UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-320

METROPOLITAN EDISON COMPANY

JERSEY CENTRAL POWER & LIGHT COMPANY

PENNSYLVANIA ELECTRIC COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U.S. Nuclear Regulatory Commission (the Commission) has issued Amendment 6 to Facility Operating License No. DPR-73, issued to the Metropolitan Edison Company, Jersey Central Power & Light Company, and Pennsylvania Electric Company, for operation of the Three Mile Island Nuclear Station, Unit 2 (the facility), located in Dauphin County, Pennsylvania. The amendment is effective as of its date of issuance.

The license is amended by revising certain Technical Specifications to permit the following:

1. Alternate procedures for containment air lock seal leak rate testing
2. Plant operation with increased ultimate heat sink temperatures
3. Removal of most orifice rod assemblies and addition of retainers on the remaining orifice rod assemblies and on the burnable poison rod assemblies
4. Replacement of the main steam safety valves.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment.

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DATE ➤						

The Commission has determined that the granting of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with this action.

For further details with respect to this action, see (1) Amendment No. 6, to Facility Operating License No. DPR-73, and (2) the Commission's related safety evaluation supporting Amendment No. 6 to Facility Operating License No. DPR-73. These items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the State Library of Pennsylvania, Commonwealth and Walnut Streets, Harrisburg, Pennsylvania 17126.

Dated at Bethesda, Maryland this 17th day of August 1978.

FOR THE NUCLEAR REGULATORY COMMISSION

Original signed by
Steven A. Varga

Steven A. Varga, Chief
Light Water Reactors Branch 4
Division of Project Management

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DATE ▶	8/14/78	8/ /78	/78	/78		

NRC FORM 318 (9-76) NRCM 0240 ☆ U. S. GOVERNMENT PRINTING OFFICE: 1976 — 626-624

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

METROPOLITAN EDISON COMPANY
JERSEY CENTRAL POWER & LIGHT COMPANY
PENNSYLVANIA ELECTRIC COMPANY

DOCKET NO. 50-320

THREE MILE ISLAND NUCLEAR STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 6
License No. DPR-73

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The issuance of this license amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the license, as amended, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the amended Facility Operating License No. DPR-73 is hereby amended by changing the Technical Specifications as indicated in the attachment to this license amendment.

   Paragraph 2.C.(2) of amended Facility Operating License No. DPR-73 is hereby amended to read as follows:

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"2.C.(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 6 are hereby incorporated in the license. Metropolitan Edison Company shall operate the facility in accordance with the Technical Specifications."

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga

Steven A. Varga, Chief
Light Water Reactors Branch No. 4
Division of Project Management

Attachment:
Changes to the Technical
Specifications

Date of Issuance:

AUG 1 7 1978

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ATTACHMENT TO LICENSE AMENDMENT NO. 6

FACILITY OPERATING LICENSE NO. DPR-73

DOCKET NO. 50-320

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages as indicated. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change. The corresponding overleaf pages are also provided to maintain document completeness.

Pages

2-2
2-3
2-5
2-6
2-7
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B 2-1
B 2-2
B 2-3
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3/4 2-13
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3/4 7-3a (added)
B 3/4 7-1
3/4 6-5
3/4 7-17

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2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 The combination of the reactor coolant core outlet pressure and outlet temperature shall not exceed the safety limit shown in Figure 2.1-1.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the point defined by the combination of reactor coolant core outlet pressure and outlet temperature has exceeded the safety limit, be in HOT STANDBY within one hour.

REACTOR CORE

2.1.2 The combination of reactor THERMAL POWER and AXIAL POWER IMBALANCE shall not exceed the safety limit shown in Figure 2.1-2 for the various combinations of two, three and four reactor coolant pump operation.

APPLICABILITY: MODE 1.

ACTION:

Whenever the point defined by the combination of Reactor Coolant System flow, AXIAL POWER IMBALANCE and THERMAL POWER has exceeded the appropriate safety limit, be in HOT STANDBY within one hour.

REACTOR COOLANT SYSTEM PRESSURE

2.1.3 The Reactor Coolant System pressure shall not exceed 2750 psig.

APPLICABILITY: MODES 1, 2, 3, 4 and 5.

ACTION:

MODES 1 and 2 - Whenever the Reactor Coolant System pressure has exceeded 2750 psig, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within one hour.

MODES 3, 4 and 5 - Whenever the Reactor Coolant System pressure has exceeded 2750 psig, reduce the Reactor Coolant System pressure to within its limit within 5 minutes.

