TABLE 3.3.2-2

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. PRIMARY CONTAINMENT ISOLATION		
a. Reactor Vessel Low Water Level		
1) Level 3	≥ 173.4 inches*	≥ 171.9 inches
2) Level 2	≥ 110.8 inches*	≥ 103.8 inches
3) Level 1	≥ 31.8 inches*	≥ 24.8 inches
b. Drywell Pressure - High	≤ 1.68 psig	≤ 1.88 psig
c. Main Steam Line		
1) Radiation - High	≤ 3.0 x full power background	≤ 3.6 x full power background
2) Pressure - Low	≥ 756 psig	≥ 736 psig
3) Flow - High	≤ ~~137.9% of rated flow/109.0~~ 115.4 psid	≤ ~~139.5% of rated flow/112.0~~ 118.4 psid
d. Main Steam Line Tunnel Temperature - High	≤ 200°F	≤ 206°F
e. Condenser Pressure - High	≤ 6.85 psia	≤ 7.05 psia
f. Turbine Bldg. Area Temperature - High	≤ 200°F	≤ 206°F
g. Deleted		
h. Manual Initiation	NA	NA

Revised, April 1992

TABLE 3.8.4.3-1 (Continued)

MOTOR-OPERATED VALVES THERMAL OVERLOAD PROTECTION

	VALVE NUMBER	SYSTEM(S) AFFECTED
	E41-F022	HPCI
	E41-F041	HPCI
	E41-F042	HPCI
	E41-F059	HPCI
	E41-F075	HPCI
	E41-F079	HPCI
	E41-F600	HPCI
7.	E51-F001	Reactor Core Isolation Cooling System (RCIC)
	E51-F002	RCIC
	E51-F007	RCIC
	E51-F008	RCIC
	E51-F010	RCIC
	E51-F012	RCIC
	E51-F013	RCIC
	E51-F019	RCIC
	E51-F022	RCIC
	E51-F029	RCIC
	E51-F031	RCIC
	E51-F045	RCIC
	E51-F046	RCIC
	E51-F059	RCIC
	E51-F062	RCIC
	E51-F084	RCIC
	E51-F095	RCIC
8.	G1154-F018	Drywell Floor Drain System
	G1154-F600	Drywell Floor Drain System
9.	G33-F001	Reactor Water Clean-Up System (RWCU)
	G33-F004	RWCU
10.	G51-F600	Torus Water Management System (TWMS)
	G51-F601	TWMS
	G51-F602	TWMS
	G51-F603	TWMS
	G51-F604	TWMS
	G51-F605	TWMS
	G51-F606	TWMS
	G51-F607	TWMS
11.	N11-F607	Main Steam System
	N11-F608	Main Steam System
	N11-F609	Main Steam System
	N11-F610	Main Steam System

ENCLOSURE 2 - PART 2

PROPOSED OPERATING LICENSE

AND

TECHNICAL SPECIFICATION REVISED PAGES

Revision 1, April 1992

(4) DECo, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use at any time any byproduct, source and special nuclear material such as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(5) DECo, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(6) DECo, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

DECo is authorized to operate the facility at reactor core power levels not in excess of 3430 megawatts thermal (100% power) in accordance with the conditions specified herein.

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A as revised through Amendment ____, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated into this license. DECo shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Antitrust Conditions

DECo shall abide by the agreements and interpretations between it and the Department of Justice relating to Article I, Paragraph 3 of the Electric Power Pool Agreement between Detroit Edison Company and Consumers Power Company as specified in a letter from DECo to the Director of Regulation, dated August 13, 1971, and the letter

2. Closed by at least one manual valve, blank flange, or deactivated automatic valve secured in its closed position, except as provided in Table 3.6.3-1 or Specification 3.6.3.

b. All primary containment equipment hatches are closed and sealed.

c. Each primary containment air lock is in compliance with the requirements of Specification 3.6.1.3.

d. The primary containment leakage rates are within the limits of Specification 3.6.1.2.

e. The suppression chamber is in compliance with the requirement of Specification 3.6.2.1.

f. The sealing mechanism associated with each primary containment penetration, e.g., welds, bellows, or O-rings, is OPERABLE.

g. The suppression chamber to reactor building vacuum breakers are in compliance with Specification 3.6.4.2.

THE PROCESS CONTROL PROGRAM

1.30 The PROCESS CONTROL PROGRAM (PCP) shall contain the provisions to assure that the SOLIDIFICATION of wet radioactive wastes results in a waste form with properties that meet the requirements of 10 CFR Part 61 and of low-level radioactive waste disposal sites. The PCP shall identify process parameters influencing SOLIDIFICATION, such as pH, oil content, H_2O content, solids content, ratio of solidification agent to waste and/or necessary additives for each type of anticipated waste, and the acceptable boundary conditions for the process parameters shall be identified for each waste type, based on laboratory scale and full scale testing or experience. The PCP shall also include an identification of conditions that must be satisfied, based on full scale testing, to assure that dewatering of bead resins, powdered resins, and filter sludges will result in volumes of free water, at the time of disposal, within the limits of 10 CFR Part 61 and of low-level radioactive waste disposal sites.

PURGE - PURGING

1.31 PURGE or PURGING is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

RATED THERMAL POWER

1.32 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3430 MWT.

TABLE 2.2.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Intermediate Range Monitor, Neutron Flux - High	≤ 120/125 divisions of of full scale	≤ 122/125 divisions of full scale
2. Average Power Range Monitor:		
a. Neutron Flux-Upscale, Setdown	≤ 15% of RATED THERMAL POWER	≤ 20% of RATED THERMAL POWER
b. Flow Biased Simulated Thermal Power-Upscale		
1) During two recirculation loop operation:		
a. Flow Biased	≤ 0.63 W+61.4%, with	≤ 0.63 W+64.3%, with
b. High Flow Clamped	a maximum of ≤ 113.5% of RATED THERMAL POWER	a maximum of ≤ 115.5% of RATED THERMAL POWER
2) During single recirculation loop operation:		
a. Flow Biased	≤ 0.63W+56.3%,**	≤ 0.63W+59.2%,**
b. High Flow Clamped	NA	NA
c. Fixed Neutron Flux-Upscale	≤ 118% of RATED THERMAL POWER	≤ 120% of RATED THERMAL POWER
d. Inoperative	NA	NA
3. Reactor Vessel Steam Dome Pressure - High	≤ 1093 psig	≤ 1113 psig
4. Reactor Vessel Low Water Level - Level 3	≥ 173.4 inches*	≥ 171.9 inches

*See Bases Figure B 3/4 3-1.

**During single recirculation loop operation, rather than adjusting the APRM Flow Biased Setpoints to comply with the single loop values, the gain of the APRMs may be adjusted for a period not to exceed 72 hours such that the final APRM readings are at least 5.1% of rated power greater than 100% times FRTP, provided that the adjusted APRM readings do not exceed 100% of RATED THERMAL POWER and a notice of adjustment is posted on the reactor control panel.

b. At least once per 31 days by:

1. Verifying the continuity of the explosive charge.

2. Determining that the concentration of boron in solution is within the limits of Figure 3.1.5-1 by chemical analysis.*

3. Verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

c. Demonstrating that, when tested pursuant to Specification 4.0.5, the minimum flow requirement of 41.2 gpm at a pressure of greater than or equal to 1215 psig is met.

d. At least once per 18 months during shutdown by:

1. Initiating one of the standby liquid control system loops, including an explosive valve, and verifying that a flow path from the pumps to the reactor pressure vessel is available by pumping demineralized water into the reactor vessel. The replacement charge for the explosive valve shall be from the same manufactured batch as the one fired or from another batch which has been certified by having one charge of that batch successfully fired. Both injection loops shall be tested in 36 months.

2. Demonstrating that the pump relief valve setpoint is less than or equal to 1400 psig and verifying that the relief valve does not actuate during recirculation to the test tank.

