NORTHEAST UTILITIES

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

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November 15, 1985

Docket No. 50-423
B11874

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

Reference: (1) T. M. Novak letter to J. F. Opeka, Final Draft Technical Specifications for Millstone Nuclear Power Station, Unit 3, dated November 8, 1985.

Gentlemen:

Millstone Nuclear Power Station, Unit No. 3
Final Safety Analysis Report Changes

Reference (1) provided the Final Draft Technical Specifications for Millstone Unit No. 3, and requested Northeast Nuclear Energy Company (NNECO) review the draft Technical Specifications to insure that they accurately reflect the plant's Final Safety Analysis Report, the NRC Staff's Safety Evaluation Report (SER) and the as-built configuration of Millstone Unit No. 3.

In NNECO's review of the draft Technical Specifications the following inconsistencies were identified in the Final Safety Analysis Report (FSAR). These inconsistencies will be corrected as shown in Enclosures 1, 2 and 3, in an upcoming Amendment to the FSAR:

- o Section 7.2.1.1.2, Reactor Trips.
  The description of the Overtemperature ΔT Trip (page 7.2-4) and the Overpower ΔT Trip (page 7.2-5).

- o Table 5.2-4, Reactor Coolant Water Chemistry Specification
  Maximum oxygen.

- o Table 6.3-1, Emergency Core Cooling System Component Parameters. The Charging and Safety Injection Pumps maximum flow rate.

- o Table 6.3-11, NPSH for ECCS Pumps. The Charging and Safety Injection pump runout.

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- o Section 10.4.9, Auxiliary Feedwater System. The contained usable water volume of the Demineralized Water Storage Tank.
- o Sections 6, 10 and 15, Auxiliary Feedwater Transients.
- o Table 6.2-65, Containment Penetration.

We trust the Staff finds these changes acceptable.

Very truly yours,

NORTHEAST NUCLEAR ENERGY COMPANY
et. al.

BY NORTHEAST NUCLEAR ENERGY COMPANY
Their Agent

J. F. Opeka

J. F. Opeka
Senior Vice President

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

Then personally appeared before me J. F. Opeka, who being duly sworn, did state that he is 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

ENCLOSURE 1

or prior to startup. This bypass action is annunciated on the control board.

d. Power range high positive neutron flux rate trip

This circuit trips the reactor when a sudden abnormal increase in nuclear power occurs in two out of four power range channels. This trip provides DNB protection against rod ejection accidents of low worth from mid-power and is always active.

e. Power range high negative neutron flux rate trip

This circuit trips the reactor when a sudden abnormal decrease in nuclear power occurs in two out of four power range channels. This trip provides protection against two or more dropped rods and is always active. Protection against one dropped rod is not required to prevent occurrence of DNB per Section 15.2.3.

Figure 7.2-1, Sheet 3, shows the logic for all of the nuclear overpower and rate trips.

2. Core Thermal Overpower Trips

The specific trip functions generated are as follows:

a. Overtemperature ΔT Trip

This trip protects the core against low DNBR and trips the reactor on coincidence as listed in Table 7.2-1, with one set of temperature measurements per loop. The setpoint for this trip is continuously calculated by analog circuitry for each loop by solving the following equation:

INSERT 'A'

$$\Delta T_{setpoint} = \Delta T_o \left[K_1 - K_2 \left(\frac{1 + \tau_1 s}{1 + \tau_2 s}\right)\left(T_{avg} - T^o_{avg}\right) + K_3\left(P - 2235\right) - f\left(\Delta\phi\right)\right] \quad (7.2\text{-}1)$$

where:

ΔT	=	Indicated ΔT at rated thermal power
T_{avg}	=	Average reactor coolant temperature (°F)
T^o_{avg}	=	Indicated T_{avg} at rated thermal power
P	=	Pressurizer pressure (psig)
K_1	=	Preset bias

e

K_2 = Preset gain which compensates for effects of temperature DNB limits

K_3 = Preset gain which compensates for the effects of pressure on the DNB limits

τ_1, τ_2 = Preset constants which compensate for piping and instrument time delay

s = Laplace transform operator (seconds $^{-1}$)

$f(\Delta\phi)$ = A function of the neutron flux difference between upper and lower long ion chambers. (Refer to Figure 7.2-2).

A separate long ion chamber unit supplies the flux signal for each overtemperature ΔT trip channel. Increases in $\Delta\phi$ beyond a pre-defined deadband result in a decrease in trip setpoint. Refer to Figure 7.2-2.

The required one pressurizer pressure parameter per loop is obtained from separate sensors connected to three pressure taps at the top of the pressurizer. Three pressurizer pressure signals are obtained from the three taps by connecting one of the taps to two pressure transmitters. Refer to Section 7.2.2.3.3 for an analysis of this arrangement.

Figure 7.2-1, Sheet 5, shows the logic for overtemperature ΔT trip function.

b. Overpower ΔT trip

This trip protects against excessive power (fuel rod rating protection) and trips the reactor on coincidence as listed in Table 7.2-1, with one set of temperature measurements per loop. The setpoint for each channel is continuously calculated using the following equation:

INSERT B'

$$\Delta T_{setpoint} = \Delta T_o \left[K_4 - K_5 \left(\frac{\tau_3 s}{1 + \tau_3 s} \right) T_{avg} - K_6 \left(T_{avg} - T' \right) - f\left(\Delta\phi\right) \right] \quad (7.2\text{-}2)$$

where:

ΔT = Indicated ΔT at rated thermal power

$f(\Delta\phi)$ = A function of the neutron flux difference between upper and lower long ion chamber section.

K_4 = A preset bias

K_5 = A constant which compensates for piping and instrument time delay

K_6 = A constant which compensates for the change in density flow and heat capacity of the water with temperature.

T' = Indicated T_{avg} at rated thermal power

T_{avg} = Average reactor coolant temperature (°F)

τ_3 = Preset time constant (seconds)

s = Laplace transform operator (seconds $^{-1}$)

The source of temperature and flux information is identical to that of the overtemperature ΔT trip and the resultant ΔT setpoint is compared to the same ΔT. Figure 7.2-1, Sheet 5, shows the logic for this trip function.

