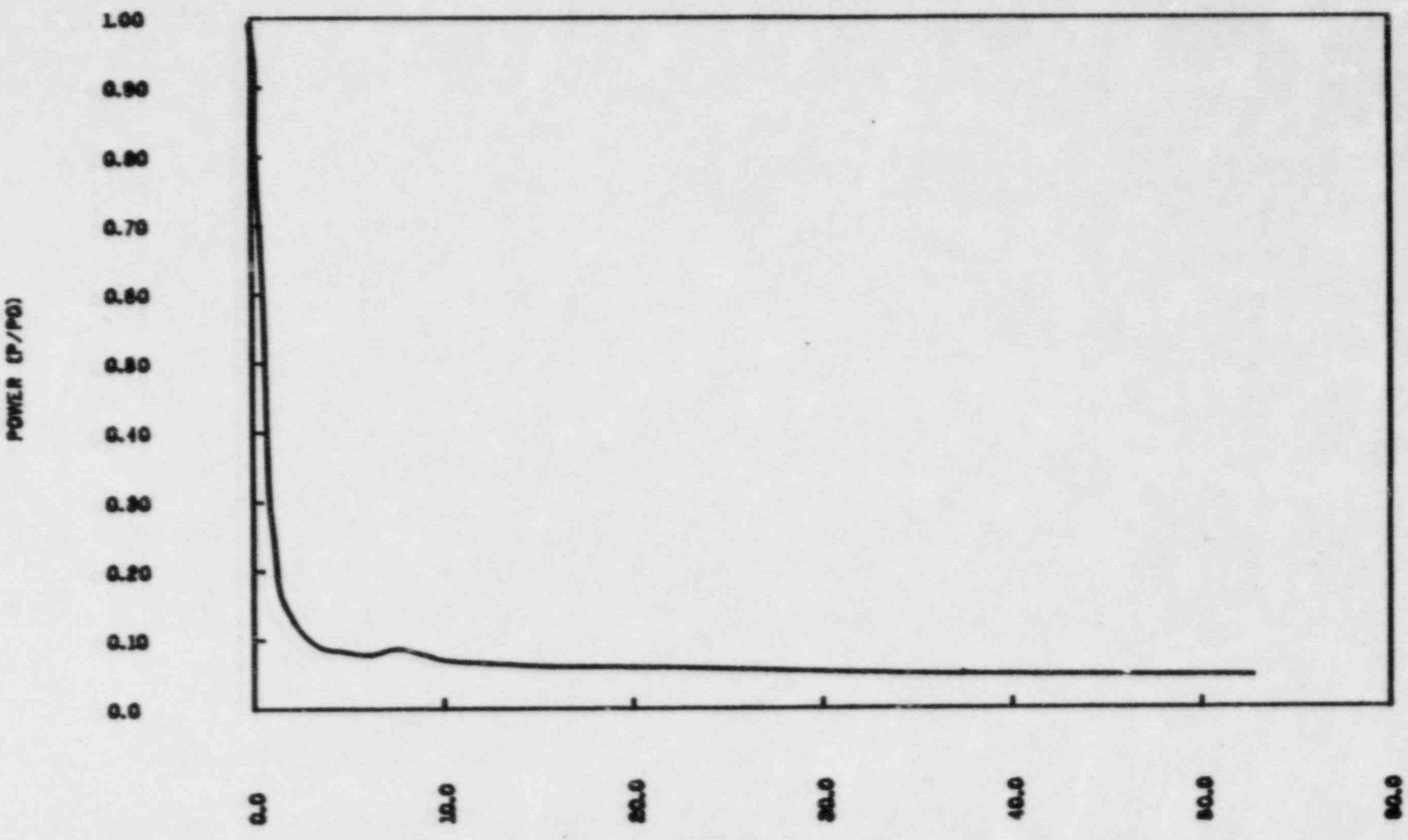


FIGURE 15.6-67
THERMAL POWER DURING BLOWDOWN (Inactive Loop Break CD=0.
N-1 LOOP OPERATION

INTERACTIVE GRAPHICS

(ENCLOSURE 2)

MILLSTONE UNIT NO. 3

DESIGN PACKAGE FOR INCORPORATION OF N-1 OPERATION FOR THE SOLID STATE PROTECTION SYSTEM

TABLE OF CONTENTS

1.0 Introduction

This package has been developed in order to identify a design for Millstone Unit 3 to convert the Solid State Protection System from a 4-loop to a 3-loop system during N-1 operation. The intent is to present a design for review by the Nuclear Regulatory Commission to facilitate regulatory review for Millstone Unit 3.

Included in this package are design basis and functional requirements for the equipment design. In addition a licensing safety evaluation of the design has been performed and is included in this design package.

2.0 Design Basis

2.1 Purpose

The purpose of this section is to provide the design basis for the modification to the Solid State Protection System such that the system can be configured for 3-loop operation.

2.2 Introduction

The Solid State Protection System initiates protective action based on measurements of primary and secondary coolant system parameters, as well as other plant conditions.

When the plant is configured with an inactive loop isolated, certain parameters within the protection system will be in a trip state. By utilizing a keylock transfer switch for the inactive loop, the system in affect will be transferred from a 4-loop to a 3-loop protection system. Each loop within the protection system will have its own keylock transfer switch.

2.3 Applicability

Millstone Unit 3 Solid State Protection System

2.4 Function

The keylock switches will provide a means of bypassing the inactive loop within the protection system. The switches shall operate without degrading the performance of the protection system or the performance of any other protection functions.

Each bypass switch will interface on a channel and loop level at the input to the universal logic cards. When the switch is in bypass, the logic that makes up the logic trip will inhibit the inactive loop.

2.5 Testing

Procedures shall be provided to demonstrate that the bypass switches are independently operable and capable of providing high reliability so as not to degrade plant safety.

2.6 Design Criteria

The incorporation of the bypass switches into the protection system will provide a means of transferring the system from a 4-loop to a 3-loop protection system configuration. The bypass switches shall be Class 1E equipment. The quality assurance program shall ensure that the function of the bypass switches is met. The general requirements of IEEE Standard 279-1971 and other applicable standards and criteria shall be met by the addition of the bypass switches. System reliability and plant availability shall be maximized through the use of reliable components and conservative design practices.

2.7 Licensing Position

The bypass switches shall meet all the requirements normally associated with protection systems.

3.0 Functional Requirements

3.1 System Description

The Solid State Protection System automatically trips the plant whenever plant conditions monitored by nuclear and/or process instrumentation reach specified limits. When the protection system logic receives the required number of trip signals from the nuclear or process instruments, it sends a signal via Logic Train A to trip reactor trip breaker A and bypass breaker B. It simultaneously sends an independent signal via Logic Train B to trip reactor trip breaker A and bypass break[illegible].

The bistable and field contact inputs associated with the solid state logic normally maintains the input relays in an energized (untripped) condition. When a bistable or contact input becomes tripped, the corresponding input relay will de-energize, applying a ground as a specific logic input. This ground is the logic ground, which is interruptable so that testing can be performed without external inputs being applied.

When a loop has been put in and out of service condition, the corresponding input relays will de-energize and put the protective system in a partial trip condition via the logic grounds.

The purpose of this functional requirement is to use individual keylock bypass switches as a means of interrupting the partial trip condition. When the affected loop is in bypass, the protection system would be changed from a four loop protection system configuration to a three loop protection system configuration. While serving as a bypass condition, the equipment must meet Class 1E requirements for the Solid State Protection System.

To insure a reliable system, high quality design, components, manufacturing quality control and testing will be used. In addition to redundant channels and trains, the design approach provides a reactor trip system which monitors numerous system variables.

3.2 Applicable Criteria and Standards

The following criteria apply to is system.

3.2.1 General Design Criteria (GDC), Appendix A to 10CFR50

GDC1	Quality Standards and Records
GDC2	Design Bases for Protection Against Natural Phenomena
GDC3	Fire Protection
GDC4	Environmental and Missile Design Bases
GDC17	Electric Power Systems
GDC18	Inspection and Testing of Electric Power Systems
GDC20	Protection System Functions
GDC21	Protection System Reliability and Testability
GDC22	Protection System Independence
GDC24	Separation of Protection and Control System

GDC25 Protection System Requirements for Reactivity Control Malfunctions

GDC29 Protection Against Anticipated Operational Occurences

3.2.2 Institute of Electrical and Electronics Engineers (IEEE) Standards

IEEE Std. 279-1971 Criteria for Protection Systems for Nuclear Power Generating Stations.

IEEE Std. 323-1974 IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations.