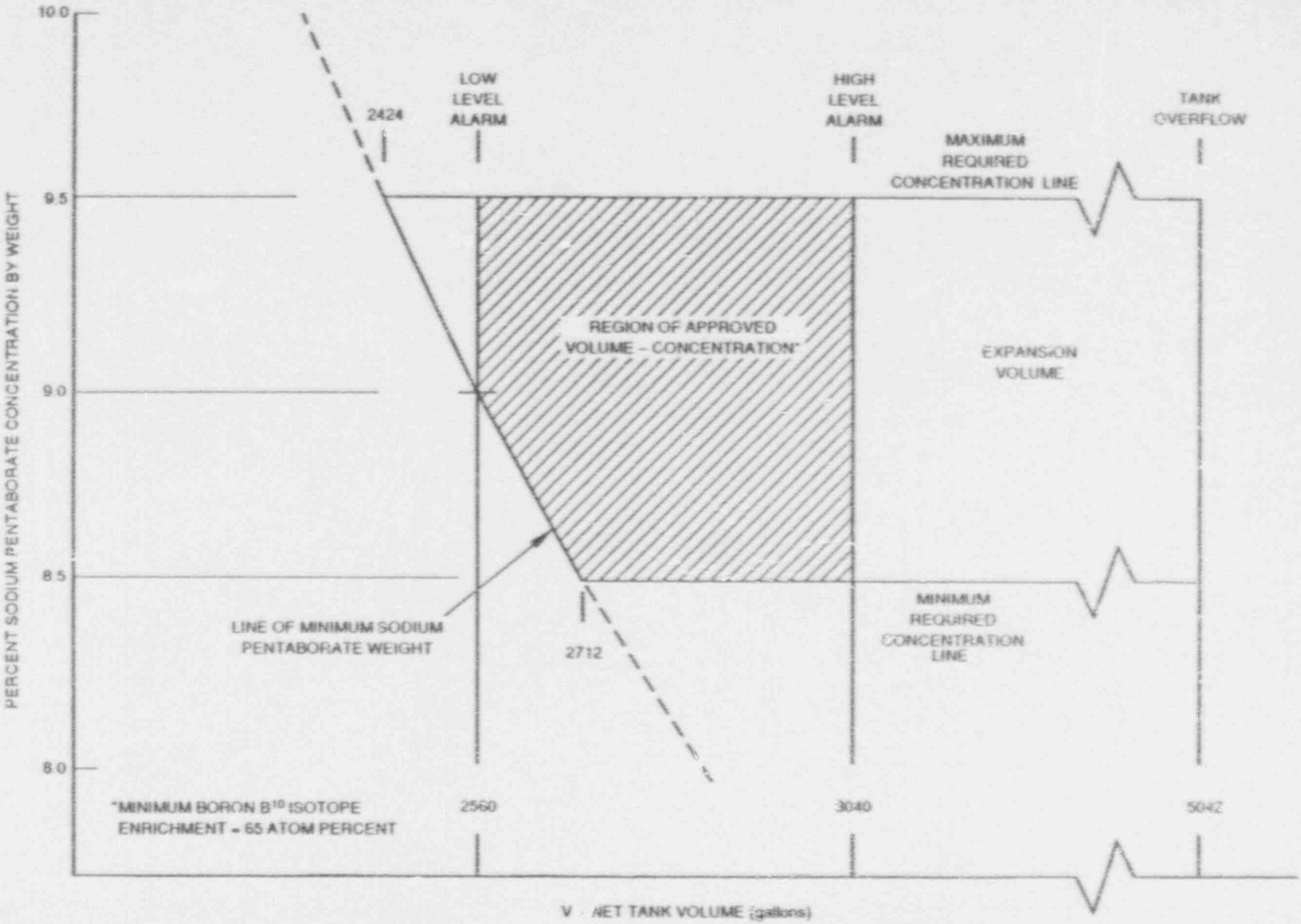
3. Demonstrating that all piping between the storage tank and the explosive valves is unblocked by pumping from the storage tank to the test tank and then draining and flushing the piping with demineralized water.**

4. Demonstrating that the storage tank heaters are OPERABLE for mixing by verifying the expected temperature rise of the sodium pentaborate solution in the storage tank after the heaters are energized.

e. At least once per 18 months sample and analyze the sodium pentaborate solution to verify that the Boron-10 Isotope enrichment exceeds 65 atom percent.

*This test shall also be performed anytime water or boron is added to the solution or when the solution temperature drops below the 48°F limit.

**This test shall also be performed whenever the solution temperature drops below the 48°F limit and may be performed by any series of sequential, overlapping or total flow path steps such that the entire flow path is included.



SODIUM PENTABORATE VOLUME/CONCENTRATION REQUIREMENTS

FIGURE 3.1.5-1

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION

3.2.1 All AVERAGE PLANAR LINEAR HEAT GENERATION RATES (APLHGRs) shall not exceed:

a. The MAPLHGR limit which has been approved for the respective fuel and lattice type as a function of the average planar exposure (as determined by the NRC approved methodology described in GESTAR-II), or

b. When hand calculations are required, the most limiting lattice type MAPLHGR limit as a function of the average planar exposure shown in the CORE OPERATING LIMITS REPORT (COLR) for the applicable fuel type.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER.

ACTION:

With an APLHGR exceeding the above limits, initiate corrective action within 15 minutes and restore APLHGR to within the required limits within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.1 All APLHGRs shall be verified to be equal to or less than the limits required by Specification 3.2.1:

a. At least once per 24 hours,

b. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and

c. Initially and at least once per 12 hours when the reactor is operating with a LIMITING CONTROL ROD PATTERN for APLHGR.

d. The provisions of Specification 4.0.4 are not applicable.

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

ACTION STATEMENTS

ACTION 1 - Be in at least HOT SHUTDOWN within 12 hours.

ACTION 2 - Verify all insertable control rods to be inserted in the core and lock the reactor mode switch in the Shutdown position within 1 hour.

ACTION 3 - Suspend all operations involving CORE ALTERATIONS and insert all insertable control rods within 1 hour.

ACTION 4 - Be in at least STARTUP within 6 hours.

ACTION 5 - Be in STARTUP with the main steam line isolation valves closed within 6 hours or in at least HOT SHUTDOWN within 12 hours.

ACTION 6 - Initiate a reduction in THERMAL POWER within 15 minutes and reduce turbine first stage pressure to ≤ 161.9 psig, equivalent to THERMAL POWER less than 30% of RATED THERMAL POWER, within 2 hours.

ACTION 7 - Verify all insertable control rods to be inserted within 1 hour.

ACTION 8 - Lock the reactor mode switch in the Shutdown position within 1 hour.

ACTION 9 - Suspend all operations involving CORE ALTERATIONS, and insert all insertable control rods and lock the reactor mode switch in the Shutdown position within 1 hour.

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

TABLE NOTATIONS

(a) A channel may be placed in an inoperable status for up to 6 hours for required surveillance without placing the trip system in the tripped condition provided at least one OPERABLE channel in the same trip system is monitoring that parameter.

(b) This function shall be automatically bypassed when the reactor mode switch is in the Run position.

(c) Unless adequate shutdown margin has been demonstrated per Specification 3.1.1, the "shorting links" shall be removed from the RPS circuitry prior to and during the time any control rod is withdrawn.*

(d) When the "shorting links" are removed, the Minimum OPERABLE Channels Per Trip System is 4 APRMs, 6 IRMs and per Specification 3.9.2, 2 SRMs.

(e) An APRM channel is inoperable if there are less than 2 LPRM inputs per level or less than 14 LPRM inputs to an APRM channel.

(f) This function is not required to be OPERABLE when the reactor pressure vessel head is removed per Specification 3.10.1.

(g) This function shall be automatically bypassed when the reactor mode switch is not in the Run position.

(h) This function is not required to be OPERABLE when PRIMARY CONTAINMENT INTEGRITY is not required.

(i) With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

(j) This function shall be automatically bypassed when turbine first stage pressure is ≤ 161.9 psig, equivalent to THERMAL POWER less than 30% of RATED THERMAL POWER.

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*Not required for control rods removed per Specification 3.9.10.1 or 3.9.10.2.