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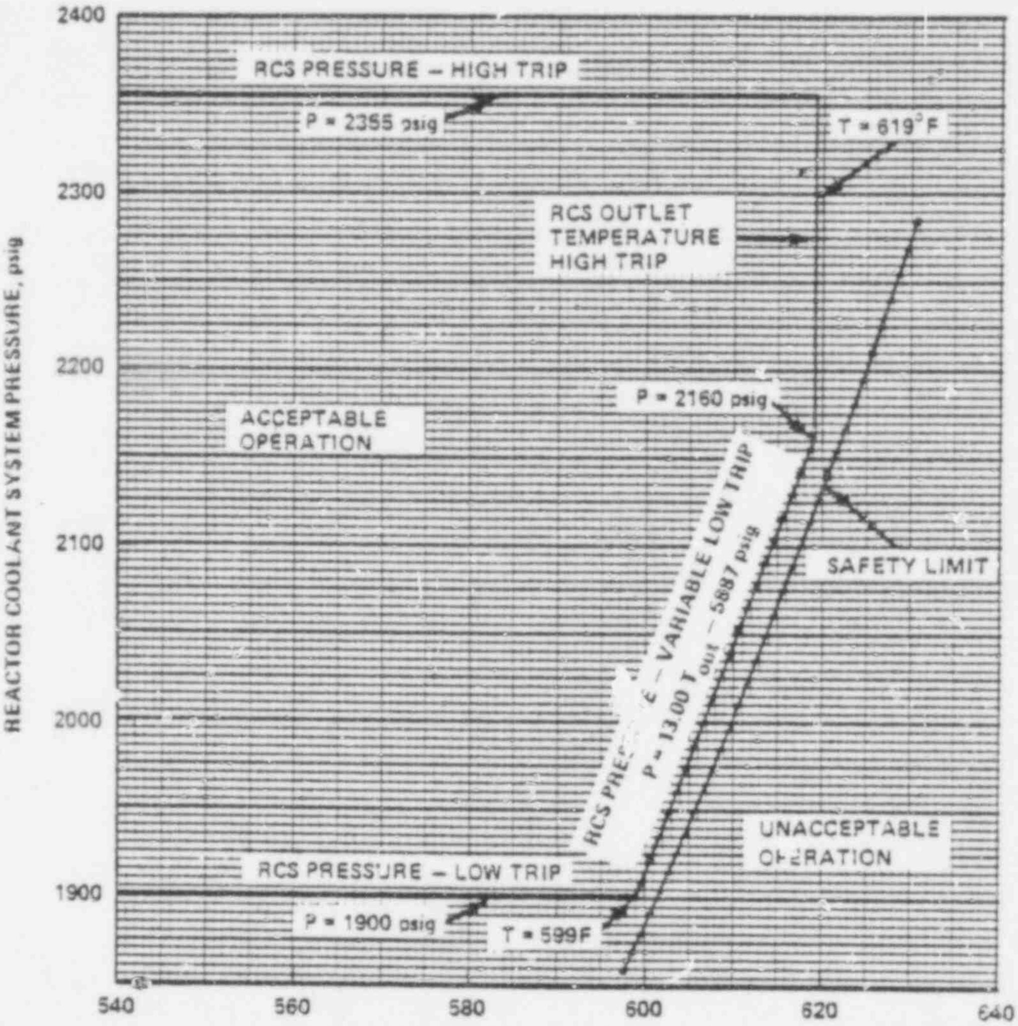
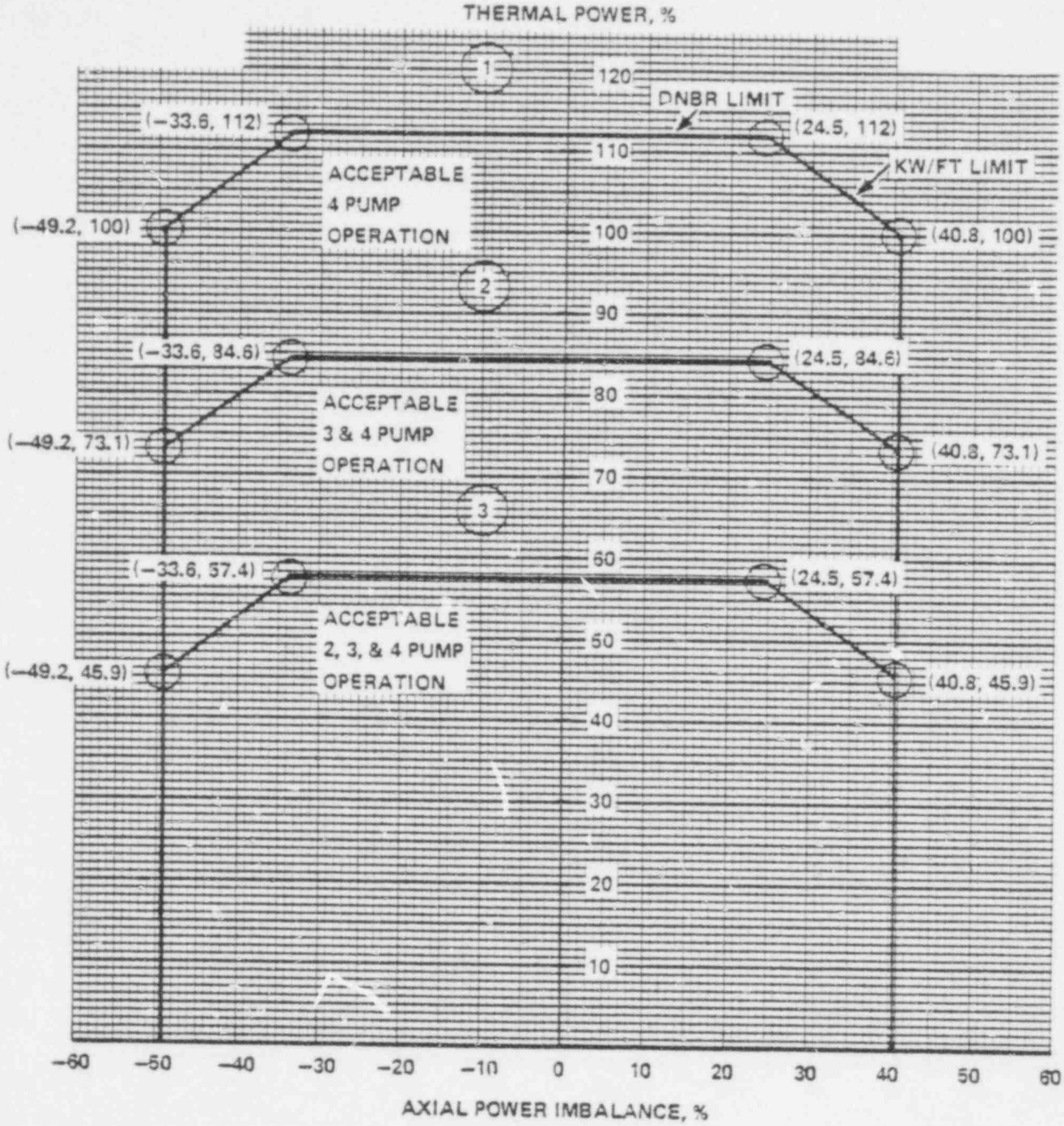


Figure 2.1–1 Reactor Core Safety Limit



CURVE	REACTOR COOLANT FLOW (GPM)
1	377,000
2	280,400
3	182,800

Figure 2.1–2 Reactor Core Safety Limits

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2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR PROTECTION SYSTEM SETPOINTS

2.2.1 The Reactor Protection System instrumentation setpoints shall be set consistent with the Trip Setpoint values shown in Table 2.2-1.

APPLICABILITY: As shown for each channel in Table 3.3-1.