IEEE Std. 344-1975 IEEE Recommended Practices for Seismic Qualification of Class 1E Equipment for Nuclear Power Stations.

IEEE Std. 384-1974 Criteria for Separation of Class 1E Equipment and Circuits.

3.2.3 Applicable Regulatory Guides

R. G. 1.53 Application of the Single Failure Criterion to Nuclear Power Plant Protection Systems.

3.3 System Diagrams

Refer to Schematic Diagrams provided in Attachment 1.

3.4 Environmental Requirements

The following environmental qualification conditions apply to the bypass switches. It must be demonstrated that the bypass switches can function during and after the following conditions and yet not cause a spurious trip to occur.

3.4.1 Plant Design Operating Conditions (Attachment 2)

Table 1 supplies plant condition parameters for the instrument rack room. The bypass switches will be located in zone CB-03 which includes the instrument rack room. The operating condition parameters include the minimum, normal and maximum expected values for temperature, relative humidity, pressure, and radiation. Zone CB-03 is not subjected to a chemical environment during normal and abnormal operating conditions.

3.4.2 Plant Operating Conditions for Design Basis Events.

The bypass switches are located in the instrument rack room and must be qualified for the environment it will be exposed to throughout the qualified equipment life. The equipment must be qualified to operate during and after seismic events in accordance with IEEE-344-1975.

3.5 Indicators, Status Lights and Controls

Bypass indication shall be provided as per 3.7 below.

3.6 Recorders

Trending information is not required.

3.7 Alarms and Annunciators

As needed to verify that the Protection System is in N-1 operation (1 annunciator per train). First out annunciation for reactor trip on improper N-1 alignment shall also be provided.

3.8 Interlocks and Permissives

For improper operation of the bypass switches an automatic reactor trip signal is generated for the following conditions:

- Any 2 loops on one train in bypass.
  2/4 reactor trip logic

- Any 2 non-identical loops on opposite train in bypass.
  2/4 reactor trip logic

3.9 Trips and Trip Logic

See figures in Attachment 1.

3.10 Accuracy

Switches do not affect setpoint accuracy.

3.11 Range

Range of process variables is not affected by this modification.

3.12 Time Response

Influence of switch contacts on time response is negligible.

3.13 Overload and Recovery Characteristics

Not applicable

3.14 Noise Levels

Not applicable

3.15 Controller Transfer Functions

Not applicable

3.16 Setpoints

As needed to satisfy the requirements in section 3.1. The transition from 4 loop to 3 loop operation shall be accomplished under strict administrative control where first the affected setpoints are changed to more restrictive valves, after which the loop to be isolated may have its affected trip channels bypassed. Loop stop valves can then be closed. Transition from 3 loop to 4 loop operation will only occur in mode 5 (cold shutdown) or Mode 6 (Refueling Operation).

3.17 Requirements for Test and Calibration

Procedures shall be provided to independently demonstrate that the bypass switches are operable at every refueling cycle. Also, two monthly surveillance procedures (4-loop & 3 loop) will be provided so as not to degrade the system below an acceptable level for the performance of safety related systems.

3.18 Power Supply

Not applicable

4.0 Design Description for Bypass Switches Within the Solid State Protective System

4.1 Introduction

This design is for Millstone Unit 3 to initiate certain functions within the protection system to enable the protection system to transfer from a four loop to a three loop system without de-grading plant safety.

When the protection and control systems are configured for three loop operation, the following will still be maintained:

- o Complete on-line testing
- o Maintain the plant within its safety analysis limits
- o Minimize system interaction
- o Maintain adherence to single failure criterion
- o Simplify operator actions when operating in a three loop configuration

4.2 The following is a description of the necessary hardware modifications to the affected loop to transfer Millstone Unit 3 protection system from a four-loop to three loop system.

4.2.1 Steam Generator Level

Since the recommended long term condition of the isolated steam generator secondary side is wet layup, this level will correspond to a level high enough on the narrow range span to actuate a 2/4 HI-HI level signal to cause main feedwater isolation and turbine trip which will result in a reactor trip. The existing cold leg stop valve position interlock logic with the output from the 2/4 Lo-Lo S/G water level signals will be removed.

The 2/4 HI-HI and 2/4 LO-LO steam generator water level channels associated with the out of service loop will be put in bypass via the loop bypass switch. Each loop for HI-HI and LO-LO level signals will be capable of being bypassed via use of keylock switches. Only one loop will be put in bypass at any one time.

The remaining steam generator level channels will provide individual 2/4 HI-HI and 2/4 LO-LO steam generator water level trip on the active loops. With the isolated steam generator bypassed, adherence to the single failure criterion is still maintained.

4.2.2 Low RCP Speed

A 2/4 low RCP speed which is interlocked with P-7 provides a reactor trip.

Each RCP will have individual keylock switches. Only one loop will be put in bypass at any one time.

The three remaining RCP's will provide a 2/3 protection signal so adherence to the single failure criterion is still be maintained.

4.2.3 Primary Coolant Flow

The loss of flow reactor trip is interlocked with permissives based on reactor power such that above $\approx$ 10% power (P-7), a trip is initiated on loss of flow in any two looops and above $\approx$ 48% power a trip in initiated on loss of flow in any loop.

Each loop provides a 2/3 loss of flow. Each loop will have individual keylock switches. Only one loop will be put in bypass at any one time.

Each of the three remaining loops will provide individual 2/3 loss of flow trip. With the isolated loop bypassed, adherence to the single failure criteria is still maintained.

P-8 will be lowered, so protection is demonstrated for loss of flow from a three loop configuration.

4.2.4 Primary Coolant Temperature

Low average reactor coolant temperature provides for two protection actions. Feedwater isolation is initiated on LO T-avg and on LO-LO T-avg (P-12), steam dump is terminated and a permissive is provided to reopen the cooldown condenser dump valves.

The 2/4 LO T-avg and LO-LO T-avg channels associated with the out-of-service loop will be put in a bypass condition. Each loop will have individual keylock switches with only one loop in bypass at any one time.

The three remaining loops will provide 2/3 logic. With the isolated loop bypassed, adherence to the single failure criterion is still maintained.

The Overpower Delta T (OPDT) and Overtemperature Delta T (OTDT) do not require a bypass switch because the indicated temperatures on the isolated loop will be the lowest. The isolated portion of the RCS will eventually cool down below the bottom of the narrow range temperature span.

Therefore, the channels associated with the isolated loops will not generate a trip signal so a bypass of these channels is not required. There is also no need to put these bistables in the trip mode since the three remaining loops will provide a 2/3 trip signal to be generated. With the isolated loop functionally disabled, adherence to the single failure criteria is still maintained.

4.2.5 Steam Line Pressure

The existing cold leg loop stop valve positon interlock with the output from the 2/3 Loop Low steamline pressure ECCS actuation and MSSV closure signals will be removed. These signals will be put in bypass via the loop bypass switch. Each loop Low Steamline pressure bistable will be capable of being bypassed via use of keylock bypass switches.

The remaining steam generator low steam line pressure channels will provide individual 2/3 ECCS actuation and MSSV closure signals on the active loops. With the isolated steam generator bypassed, adherence to the single failure criterion is still maintained.

High (negative) steam line pressure rate signals do not require a bypass switch because during plant cooldown the isolated loop would already be depressurized.

4.2.6 Bypass Switches

Each protection train will be provided with four individual keylock switches. The switches will be located within the protection system and will be locked in the normal position. When going to 3 loop operation, the individual key for the affected loop will be inserted to go to bypass. An annunciation will be provided on the main control board confirming 3 loop operation for the protection system.

To prevent more than two loops being put in bypass which would degrade the protection system below an acceptable level, the following reactor trips are provided:

- o Any two loops on one train in bypass will provide an automatic reactor trip.
  (i.e., loop 1 and loop 2 in bypass on train A)

- o Any two non-identical loops on opposite trains in bypass will provide an automatic reactor trip.
  (i.e., loop 1 in bypass on train A and loop 2 in bypass on train B)

The bypass switch contacts will be wired in at the channel level of the protective system for RCP Low speed, Lo-Tavg and Lo-Lo Tavg signals. For steam generator Low Steamline pressure, Lo-Lo water level, Hi-Hi water level, the bypass switch contacts will be wired in at the Loop Level. The inhibit is located at the input to the universal logic boards that generate the trip logic.

4.3 Abnormal Indications

During three-loop operation, the following instrumentation on the main control board associated with the loop out-of-service will be in an abnormal status based on system parameters.