TABLE 3.3.2-2

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. PRIMARY CONTAINMENT ISOLATION		
a. Reactor Vessel Low Water Level		
1) Level 3	≥ 173.4 inches*	≥ 171.9 inches
2) Level 2	≥ 110.8 inches*	≥ 103.8 inches
3) Level 1	≥ 31.8 inches*	≥ 24.8 inches
b. Drywell Pressure - High	≤ 1.68 psig	≤ 1.88 psig
c. Main Steam Line		
1) Radiation - High	≤ 3.0 x full power background	≤ 3.6 x full power background
2) Pressure - Low	≥ 756 psig	≥ 736 psig
3) Flow - High	≤ 115.4 psid	≤ 118.4 psid
d. Main Steam Line Tunnel Temperature - High	≤ 200°F	≤ 206°F
e. Condenser Pressure - High	≤ 6.85 psia	≤ 7.05 psia
f. Turbine Bldg. Area Temperature - High	≤ 200°F	≤ 206°F
g. Deleted		
h. Manual Initiation	NA	NA

TABLE 3.3.6-2

CONTROL ROD BLOCK INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. ROD BLOCK MONITOR		
a. Upscale	As specified in the CORE OPERATING LIMITS REPORT	As specified in the CORE OPERATING LIMITS REPORT
b. Inoperative	NA	NA
c. Downscale	≥ 5% of RATED THERMAL POWER	≥ 3% of RATED THERMAL POWER
2. APRM		
a. Flow Biased Neutron Flux - High		
1) During two recirculation loop operation	≤ 0.63 W + 55.6%* with a maximum of 108%	≤ 0.63 W + 58.5%* with a maximum of 110%
2) During single recirculation loop operation	≤ 0.63 W + 50.5%#*	≤ 0.63 W + 53.4%#*
b. Inoperative	NA	NA
c. Downscale	≥ 5% of RATED THERMAL POWER	≥ 3% of RATED THERMAL POWER
d. Neutron Flux - Upscale, Setdown	≤ 12% of RATED THERMAL POWER	≤ 14% of RATED THERMAL POWER
3. SOURCE RANGE MONITORS		
a. Detector not full in	NA	NA
b. Upscale	≤ 1.0×10^5 cps	≤ 1.5×10^5 cps
c. Inoperative	NA	NA
d. Downscale	≥ 3 cps**	≥ 2 cps**

*The APRM rod block function is varied as a function of recirculation loop drive flow (W).

**May be reduced to ≥ 0.7 cps provided the signal-to-noise ratio ≥ 20.

#During single recirculation loop operation, rather than adjusting the APRM Flow Biased Setpoints to comply with the single loop values, the gain of the APRMs may be adjusted for a period not to exceed 72 hours such that the final APRM readings are at least 5.1% of rated power greater than 100% times FRIP, provided that the adjusted APRM readings do not exceed 100% of RATED THERMAL POWER and a notice of adjustment is posted on the reactor control panel.

3/4.4 REACTOR COOLANT SYSTEM
3/4.4.1 RECIRCULATION SYSTEM
RECIRCULATION LOOPS
LIMITING CONDITION FOR OPERATION

3.4.1.1 Two reactor coolant system recirculation loops shall be in operation.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2*.

ACTION:

a. With one reactor coolant system recirculation loop not in operation:

   1. Within 4 hours:

      a) Place the individual recirculation pump flow controller for the operating recirculation pump in the Manual mode.

      b) Reduce THERMAL POWER to less than or equal to 67.2% of RATED THERMAL POWER.

      c) Limit the speed of the operating recirculation pump to less than or equal to 75% of rated pump speed.

      d) Increase the MINIMUM CRITICAL POWER RATIO (MCPR) Safety Limit by 0.01 to 1.08 per Specification 2.1.2.

      e) Reduce the Average Power Range Monitor (APRM) Scram and Rod Block Trip Setpoints and Allowable Values to those applicable for single recirculation loop operation# per Specifications 2.2.1 and 3.3.6.

      f) Perform Surveillance Requirement 4.4.1.1.4 if THERMAL POWER is less than or equal to 30% of RATED THERMAL POWER or the recirculation loop flow in the operating loop is less than or equal to 50% of rated loop flow.

   2. The provisions of Specification 3.0.4 are not applicable.

   3. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

b. With no reactor coolant system recirculation loop in operation while in OPERATIONAL CONDITION 1, immediately place the Reactor Mode Switch in the SHUTDOWN position.

c. With no reactor coolant system recirculation loops in operation, while in OPERATIONAL CONDITION 2, initiate measures to place the unit in at least HOT SHUTDOWN within the next 6 hours.

---

*See Special Test Exception 3.10.4.

#APRM gain adjustments may be made in lieu of adjusting the APRM Flow Biased Setpoints to comply with the single loop values for a period of up to 72 hours.

SURVEILLANCE REQUIREMENTS

4.4.1.1.1 Each pump discharge valve shall be demonstrated OPERABLE by cycling each valve through at least one complete cycle of full travel during each STARTUP* prior to THERMAL POWER exceeding 25% of RATED THERMAL POWER.

4.4.1.1.2 Each pump MG set scoop tube mechanical and electrical stop shall be demonstrated OPERABLE with overspeed setpoints less than or equal to 110% and 107%, respectively, of rated core flow, at least once per 18 months.

4.4.1.1.3 With one reactor coolant system recirculation loop not in operation, at least once per 12 hours verify that:

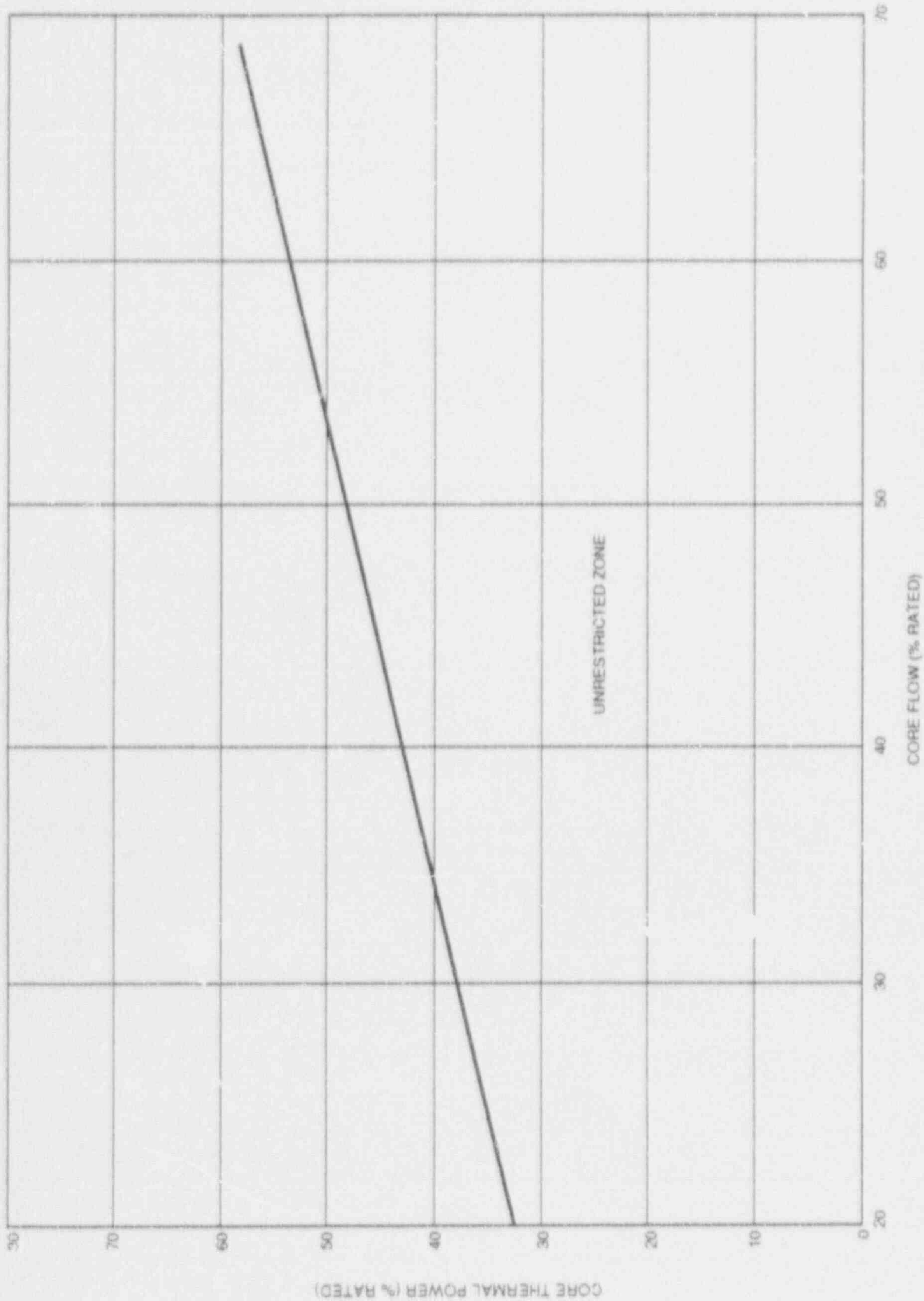
a. THERMAL POWER is less than or equal to 67.2% of RATED THERMAL POWER, and
b. The individual recirculation pump flow controller for the operating recirculation pump is in the Manual mode, and
c. The speed of the operating recirculation pump is less than or equal to 75% of rated pump speed.

