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

GEORGIA POWER COMPANY

OGLETHORPE POWER CORPORATION

MUNICIPAL ELECTRIC AUTHORITY OF GEORGIA

CITY OF DALTON, GEORGIA

DOCKET NO. 50-366

EDWIN I. HATCH NUCLEAR PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 39
License No. NPF-5

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

   A. The applications for amendment by Georgia Power Company, et al., (the licensee) dated January 23, 1984, as supplemented April 3, 1984, June 7, 14 and 15, 1984; February 6, 1984, as supplemented April 3, 1984, June 20 and 27, 1984; and April 3, 1984, comply 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 applications, 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 license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-5 is hereby amended to read as follows:

Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 39 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Gus C. Lainas, Assistant Director
for Operating Reactors
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: July 13, 1984

ATTACHMENT TO LICENSE AMENDMENT NO. 39

FACILITY OPERATING LICENSE NO. NPF-5

DOCKET NO. 50-366

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

Remove	Insert
2-4	2-4
B 2-10	B 2-10
B 2-12	B 2-12
B 2-13	B 2-13
3/4 1-17	3/4 1-17
3/4 2-1	3/4 2-1
-----	3/4 2-4g
-----	3/4 2-4h
-----	3/4 2-4i
3/4 2-5	3/4 2-5
3/4 2-6	3/4 2-6
3/4 2-7	3/4 2-7
3/4 2-7a	3/4 2-7a
3/4 2-7b	3/4 2-7b
3/4 2-7c	3/4 2-7c
-----	3/4 2-7d
3/4 3-2	3/4 3-2
3/4 3-7	3/4 3-7
3/4 3-8	3/4 3-8
3/4 3-11	3/4 3-11
3/4 3-12	3/4 3-12
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Remove	Insert
3/4 3-31	3/4 3-31
3/4 3-32	3/4 3-32
3/4 3-34	3/4 3-34
3/4 3-35	3/4 3-35
3/4 3-36	3/4 3-36
3/4 3-39	3/4 3-39
3/4 3-40	3/4 3-40
-----	3/4 3-40a
3/4 3-42	3/4 3-42
3/4 3-54	3/4 3-54
3/4 4-18	3/4 4-18
3/4 6-41	3/4 6-41
3/4 9-4	3/4 9-4
B 3/4 1-4	B 3/4 1-4
-----	B 3/4 1-4a
B 3/4 2-1	B 3/4 2-1
B 3/4 2-3	B 3/4 2-3
B 3/4 2-4	B 3/4 2-4
B 3/4 2-5	B 3/4 2-5
B 3/4 2-6	B 3/4 2-6
B 3/4 3-6	B 3/4 3-6
B 3/4 9-1	B 3/4 9-1
5-3	5-3

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

2.2.1 The reactor protection system instrumentation setpoints shall be set consistent with the Trip Setpoint values shown in Table 2.2.1-1.

APPLICABILITY: As shown for each channel in Table 3.3.1-1.

ACTION:

With a reactor protection system instrumentation setpoint less conservative than the value shown in the Allowable Values column of Table 2.2.1-1, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1 until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

TABLE 2.2.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Intermediate Range Monitor, Neutron Flux-High (2C51-K601 A,B,C,D,E,F,G,H)	≤ 120/125 divisions of full scale	≤ 120/125 divisions of full scale
2. Average Power Range Monitor: (2C51-K605 A,B,C,D,E,F)		
a. Neutron Flux-Upscale, 15%	≤ 15/125 divisions of full scale	≤ 20/125 divisions of full scale
b. Flow Referenced Simulated Thermal Power-Upscale	≤ (0.58 W + 59%), with a maximum ≤ 113.5% of RATED THERMAL POWER	≤ (0.58 W + 62%), with a maximum ≤ 115.5% of RATED THERMAL POWER
c. Fixed Neutron Flux-Upscale, 118%	≤ 118% of RATED THERMAL POWER	≤ 120% of RATED THERMAL POWER
3. Reactor Vessel Steam Dome Pressure - High (2B21-N678 A,B,C,D)	≤ 1054 psig	≤ 1054 psig
4. Reactor Vessel Water Level - Low (Level 3) (2B21-N680 A,B,C,D)	≥ 8.5 inches above instrument zero*	≥ 8.5 inches above instrument zero*
5. Main Steam Line Isolation Valve - Closure (NA)	≤ 10% closed	≤ 10% closed
6. Main Steam Line Radiation - High (2D11-K603A,B,C,D)	≤ 3 x full power background	≤ 3 x full power background
7. Drywell Pressure - High (2C71-N650A,B,C,D)	≤ 1.85 psig	≤ 1.85 psig