ACTION:

With a Reactor Protection System instrumentation setpoint less conservative than the value shown in the Allowable Values column of Table 2.2-1, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1.1 until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

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TABLE 2.2-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT		TRIP SETPOINT	ALLOWABLE VALUES
1.	Manual Reactor Trip	Not Applicable	Not Applicable
2.	Nuclear Overpower	$\leq$ 105.5% of RATED THERMAL POWER with four pumps operating	$\leq$ 105.6% of RATED THERMAL POWER with four pumps operating#
		$\leq$ 78.1% of RATED THERMAL POWER with three pumps operating	$\leq$ 78.2% of RATED THERMAL POWER with three pumps operating#
		$\leq$ 50.9% of RATED THERMAL POWER with one pump operating in each loop	$\leq$ 51.0% of RATED THERMAL POWER with one pump operating in each loop#
3.	RCS Outlet Temperature-High	$\leq$ 619°F	$\leq$ 619.08°F#
4.	Nuclear Overpower Based on RCS Flow and AXIAL POWER IMBLANCE(1)	Trip Setpoint not to exceed the limit line of Figure 2.2-1.	Allowable Values not to exceed the limit line of Figure 2.2-2.#
5.	RCS Pressure-Low(1)	$\geq$ 1900 psig	$\geq$ 1899.0 psig*; $\geq$ 1891.5 psig **
6.	RCS Pressure-High	$\leq$ 2355 psig	$\leq$ 2356.0 psig*; $\leq$ 2363.5 psig **
7.	RCS Pressure-Variable Low(1)	$\geq$ (13.00 T_{out}°F - 5887) psig	$\geq$ (13.00 T_{out}°F - 5887.64) psig#

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TABLE 2.2-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTION UNIT	TRIP SETPOINT	ALLOWABLE VALUES
8. Nuclear Overpower based on Pump Monitors(1)	≤ 125% of RATED THERMAL POWER with three pumps operating	≤ 125% of RATED THERMAL POWER with three pumps operating#
	≤ 56.9% of RATED THERMAL POWER with one pump operating in each loop	≤ 57.18% of RATED THERMAL POWER with one pump operating in each loop#
	≤ 0% of RATED THERMAL POWER with two pump operating in one loop and no pump operating in the other loop	≤ 0.28% of RATED THERMAL POWER with two pumps operating in one loop and no pump operating in the other loop#
	≤ 0% of RATED THERMAL POWER with no pumps operating or only one pump operating	≤ 0.28% of RATED THERMAL POWER with no pumps operating or only one pump operating#
9. Reactor Containment Vessel	≤ 4 psig	≤ 4 psig#

(1) Trip may be manually bypassed when RCS pressure ≤ 1720 psig by actuating Shutdown Bypass provided that:

a. The Nuclear Overpower Trip Setpoint is ≤ 5% of RATED THERMAL POWER
b. The Shutdown Bypass RCS Pressure - High Trip Setpoint of ≤ 1720 psig is imposed, and
c. The Shutdown Bypass is removed when RCS Pressure > 1800 psig.

*Allowable value for Channel Functional Test.
**Allowable value for Channel Calibration.
#Allowable value for Channel Functional Test and Channel Calibration.

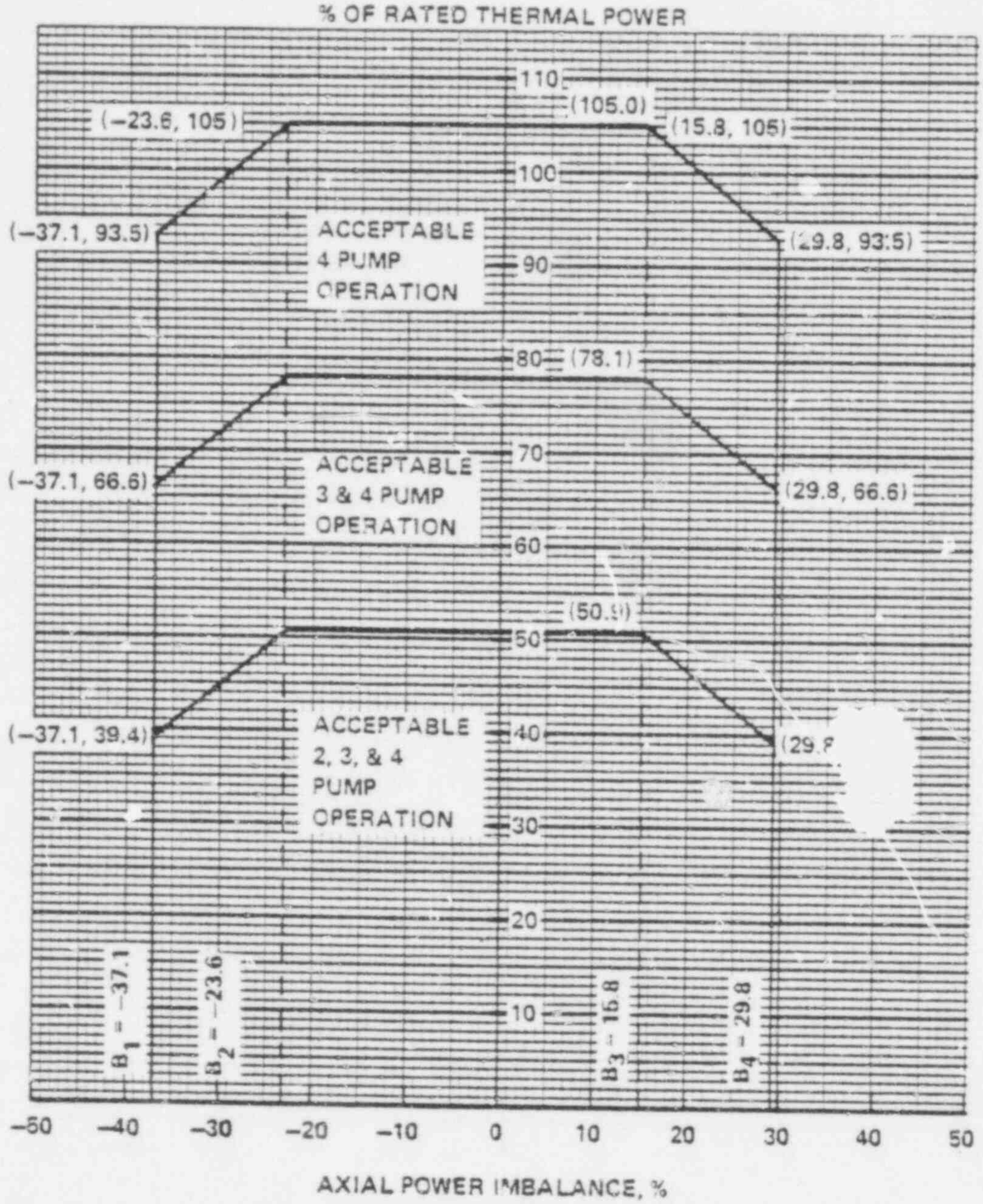

Figure 2.2−1 Trip Setpoint for Nuclear Overpower Based on RCS Flow and Axial Power Imbalance