4.4.1.1.4 With one reactor coolant system loop not in operation with THERMAL POWER less than or equal to 30% of RATED THERMAL POWER or with recirculation loop flow in the operating loop less than or equal to 50% of rated loop flow, verify the following differential temperature requirements are met within no more than 15 minutes prior to either THERMAL POWER increase or recirculation flow increase:

a. Less than or equal to 145°F between reactor vessel steam space coolant and bottom head drain line coolant, and
b. Less than or equal to 50°F between the reactor coolant within the loop not in operation and the coolant in the reactor pressure vessel**, and
c. Less than or equal to 50°F between the reactor coolant within the loop not in operation and the operating loop.**

*If not performed within the previous 31 days.

**Requirement does not apply when the recirculation loop not in operation is isolated from the reactor pressure vessel.



THERMAL POWER VERSUS CORE FLOW

FIGURE 3.4.1.4-1

3/4.4.2 SAFETY/RELIEF VALVES

SAFETY/RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.2.1 The safety valve function of at least 11 of the following reactor coolant system safety/relief valves shall be OPERABLE with the specified code safety valve function lift settings:*

- 5 safety/relief valves @ 1135 psig ±1%
- 5 safety/relief valves @ 1145 psig ±1%
- 5 safety/relief valves @ 1155 psig ±1%

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With the safety valve function of less than 11 of the above safety/relief valves OPERABLE, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

b. With one or more safety/relief valves stuck open, provided that suppression pool average water temperature is less than 95°F, close the stuck open safety/relief valve(s); if unable to close the stuck open valve(s) within 2 minutes or if suppression pool average water temperature is 95°F or greater, place the reactor mode switch in the Shutdown position.

c. With one or more safety/relief valve position indicators inoperable, restore the inoperable indicator(s) to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.2.1.1 The valve position indicator for each safety/relief valve shall be demonstrated OPERABLE with the pressure setpoint of each of the tail-pipe pressure switches verified to be 30 ± 5 psig by performance of a CHANNEL CALIBRATION at least once per 18 months.

4.4.2.1.2 At least 1/2 of the safety relief valves shall be set pressure tested at least once per 18 months, such that all 15 safety relief valves are set pressure tested at least once per 40 months.

*The lift setting pressure shall correspond to ambient conditions of the valves at nominal operating temperatures and pressures.

3.4.3.2 Reactor coolant system leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE.

b. 5 gpm UNIDENTIFIED LEAKAGE.

c. 25 gpm total leakage averaged over any 24-hour period.

d. 1 gpm leakage at a reactor coolant system pressure of 1045 ± 10 psig from any reactor coolant system pressure isolation valve specified in Table 3.4.3.2-1.

e. 2 gpm increase in UNIDENTIFIED LEAKAGE within any 4-hour period.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

b. With any reactor coolant system leakage greater than the limits in b and/or c, above, reduce the leakage rate to within the limits within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. With any reactor coolant system pressure isolation valve leakage greater than the above limit, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least one other closed manual, deactivated automatic, or check* valve, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

d. With one or more of the high/low pressure interface valve leakage pressure monitors shown in Table 3.4.3.2-2 inoperable, restore the inoperable monitor(s) to OPERABLE status within 7 days or verify the pressure to be less than the alarm setpoint at least once per 12 hours; restore the inoperable monitor(s) to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

e. With any reactor coolant system UNIDENTIFIED LEAKAGE increase greater than 2 gpm within any 4-hour period, identify the source of leakage increase as not service sensitive Type 304 or 316 austenitic stainless steel within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

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*Which has been verified not to exceed the allowable leakage limit at the last refueling outage or after the last time the valve was disturbed, whichever is more recent.

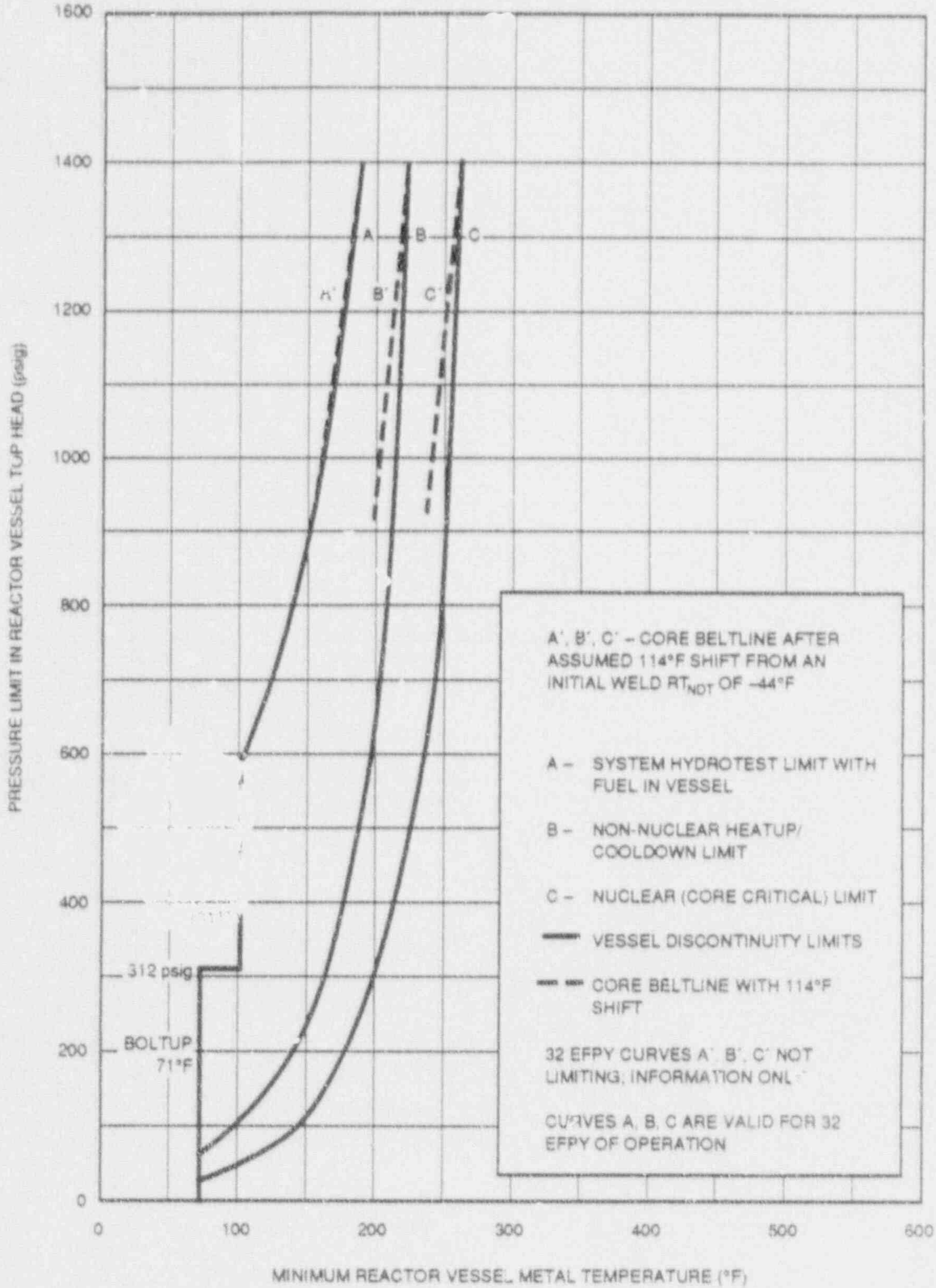


FIGURE 3.4.6.1-1

MINIMUM REACTOR PRESSURE VESSEL METAL TEMPERATURE VS. REACTOR VESSEL PRESSURE

REACTOR COOLANT SYSTEM

REACTOR STEAM DOME

LIMITING CONDITION FOR OPERATION

3.4.6.2 The pressure in the reactor steam dome shall be less than 1045 psig.

APPLICABILITY: OPERATIONAL CONDITIONS 1* and 2*.