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

2.2 LIMITING SAFETY SYSTEM SETTINGS

BASES

2.2.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

The Reactor Protection System Instrumentation Setpoints specified in Table 2.2.1-1 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 set 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.

1. Intermediate Range Monitor, Neutron Flux

The IRM system consists of 8 chambers, 4 in each of the reactor trip systems. The IRM is a 5 decade 10 range instrument. The trip setpoint of 120 divisions of scale is active in each of the 10 ranges. Thus, as the IRM is ranged up to accommodate the increase in power level, the trip setpoint is also ranged up. The IRM instruments provide for overlap with both the APRM and SRM systems.

The most significant source of reactivity changes during the power increase are due to control rod withdrawal. In order to ensure that the IRM provides the required protection, a range of rod withdrawal accidents have been analyzed, Section 7.5 of the FSAR. The most severe case involves an initial condition in which the reactor is just subcritical and the IRM's are not yet on scale. Additional conservatism was taken in this analysis by assuming the IRM channel closest to the rod being withdrawn is bypassed. The results of this analysis show that the reactor is shutdown and peak power is limited to 1% of RATED THERMAL POWER, thus maintaining MCPR above 1.07. Based on this analysis, the IRM provides protection against local control rod errors and continuous withdrawal of control rods in sequence and provides backup protection for the APRM.

2. Average Power Range Monitor

For operation at low pressure and low flow during STARTUP, the APRM scram setting of 15/125 divisions of full scale neutron flux provides adequate thermal margin between the setpoint and the Safety Limits. The margin accommodates the anticipated maneuvers associated with power plant startup. Effects of increasing pressure at zero or low void content are minor and cold water from sources available during startup is not much colder than that already in the system. Temperature coefficients are small and control rod patterns are constrained by the RSCS and RWM.

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS (Continued)

Average Power Range Monitor (Continued)

Of all the possible sources of reactivity input, uniform control rod withdrawal is the most probable cause of significant power increase. Because the flux distribution associated with uniform rod withdrawals does not involve high local peaks and because several rods must be moved to change power by a significant amount, the rate of power rise is very slow. Generally the heat flux is in near equilibrium with the fission rate. In an assumed uniform rod withdrawal approach to the trip level, the rate of power rise is not more than 5% of RATED THERMAL POWER per minute and the APRM system would be more than adequate to assure shutdown before the power could exceed the Safety Limit. The 15% neutron flux trip remains active until the mode switch is placed in the Run position.

The APRM flux scram trip in the Run mode consists of a flow referenced simulated thermal power scram setpoint and a fixed neutron flux scram setpoint. The APRM flow referenced neutron flux signal is passed through a filtering network with a time constant which is representative of the fuel dynamics. This provides a flow referenced signal that approximates the average heat flux or thermal power that is developed in the core during transient or steady-state conditions.

The APRM flow referenced simulated thermal power scram trip setting at full recirculation flow is adjustable up to 113.5% of RATED THERMAL POWER. This reduced flow referenced trip setpoint will result in an earlier scram during slow thermal transients, such as the loss of 100°F feedwater heating event, than would result with the 118% fixed neutron flux scram trip. The lower flow referenced scram setpoint therefore decreases the severity, ΔCPR, of a slow thermal transient and allows lower operating limits if such a transient is the limiting abnormal operational transient during a certain exposure interval in the fuel cycle.