- 1 channel LOW-LOW TAVG in trip mode (annunciator lit)
- 1 channel LOW TAVG in trip mode (trip light lit)
- 1 LOOP LOW RCS FLOW in trip mode (trip light and annunciator lit)
- 1 RCP LOW SHAFT SPEED in trip mode (trip light and annunciator lit)
- 1 loop hot and cold leg LSV showing CLOSED position indication
- 1 LOOP LOW STEAMLINE PRESSURE-ECCS ACTUATION in trip status (trip lights and annunciator lit)
- 1 loop STEAMLINE ISOLATION VALVE POSITON CLOSED (annunciator lit)
- 1 loop HI-HI SG WATER LEVEL in trip status (trip lights and annunciator lit) when loop is placed in wet layup. The associated WR and NR SG water level at or near 100% span.
- 1 loop FCVs and FIVs indicated CLOSED position
- 1 loop AFW FCVs indicate CLOSED position
- 1 loop SG WATER LEVEL DEVIATION (annunciator lit)
- 1 loop SG blowdown FCVs indicate CLOSED position
- 1 loop Tavg measurement in BYPASS position OR remote load dispatch/load reference increase BLOCKED (status lamp lit)
- 1 loop RCP instrumentation indicating stopped RCP
- 1 loop WR and NR temperatures indicating low
- 1 loop bypass valve indicates OPEN position
- 1 loop drain valve indicate OPEN position until cold conditions are reached and seal injection flow is secured.

- 1 loop seal injection flow rate indicates higher than normal flow rate during loop cooldown and depressurization OR zero flow after cold conditions (less than 200° F)are reached.
- 3 loops indicated Delta T are higher than expected for 4 loop operation for a given core power as indicated by power range neutron flux.

IF AN ISOLATED LOOP IS ASSOCIATED WITH AN RCP THAT PROVIDES PRESSURIZER SPRAY, ASSOCIATED ABNORMAL INDICATIONS ARE:

- 1 pressurizer spray line indicating low spray temperature

4.4 Summary

4.4.1 The following actions are required with regard to the protection systems when operating with a loop out of service:

1. The High Neutron Flux (Power Range) signal would be lowered during three locp operation to just above the three loop maximum permissible power level.
2. Readjustment of overtemperature delta T and overpower delta T terms K1 and K4.
3. Decreasing the P-8 interlock setpoint for a reactor trip on low flow for 3-loop operation.
4. Decrease high limit on programmed T-avg. for Rod Control System.

4.4.2 The following systems were reviewed on the affect of system performance due to 3-loop operation which may or may not require adjustments to the operating mode.

1. Rod control and steam dump control functions use HI auctioneered T-avg, and HI auctioneered delta T signals which are loop condition dependent. High auctioneered functions would not require bypassing since isolated loop T-avg would be lowest. Isolated loop indicated delta T would be zero in steady state. It is not limiting during loop isolation because any large temperature differences would be at conditions below the bottom of the NR temperature span. Automatic remote dispatching uses Low auctioneered Tavg signals. These signals will be ignored by the auctioneering units if inhibited by input blocking or if the operator uses the bypass switch on the main control board.

2. The High Limit on programmed T-avg for the rod control system would be lowered to a value corresponding to the maximum permissible three loop power level. The Low Limit and Temperature Gain modules would be left at their four loop operational values.

3. The Pressurizer Programmed Water Level Temperature Gain and High Limit may be changed to reflect lower total coolant expansion in three loop operation with part of the RCS isolated. However, the basis for this change would be to limit makeup and letdown water handling and is not required for safety or reactor operability reasons.

4. Design limit load rejection transients have been examined. It has been determined that even with the potential for reduced pressurizer spray capacity that there is no increased possibility for challenging pressurizer PORV's during three loop operation at Millstone 3. Therefore, no change in pressurizer spray and relief setpoints are contemplated during three loop operation.

5. The rod insertion limits alarm setpoints may require a change in their respective gains and limits since a given indicated delta T corresponds to a lower power level. There is currently no safety analysis related need for changing these limits. However, fuel characteristics may dictate this change.

6. No change to Delta I and F Delta I functions (OTDT) are contemplated during three loop operation, however, fuel characteristics may dictate this change.

4.4.3 The affect of 3 Loop operation with respect to the protection system are the following:

1. The logic for the overtemperature and overpower delta T trips is reduced from 2/4 to 2/3.

   No bypass is required for the out-of-service loop.

2. The logic for LO T-avg. is reduced from 2/4 to 2/3.

   Manual bypass switch is required to bypass the out-of-service loop.

3. The logic for LO-LO T-avg. reduced from 2/4 to 2/3.

   Manual bypass switch is required to bypass the out-of-service loop.

4. The logic for HI-HI steam generator water level (2/4) will be placed in bypass via a Loop Level inhibit to preclude the potential for inadvertent turbine trip and feedwater isolation.

   Manual bypass switch is required to bypass the out-of-service loop.

5. The logic for LO-LO steam generator water level (2/4) will be revised from using an interlock with the cold leg loop stop valve position to using Loop Level inhibits. This logic will be placed in bypass to preclude the potential for inadvertent reactor trip and auxiliary feedwater actuation.

   Manual bypass switch is required to bypass the out-of-service loop.

6. The logic for Low RCP speed is reduced from 2/4 to 2/3.

   Manual bypass switch is required to bypass the out-of-service loop.

7. The logic for primary coolant flow is reduced from 2/3 on any of four Loops to 2/3 on any of three Loops.

   Manual bypass switch is required to bypass the out-of-service loop.

8. The logic for Low Steamline Pressure (2/3) ECCS actuation and main steam isolation will be revised from using an interlock with the cold leg loop stop valve position to using Loop Level inhibits. This logic will be placed in bypass to preclude the potential for inadvertent reactor trip, safety injection actuation, and main steamline isolation.

   Manual bypass switch is required to bypass the out-of-service loop.

9. The logic for High (negative) steamline pressure rate (2/3) main steam isolation is not required to be modified since during plant cooldown the isolated loop would already be depressurized.

   No manual bypass switch is required for the out-of-service loop.

Secondary System

The secondary side of the isolated loop would be kept filled and the steam generator would be maintained in a wet long term layup condition as determined by secondary side chemistry requirements.

The loop would be isolated from the main and auxiliary feedwater system and the main steam system by closing the appropriate valves.

The turbine-driven auxiliary feedwater pump (AFW) pump which can receive steam from three of the four loops will receive steam from either of two operating loops during 3 loop operation. Thus, there is still redundancy in steam sources for the AFW pump turbine and, in addition, the two motor driven AFW pumps, powered from two separate class IE power sources would be available. The reliability of the AFW system is acceptable for 3 Loop operation.

5.0 Conformance to Applicable Safety Criteria

5.1 Safety Evaluation of Manual Bypass Switches

The methodology used to perform a safety evaluation of the bypass switch design has been to review the block design schematic (FSAR, Sec. 7.2) and the attached functional diagrams (attached) using the design basis and the functional requirements as a bases. The conformance of the design to the specific criteria is discussed in the following pages in a format that follows Section 4 of the IEEE 279-1971 criteria.

5.1.1 Conformance to General Functional Requirements (Paragraph 4.1 of IEEE 279-1971)

The protection system automatically initiates appropriate protective action whenever a condition monitored by the system reaches a preset level. Although the protection system continues to furnish the primary protective function, the bypass switch is being incorporated into the protective system to inhibit the system from acknowledging an inactive loop as a partial trip condition. The addition of the bypass switches will not degrade the protective system below an acceptable level. Specific functional requirements for the bypass switches are included herein and these requirements will be satisfied by implementation and finalization of the design.

5.1.2 Conformance to the Single Failure Criterion (Paragraph 4.2 of IEEE 279-1971, Regulatory Guide 1.53)

The protection system is designed to provide two, three, or four instrumentation channels for each protective function and two logic train circuits. The bypass switches for the channels will be electrically isolated and physically separated. Thus, when the inactive loop is put in bypass, any single failure within a channel or train will not prevent protective action at the system level when required.

To prevent the occurrence of common mode failures, four individual bypass switches will be provided for each loop within a train. Additional measures as functional, physical separation as well as administrative controls will minimize common mode failures.

5.1.3 Conformance to the Requirements for Quality Components and Modules (Paragraph 4.3 of IEEE 279-1971, GDC1)

Components and wiring will be of a quality that is consistent with the protective system on Millstone Unit 3. Refer to Chapter 17 of the Millstone Unit 3 FSAR for the Quality Assurance program.