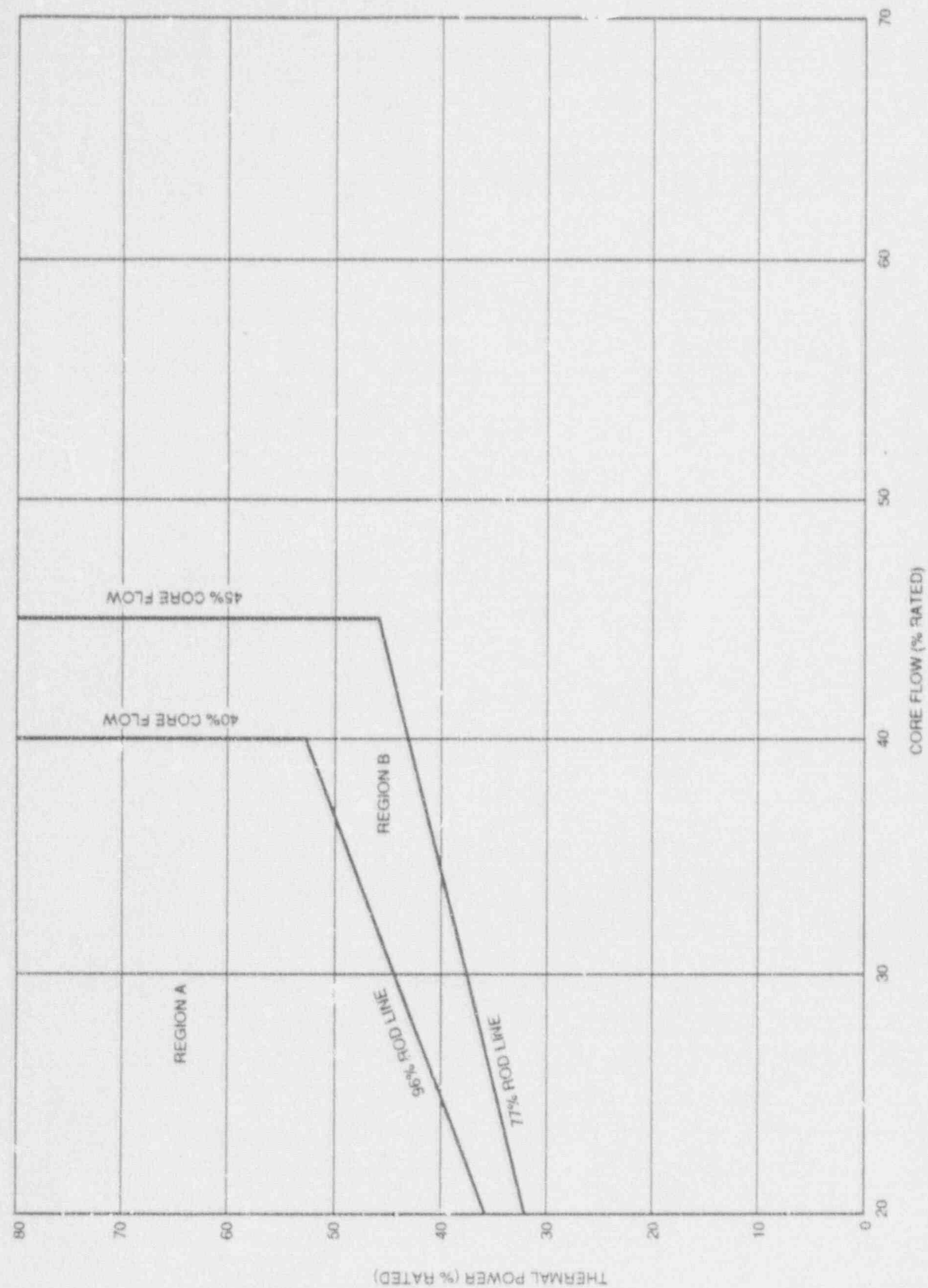
ACTION:

With the reactor steam dome pressure exceeding 1045 psig, reduce the pressure to less than 1045 psig within 15 minutes or be in at least HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.2 The reactor steam dome pressure shall be verified to be less than 1045 psig at least once per 12 hours.

*Not applicable during anticipated transients.



THERMAL POWER VERSUS CORE FLOW

FIGURE 3.4.10-1

3. For the HPCI system, verifying that the HPCI pump flow controller is in the correct position.

b. Verifying that, when pursuant to Specification 4.0.5:

1. The two CSS pumps in each subsystem together develop a flow of at least 6350 gpm against a test line pressure of greater than or equal to 270 psig, corresponding to a reactor vessel pressure of ≥ 100 psig.

2. Each LPCI pump in each subsystem develops a flow of at least 10,000 gpm against a test line pressure of ≥ 230 psig, corresponding to a reactor vessel to primary containment differential pressure of ≥ 20 psig.

3. The HPCI pump develops a flow of at least 5000 gpm in the test flow path with a system head corresponding to reactor vessel operating pressure including injection line losses when steam is being supplied to the turbine at 1025 +20, -80 psig.*

c. At least once per 18 months:

1. For the CSS, the LPCI system, and the HPCI system, performing a system functional test which includes simulated automatic actuation of the system throughout its emergency operating sequence and verifying that each automatic valve in the flow path actuates to its correct position. Actual injection of coolant into the reactor vessel may be excluded from this test.

2. For the HPCI system, verifying that:

a) The system develops a flow of at least 5000 gpm in the test flow path with a system head corresponding to reactor vessel operating pressure including injection line losses when steam is being supplied to the turbine at 165 + 50, -0 psig.*

b) The suction for the HPCI system is automatically transferred from the condensate storage tank to the suppression chamber on a condensate storage tank water level - low signal and on a suppression chamber - water level high signal.

3. Performing a CHANNEL CALIBRATION of the CSS and the LPCI system discharge line "keep filled" alarm instrumentation.

4. Performing a CHANNEL CALIBRATION of the CSS header ΔP instrumentation and verifying the setpoint to be ≤ the allowable value of 1.0 psid.

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*The provisions of Specification 4.0.4 are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test.

PLANT SYSTEMS

3/4.7.4 REACTOR CORE ISOLATION COOLING SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.4 The reactor core isolation cooling (RCIC) system shall be OPERABLE with an OPERABLE flow path capable of taking suction from the suppression pool and transferring the water to the reactor pressure vessel.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3 with reactor steam dome pressure greater than 150 psig.

ACTION:

With the RCIC system inoperable, operation may continue provided the HPCI system is OPERABLE; restore the RCIC system to OPERABLE status within 14 days, otherwise be in at least HOT SHUTDOWN within the next 12 hours and reduce reactor steam dome pressure to less than or equal to 150 psig within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.7.4 The RCIC system shall be demonstrated OPERABLE:

a. At least once per 31 days by:

1. Verifying by venting at the high point vents that the system piping from the pump discharge valve to the system isolation valve is filled with water.

2. Verifying that each valve (manual, power-operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

3. Verifying that the pump flow controller is in the correct position.

b. At least once per 92 days by verifying that the RCIC pump develops a flow of greater than or equal to 600 gpm in the test flow path with a system head corresponding to reactor vessel operating pressure including injection line losses when steam is being supplied to the turbine at 1025 + 20, - 80 psig.*

*The provisions of Specification 4.0.4 are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test.