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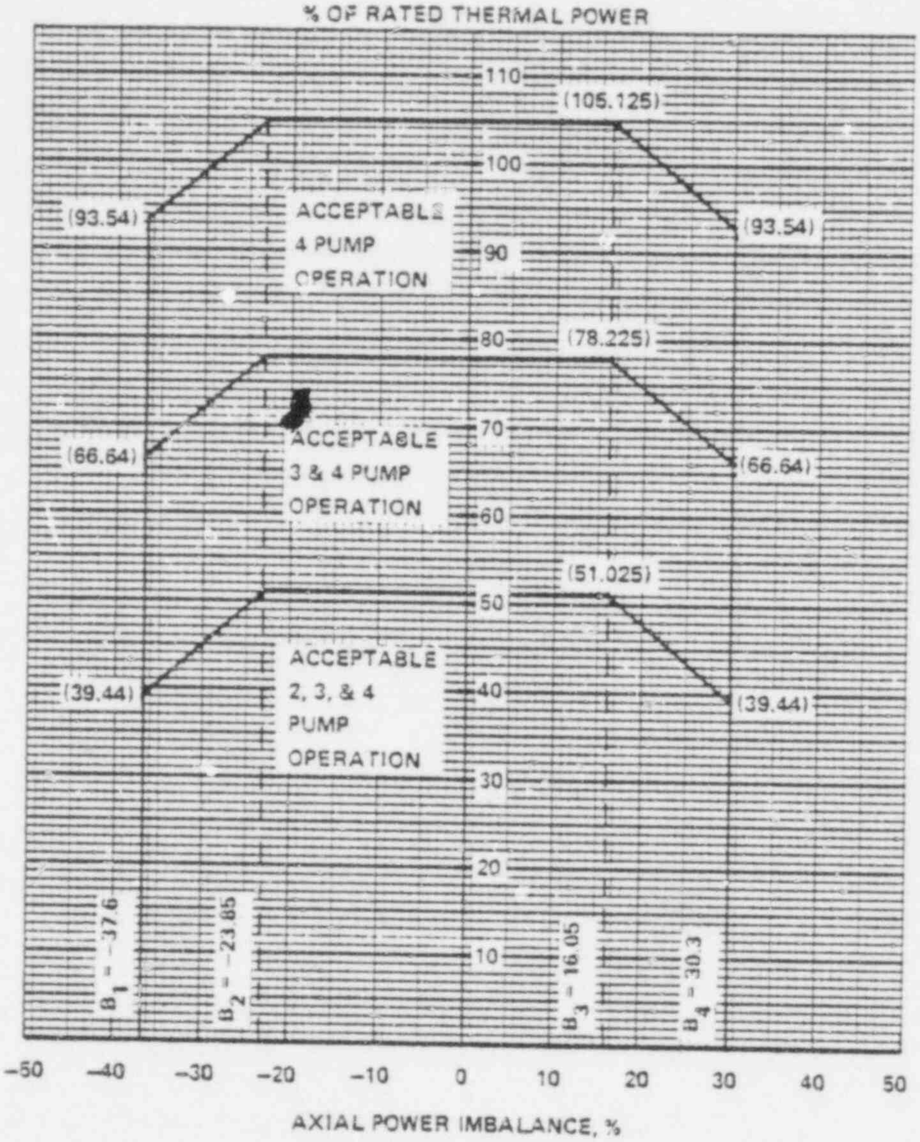

Figure 2.2-2 Allowable Value for Nuclear Overpower
Based on RCS Flow and Axial Power Imbalance

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2.1 SAFETY LIMITS

BASES

2.1.1 and 2.1.2 REACTOR CORE

The restrictions of this safety limit prevent overheating of the fuel cladding and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime would result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and Reactor Coolant Temperature and Pressure have been related to DNB through the BAW-2 DNB flux correlation. The DNB correlation has been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, DNBR, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB.

The minimum value of the DNBR during steady state operation, normal operational transients, and anticipated transients is limited to 1.30. This value corresponds to a 95 percent probability at a 95 percent confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all operating conditions.

The curve presented in Figure 2.1-1 represents the conditions at which a minimum DNBR of 1.30 is predicted for the maximum possible thermal power 112% when the reactor coolant flow is 377,000 gpm, which is 102% of the design flow rate for four operating reactor coolant pumps. This curve is based on the following nuclear power peaking factors with potential fuel densification effects:

$$F_Q^N = 2.67; \quad F_{\Delta H}^N = 1.78; \quad F_Z^N = 1.50$$

The design limit power peaking factors are the most restrictive calculated at full power for the range from all control rods fully withdrawn to minimum allowable control rod withdrawal, and form the core DNBR design basis.

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The reactor trip envelope appears to approach the safety limit more closely than it actually does because the reactor trip pressures are measured at a location where the indicated pressure is about 30 psi less than core outlet pressure, providing a more conservative margin to the safety limit.

The curves of Figure 2.1-2 are based on the more restrictive of two thermal limits and include the effects of potential fuel densification:

1. The 1.30 DNBR limit produced by a nuclear power peaking factor of $F_Q^N = 2.67$ or the combination of the radial peak, axial peak and position of the axial peak that yields no less than a 1.30 DNBR.

2. The combination of radial and axial peak that causes central fuel melting at the hot spot. The limit is 21.0 kw/ft.

Power peaking is not a directly observable quantity and therefore limits have been established on the basis of the reactor power imbalance produced by the power peaking.

The specified flow rates for curves 1, 2, and 3 of Figure 2.1-2 correspond to the expected minimum flow rates with four pumps, three pumps, and one pump in each loop, respectively.