5.1.4 Conformance to the Requirements for Equipment Qualification (Paragraph 4.4 of IEEE 279-1971, GDC-2, GDC-4, IEEE 323-1974, IEEE 344-1975)

The bypass switches will be environmentally and seismically qualified in accordance with IEEE-323-1974 and 344-1975 to meet the accident conditions through which these switches must operate to mitigate the consequences of an accident.

5.1.5 Conformance to the Requirements to Maintain Channel Integrity (Paragraph 4.5 of IEEE 279-1971, GDC2, GDC3, GDC4)

The bypass switch functions implementation will be designed to maintain its capability to bypass the inactive loop and not degrade the protection system below an acceptable level following natural phenomena credible to the plant site. The functional capability of the switches will be maintained despite degraded conditions that may exist in the plant due to credible events. The equipment will be environmentally and seismically qualified as discussed in the preceeding subsections. The modification to the existing protection systems will be accomplished totally within the confines of the cabinets.

5.1.6 Conformance to the Requirements to Maintain Channel Independence (Paragraph 4.6 of IEEE 279-1971, GDC22, IEEE 384-1974)

Wiring and component location for the switches of the bypass function will employ physical separation or barriers to ensure independence of the circuits to the extent that is equivalent to the existing independent measures employed by the protection system. The modification to the protection system will be accomplished totally within the cabinets. By staying within the cabinets, the separation provisions of the presently existing switchgear can be maintained.

5.1.7 Conformance to the Requirements concerning Control and Protection System Interaction (Paragraph 4.7 of IEEE 279-1971, GDC24)

The protection system is designed to be independent of the control system. In certain applications the control signals and other non-protective functions are derived from individual protective channels through isolation amplifiers. The bypass switch modification does not compromise this interface and therefore control protection interaction considerations are not sensitive to this modification.

5.1.8 Conformance to the Requirements concerning the Deviation of System Inputs (Paragraph 4.8 of IEEE 279-1971)

Protection system inputs are derived from signals that are direct measurements of the desired variables. The bypass switch modification is not sensitive to this compliance.

5.1.9 Conformance to the Requirements to Provide Capability for Sensor Checks (Paragraph 4.9 of IEEE 279-1971)

Compliance to this requirement is not sensitive to the bypass switch modification.

5.1.10 Conformance to the Requirements to Provide Capability for Test and Calibration (Paragraph 4.10 of IEEE 279-1971, GDC17, GDC21)

The bypass switch modification makes use of and shares on-line testing provisions which are already available as part of the in-place protective systems. These features provide testability during full power operation while the plant is at power.

During normal operation the bypass switches will be keylocked in the normal position. At every refueling outage the components will be tested to verify they function so as not to degrade the protection system below an acceptable level.

During 3-loop operation the affected loop will be in a bypass condition. Procedures will be provided to verify the bypass switches are performing the intended function which will be integrated into the existing on-line testing provisions of the protection system.

5.1.11 Conformance to the Requirements on Channel Bypass or Removal from Operation (Paragraph 4.11 of IEEE 279-1971)

The protection system is designed to permit periodic testing of the analog channel portion of the protection system without initiating a protective action unless a trip condition actually exists. The bypass switch modification is not sensitive to this requirement because of the coincidence logic required for a reactor trip.

5.1.12 Conformance to the Requirements on Operating Bypass (Paragraph 4.12 of IEEE 279-1971)

Where operating requirements necessitate manual bypass of the inactive loop, the design is such that switches are put in bypass manually. Indication will be provided in the control room when the inactive loop has been administratively bypassed.

The transition from 4 loop to 3 loop operations shall be accomplished under strict administrative control.

Setpoints will be changed to more restrictive valves, after which the loop to be isolated may have its affected trip channels bypassed. Loop stop valves will then be closed.

Transition from 3 loop to 4 loop operations shall be accomplished under strict administrative control. This transition will only occur in Mode 5 (Cold Shutdown) or Mode 6 (Refueling Operation).

5.1.13 Conformance to the Requirements on Indication of Bypass (Paragraph 4.13 of IEEE 279-1971)

When a bypass switch is in the bypass position for the inactive loop, the bypass condition will automatically indicate to the reactor operator in the main control room by a separate annunciator. The circuitry will not allow more than one loop to be put in bypass at the same time.

5.1.14 Conformance to the Requirements on Access to Means for Bypassing (Paragraph 4.14 of IEEE 279-1971)

The design provides for strict administrative control of access to the means for manually bypassing protective channels.

5.1.15 Conformance to the Requirements Governing Access to Setpoint Adjustments, Calibration, and Test Points (Paragraph 4.18 of IEEE 279-1971)

Redundant sets of four keylock bypass switches will be added within the protection system. These four are:

a. "Normal, Bypass", loop 1 switch (Train A & B)
b. "Normal, Bypass", loop 2 switch (Train A & B)
c. "Normal, Bypass", loop 3 switch (Train A & B)
d. "Normal, Bypass", loop 4 switch (Train A & B)

Compliance to the requirements that access to and use of these switches will be under strict administrative control to govern there use.

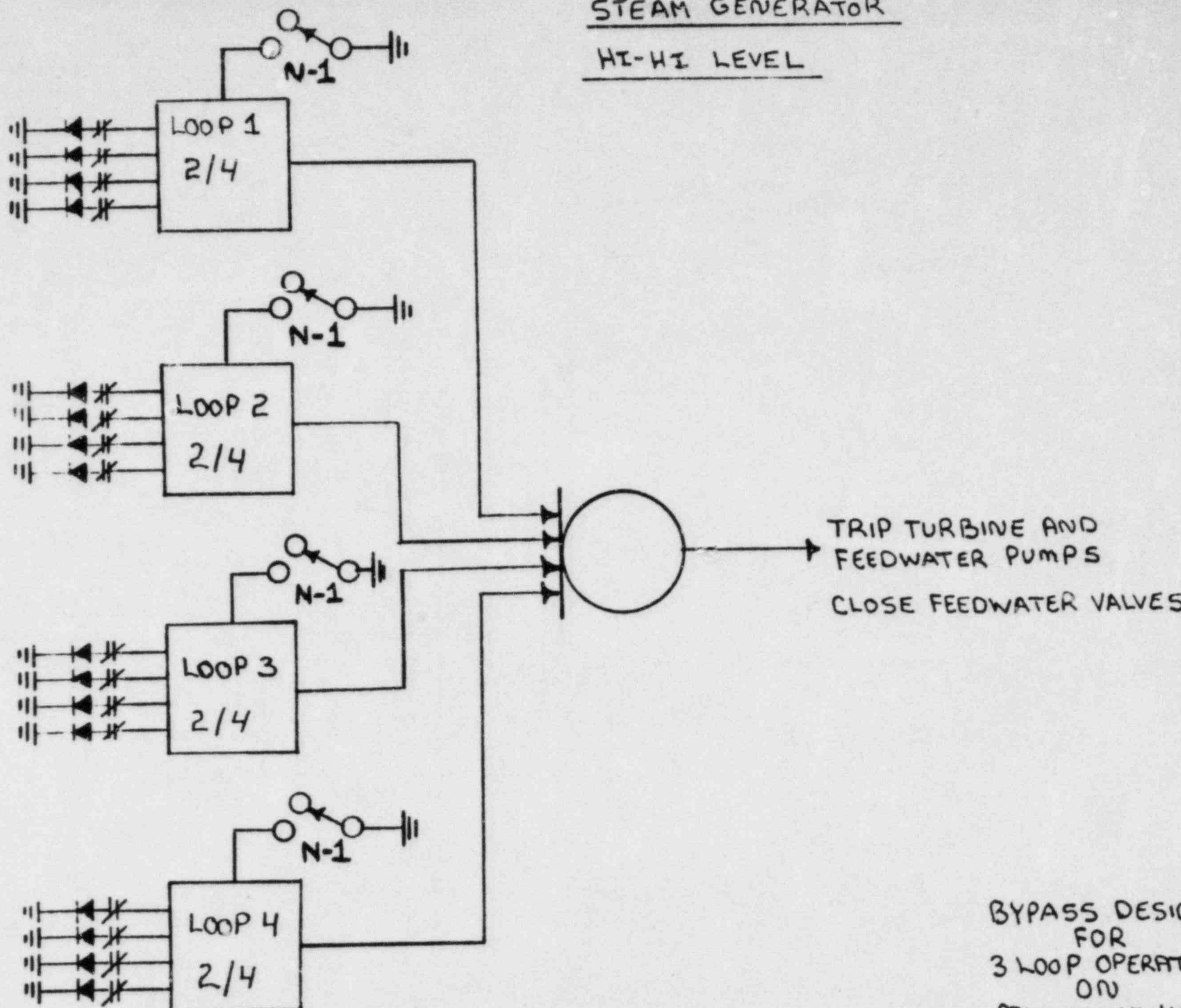
5.1.16 Conformance to the remaining requirements of IEEE 279-1971 listed as follows is not sensitive to the manual bypass switches.

a. Multiple Setpoints (Paragraph 4.14)
b. Completion of Protection Action once it is Initiated (Paragraph 4.16)
c. Manual Initiation (Paragraph 4.17)
d. Identification of Protective Actions (Paragraph 4.19)
e. Information Readout (Paragraph 4.20)
f. System Repair (Paragraph 4.21)
g. Identification (Paragraph 4.22)

5.1.17 Concerning conformance to 10CFR 50 GDC's 20, (Protection System Functions), GDC 25 (Protection System Requirements for Reactivity Control Malfunctions), and GDC 29 (Protection Against Anticipated Operational Occurrences), compliance with this criteria is not sensitive to the manual bypass switches.