TABLE 3.8.4.3-1 (Continued)

MOTOR-OPERATED VALVES THERMAL OVERLOAD PROTECTION

VALVE NUMBER	SYSTEM(S) AFFECTED
E41-F022	HPCI
E41-F041	HPCI
E41-F042	HPCI
E41-F059	HPCI
E41-F075	HPCI
E41-F079	HPCI
E41-F600	HPCI
7. E51-F001	Reactor Core Isolation Cooling System (RCIC)
E51-F002	RCIC
E51-F007	RCIC
E51-F008	RCIC
E51-F010	RCIC
E51-F012	RCIC
E51-F013	RCIC
E51-F019	RCIC
E51-F022	RCIC
E51-F029	RCIC
E51-F031	RCIC
E51-F045	RCIC
E51-F046	RCIC
E51-F059	RCIC
E51-F062	RCIC
E51-F084	RCIC
E51-F095	RCIC
8. G1154-F018	Drywell Floor Drain System
G1154-F600	Drywell Floor Drain System
9. G33-F001	Reactor Water Clean-Up System (RWCU)
G33-F004	RWCU
10. G51-F600	Torus Water Management System (TWMS)
G51-F601	TWMS
G51-F602	TWMS
G51-F603	TWMS
G51-F604	TWMS
G51-F605	TWMS
G51-F606	TWMS
G51-F607	TWMS
11. N11-F607	Main Steam System
N11-F608	Main Steam System
N11-F609	Main Steam System
N11-F610	Main Steam System

3/4.1.5 STANDBY LIQUID CONTROL SYSTEM

The design objective of the Standby Liquid Control (SLC) System is two fold. One objective is to provide backup capability for bringing the reactor from full power to a cold, Xenon-free shutdown, assuming that the withdrawn control rods remain fixed in the rated power pattern. The second objective of the SLC System is to meet the requirement of the ATWS Rule, specifically 10 CFR 50.62 paragraph (c)(4) which states that, in part:

"Each boiling water reactor must have standby liquid control system (SLCS) with a minimum flow capacity and boron content equivalent in control capacity to 86 gallons per minute of 13 weight percent sodium pentaborate solution."

The SLC System uses enriched Boron-10 (contained in the Sodium pentaborate solution) to comply with 10 CFR 50.62 paragraph (c)(4). The methods used to determine compliance with the ATWS Rule are in accordance with Reference 2.

To meet both objectives, it is necessary to inject a minimum quantity of 2560 net gallons of 65 atom percent Boron-10 enriched sodium pentaborate in a solution having a concentration of no less than 9.0 weight percent (see Figure 3.1.5-1 for equivalent volumes and concentration ranges). The equivalent concentration of natural boron required to shutdown the reactor is 720 parts per million (ppm) in the 70°F moderator, including the Recirculation loops and with the RHR Shutdown Cooling Subsystems in operation. In addition to this, a 25 percent margin is provided to allow for leakage and imperfect mixing (900 ppm). The pumping rate of 41.2 gpm provides a negative reactivity insertion rate over the permissible sodium pentaborate solution volume range, which adequately compensates for the positive reactivity effects due to moderator temperature reduction and xenon decay during shutdown. The temperature requirement is necessary to ensure that the sodium pentaborate remains in solution.

With redundant pumps and explosive injection valves and with a highly reliable control rod scram system, operation of the reactor is permitted to continue for short periods of time with the system inoperable or for longer periods of time with one of the redundant components inoperable. The SLC tank heaters are only required when mixing sodium pentaborate and/or water to establish the required solution operating parameters during additions to the SLC tank. Normal operation of the SLCS does not depend on these tank heaters to maintain the solution above its saturation temperature. Technical requirements have been placed on the tank heater circuit breakers to ensure that their failure will not degrade other SLC components (see Specification 3/4.8.4.5).

Surveillance requirements are established on a frequency that assures a high reliability of the system. Once the solution is established, boron concentration will not vary unless more boron or water is added, thus a check on the temperature and volume once each 24 hours assures that the solution is available for use. Analysis of Boron-10 enrichment each 18 months provides sufficient assurance that the minimum enrichment of Boron-10 will be maintained.

BASES

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE (Continued)

Power and flow dependent adjustments are provided in the COLR to assure that the fuel thermal-mechanical design criteria are preserved during abnormal transients initiated from off-rated conditions.

3/4.2.3 MINIMUM CRITICAL POWER RATIO (Continued)

Details on how evaluations are performed, on the methods used, and how the MCPR limit is adjusted for operation at less than rated power and flow conditions are given in References 1 and 3 and the CORE OPERATING LIMITS REPORT.

At THERMAL POWER levels less than or equal to 25 percent of RATED THERMAL POWER, the reactor will be operating at minimum recirculation pump speed and the moderator void content will be very small. For all designated control rod patterns which may be employed at this point, operating plant experience indicates that the resulting MCPR value is in excess of requirements by a considerable margin. During initial startup testing of the plant, a MCPR evaluation will be made at 25 percent of RATED THERMAL POWER level with minimum recirculation pump speed. The MCPR margin will thus be demonstrated such that future MCPR evaluation below this power level will be shown to be unnecessary. The daily requirement for calculating MCPR when THERMAL POWER is greater than or equal to 25 percent of RATED THERMAL POWER is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement for calculating MCPR when a limiting control rod pattern is approached ensures that MCPR will be known following a change in THERMAL POWER or power shape, regardless of magnitude, that could place operation at a thermal limit.

3.4.2.4 LINEAR HEAT GENERATION RATE

The thermal expansion rate of UO_2 pellets and Zircalloy cladding are different in that, during heatup, the fuel pellet could come into contact with the cladding and create stress. If the stress exceeds the yield stress of the cladding material, the cladding will crack. The LHGR limit assures that at any exposure, 1% plastic strain on the clad is not exceeded. This limit is a function of fuel type and is presented in the CORE OPERATING LIMITS REPORT.

References:

1. "General Electric Standard Application for Reactor Fuel", NEDE-24011-P-A (the approved version at the time the reload analyses are performed shall be identified in the COLR).

2. "The GESTR-LOCA and SAFER Models for the Evaluation of the Loss-of-Coolant Accident - SAFER/GESTR Application Methodology", NEDE 23785-1-PA (the approved version at the time the reload analyses are performed shall be identified in the COLR).

3. "Fermi 2 Maximum Extended Operating Domain Analysis", NEDC-31843P, July 1990.

3/4.4.1 RECIRCULATION SYSTEM

The impact of single recirculation loop operation upon plant safety is assessed and shows that single-loop operation is permitted at power level is up to 67.2% of RATED THERMAL POWER if the MCPR fuel cladding safety limit is increased as noted by Specification 2.1.2. APRM scram and control rod block setpoints (or APRM gains) are adjusted as noted in Tables 2.2.1-1 and 3.3.6-2, respectively. A time period of 4 hours is allowed to make these adjustments following the establishment of single loop operation since the need for single loop operation often cannot be anticipated. MCPR operating limits adjustments in Specification 3.2.3 for different plant operating situations are applicable to both single and two recirculation loop operation.

To prevent potential control system oscillations from occurring in the recirculation flow control system, the operating mode of the recirculation flow control system must be restricted to the manual control mode for single-loop operation.

Additionally, surveillance on the pump speed of operating recirculation loop is imposed to exclude the possibility of excessive core internals vibration. The surveillance on differential temperatures below 30% THERMAL POWER or 50% rated recirculation loop flow is to prevent undue thermal stress on vessel nozzles, recirculation pump and vessel bottom head during a power or flow increase following extended operation in the single recirculation loop mode.

An inoperable jet pump is not, in itself, a sufficient reason to declare a recirculation loop inoperable, but it does, in case of a design-basis-accident, increase the blowdown area and reduce the capability of reflooding the core; thus, the requirement for shutdown of the facility with a jet pump inoperable. Jet pump failure can be detected by monitoring jet pump performance on a prescribed schedule for significant degradation.

Recirculation pump speed mismatch limits are in compliance with the ECCS LOCA analysis design criteria for two recirculation loop operation. The limits will ensure an adequate core flow coastdown from either recirculation loop following a LOCA.

In the case where the mismatch limits cannot be maintained during two loop operation, continued operation is permitted in a single recirculation loop mode.