The curve of Figure 2.1-1 is the most restrictive of all possible reactor coolant pump-maximum thermal power combinations shown in BASES Figure 2.1. The curves of BASES Figure 2 represent the conditions at which a minimum DNBR of 1.30 is predicted at the maximum possible thermal power for the number of reactor coolant pumps in operation or the local quality at the point of minimum DNBR is equal to 22%, whichever condition is more restrictive.

Using a local quality limit of 22% at the point of minimum DNBR as a basis for curve 3 of BASES Figure 2.1 is a conservative criterion even though the quality at the exit is higher than the quality at the point of minimum DNBR.

The DNBR as calculated by the BAW-2 DNB correlation continually increases from point of minimum DNBR, so that the exit DNBR is always higher. Extrapolation of the correlation beyond its published quality range of 22% is justified on the basis of experimental data.

For each curve of BASES Figure 2.1, a pressure-temperature point above and to the left of the curve would result in a DNBR greater than 1.30 or a local quality at the point of minimum DNBR less than 22% for that particular reactor coolant pump situation. The 1.30 DNBR curve for four pump operation is more restrictive than any other reactor coolant pump situation because any pressure/temperature point above and to the left of the four pump curve will be above and to the left of the other curves.

2.1.3 REACTOR COOLANT SYSTEM PRESSURE

The restriction of this Safety Limit protects the integrity of the Reactor Coolant System from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.

The reactor pressure vessel and pressurizer are designed to Section III of the ASME Boiler and Pressure Vessel Code which permits a maximum transient pressure of 110%, 2750 psig, of design pressure. The Reactor Coolant System piping, valves and fittings, are designed to ANSI B 31.7, 2/68 Edition. Reactor Coolant System valves are designed to ANSI B 16.5-1963, MSSP-61 and MSSP-66. The maximum transient pressure for the Reactor Coolant System valves is permitted by ASME to be 110%, 2750 psig of design pressure. The Safety Limit of 2750 psig is therefore consistent with the design criteria and associated code requirements.

The entire Reactor Coolant System is hydrotested at 3125 psig, 125% of design pressure, to demonstrate integrity prior to initial operation.

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BASES

2.2.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

The Reactor Protection System Instrumentation Trip Setpoint specified in Table 2.2-1 are the values at which the Reactor Trips are set for each parameter. The Trip Setpoints have been selected to ensure that the reactor core and reactor coolant system are prevented from exceeding their safety limits. Operation with a trip setpoint less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that each Allowable Value is equal to or less than the drift allowance assumed for each trip in the safety analyses.

The Shutdown Bypass provides for bypassing certain functions of the Reactor Protection System in order to permit control rod drive tests, zero power PHYSICS TESTS and certain startup and shutdown procedures. The purpose of the Shutdown Bypass RCS Pressure-High trip is to prevent normal operation with Shutdown Bypass activated. This high pressure trip setpoint is lower than the normal low pressure trip setpoint so that the reactor must be tripped before the bypass is initiated. The Nuclear Overpower Trip Setpoint of $\leq$ 5.0% prevents any significant reactor power from being produced. Sufficient natural circulation would be available to remove 5.0% of RATED THERMAL POWER if none of the reactor coolant pumps were operating.

Manual Reactor Trip

The Manual Reactor Trip is a redundant channel to the automatic Reactor Protection System instrumentation channels and provides manual reactor trip capability.

Nuclear Overpower

A Nuclear Overpower trip at high power level (neutron flux) provides reactor core protection against reactivity excursions which are too rapid to be protected by temperature and pressure protective circuitry.

During normal station operation, reactor trip is initiated when the reactor power level reaches 105.5% of rated power. Due to calibration and instrument errors, the maximum actual power at which a trip would be actuated could be 112%, which was used in the safety analysis.

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RCS Outlet Temperature - High

The RCS Outlet Temperature High trip $\leq$ 619°F prevents the reactor outlet temperature from exceeding the design limits and acts as a backup trip for all power excursion transients.

Nuclear Overpower Based on RCS Flow and AXIAL POWER IMBALANCE

The power level trip setpoint produced by the reactor coolant system flow is based on a flux-to-flow ratio which has been established to accomodate flow decreasing transients from high power where protection is not provided by the Nuclear Overpower Based on Pump Monitors channels.

The power level trip setpoint produced by the power-to-flow ratio provides both high power level and low flow protection in the event the reactor power level increases or the reactor coolant flow rate decreases. The power level setpoint produced by the power-to-flow ratio provides overpower DNB protection for all modes of pump operation. For every flow rate there is a maximum permissible power level, and for every power level there is a minimum permissible low flow rate. Typical power level and low flow rate combinations for the pump situations of Table 2.2-1 are as follows:

1. Trip would occur when four reactor coolant pumps are operating if power is 105.0% and reactor flow rate is 100%, or flow rate is 95.2% and power level is 100%.

2. Trip would occur when three reactor coolant pumps are operating if power is 78.2% and reactor flow rate is 74.4%, or flow rate is 71.4% and power is 75%.

3. Trip would occur when one reactor coolant pump is operating in each loop (total of two pumps operating) if the power is 51.0% and reactor flow rate is 48.5% or flow rate is 46.6% and the power level is 48.5%.

For safety calculations the maximum calibration and instrumentation errors for the power level were used.

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BASES

The AXIAL POWER IMBALANCE boundaries are established in order to prevent reactor thermal limits from being exceeded. These thermal limits are either power peaking kw/ft limits or DNBR limits. The AXIAL POWER IMBALANCE reduces the power level trip produced by the flux-to-flow ratio such that the boundaries of Figure 2.2-1 are produced. The flux-to-flow ratio reduces the power level trip and associated reactor power-reactor power-imbalance boundaries by 1.06% for a 1% flow reduction.

RCS Pressure - Low, High and Variable Low

The High and Low trips are provided to limit the pressure range in which reactor operation is permitted.

During a slow reactivity insertion startup accident from low power or a slow reactivity insertion from high power, the RCS Pressure-High setpoint is reached before the Nuclear Overpower Trip Setpoint. The trip setpoint for RCS Pressure-High, 2355 psig, has been established to maintain the system pressure below the safety limit, 2750 psig, for any design transient. The RCS Pressure-High trip is backed up by the pressurizer code safety valves for RCS over pressure protection, and is therefore set lower than the set pressure for these valves, 2500 psig. The RCS Pressure-High trip also backs up the Nuclear Overpower trip.