3. Reactor Coolant System Pressurizer Pressure and Water Level Trips

The specific trip functions generated are as follows:

a. Pressurizer low pressure trip

The purpose of this trip is to protect against low pressure which could lead to DNB. The parameter being sensed is reactor coolant pressure as measured in the pressurizer. Above P-7 the reactor is tripped when the pressurizer pressure measurements (compensated for rate of change) fall below preset limits. This trip is blocked below P-7 to permit startup. The trip logic and interlocks are given in Table 7.2-1.

The trip logic is shown on Figure 7.2-1, Sheet 6.

b. Pressurizer high pressure trip

The purpose of this trip is to protect the reactor coolant system against system overpressure.

The same sensors and transmitters used for the pressurizer low pressure trip are used for the high pressure trip except that separate bistables are used for trip. These bistables trip when uncompensated pressurizer pressure signals exceed preset limits on coincidence as listed in Table 7.2-1. There are no interlocks or permissives associated with this trip function.

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TABLE 2.2-1 (Continued)

TABLE NOTATIONS

NOTE 1: OVERTEMPERATURE ΔT

$$\Delta T \frac{(1 + \tau_1 S)}{(1 + \tau_2 S)} \left(\frac{1}{1 + \tau_3 S}\right) \leq \Delta T_o \left\{K_1 - K_2 \frac{(1 + \tau_4 S)}{(1 + \tau_5 S)} \left[T \left(\frac{1}{1 + \tau_6 S}\right) - T'\right] + K_3(P - P') - f_1(\Delta I)\right\}$$

Where:	ΔT	=	Measured ΔT by RTD Manifold Instrumentation;
	$\frac{1 + \tau_1 S}{1 + \tau_2 S}$	=	Lead-lag compensator on measured ΔT;
	τ_1, τ_2	=	Time constants utilized in lead-lag compensator for ΔT, τ_1 = 8 s, τ_2 = 3 s;
	$\frac{1}{1 + \tau_3 S}$	=	Lag compensator on measured ΔT;
	τ_3	=	Time constants utilized in the lag compensator for ΔT, τ_3 = 0 s;
	ΔT_o	=	Indicated ΔT at RATED THERMAL POWER;
	K_1	=	1.08 (Four Loops Operating); 1.01 (Three Loops Operating);
	K_2	=	0.01313;
	$\frac{1 + \tau_4 S}{1 + \tau_5 S}$	=	The function generated by the lead-lag compensator for T_{avg} dynamic compensation;
	τ_4, τ_5	=	Time constants utilized in the lead-lag compensator for T_{avg}, τ_4 = 33 s, τ_5 = 4 s;
	T	=	Average temperature, °F;
	$\frac{1}{1 + \tau_6 S}$	=	Lag compensator on measured T_{avg};
	τ_6	=	Time constant utilized in the measured T_{avg} lag compensator, τ_6 = 0 s;

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TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 1: (Continued)

T' $\leq$ 587.1°F (Nominal T_{avg} at RATED THERMAL POWER);

K_3 = 0.000603/psia;

P = Pressurizer pressure, psia;

P' = 2250 psia (Nominal RCS operating pressure);

S = Laplace transform operator, s^{-1};

and $f_1(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range neutron ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

(1) For $q_t - q_b$ between -30% and + 10%, $f_1(\Delta I) = 0$, where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER;

(2) For each percent that the magnitude of $q_t - q_b$ exceeds -30%, the ΔT Trip Setpoint shall be automatically reduced by 3.6% of its value at RATED THERMAL POWER; and

(3) For each percent that the magnitude of $q_t - q_b$ exceeds +10%, the ΔT Trip Setpoint shall be automatically reduced by 2.0% of its value at RATED THERMAL POWER.

NOTE 2: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 2.1% ΔT span (Four Loop Operation); 4.1% ΔT span (Three Loop Operation).

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Page 1 of 2

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 3: OVERPOWER ΔT

$$\Delta T \frac{(1 + \tau_1 S)}{(1 + \tau_2 S)} \left(\frac{1}{1 + \tau_3 S}\right) \leq \Delta T_0 \left\{K_4 - K_5 \left(\frac{\tau_7 S}{1 + \tau_7 S}\right) \left(\frac{1}{1 + \tau_6 S}\right) T - K_6 \left[T \left(\frac{1}{1 + \tau_6 S}\right) - T''\right] - f_2(\Delta I)\right\}$$

Where:

ΔT = As defined in Note 1,

$\frac{1 + \tau_1 S}{1 + \tau_2 S}$ = As defined in Note 1,

τ_1, τ_2 = As defined in Note 1,

$\frac{1}{1 + \tau_3 S}$ = As defined in Note 1,

τ_3 = As defined in Note 1,

ΔT_0 = As defined in Note 1,

K_4 = 1.09,

K_5 = 0.02/°F for increasing average temperature and 0 for decreasing average temperature,

$\frac{\tau_7 S}{1 + \tau_7 S}$ = The function generated by the rate-lag compensator for T_{avg} dynamic compensation,

τ_7 = Time constants utilized in the rate-lag compensator for T_{avg}, τ_7 = 10 s,

$\frac{1}{1 + \tau_6 S}$ = As defined in Note 1,

τ_6 = As defined in Note 1,

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TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 3: (Continued)

K_6	=	0.00129/°F for T > T" and K_6 = 0 for T ≤ T",
T	=	As defined in Note 1,
T"	=	Indicated T_{avg} at RATED THERMAL POWER (Calibration temperature for ΔT instrumentation, ≤ 587.1°F),
S	=	As defined in Note 1, and
$f_2(\Delta I)$	=	0 for all ΔI.

NOTE 4: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 3.4% ΔT span.

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TABLE 5.2-4

REACTOR COOLANT WATER CHEMISTRY SPECIFICATION

Electrical conductivity	Determined by the concentration of boric acid and alkali present, expected range is <1 to 40 Mhos/cm at 25°C.
Solution pH	Determined by the concentration of boric acid and alkali present, expected values range between 4.2 (high boric acid concentration) to 10.5 (low boric acid concentration) at 25°C; values will be 5.0 or greater at normal operating temperatures.
Oxygen, maximum (ppm)(1)	~~0.005~~ 0.1
Chloride, maximum (ppm)(2)	0.15
Fluoride, maximum (ppm)(2)	0.15
Hydrogen (cc(STP)/Kg H_2O)(3)	25 to 50
Total suspended solids, maximum (ppm)(4)	1.0
pH control agent (Li7OH) (ppm)(5)	0.7 to 2.2 as Li
Boric acid (ppm B)	Variable from 0 to approximately 4,000
Silica, maximum (ppm)(6)	0.2
Aluminum, maximum (ppm)(6)	0.05
Calcium (ppm)(6)	0.05 maximum
Magnesium (ppm)(6)	0.05 maximum

NOTES:

1. Oxygen concentration must be controlled to less than 0.1 ppm in the reactor coolant at temperatures above 180°F by scavenging with hydrazine. During power operation with the specified hydrogen concentration maintained in the coolant, the residual oxygen concentration ~~must not exceed 0.005 ppm~~ Control value becomes ≤0.005.
2. Halogen concentrations must be maintained below the specified values at all times regardless of system temperature.