In order to prevent undue stress on the vessel nozzles and bottom head region, the recirculation loop temperatures shall be within 50°F of each other prior to startup of an idle loop. The loop temperature must also be within 50°F of the reactor pressure vessel coolant temperature to prevent thermal shock to the recirculation pump and recirculation nozzles.

3/4.4.1 RECIRCULATION SYSTEM (Continued)

Sudden equalization of a temperature difference greater than 145°F between the reactor vessel bottom head coolant and the coolant in the upper region of the reactor vessel by increasing core flow rate would cause undue stress in the reactor vessel bottom head.

Requirements are imposed to prohibit idle loop startup above the 77% rod line to minimize the potential for initiating core thermal-hydraulic instability.

3/4.4.2 SAFETY/RELIEF VALVES

The safety valve function of the safety/relief valves operate to prevent the reactor coolant system from being pressurized above the Safety Limit of 1325 psig in accordance with the ASME Code. A total of 11 OPERABLE safety/relief valves is required to limit reactor pressure to within ASME III allowable values for the worst case upset transient.

Demonstration of the safety/relief valve lift settings will occur only during shutdown and will be performed in accordance with the provisions of Specification 4.0.5.

The low-low set system ensures that a potentially high thrust load (designated as load case C.3.3) on the SRV discharge lines is eliminated during subsequent actuations. This is achieved by automatically lowering the closing setpoint of two valves and lowering the opening setpoint of two valves following the initial opening. Sufficient redundancy is provided for the low-low set system such that failure of any one valve to open or close at its reduced setpoint does not violate the design basis.

3/4.4.10 CORE THERMAL HYDRAULIC STABILITY

BWR cores typically operate with the presence of global flux noise in a stable mode which is due to random boiling and flow noise. As the power/flow conditions are changed, along with other system parameters (pressure, subcooling, power distribution, etc.) the thermal hydraulic/reactor kinetic feedback mechanism can be enhanced such that random perturbations may result in sustained limit cycle or divergent oscillations in power and flow.

Two major modes of oscillations have been observed in BWRs. The first mode is the fundamental or core-wide oscillation mode in which the entire core oscillates in phase in a given axial plane. The second mode involves regional oscillation in which one half of the core oscillates 180 degrees out of phase with the other half. Studies have indicated that adequate margin to the Safety Limit Minimum Critical Power Ratio (SLMCPR) may not exist during regional oscillations.

Region A and B of Figure 3.4.10-1 represent the least stable conditions of the plant (high power/low flow). Region A and B are usually entered as the result of a plant transient (for example, recirculation pump trips) and therefore are generally not considered part of the normal operating domain. Since all stability events (including test experience) have occurred in either Region A or B, these regions are avoided to minimize the possibility of encountering oscillations and potentially challenging the SLMCPR. Therefore, intentional operation in Regions A or B is not allowed. It is recognized that during certain abnormal conditions within the plant, it may become necessary to enter Region A or B for the purpose of protecting equipment which, were it to fail, could impact plant safety or for the purpose of protecting a safety or fuel operating limit. In these cases, the appropriate actions for the region entered would be performed as required.

Most oscillations that have occurred during testing and operation have occurred at or above the 96% rod line with core flow near natural circulation. This behavior is consistent with analysis which predict reduced stability margin with increasing power or decreasing flow. As core flow is increased or power decreased, the probability of oscillations occurring will decrease. Region A of Figure 3.4.10-1 bounds the majority of the stability events and tests observed in GE BWRs. Since Region A represents the least stable region of the power/flow operating domain, the potential to rapidly encounter large magnitude core thermal hydraulic oscillations is increased. During transients, the operator may not have sufficient time to manually insert control rods to mitigate the oscillations before they reach an unacceptable magnitude. Therefore, the prompt action of manually scramming the plant when Region A is entered is required to ensure protection of the SLMCPR.

3/4.6 CONTAINMENT SYSTEMS

BASES

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 PRIMARY CONTAINMENT INTEGRITY

PRIMARY CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the safety analyses. This restriction, in conjunction with the leakage rate limitation, will limit the SITE BOUNDARY radiation doses to within the limits of 10 CFR Part 100 during accident conditions.

PRIMARY CONTAINMENT INTEGRITY is demonstrated by leak rate testing and by verifying that all primary containment penetrations not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by locked valves, blank flanges or deactivated automatic valves secured in the closed position. For test, vent and drain connections which are part of the containment boundary, a threaded pipe cap with acceptable sealant in addition to the containment isolation valve(s) provides protection equivalent to a blank flange.

3/4.6.1.2 PRIMARY CONTAINMENT LEAKAGE

The limitations on primary containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the safety analyses at the peak accident pressure of 56.5 psig, P_a. Updated analysis demonstrates maximum expected pressure is less than 56.5 psig. As an added conservatism, the measured overall integrated leakage rate is further limited to less than or equal to 0.75 L_a during performance of the periodic tests to account for possible degradation of the containment leakage barriers between leakage tests.

Operating experience with the main steam line isolation valves has indicated that degradation has occasionally occurred in the leak tightness of the valves; therefore the special requirement for testing these valves.

The surveillance testing for measuring leakage rates is consistent with the requirements of Appendix J of 10 CFR Part 50 with the exception of exemptions granted for main steam isolation valve leak testing and testing the airlocks after each opening and analyzing the Type A test data.

Appendix J to 10 CFR Part 50, Paragraph III.A.3, requires that all Type A tests be conducted in accordance with the provisions of N45.4-1972, "Leakage-Rate Testing of Containment Structures for Nuclear Reactors." N45.4-1972 requires that Type A test data be analyzed using point-to-point or total time analytical techniques. Specification 4.6.1.2a. requires use of the mass plot analytical technique. The mass plot method is considered the better analytical technique, since it yields a confidence interval which is a small fraction of the calculated leak rate; and the interval decreases as more data sets are added to the calculation. The total time and point-to-point techniques may give confidence intervals, which are large fractions of the calculated leak rate, and the intervals may increase as more data sets are added.

PRIMARY CONTAINMENT AIR LOCKS (Continued)

3.6.1.2. The specification makes allowances for the fact that there may be long periods of time when the air locks will be in a closed and secured position during reactor operation. Only one closed door in each air lock is required to maintain the integrity of the containment.

3/4.6.1.4 MSIV LEAKAGE CONTROL SYSTEM

Calculated doses resulting from the maximum leakage allowance for the main steamline isolation valves in the postulated LOCA situations would be a small fraction of the 10 CFR Part 100 guidelines, provided the main steam line system from the isolation valves up to and including the turbine condenser remains intact. Operating experience has indicated that degradation has occasionally occurred in the leak tightness of the MSIVs such that the specified leakage requirements have not always been maintained continuously. The requirement for the leakage control system will reduce the untreated leakage from the MSIVs when isolation of the primary system and containment is required.

3/4.6.1.5 PRIMARY CONTAINMENT STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment will be maintained comparable to the original design standards for the life of the unit. Structural integrity is required to ensure that the containment will withstand the maximum pressure of 56.5 psig in the event of a LOCA. A visual inspection in conjunction with Type A leakage tests is sufficient to demonstrate this capability.

3/4.6.1.6 DRYWELL AND SUPPRESSION CHAMBER INTERNAL PRESSURE

The limitations on drywell and suppression chamber internal pressure ensure that the containment peak pressure of less than 56.5 psig does not exceed the maximum allowable pressure of 62 psig during LOCA conditions or that the external pressure differential does not exceed the design maximum external pressure differential of 2 psid.

3/4.6.1.7 DRYWELL AVERAGE AIR TEMPERATURE

The limitation on drywell average air temperature ensures that the containment peak air temperature does not exceed the design temperature of 340°F during LOCA conditions and is consistent with the safety analysis.

3/4.6.1.8 DRYWELL AND SUPPRESSION CHAMBER PURGE SYSTEM

The drywell and suppression chamber purge supply and exhaust isolation valves are maintained closed during a majority of the plant operating time. Maintaining these valves closed (even though they have been qualified to close against the buildup of pressure in primary containment in the event of DBA/LOCA) reduces the potential for release of excessive quantities of radioactive material.