STEAM GENERATOR

HI-HI LEVEL

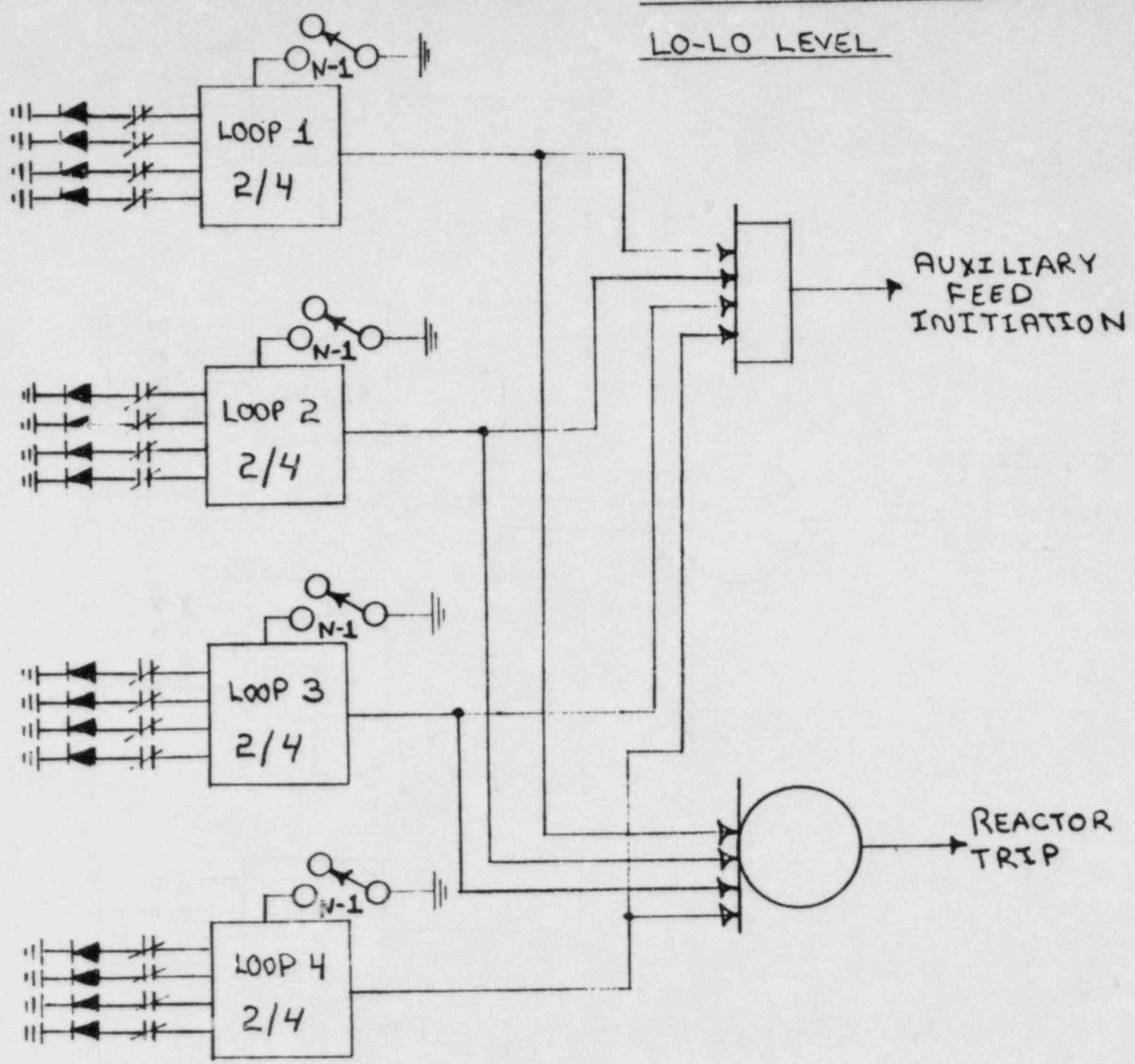


BYPASS DESIGN
FOR
3 LOOP OPERATION
ON
MILLSTONE UNIT 3

(ATTACHMENT 1)
Figure 1

STEAM GENERATOR

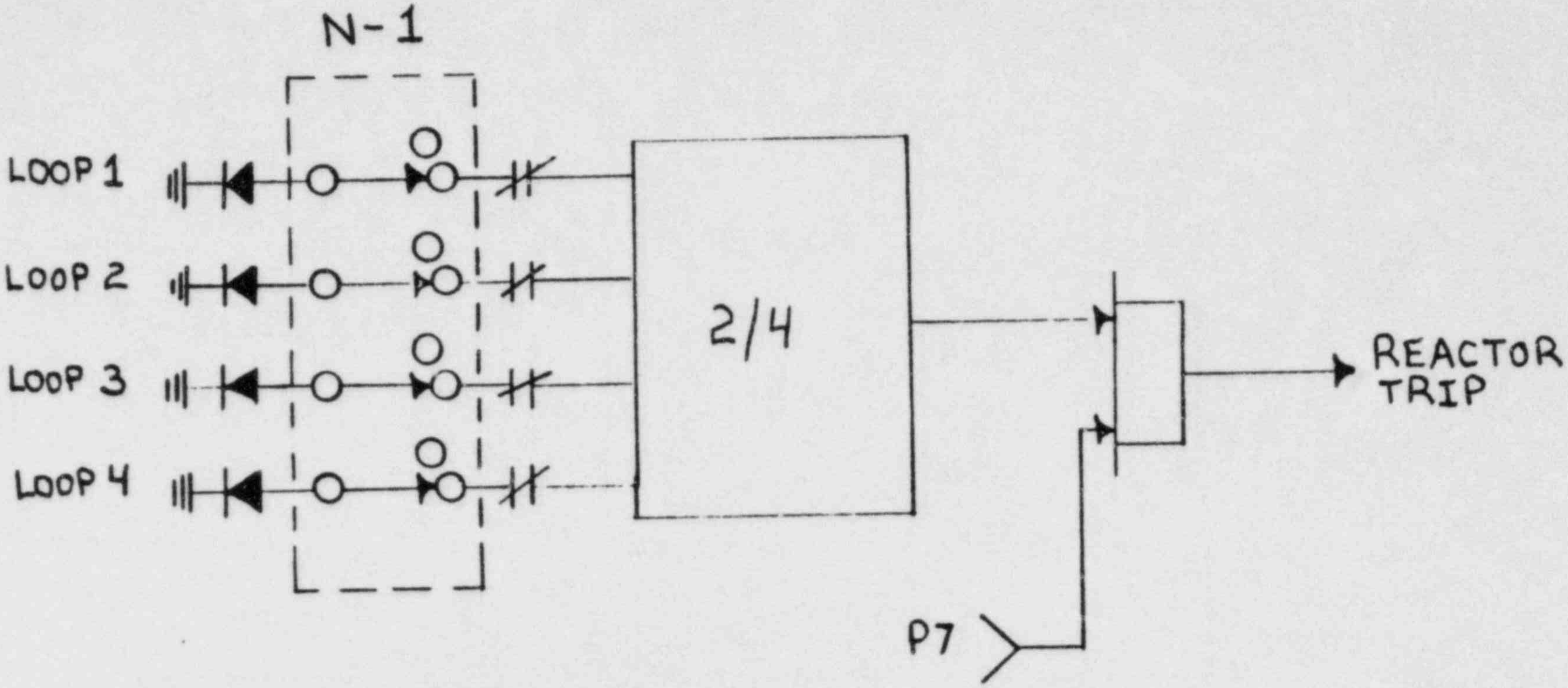
LO-LO LEVEL



BYPASS DESIGN
FOR
3 LOOP OPERATION
ON
MILLSTONE UNIT 3

(ATTACHMENT 1)
FIGURE 2

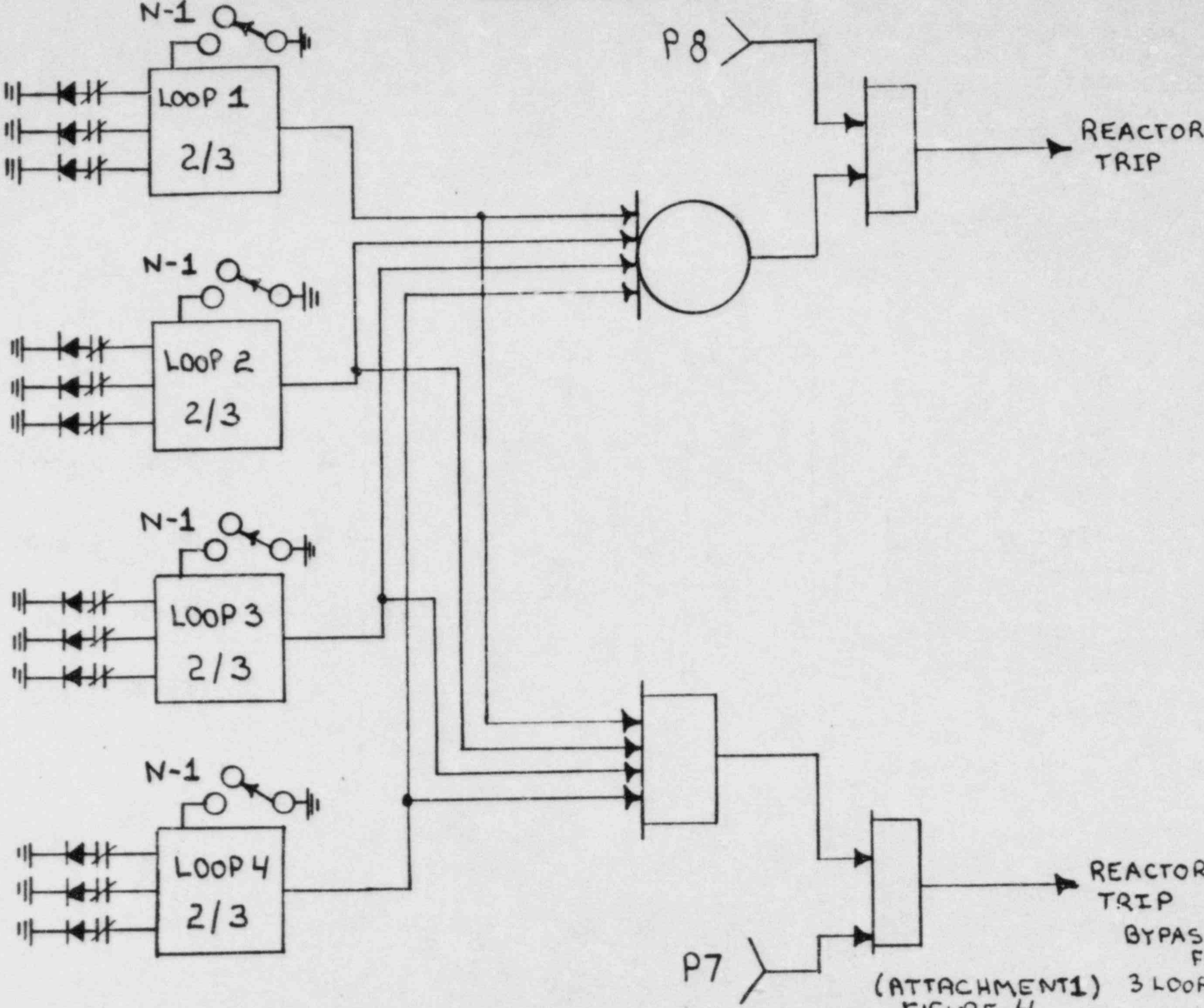
RCP LOW SPEED



BYPASS DESIGN
FOR
3 LOOP OPERATION
ON
MILLSTONE UNIT 3

(ATTACHMENT 1)
FIGURE 3

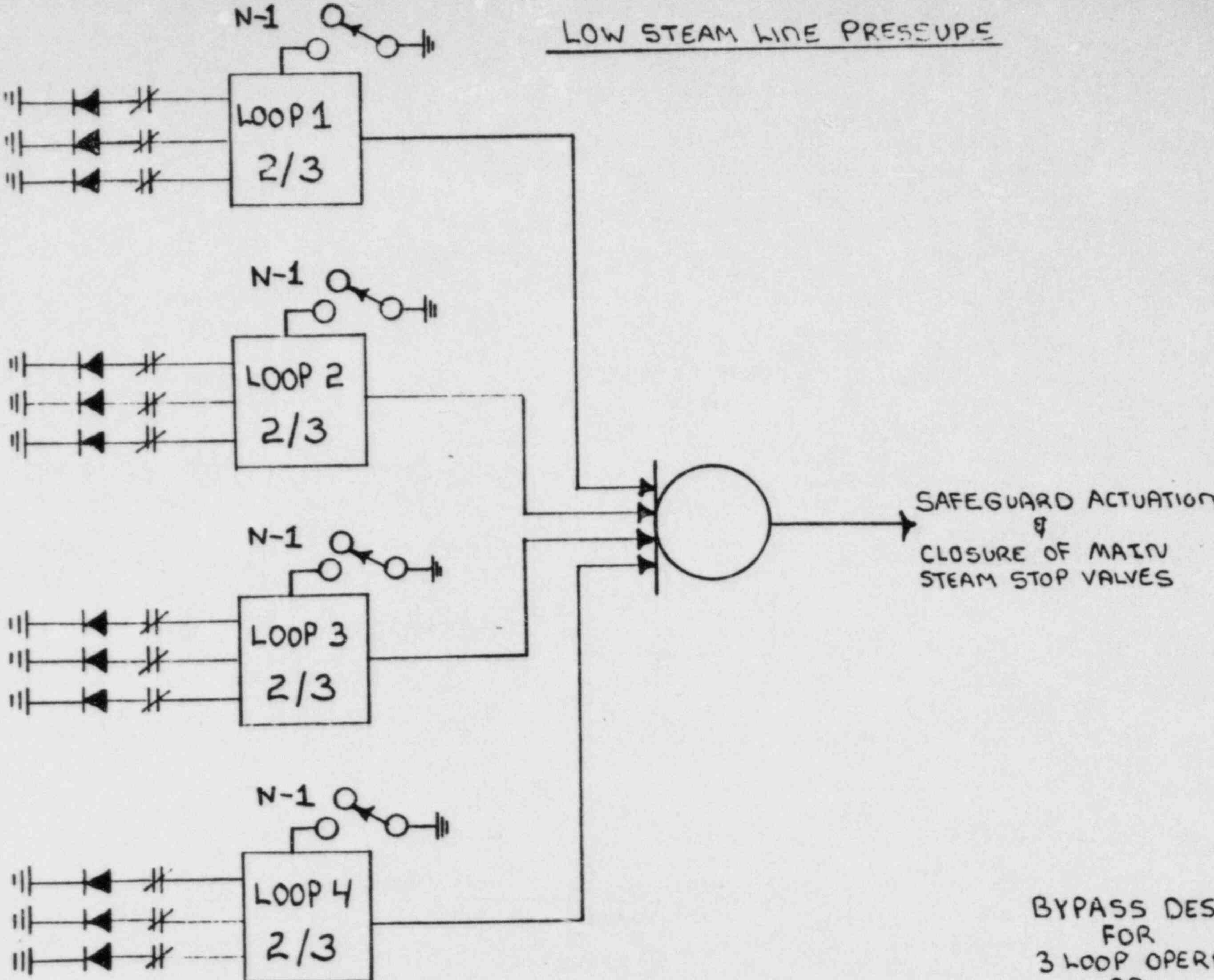
LOSS OF FLOW



(ATTACHMENT1)
FIGURE 4