TABLE 6.3-1

EMERGENCY CORE COOLING SYSTEM
COMPONENT PARAMETERS

<u>Accumulators</u>		
Number	4	
Design Pressure (psig)	700	
Design Temperature (°F)	300	
Operating Temperature (°F)	80	
Normal Operating Pressure (psig)	650	
Minimum Operating Pressure (psig)	600	
Total Volume (ft^3)	1,350 each	
Minimum Water Volume (ft^3)	950 each	
Volume N2 Gas (ft^3)	400	
Boric Acid Concentration, nominal (ppm)	2,000	
Boric Acid Concentration, minimum (ppm)	1,900	
Relief Valve Sepoint (psig)	700	
<u>Charging Pumps</u>		
Number	3	
Design Pressure (psig)	2,800	
Design Temperature (°F)	300	
*Design Flow Rate (gpm)	150	
Design Head (ft)	5,800	
Maximum Flow Rate (gpm)	~~550~~ 560	
Head at Maximum Flow Rate (ft)	1,400	
Discharge Head at Shutoff (ft)	6,000	
**Motor Rating (bhp)	600	
Required NPSH (ECCS) Maximum Flowrate (ft)	***	440.30
Available NPSH	***	
<u>Safety Injection Pumps</u>		
Number	2	
Design Pressure (psig)	1,750	
Design Temperature (°F)	300	
Design Flow Rate (gpm)	425	
Design Head (ft)	2,850	"A" pump 670
Maximum Flow Rate (gpm)	~~[illegible]~~ "B" pump 650	440.30
Head at Maximum Flow Rate (ft)	1,650	
Discharge Head (ft)	3,545	
**Motor Rating (bhp)	450	
Required NPSH	***	
Available NPSH	***	440.30
<u>Residual Heat Removal Pumps</u>		
(See Section 5.4.7 for design parameters)		
Required NPSH	***	
Available NPSH	***	440.30

TABLE 6.3-11

NPSH FOR ECCS PUMPS

	Charging	SI	RHS	
Elevation head (ft) (Z)	17.0	21.1	35.6	8
Pipe losses (ft) (H_f)	12.5	5.4	20.5	
$P_t - P_v$ (ft)	33.5	33.5	33.5	440.30
Available NPSH (ft)	38.2	49.2	48.6	
Pump flow (gpm per pump)	560	670		8
Pump runout	~~560~~	~~675~~	5500	
System runout	410	445	4850	
Required NPSH (ft)	18.0	16.0	25.0	
Available minus required NPSH (ft) (NPSH margin)	20.2	33.2	23.6	8

(360,000)

This provides a guaranteed source of cooling water under all conditions.

12 During safety related operation, each auxiliary feedwater pump takes suction through a separate supply line directly from the DWST. The DWST, sized at ~~320,000~~ total usable gallons, has sufficient capacity to satisfy the design basis of the auxiliary feedwater system. The total tank capacity includes additional unusable volume necessary to prevent vortexing at the auxiliary feedwater pump suction nozzles. Makeup is provided to the DWST from the water treating system (Section 9.2.3).

334,000

12 An additional source of water is provided to each auxiliary feedwater pump suction by the condensate storage tank (Section 9.2.6). This source is not safety related and, therefore, is not considered available for safety related purposes. The normally closed air-operated valve connecting the condensate storage tank and each auxiliary feedwater pump suction is under administrative control.

The service water system is available as a long term safety grade source of auxiliary feedwater for the steam generators. Before the auxiliary feedwater pumps can take suction from the service water system, spool pieces must be added to connect the service water system to the auxiliary feedwater system. These spool pieces are provided, in lieu of permanent piping, to preclude inadvertant discharging of service water to the steam generators.

A connection from the domestic water system to the DWST fill line is provided to satisfy the requirement that auxiliary feedwater be available to support 72 hours of hot standby followed by a subsequent cooldown in the event of damaging fires. A removable spool piece must be placed in line prior to use of the domestic water system. During this activity water is available in the DWST and the condensate storage tank.

Feedwater from the steam generator auxiliary feedwater pumps is pumped to each steam generator through normally open control valves. Flow is monitored in each line connecting to the feedwater system (Section 10.4.2). Each control valve is manually adjusted from the control room as dictated by the steam generator water level and auxiliary feedwater flow rate. The control valves can also be manually adjusted from the auxiliary shutdown panel. In the event of a loss of power, these valves will remain open. The control valves will be equipped with handwheels and may be adjusted by hand.

Auxiliary feedwater flow to the steam generators is limited by flow venturis located in each auxiliary feedwater line. These venturis are sized to cavitate in order to maintain the minimum required flows to the intact steam generators and to prevent runout flow to a depressurized steam generator.

The auxiliary feedwater is discharged to the steam generators through a connection in each main feedwater line inside the containment structure and downstream of the main feedwater stop-check valves.

ENCLOSURE 2

AUXILIARY FEEDWATER TRANSIENTS

FSAR SECTION 6

CHANGES

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

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

1. Feedwater flow to the steam generators is reduced. Since 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 functions to ensure the availability of adequate feedwater to the unaffected steam generators so that:

1. No substantial overpressurization of the RCS occurs (less than 110 percent of design pressures); and

2. Sufficient liquid in the RCS is maintained so that the core remains in place and geometrically intact with no loss of core cooling capability.

NEW

The engineered safety systems assumed to function are the auxiliary feedwater system and the safety injection system. ~~For the auxiliary~~ 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; flow from the two motor-driven pumps delivers 480 gpm to the three intact steam generators. ~~([illegible] gpm in the case of N-1 loop operation). driven auxiliary feedwater pump delivers a minimum of [illegible] gpm to the three nonfaulted steam generators. This assumption is conservative because it maximizes the purge time in the feedwater lines before auxiliary feedwater enters the non faulted steam generators.~~

X

A safety injection signal from either low steamline pressure or high containment pressure initiates flow of cold borated water into the RCS. The amount of safety injection flow is a function of RCS pressure.