DRYWELL AND SUPPRESSION CHAMBER PURGE SYSTEM (Continued)

Purging or venting through the Standby Gas Treatment System (SGTS) imposes a vulnerability factor on the integrity of the SGTS. Should a LOCA occur while the purge pathway is through the SGTS the associated pressure surge, before the purge valves close, may adversely affect the integrity of the SGTS charcoal filters. Therefore, PURGING or VENTING through the SGTS is limited to 90 hours per 365 days. This time limit is not imposed when venting through the SGTS with the 1-inch valves or when PURGING or VENTING through the Reactor Building Ventilation System with any of the purge valves.

Leakage integrity tests with a maximum allowable leakage rate for purge supply and exhaust isolation valves will provide early indication of resilient material seal degradation and will allow the opportunity for repair before gross leakage failure develops. The 0.60 L_a leakage limit shall not be exceeded when the leakage rates determined by the leakage integrity tests of these valves are added to the previously determined total for all valves and penetrations subject to Type B and C tests.

The 6, 10, 20, and 24 inch purge valves are generally configured in a three (3) valve arrangement at each of the associated purge penetrations. The valves are leak tested by pressurizing between the three valves and a total leakage is determined as opposed to a single valve leakage. Verifying that the measured leakage rate is less than 0.5 L_a for this multi-valve arrangement is more conservative than a limit of 0.5 L_a for a single valve.

3/4.6.2 DEPRESSURIZATION SYSTEMS

The specifications of this section ensure that the primary containment pressure will not exceed the maximum allowable pressure of 62 psig during primary system blowdown from full operating pressure.

The suppression chamber water provides the heat sink for the reactor coolant system energy release following a postulated rupture of the system. The suppression chamber water volume must absorb the associated decay and structural sensible heat released during reactor coolant system blowdown from 1045 psig. Since all of the gases in the drywell are purged into the suppression chamber air space during a loss-of-coolant accident, the pressure of the liquid must not exceed 62 psig, the suppression chamber maximum pressure. The design volume of the suppression chamber, water and air, was obtained by considering that the total volume of reactor coolant to be condensed is discharged to the suppression chamber and that the drywell volume is purged to the suppression chamber.

DEPRESSURIZATION SYSTEMS (Continued)

Using the minimum or maximum water volumes given in this specification, containment pressure during the design basis accident is less than 56.5 psig which is below the maximum allowable pressure of 62 psig. Maximum water volume of 124,220 ft^3 results in a downcomer submergence of 3'4" and the minimum volume of 121,080 ft^3 results in a submergence of 3'0". The maximum temperature at the end of the blowdown tested during the Humboldt Bay and Bodega Bay tests was 170°F.

Should it be necessary to make the suppression chamber inoperable, this shall only be done as specified in Specification 3.5.3.

Under full power operation conditions, a design basis accident blowdown from an initial suppression chamber water temperature of 95°F results in a water temperature of approximately 135°F in the short term following the blowdown. At this temperature and atmospheric pressure, the available NPSH exceeds that required by both the RHR and core spray pumps, thus there is no dependency on containment overpressure during the accident injection phase. If both RHR loops are used for containment cooling, there is no dependence on containment overpressure for post-LOCA operations.

The large thermal capacitance of the suppression pool is also utilized during plant transients requiring safety/relief valve (SRV) actuation. Steam is discharged from the main steam lines through the SRVs and their accompanying discharge lines into the suppression pool where it is condensed, resulting in an increase in the temperature of the suppression pool water. Although stable steam condensation is expected at all pool temperatures, NUREG-0783 imposes a local temperature limit shown in Figure B 3/4.6.2-1 in the vicinity of the T-type quencher discharge device. The limiting plant transients with respect to heat input to the suppression pool have been analyzed. The conservative analysis showed that limiting the average water temperature to less than or equal to 170°F will result in local pool temperatures below the condensation stability limit of Bases Figure B 3/4.6.2-1.

Experimental data indicate that excessive steam condensing loads can be avoided if the peak local temperature of the suppression pool is maintained below 200°F during any period of relief valve operation. Specifications have been placed on the envelope of reactor operating conditions so that the reactor can be depressurized in a timely manner to avoid the regime of potentially high suppression chamber loadings.

Because of the large volume and thermal capacity of the suppression pool, the volume and temperature normally change very slowly and monitoring these parameters daily is sufficient to establish any temperature trends. By requiring the suppression pool temperature to be frequently recorded during periods of significant heat addition, the temperature trends will be closely followed so that appropriate action can be taken. The requirement for an external visual

3/4.7.9 MAIN TURBINE BYPASS SYSTEM AND MOISTURE SEPARATOR REHEATER

The main turbine bypass system is an active bypass system designed to open the bypass valves in the event of a turbine trip to decrease the severity of the pressure transient. Each valve is sized to pass approximately 12½ percent reactor steam flow in the full-open position for a controlled total bypass of approximately 25 percent reactor steam flow. The main turbine bypass system is required to be OPERABLE consistent with the assumptions of the Feedwater Controller Failure analysis.

The primary purpose of the moisture separator reheater is to improve cycle efficiency by using primary system steam to heat the high pressure turbine exhaust before it enters the low-pressure turbines. In doing so, it also provides a passive steam bypass flow of about 10 percent that mitigates the early effects of over-pressure transients. The moisture separator reheater is required to be OPERABLE consistent with the assumptions of the Main Turbine Trip with Turbine Bypass Failure analysis and the Feedwater Controller Failure analysis.

The operation with one or both of the main turbine bypasses inoperable or the moisture separator reheater inoperable to perform preventive or corrective maintenance above 25 percent RATED THERMAL POWER, requires, after one hour, the evaluation of the MCPR in accordance with Specification 3.2.3. If the MCPR is within the bounds established by Specification 3.2.3, power increases to or operation above 25 percent RATED THERMAL POWER is allowed.

3/4.7.11 APPENDIX R ALTERNATIVE SHUTDOWN AUXILIARY SYSTEMS

The systems identified in this section are those utilized for Appendix R Alternative shutdown but not included in other sections of the Technical Specifications. The ACTION statements assure that the auxiliary systems will be OPERABLE or that acceptable alternative means are established to achieve the same objective.

There are four independent Combustion Turbine-Generator units onsite. CTG 11 Unit 1 has a diesel engine starter and thus can be started independently from offsite power. CTG 11 Units 2, 3, and 4 have AC-motor starters and rely on a 480-volt AC feed. The phrase "alternative source of power", as used in Specification 3.7.11, ACTION b.2, is defined as a source of power that is not reliant on offsite power for starting (if required) or operating (if already running) and capable of supplying the required loads on the 4160-volt busses associated with the Alternative Shutdown System.

One of the two installed Standby Feedwater Pumps and one of the two listed Drywell Cooling Units are necessary for Appendix R Alternative shutdown. Therefore unlimited operation with one of the two components inoperable is justified provided increased surveillance is performed on the components which remain OPERABLE.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional Administrator of the Regional Office of the NRC within the time period specified for each report.

CORE OPERATING LIMITS REPORT

6.9.3 Selected cycle specific core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT (COLR) before each reload cycle or any remaining part of a reload cycle. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC in General Electric Company reports NEDE-24011-P-A and NEDE-23785-1-PA. The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met. The COLR, including any mid-cycle revisions or supplement thereto, shall be submitted upon issuance to the NRC Document Control Desk, with copies to the Regional Administrator and Resident Inspector prior to use.

6.10 RECORD RETENTION

6.10.1 In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.

6.10.2 The following records shall be retained for at least 5 years:

a. Records and logs of unit operation covering time interval at each power level.

b. Records and logs of principal maintenance activities, inspections, repair, and replacement of principal items of equipment related to nuclear safety.

c ALL REPORTABLE EVENTS.

d. Records of surveillance activities, inspections, and calibrations required by these Technical Specifications.

e. Records of changes made to the procedures required by Specification 6.8.1.

f. Records of radioactive shipments.

g. Records of sealed source and fission detector leak tests and results.

h. Records of annual physical inventory of all sealed source material of record.

ENCLOSURE 3

POWER UPRATE SAFETY ANALYSIS

DETROIT EDISON CO.

FERMI 2 - 91-150, SEPTEMBER 1991