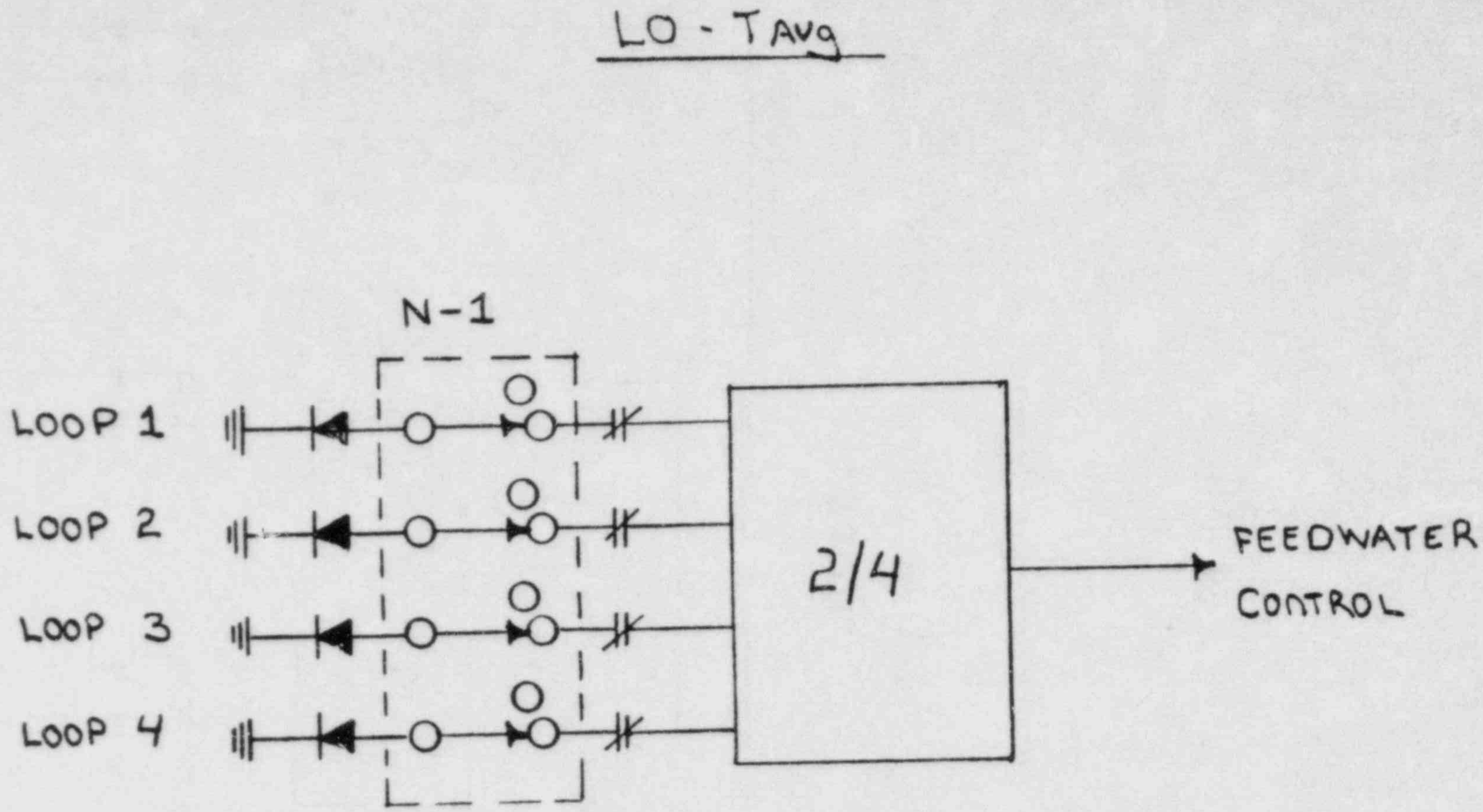
BYPASS DESIGN
FOR
3 LOOP OPERATION

LOW STEAM LINE PRESSURE



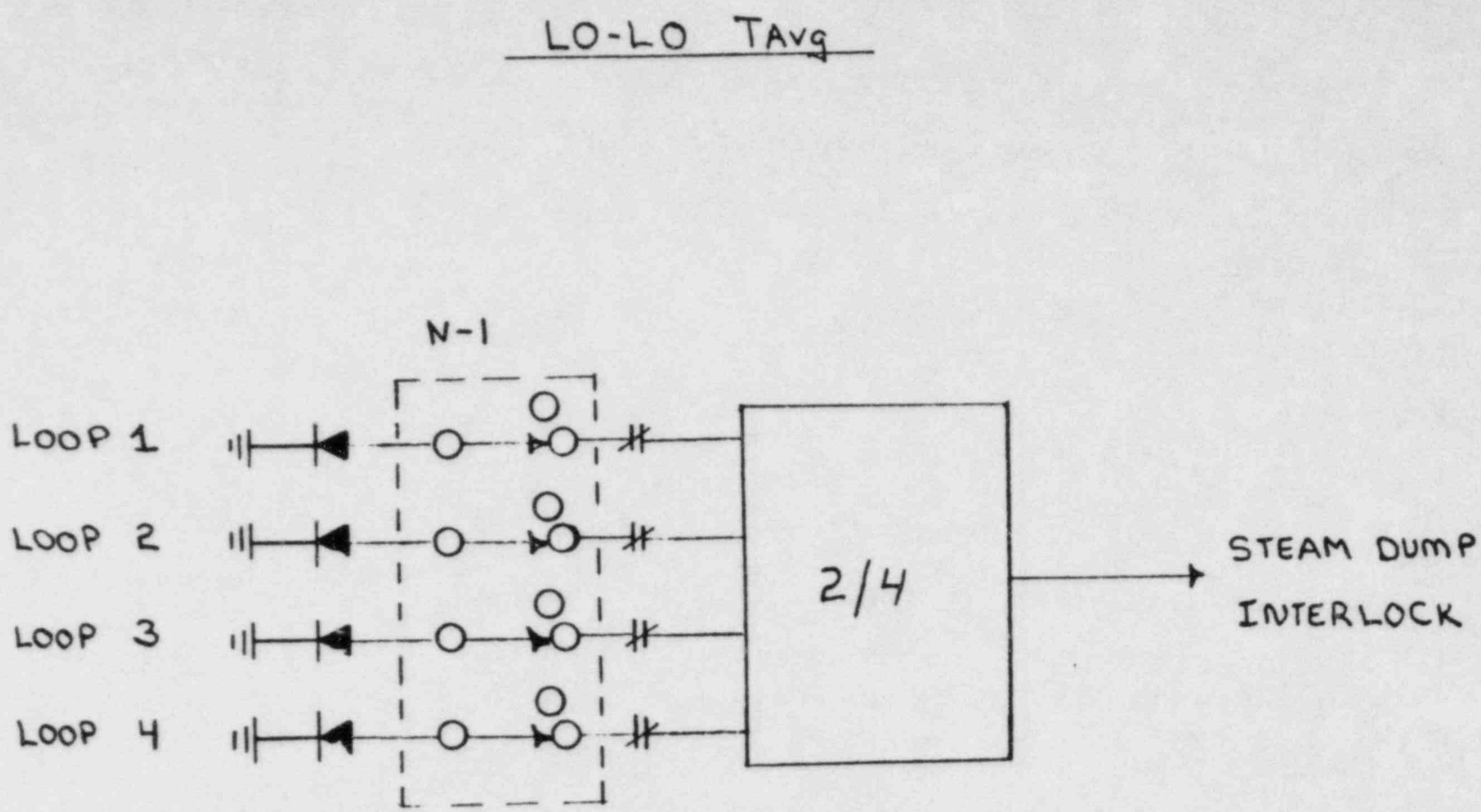
BYPASS DESIGN
FOR
3 LOOP OPERATION
ON
MILLSTONE UNIT 3

(ATTACHMENT 1)
FIGURE 5



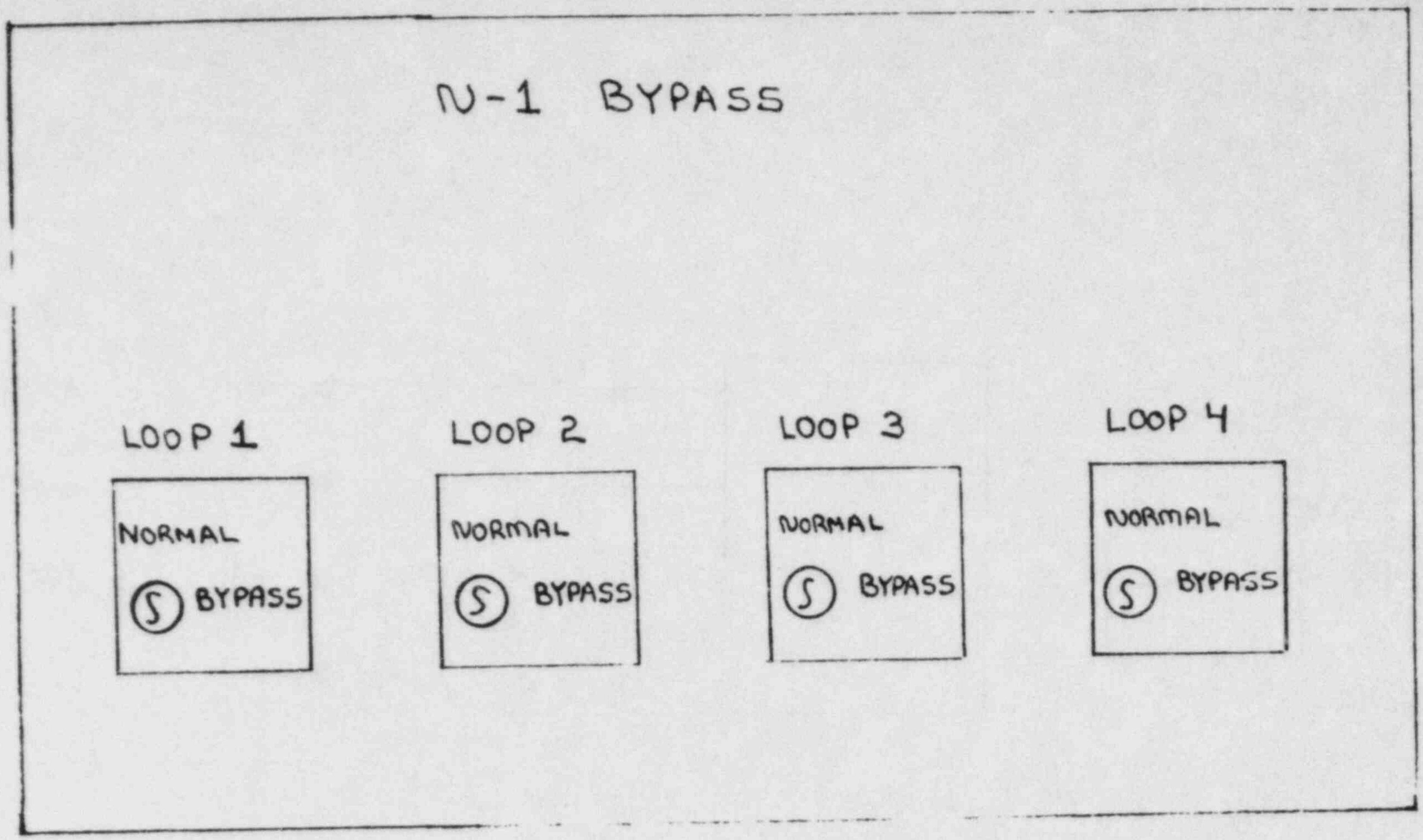
BYPASS DESIGN
FOR
3 LOOP OPERATION
ON
MILLSTONE UNIT 3

(ATTACHMENT 1)
FIGURE 6



BYPASS DESIGN
FOR
3 LOOP OPERATION
ON
MILLSTONE UNIT 3

(ATTACHMENT 1)
FIGURE 7



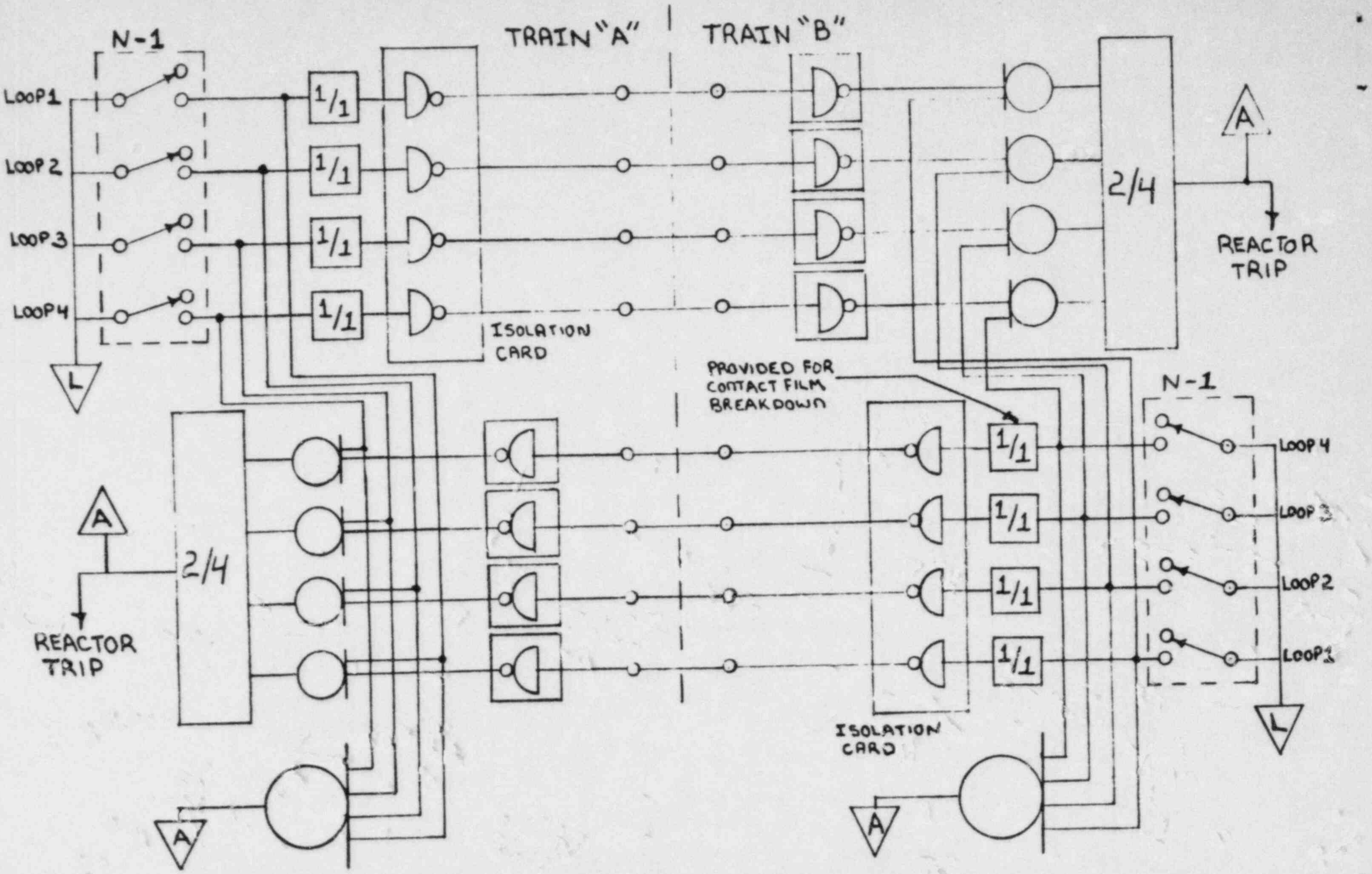
1) KEYLOCK SWITCHES (FOUR INDIVIDUAL KEYS) per TRAIN