Conclusions

Results of the analyses show that for the postulated feedwater line rupture, the assumed auxiliary feedwater system capacity is adequate to remove decay heat, to prevent overpressurizing the RCS, and to

FSAR SECTION 10

CHANGES

   c. The capability to isolate components, subsystems, or piping, if required, so that the system safety function will be maintained.

7. General Design Criterion 45, for design provisions to permit periodic inservice inspection of system components and equipment.

8. General Design Criterion 46, for design provisions to permit appropriate functional testing of the system and components to ensure structural integrity and leaktightness, operability and performance of active components, and capability of the integrated system to function as intended during normal, shutdown, and accident conditions.

9. Regulatory Guide 1.26, for the quality group classification of system components (Section 3.2).

10. Regulatory Guide 1.29, for seismic design classification of system components (Section 3.2).

11. Regulatory Guide 1.62, for design provisions made for manual initiation of each protective action.

12. Regulatory Guide 1.102, for the protection of structures, systems, and components important to safety from the effects of flooding.

13. Regulatory Guide 1.117, for the protection of structures, systems, and components important to safety from the effects of tornado missiles.

14. Branch Technical Positions APCSB 3-1 and MEB 3-1, for breaks in high and moderate energy piping systems outside containment.

15. Branch Technical Position ASB 10-1, for auxiliary feedwater pump drive and power supply diversity.

Replace with INSERT (A)

16. The auxiliary feedwater system is designed to supply a minimum of 470 gpm total flow to at least two steam generators even with the occurrence of a single failure for the following transients:

    a. loss of normal feedwater;

    b. loss of offsite power followed by reactor trip (results in a loss of normal feedwater);

    c. secondary system pipe rupture; 12

    d. cooldown following steam generator tube rupture; and

    e. loss-of-coolant accident, small break.

INSERT (A)

16. The auxiliary feedwater system is designed to supply a minimum of 520 gpm to four steam generators even with the occurrence of a single failure following any condition II event (i.e., loss of normal feedwater, blackout).

16.a. The auxiliary feedwater ~~a minimum~~ system is designed to supply a minimum of 480 gpm to three effective steam generators even with the occurrence of a single failure following any condition III or IV event (i.e., secondary side rupture, small break loss-of-coolant accident, or cooldown following steam generator ~~tupe~~ tube rupture).

FSAR SECTION 15

CHANGES

15.2.7 Loss of Normal Feedwater Flow

15.2.7.1 Identification of Causes and Accident Description

A loss of normal feedwater (from pump failures, valve malfunctions, or loss of offsite ac power) results in a reduction in capability of the secondary system to remove the heat generated in the reactor core. If an alternative supply of feedwater were not supplied to the 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 from the RCS. Since the plant is tripped well before the steam generator heat transfer capability is reduced, the primary system variables never approach a DNB condition.

INSERT A >

FOR THE CASE WHERE THREE LOOPS ARE OPERATING INITIALLY, ~~The~~ the worst postulated loss of normal feedwater event is one initiated by a loss of offsite ac power as described in Section 15.2.6. This is due to the decreased capability of the reactor coolant to remove residual core heat as a result of the reactor coolant pump coastdown.

~~As stated in Section 15.2.6.1,~~ The following events occur upon loss of ~~ac power,~~ NORMAL FEEDWATER

1. Plant vital instruments are supplied from emergency dc power sources FOR THE CASES ANALYZED WITH A LOSS OF OFFSITE POWER.
2. As the steam system pressure rises following the trip, the steam generator power operated relief valves are automatically opened to the atmosphere. Steam dump to the condenser is assumed not to be available. If the steam flow 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.

INSERT B >

~~A loss of normal feedwater caused by a loss of offsite ac power is 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~~ A full analysis of the system transient is presented below 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 from the reactor core, and returning the plant to a safe condition.

INSERT C >

15.2.7.2 Analysis of Effects and Consequences

Method of Analysis

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

Assumptions made in the analysis are as follows.

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

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

3. FOR THE CASES WITH A LOSS OF OFFSITE POWER, A heat transfer coefficient in the steam generator associated with RCS natural circulation.

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

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

6. Auxiliary feedwater is delivered to FOUR ~~two~~ steam generators for N-loop analysis. The most limiting auxiliary feedwater system configuration for N-1 loop operation is for auxiliary feedwater to be supplied to one active steam generator. 12

7. Secondary system steam relief is achieved through the self-actuated safety valves. Note that steam relief will, in 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 been assumed unavailable.

8. The initial reactor coolant average temperature is 6.5°F 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.

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

As such, the assumptions used in this analysis are designed to 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 above.

One such assumption is the loss of external (offsite) ac power. This 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 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 complete loss of forced reactor coolant flow.

If ac power IS MAINTAINED ~~were not lost~~ for this incident the reactor coolant flow ~~would~~ remains at its normal value and the reactor would trip via the 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 be manually tripped at some later time to reduce heat addition to the RCS. The auxiliary feedwater system has sufficient capacity, even assuming the worst single failure, to preclude filling the pressurizer should the pumps not be tripped.

Plant characteristics and initial conditions are further discussed in Section 15.0.3. Plant systems and equipment which are available to 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 reactor control systems are not required to function during this transient. The reactor protection system is required to function following a loss of normal feedwater as analyzed here. The auxiliary feedwater system is required to deliver a minimum auxiliary feedwater flow rate. No single active failure will prevent operation of any system required to function. A discussion of ATWT considerations is presented in WCAP-8330.

<u>Results</u>

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

Following the reactor and turbine trip from full load, the water 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 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 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 or safety valves. From Figure 15.2-10, 10A, and 10B, it can be seen that at no time

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 Table 15.2-1. As shown on Figures 15.2-9, ~~and~~ 15.2-10, 15.2.9A,B and 15.2.10 A,B the plant approaches a stabilized condition following reactor trip and auxiliary feedwater initiation. Standard plant shutdown procedures may be followed to further cool down the plant.

15.2.7.3 Conclusions

Results of the analysis show that a loss of normal feedwater does not adversely affect the core, the RCS, or the steam system since the 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 maintained above the tubesheets.