The APRM fixed neutron flux signal does not incorporate the time constant, but responds directly to instantaneous neutron flux. This scram setpoint scrams the reactor during fast power increase transients if credit is not taken for a direct (position) scram, and also serves to scram the reactor if credit is not taken for the flow referenced simulated thermal power scram.

The APRM setpoints were selected to provide adequate margin for the Safety Limits and yet allow operating margin that reduces the possibility of unnecessary shutdown.

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS (Continued)

3. Reactor Vessel Steam Dome Pressure-High

High pressure in the nuclear system could cause a rupture to the nuclear system process barrier resulting in the release of fission products. A pressure increase while operating will also tend to increase the power of the reactor by compressing voids thus adding reactivity. The trip will quickly reduce the neutron flux, counteracting the pressure increase by decreasing heat generation. The trip setting is slightly higher than the operating pressure to permit normal operation without spurious trips. The setting provides for a wide margin to the maximum allowable design pressure and takes into account the location of the pressure measurement compared to the highest pressure that occurs in the system during a transient. This trip setpoint is effective at low power/flow conditions when the turbine stop valve closure trip is bypassed. For a turbine trip under these conditions, the transient analysis indicated a considerable margin to the thermal hydraulic limit.

4. Reactor Vessel Water Level-Low

The reactor vessel water level trip setpoint was chosen far enough below the normal operating level to avoid spurious trips but high enough above the fuel to assure that there is adequate protection for the fuel and pressure barriers.

5. Main Steam Line Isolation Valve-Closure

The main steam line isolation valve closure trip was provided to limit the amount of fission product release for certain postulated events. The MSIVs are closed automatically from measured parameters such as high steam flow, high steam line radiation, low reactor water level, high steam tunnel temperature and low steam line pressure. The MSIV closure scram anticipates the pressure and flux transients which could follow MSIV closure, and thereby protects reactor vessel pressure and fuel thermal/hydraulic Safety Limits.

6. Main Steam Line Radiation-High

The main steam line radiation detectors are provided to detect a gross failure of the fuel cladding. When the high radiation is detected a trip is initiated to reduce the continued failure of fuel cladding. At the same time the main steam line isolation valves are closed to limit the release of fission products. The trip setting is high enough above background radiation levels to prevent spurious trips yet low enough to promptly detect gross failures in the fuel cladding.

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS (Continued)

7. Drywell Pressure-High

High pressure in the drywell could indicate a break in the nuclear process systems. The reactor is tripped in order to minimize the possibility of fuel damage and reduce the amount of energy being added to the coolant. The trip setting was selected as low as possible without causing spurious trips.

8. Scram Discharge Volume Water Level-High

The scram discharge volume receives the water displaced by the motion of the control rod drive pistons during a reactor scram. Should this volume fill up to a point where there is insufficient volume to accept the displaced water, control rod insertion would be hindered. The reactor is therefore tripped when the water level has reached a point high enough to indicate that it is indeed filling up, but the volume is still great enough to accommodate the water from the movement of the rods when they are tripped.

9. Turbine Stop Valve-Closure

The turbine stop valve closure trip anticipates the pressure, neutron flux, and heat flux increases that would result from closure of the stop valves. With a trip setting of 10% of valve closure from full open, the resultant increase in heat flux is such that adequate thermal margins are maintained even during the worst case transient that assumes the turbine bypass valves remain closed. This scram is bypassed when the turbine steam flow is below that corresponding to 30% of RATED THERMAL POWER, as measured by turbine first stage pressure.

10. Turbine Control Valve Fast Closure, Trip Oil Pressure-Low

The turbine control valve fast closure trip anticipates the pressure, neutron flux, and heat flux increase that could result from fast closure of the turbine control valves due to load rejection coincident with failures of the turbine bypass valves. The Reactor Protection System initiates a trip when fast closure of the control valves is initiated by the fast acting solenoid valves and in less than 30 milliseconds after the start of control valve fast closure. This is achieved by the action of the fast acting solenoid valves in rapidly reducing hydraulic trip oil pressure at the main turbine control valve actuator disc dump valves. This loss of pressure is sensed by

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS (Continued)

Turbine Control Valve Fast Closure, Trip Oil Pressure-Low (Continued)

pressure switches whose contacts form the one-out-of-two-twice logic input to the Reactor Protection System. This trip setting, a nominally 50% greater closure time and a different valve characteristic from that of the turbine stop valve, combine to produce transients very similar to that for the stop valve. No significant change in MCPR occurs. Relevant transient analyses are discussed in Section 15 of the Final Safety Analysis Report. This scram is bypassed when turbine steam flow is below that corresponding to 30% of RATED THERMAL POWER, as measured by turbine first stage pressure.