2) ANY TWO IN "BYPASS" PROVIDES REACTOR TRIP

3) SWITCHES LOCATED on SPRAY TEST PANEL IN
EACH SSPS LOGIC CABINET.

BYPASS DESIGN
FOR
3 LOOP OPERATION
ON
MILLSTONE UNIT 3

(ATTACHMENT 1)
FIGURE 9



1) ANY TWO LOOPS IN "BYPASS" ON SAME TRAIN PROVIDES A REACTOR TRIP & FIRST OUT ANNUNCIATION.
2) ANY TWO NON-IDENTICAL LOOPS ON OPPOSITE TRAIN IN "BYPASS" PROVIDES A REACTOR TRIP & FIRST OUT ANNUNCIATION.
3) ANY LOOP IN "BYPASS" PROVIDES ALARM FOR 3 LOOP OPERATION.

BYPASS DESIGN
FOR
3 LOOP OPERATION
ON
MILLSTONE UNIT 3

(ATTACHMENT 1)

Attachment 2

CONTROL BUILDING - Elevation 47 ft. 6 in.

Zone: CB-03 Control Room, Instrument Rack Room, Computer Room

Normal Environment (40-year life)

Temperature:

Range:	70 - 80° F
Normal Maximum Average:	80° F
Maximum Normal Excursion:	N/A
Maximum Abnormal Excursion:	N/A

Pressure: Atmosphere

Relative Humidity: 30 - 60%

Radiation Dose (RADS) 40-year life: 70

One Time Accident Environment: Same as Normal Environment described above

Accident Radiation Dose (RADS): 5

Radiation Dose (RADS) - 40-year life plus accident: 75