15.2.7.4 Radiological Consequences

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

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
		High pressurizer pressure reactor trip point reached	5.2	6.5
		Initiation of steam release from steam generator safety valves	7.0	9.0
		Rods begin to drop	7.2	8.5
		Peak pressurizer pressure occurs	8.0	10.0
		Minimum DNBR occurs	(1)	(1)
12	4. Without pressurizer control (maximum moderator feedback)	Turbine trip, loss of main feed flow	0.0	0.0
		High pressurizer pressure reactor trip point reached	5.2	6.5
		Initiation of steam release from steam generator safety valves	7.0	9.0
		Rods begin to drop	7.2	8.5
		Peak pressurizer pressure occurs	8.0	9.0
		Minimum DNBR occurs	(1)	(1)
	Loss of normal feedwater flow WITH LOSS OF OFFSITE POWER	Main feedwater flow stops	10.0	~~10.0~~ (later)
		Low steam generator water level trip	64.9 ~~65.2~~	~~67.4~~
		Rods begin to drop	66.9 ~~67.2~~	~~69.4~~

TABLE 15.2-1 (Cont)

Accident	Event	N-Loop Time (sec)	N-1 Loop Time (sec)
	Reactor coolant pumps begin to coastdown	68.9 ~~67.2~~	~~69.4~~
	Peak water level in pressurizer occurs	69.0 ~~70.0~~	-
	Four ~~Two~~ steam generators begin to receive ~~550~~ 520 gpm of auxiliary feed ~~from one motor-driven auxiliary feedwater pump~~	124.9 ~~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	345 ~~~2000~~	~~~1800~~
INSERT E			
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	15.4	16.4
	Rods begin to drop	17.4	18.4
	Auxiliary feedwater is delivered to three intact steam generators	75.4	76.4
	Borated safety injection flow enters cold legs	108.3	79.9

12

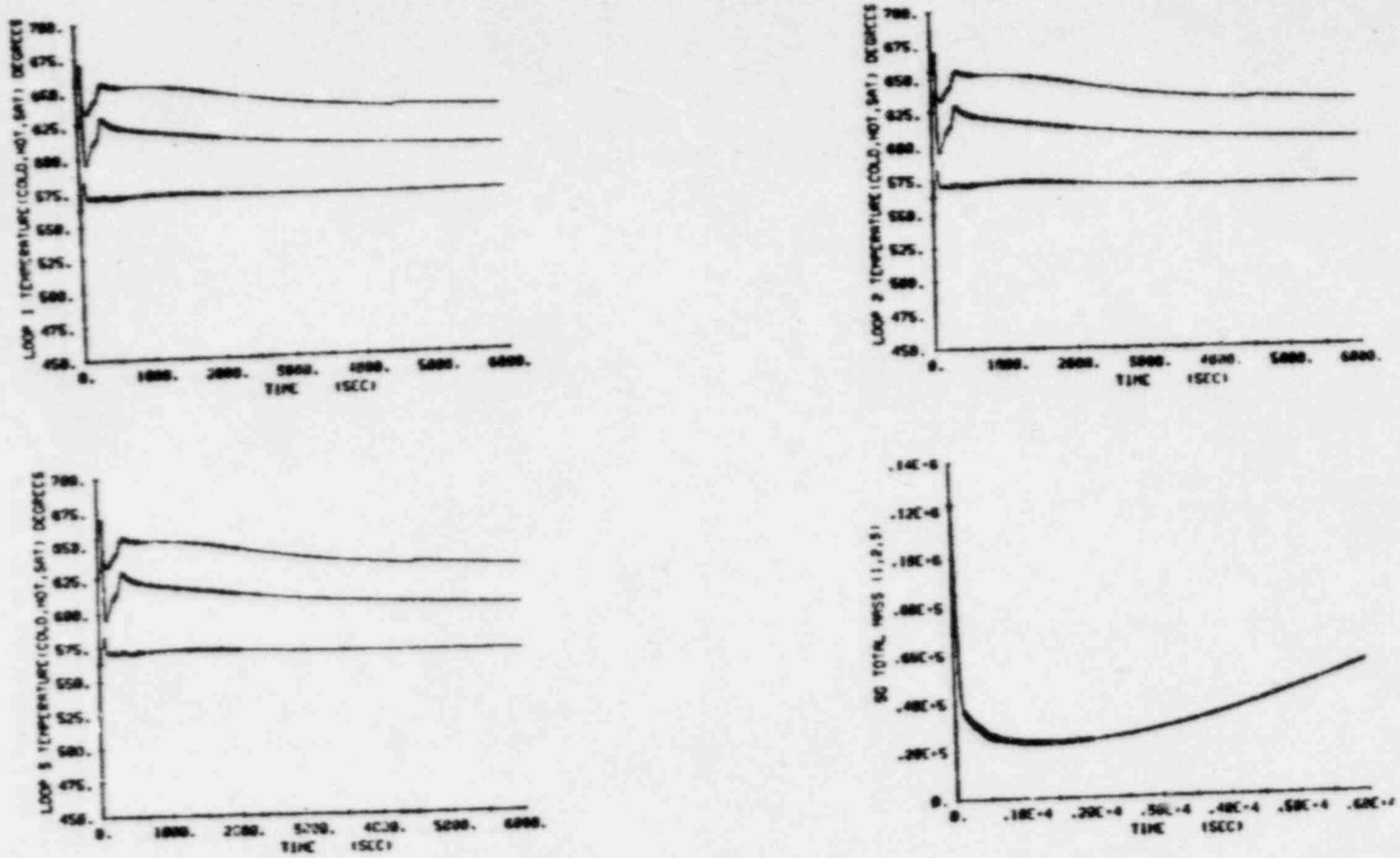


FIGURE 15.2-10B
CORE AVERAGE TEMPERATURE TRANSIENT
AND STEAM GENERATOR WATER VOLUME
TRANSIENT FOR LOSS OF NORMAL FEEDWATER
WITH LOSS OF OFFSITE POWER
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT
Four Loop Operation

INSERT A

Two cases are presented for a loss of normal feedwater event with four loops operating initially. The first is the case where offsite AC power is maintained and the second is the case where offsite AC power is lost which results in reactor coolant pump coastdown as described in Section 15.2.6.

INSERT B

The reactor trip on low-low narrow range level in any steam generator provides the necessary protection against a loss of normal feedwater.