The RCS Pressure-Low, 1800 psig, and RCS Pressure-Variable Low, ($13.00\ T_{out}$ °F-5837) psig, Trip Setpoints have been established to maintain the DNB ratio greater than or equal to 1.30 for those design accidents that result in a pressure reduction. It also prevents reactor operation at pressures below the valid range of DNB correlation limits, protecting against DNB.

Due to the calibration and instrumentation errors, the safety analysis used a RCS Pressure-Variable Low Trip Setpoint of ($13.00\ T_{out}$ °F-5927) psig.

Nuclear Overpower Based on Pump Monitors

In conjunction with the power/imbalance/flow trips the Nuclear Overpower Based On Pump Monitors trip prevents the minimum core DNBR from decreasing below 1.30 by tripping the reactor due to the loss of reactor coolant pump(s). The pump monitors also restrict the power level for the number of pumps in operation.

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BASES

Reactor Containment Vessel Pressure - High

The Reactor Containment Vessel Pressure-High Trip Setpoint $\leq$ 4 psig, provides positive assurance that a reactor trip will occur in the unlikely event of a steam line failure in the containment vessel or a loss-of-coolant accident, even in the absence of a RCS Pressure -Low trip.

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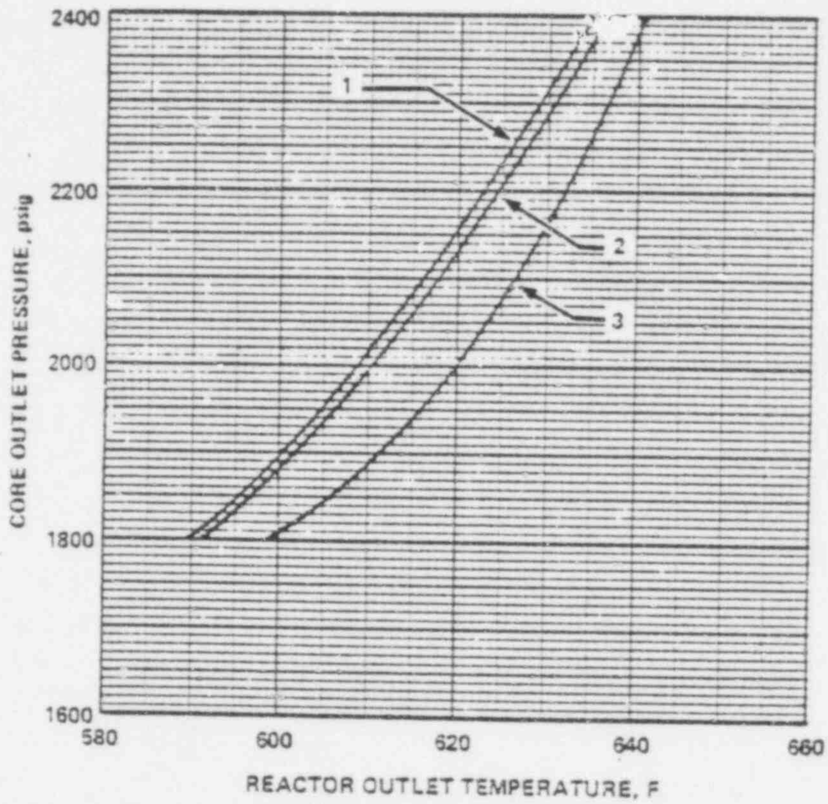

CURVE	REACTOR COOLANT FLOW (GPM)	POWER	PUMPS OPERATING (TYPE OF LIMIT)
1	377,000 (100%)	112%	FOUR PUMPS (DNBR LIMIT)
2	280,40[illegible] (74.4%)	84.6%	THREE PUMPS (DNBR LIMIT)
3	182,800 (48.5%)	57.4%	ONE PUMP IN EACH LOOP (QUALITY LIMIT)

Bases Figure 2.1 Pressure/Temperature Limits at Maximum Allowable Power for Minimum DNBR

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TABLE 3.2-2

DNB MARGIN

Parameter	Four Reactor Coolant Pumps Operating	Three Reactor Coolant Pumps Operating	One Reactor Coolant Pump Operating in Each Loop
	LIMITS		
Reactor Coolant Hot Leg Temperature, T_H °F	$\leq$ 609.3	$\leq$ 609.3[1]	$\leq$ 609.3
Reactor Coolant Pressure, psig[2]	$\geq$ 2060.4	$\geq$ 2056.4[1]	$\geq$ 2091.4
Reactor Coolant Flow Rate, gpm	$\geq$ 377,000	$\geq$ 280,400	$\geq$ 182,800

[1] Applicable to the loop with 2 Reactor Coolant Pumps Operating.

[2] Limit not applicable during either a THERMAL POWER ramp increase in excess of 5% of RATED THERMAL POWER per minute or a THERMAL POWER step increase of greater than 10% of RATED THERMAL POWER.

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TABLE 3.3-1 (Continued)

TABLE NOTATION

*With the control rod drive trip breakers in the closed position and the control rod drive system capable of rod withdrawal.

**When Shutdown Bypass is actuated.

#The provisions of Specification 3.0.4 are not applicable.

##High voltage to detector may be de-energized above 10^{-10} amps on both Intermediate Range channels.

(a) Trip may be manually bypassed when RCS pressure $\leq$ 1820 psig by actuating Shutdown Bypass provided that:

(1) The Nuclear Overpower Trip Setpoint is $\leq$ 5% of RATED THERMAL POWER.

(2) The Shutdown Bypass RCS Pressure--High Trip Setpoint of $\leq$ 1820 psig is imposed.

(3) The Shutdown Bypass is removed when RCS pressure > 1900 psig.

(b) Trip may be bypassed during testing pursuant to Special Test Exception 3.10.3.

ACTION STATEMENTS

ACTION 1 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and/or open the control rod drive trip breakers.