11. Reactor Mode Switch In Shutdown Position

The reactor mode switch Shutdown position trip is a redundant channel to the automatic protective instrumentation channels and provides additional manual reactor trip capability.

12. Manual Scram

The Manual Scram is a redundant channel to the automatic protective instrumentation channels and provides manual reactor trip capability.

REACTIVITY CONTROL SYSTEMS

ROD BLOCK MONITOR

LIMITING CONDITION FOR OPERATION

3.1.4.3 Both Rod Block Monitor (RBM) channels shall be OPERABLE.

APPLICABILITY: CONDITION 1, when THERMAL POWER is greater than the preset power level of the RWM and RSCS and when:

a. THERMAL POWER is < 90% of RATED THERMAL POWER and the MCPR is less than 1.70, or

b. THERMAL POWER is ≥ 90% of RATED THERMAL POWER and the MCPR is less than 1.40.

ACTION:

a. With one RBM channel inoperable, POWER OPERATION may continue provided that the inoperable RBM channel is restored to OPERABLE status within 24 hours; otherwise, trip at least one rod block monitor channel within the next hour.

b. With both RBM channels inoperable, trip at least one rod block monitor channel within one hour.

SURVEILLANCE REQUIREMENTS

4.1.4.3 a. With both RBM channels OPERABLE, surveillance requirements are given in Specification 4.3.5.

b. With one RBM channel INOPERABLE, the other channel shall be demonstrated OPERABLE by performance of a CHANNEL FUNCTIONAL TEST prior to withdrawal of control rods.

3/4.1.5 STANDBY LIQUID CONTROL SYSTEM

LIMITING CONDITION FOR OPERATION

3.1.5 The standby liquid control system shall be OPERABLE with:

a. An OPERABLE flow path from the storage tank to the reactor core containing two pumps and two inline explosive injection valves, and

b. The contained solution concentration and the solution temperature are within the Operating Range of Figure 3.1.5-1.

APPLICABILITY: CONDITIONS 1, 2, and 5*.

ACTION:

a. In CONDITION 1 or 2:

1. With one pump and/or one explosive valve inoperable, restore the inoperable pump and/or explosive valve to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.

2. With the standby liquid control system inoperable, restore the system to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours.

b. In CONDITION 5*:

1. With one pump and/or one explosive valve inoperable, restore the inoperable pump and/or explosive valve to OPERABLE status within 30 days or fully insert all insertable control rods within the next hour.

2. With the standby liquid control system inoperable, fully insert all insertable control rods within one hour.

3. The provisions of Specification 3.0.3 and 3.0.4 are not applicable.

*With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.11.1 or 3.9.11.2.

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 be equal to or less than the applicable APLHGR limit, which is a function of fuel type and AVERAGE PLANAR EXPOSURE. The APLHGR limit is given by the applicable rated-power, rated-flow limit taken from Figures 3.2.1-1 through 3.2.1-9, multiplied by the smaller of either:

a. The factor given by Figure 3.2.1-10, or

b. The factor given by Figure 3.2.1-11.

APPLICABILITY: CONDITION 1, when THERMAL POWER $\geq$ 25% of RATED THERMAL POWER.