The Auxiliary Feedwater System is started automatically as discussed in Section 15.2.6.2.

INSERT C

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

INSERT D

Figures 15.2.9B and 15.2.10B are for four loops in operation initially with a loss of offsite power.

INSERT E

Loss of normal feedwater flow with offsite power

EVENT	N-LOOP TIME
Main feedwater flow stops	10.0
Low steam generator water level trip	64.9
Rods begin to drop	66.9
Peak water level in the Pressurizer occurs	69.0
Four steam generators begin to receive 520 gpm of auxiliary feedwater	124.9
Core decay heat decreases to auxiliary feedwater heat removal capacity	3076
Peak water level in Pressurizer occurs	3236

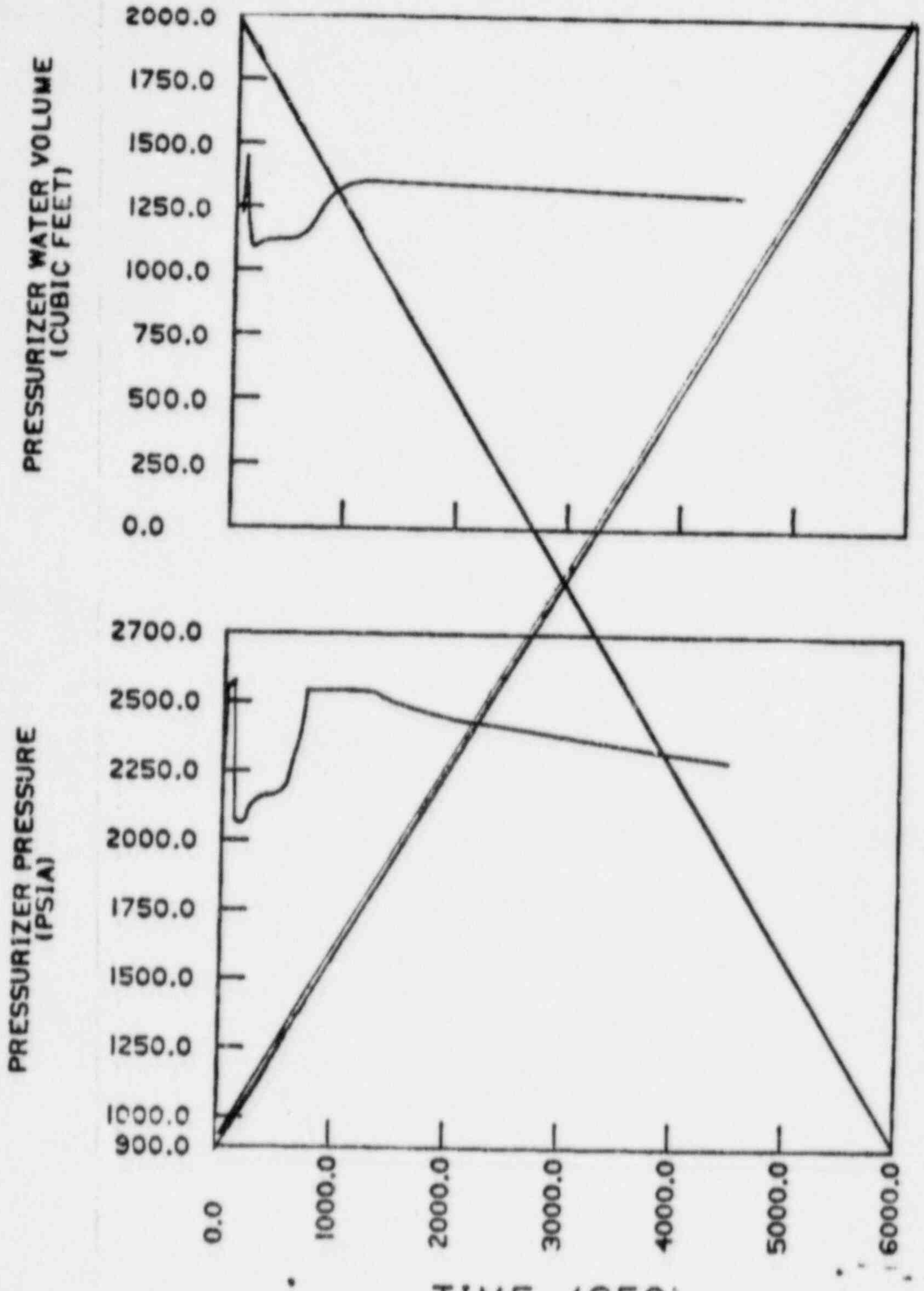


FIGURE 15.2-9
PRESSURIZER PRESSURE AND WATER VOLUME TRANSIENTS FOR LOSS OF NORMAL FEEDWATER
MILLSTONE NUCLEAR POWER STATION UNIT 3
FINAL SAFETY ANALYSIS REPORT