ACTION 2 - With the number of OPERABLE channels one less than the Total Number of Channels STARTUP and/or POWER OPERATION may proceed provided all of the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within one hour.

b. The Minimum Channels OPERABLE requirement is met; however, one additional channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.1.1.1,

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TABLE 3.3-1 (Continued)

ACTION STATEMENTS (Continued)

and the inoperable channel above may be bypassed for up to 30 minutes in any 24 hour period when necessary to test the trip breaker associated with the logic of the channel being tested per Specification 4.3.1.1.1.

c. Either, THERMAL POWER is restricted to $\leq$ 75% of RATED THERMAL POWER and the Nuclear Overpower Trip Setpoint is reduced to $\leq$ 85% of RATED THERMAL POWER within 4 hours or the QUADRANT POWER TILT is monitored at least once per 12 hours.

ACTION 3 - With the number of OPERABLE channels one less than the Total Number of Channels STARTUP and POWER OPERATION may proceed provided both of the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within one hour.

b. The Minimum Channels OPERABLE requirement is met; however, one additional channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.1.1.1, and the inoperable channel above may be bypassed for up to 30 minutes in any 24 hour period when necessary to test the trip breaker associated with the logic of the channel being tested per Specification 4.3.1.1.1.

ACTION 4 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement and with the THERMAL Power level:

a. $\leq$ 5% of RATED THERMAL POWER restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above 5% of RATED THERMAL POWER.

b. > 5% of RATED THERMAL POWER, POWER OPERATION may continue.

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TABLE 3.3-3 (Continued)

TABLE NOTATION

* Trip function may be bypassed in this MODE with RCS pressure below 1920 psig. Bypass shall be automatically removed when RCS pressure exceeds 1950 psig.

** 3 channels per Automatic Actuation Logic, Each R. B. Pressure High Channel trips one Safety Injection Channel and one R. B. Cooling & Isolation Channel.

*** 3 channels per Automatic Actuation Logic, R. B. Spray Valves are actuated by R. B. Cooling and Isolation.

**** Trip function may be bypassed in this mode with steam generator pressure $\leq$ 800 psig. Bypass shall be removed when steam generator pressure $\geq$ 800 psig.

\# The provisions of Specification 3.0.4 are not applicable.

TABLE 3.3-3 (Continued)

ACTION STATEMENTS

ACTION 9 - With the number of OPERABLE Channels one less than, the Total Number of Channels, restore the inoperable channel to OPERABE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION 10 - With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed until performance of the next required CHANNEL FUNCTIONAL TEST provided the inoperable channel is placed in the tripped condition within 1 hour.

ACTION 11 - With the number of OPERABLE channels one less than the Total Number of Channels, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the next 30 hours; however, one channel may be bypassed for up to 1 hour for surveillance testing per Specification 4.3.2.1.1.

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CONTAINMENT AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.1.3 Each containment air lock shall be OPERABLE with:

a. Both doors closed except when the air lock is being used for normal transit entry and exit through the containment, then at least one air lock door shall be closed.

b. An overall air lock leakage rate of $\leq 0.05\ L_a$ at P_a, 56.2 psig.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With an air lock inoperable, maintain at least one door closed; restore the air lock to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

a.* After each opening, except when the air lock is being used for multiple entries, then at least once per 72 hours, by verifying $\leq 0.01\ L_a$ seal leakage when the volume between the door seals is stabilized to a pressure of 10 psig.

b. At least once per 6 months by conducting an overall air lock leakage test at P_a, 56.2 psig, and by verifying that the overall air lock leakage rate is within its limit.

c. At least once per 6 months by verifying that only one door in each air lock can be opened at a time.

*Exemption to Appendix "J" of 10 CFR 50.

INTERNAL PRESSURE

LIMITING CONDITION FOR OPERATION

3.6.1.4 Primary containment internal pressure shall be maintained between -2 and +3 psig.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the containment internal pressure outside of the limits above, restore the internal pressure to within the limits within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.4 The primary containment internal pressure shall be determined to within the limits at least once per 12 hours.

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3/4.7 PLANT SYSTEMS

3/4.7.1 TURBINE CYCLE

SAFETY VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.1 All main steam line code safety valves shall be OPERABLE with lift settings as specified in Table 3.7-4.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With one or more main steam line code safety valves inoperable, operation in MODES 1, 2 and 3 may proceed provided, that within 4 hours, either the inoperable valve is restored to OPERABLE status or the Nuclear Overpower Trip Setpoint is reduced per Table 3.7-1; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.1 No additional Surveillance Requirements other than those required by Specification 4.0.5.

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TABLE 3.7-1

MAXIMUM ALLOWABLE NUCLEAR OVERPOWER TRIP SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES

Maximum Number of Inoperable Safety Valves on Any Steam Generator	Maximum Allowable Nuclear Overpower Trip Setpoint (Percent of RATED THERMAL POWER)
1	94.9
2	84.4
3	73.8
4	63.3
5	52.7

TABLE 3.7-4

STEAM LINE SAFETY VALVES PER STEAM GENERATOR

VALVE NUMBER	LIFT SETTING (± 1%)*	ORIFICE SIZE (square inches)
Steam Generator A		
a. MS-R21A	1050 psig	16
b. MS-R22A	1050 psig	16
c. MS-R26A	1050 psig	16
d. MS-R27A	1050 psig	16
e. MS-R23A	1065 psig	16
f. MS-R28A	1065 psig	16
g. MS-R24A	1075 psig	16
h. MS-R29A	1075 psig	16
i. MS-R25A	1102 psig	16
j. MS-R30A	1102 psig	16
Steam Generator B		
a. MS-R21B	1050 psig	16

*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

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TABLE 3.7-4 (Continued)

STEAM LINE SAFETY VALVES PER STEAM GENERATOR

VALVE NUMBER	LIFT SETTING (± 1%)*	ORIFICE SIZE (square inches)
b. MS-R22B	1050 psig	16
c. MS-R26B	1050 psig	16
d. MS-R26B	1050 psig	16
e. MS-R23B	1065 psig	16
f. MS-R28B	1065 psig	16
g. MS-R24B	1075 psig	16
h. MS-R29B	1075 psig	16
i. MS-R25B	1102 psig	16
j. MS-R30B	1102 psig	16

*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

EMERGENCY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.2 Three independent steam generator emergency feedwater pumps and associated flow paths shall be OPERABLE with:

a. Two emergency feedwater pumps, each capable of being powered from an OPERABLE emergency bus, and

b. One emergency feedwater pump capable of being powered from an OPERABLE steam supply system.