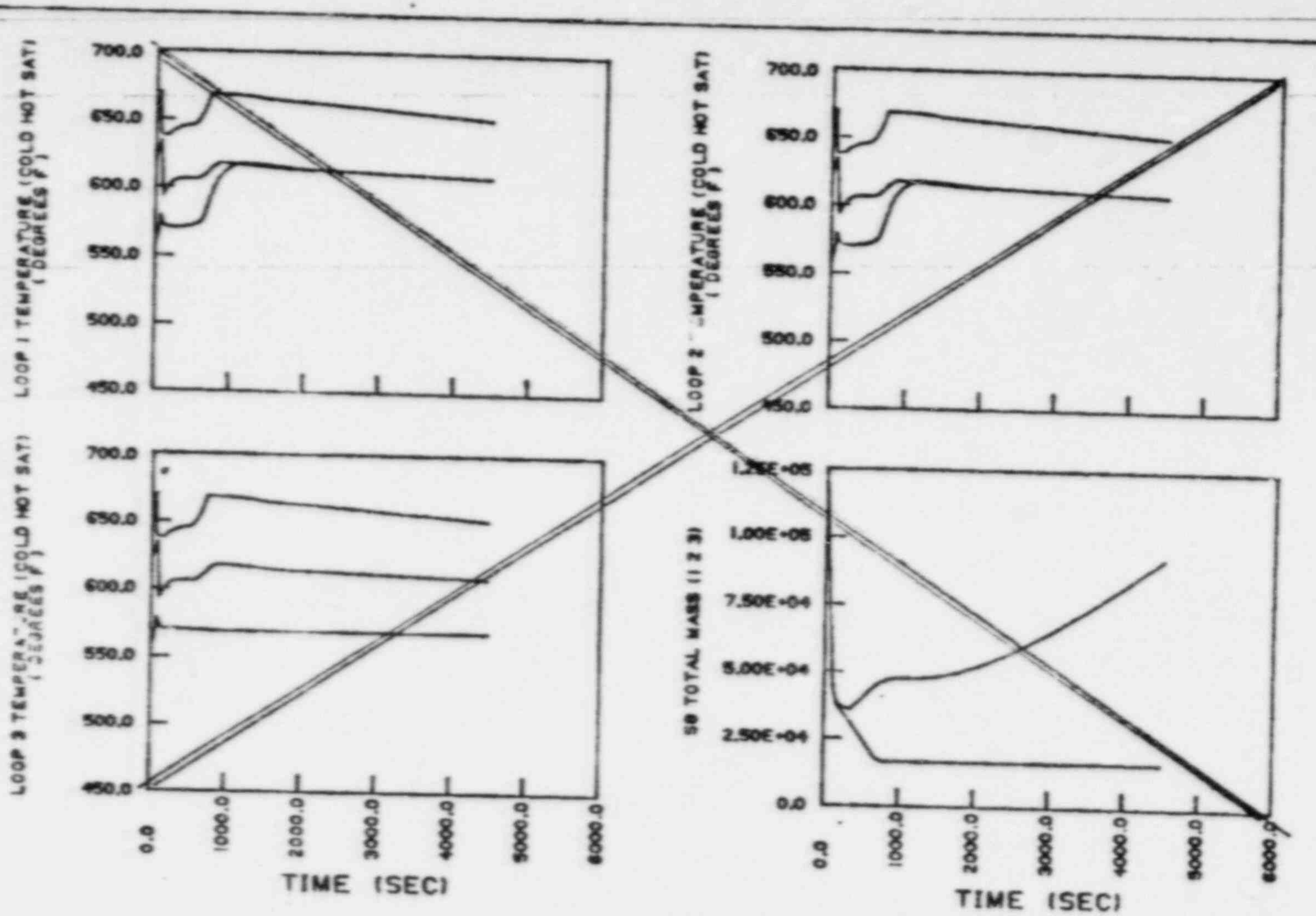
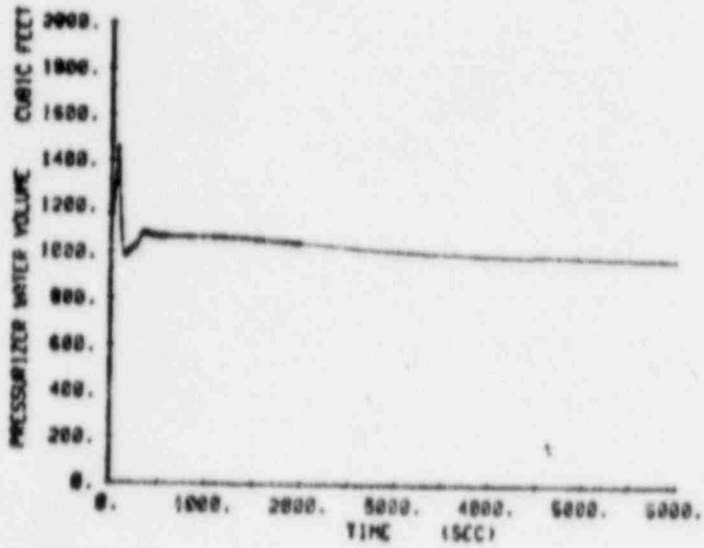


FIGURE 15.2-10
CORE AVERAGE TEMPERATURE TRANSIENT AND STEAM GENERATOR WATER VOLUME TRANSIENT FOR LOSS OF NORMAL FEEDWATER
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT



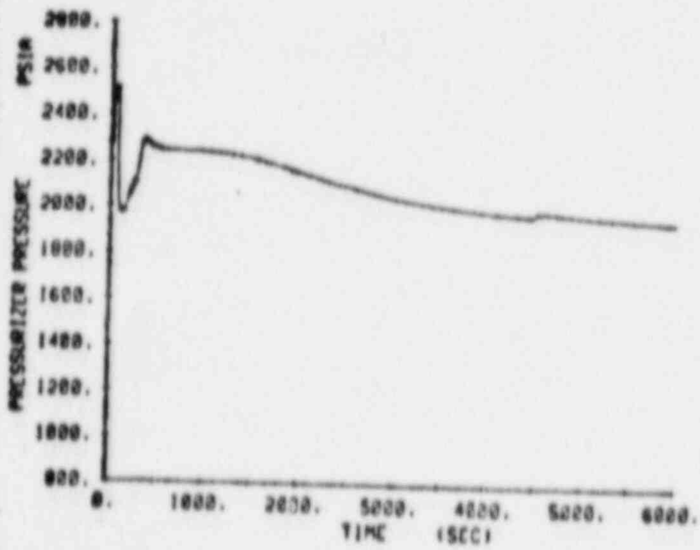


FIGURE 15.2-98
PRESSURIZER PRESSURE AND
WATER VOLUME TRANSIENTS
FOR LOSS OF NORMAL FEEDWATER
WITH LOSS OF OFFSITE POWER
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT
Four Loop Operation

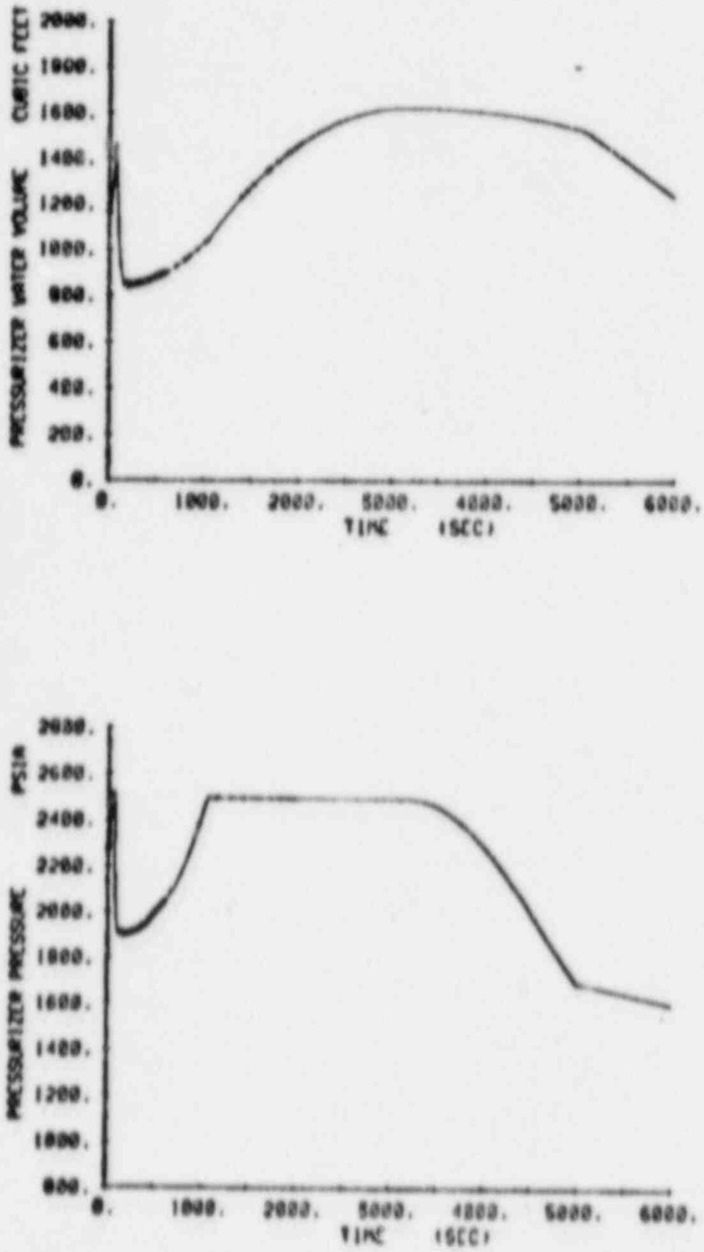


FIGURE 15.2-9A
PRESSURIZER PRESSURE AND
WATER VOLUME TRANSIENTS
FOR LOSS OF NORMAL FEEDWATER
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT
Four Loop Operation

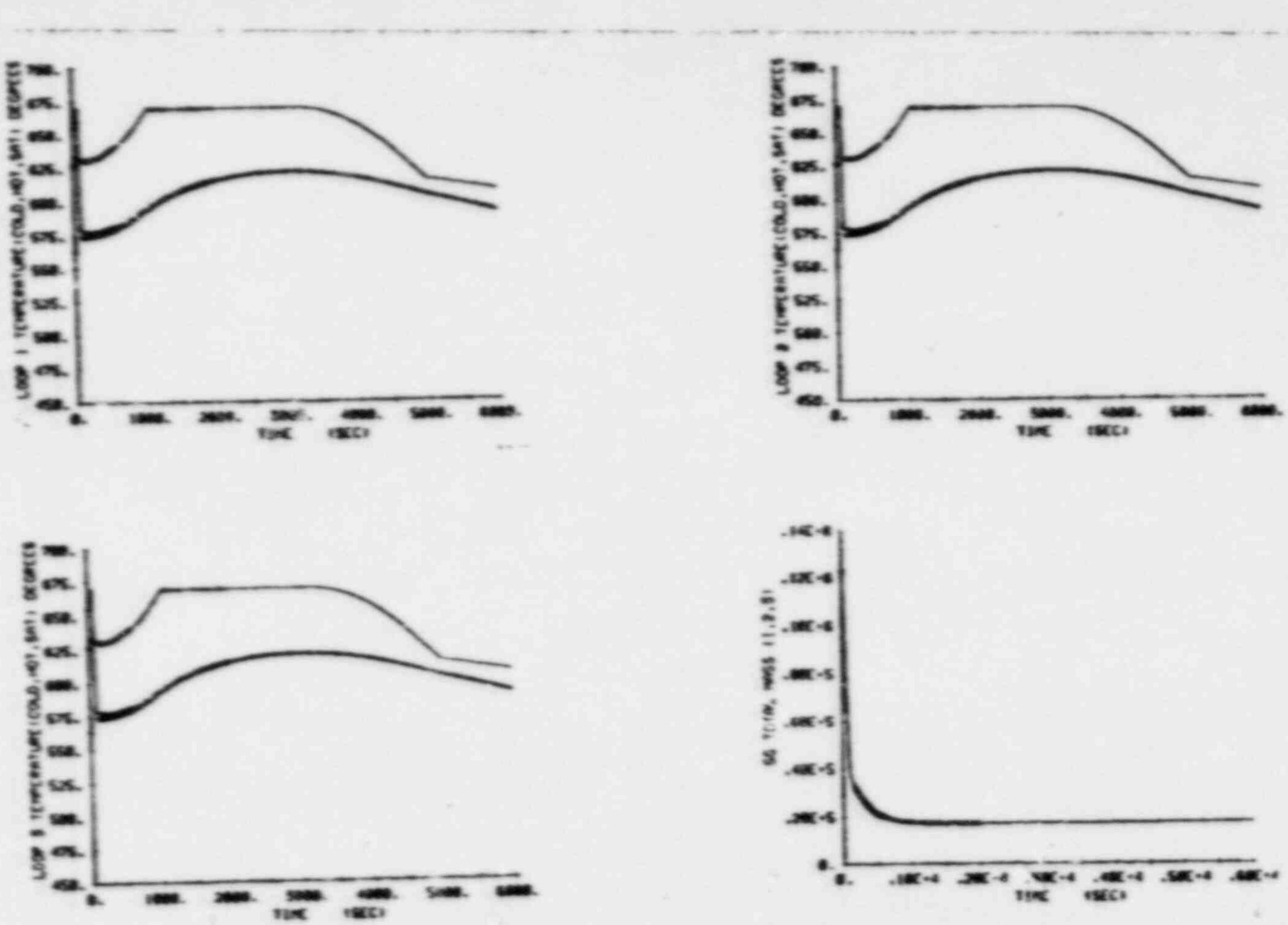


FIGURE 15.2-10
CORE AVERAGE TEMPERATURE TRANSIENT AND STEAM GENERATOR WATER VOLUME TRANSIENT FOR LOSS OF NORMAL FEEDWATER
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT
Four Loop Operation

ENCLOSURE 3

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