APPLICABILITY: MODES 1, 2 and 3*.

ACTION:

a. With one emergency feedwater system inoperable, restore the inoperable system to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.2 Each emergency feedwater system shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by:

1. Verifying that each steam turbine driven pump develops a discharge pressure of $\geq$ 1070 psig when the secondary steam supply pressure is greater than 200 psig.

*Automatic actuation of emergency feedwater system may be blocked when OTSG steam Pressure $\leq$ 800 psig.

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3/4.7.5 ULTIMATE HEAT SINK

LIMITING CONDITION FOR OPERATION

3.7.5.1 The ultimate heat sink shall be OPERABLE with:

a. A minimum water level at or above elevation 271 feet Mean Sea Level, USGS datum.

b. An average water temperature of $\leq$ 95°F*.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the requirements of the above specification not satisfied, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.5.1 The ultimate heat sink shall be determined OPERABLE:

a. At least once per 24 hours by verifying the average water temperature and water level to be within their limits.

b. By conducting hydrological surveys, and performing any needed dredging, in accordance with Section 2.4.9 of FSAR.

*The temperature Requirement of 95°F will become effective upon installation of impellers on the control Building Booster Pumps NR-P-2A/B which will increase the water flow by approximately 20% over the originally designed flow rate. Until that time, the ultimate heat sink average temperature shall be $\leq$ 90°.

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PLANT SYSTEMS

3/4.7.6 FLOOD PROTECTION

LIMITING CONDITION FOR OPERATION

3.7.6.1 Flood protection shall be provided for all safety related systems, components and structures when the water level of the Susquehanna River exceeds 301 feet Mean Sea Level USGS datum, at the river water intake structure of Three Mile Island Nuclear Station, Unit 1.

APPLICABILITY: At all times.

ACTION:

a. With the water level at the Unit 1 Intake Structure approaching 301 ft. Mean Sea Level USGS datum:

   1. Initiate patrol and inspection of the dikes surrounding the site for signs of deterioration such as undermining or excessive seepage.

   2. Inform the Station/Unit Superintendent and as directed by him:

      a) Prepare all flood panels and door seals for installation,

      b) Check all building floor drains and pumps to ensure proper operation,

      c) Commence daily soundings of the Intake Screen House Floor,

      d) Check all water tight doors to ensure proper operation,

      e) Fill all outdoor storage tanks to inhibit flotation, and

      f) Arrange for alternate supplies of diesel fuel oil and ensure fuel storage tanks are filled.

b. With the water level at the Unit 1 Intake Structure exceeding 301 ft. and approaching 302 ft. Mean Sea Level USGS datum:

   1. Ensure all door seals and flood panels are installed and all water tight doors are closed within 2 hours,

   2. Inform the Station/Unit Superintendent and prepare to place the Unit in HOT SHUTDOWN.

c. With the water level at the Unit 1 Intake Structure above 302 ft. Mean Sea Level datum:

   1. Be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

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3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES

The OPERABILITY of the main steam line code safety valves ensures that the secondary system pressure will be limited to within 110% of its design pressure during the most severe anticipated system operational transient. The maximum relieving capacity is associated with a turbine trip from 100% RATED THERMAL POWER coincident with an assumed loss of condenser heat sink (i.e., no steam bypass to the condenser).

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Vessel Code, 1968 Edition. The total relieving capacity for all valves on all of the steam lines is $\geq 14.68 \cdot 10^6$ lbs/hr which is 120 percent of the total secondary steam flow of 12.[illegible] $\cdot 10^6$ lbs/hr at 100% RATED THERMAL POWER.

STARTUP and/or POWER OPERATION is allowable with safety valves inoperable within the limitations of the ACTION requirements on the basis of the reduction in secondary system steam flow and THERMAL POWER required by the reduced reactor trip settings of the Nuclear Overpower channels. The reactor trip setpoint reductions are derived on the following bases:

$$SP = \frac{(X) - (Y)(V)}{X} \times 105.5$$

where:

SP = reduced Nuclear Overpower Trip Setpoint in percent of RATED THERMAL POWER

V = maximum number of inoperable safety valves per steam generator

105.5 = Nuclear Overpower Trip Setpoint specified in Table 2.2.1

X = Total relieving capacity of all safety valves per steam generator in lbs/hour

Y = Maximum relieving capacity of any one safety valve in lbs/hour

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BASES

3/4.7.1.2 EMERGENCY FEEDWATER SYSTEMS

The OPERABILITY of the emergency feedwater systems ensures that the Reactor Coolant System can be cooled down to less than 280°F from normal operating conditions in the event of a total loss of offsite power.

Each electric driven emergency feedwater pump is capable of delivering a total feedwater flow of 470 gpm at a pressure of 1133 psig to the entrance of the steam generators. Each steam driven auxiliary feedwater pump is capable of delivering a total feedwater flow of 940 gpm at a pressure of 1133 psig to the entrance of the steam generators. This capacity is sufficient to ensure that adequate feedwater flow is available to remove decay heat and reduce the Reactor Coolant System temperature to less than 280½F where the Decay Heat Removal System may be placed into operation.

3/4.7.1.3 CONDENSATE STORAGE TANKS

The OPERABILITY of the condensate storage tanks with the minimum water volume ensures that sufficient water is available for cooldown of the Reactor Coolant System to less than 280°F in the event of a total loss of offsite power or of the main feedwater system. The minimum water volume is sufficient to maintain the RCS at HOT STANDBY conditions for 13 hours with steam discharge to atmosphere concurrent with loss of offsite power. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

3/4.7.1.4 ACTIVITY

The limitations on secondary system specific activity ensure that the resultant offsite radiation dose will be limited to a small fraction of 10 CFR Part 100 limits in the event of a steam line rupture. This dose includes the effects of a coincident 1.0 GPM primary to secondary tube leak in the steam generator of the affected steam line. These values are consistent with the assumptions used in the safety analyses.

3/4.7.1.5 MAIN STEAM LINE ISOLATION VALVES

The OPERABILITY of the main steam line isolation valves ensures that steam generator blowdown will not continue indefinitely in the event of a steam line rupture. This restriction is required to 1) limit the positive

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