ACTION:

With an APLHGR exceeding the limits of Figures 3.2.1-1 through 3.2.1-9, as adjusted per Figures 3.2.1-10 and 3.2.1-11, initiate corrective action within 15 minutes and continue corrective action so that the APLHGR meets 3.2.1 within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

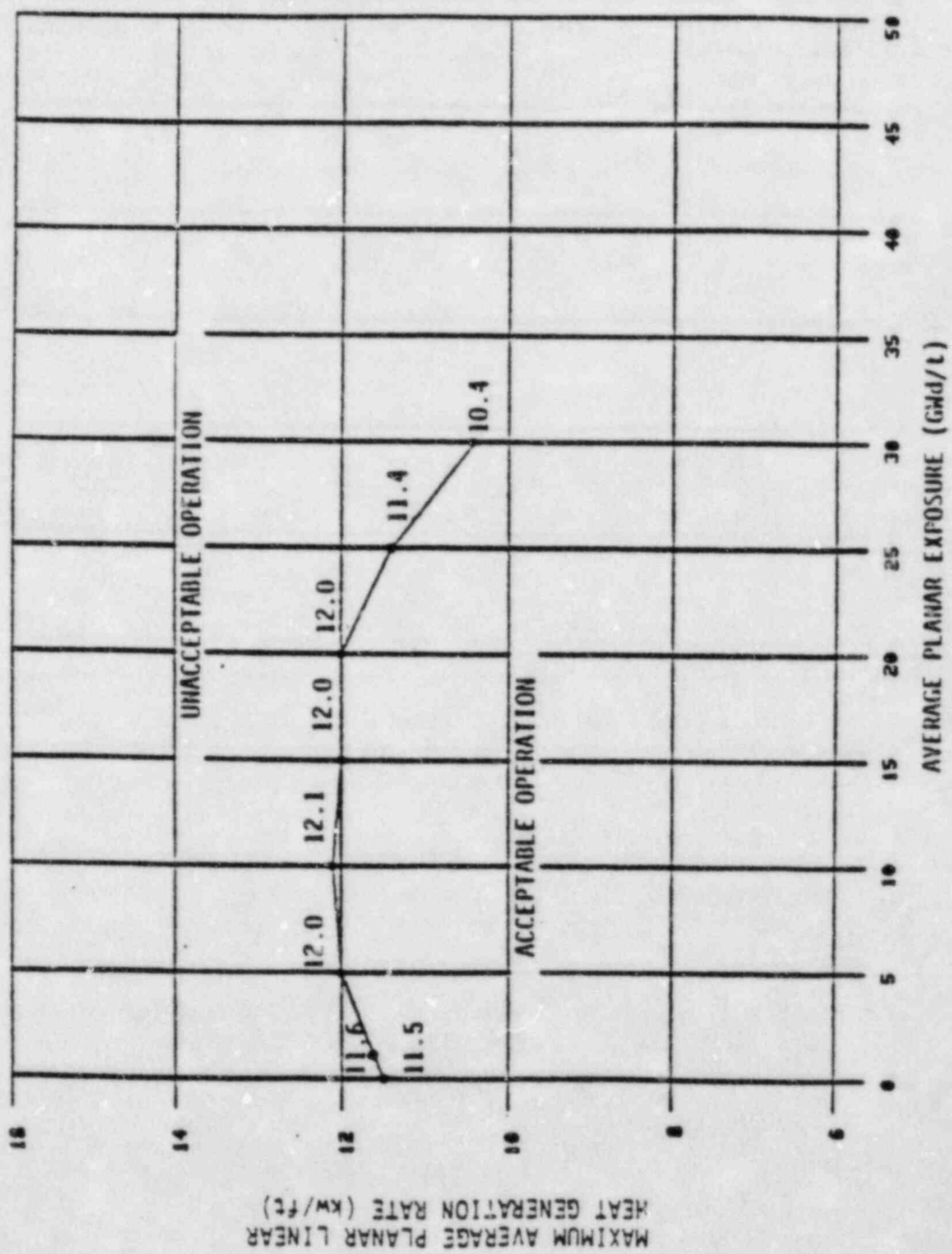
SURVEILLANCE REQUIREMENTS

4.2.1 All APLHGRs shall be verified to be equal to or less than the applicable limit determined from Figures 3.2.1-1 through 3.2.1-9, as adjusted per Figure 3.2.1-10 and 3.2.1-11:

a. At least once per 24 hours,

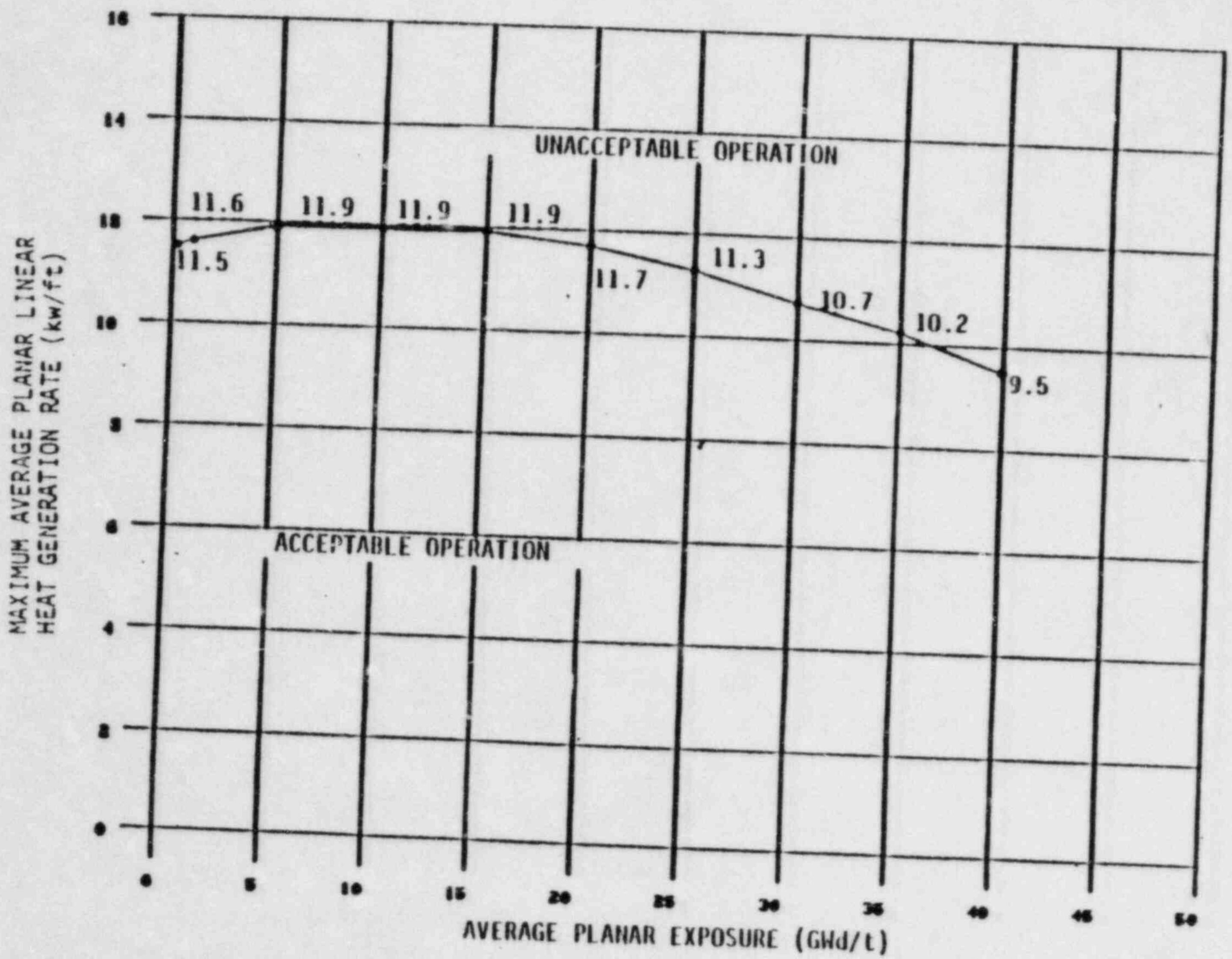
b. Whenever THERMAL POWER has been increased by at least 15% of RATED THERMAL POWER and steady state operating conditions have been established, and

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

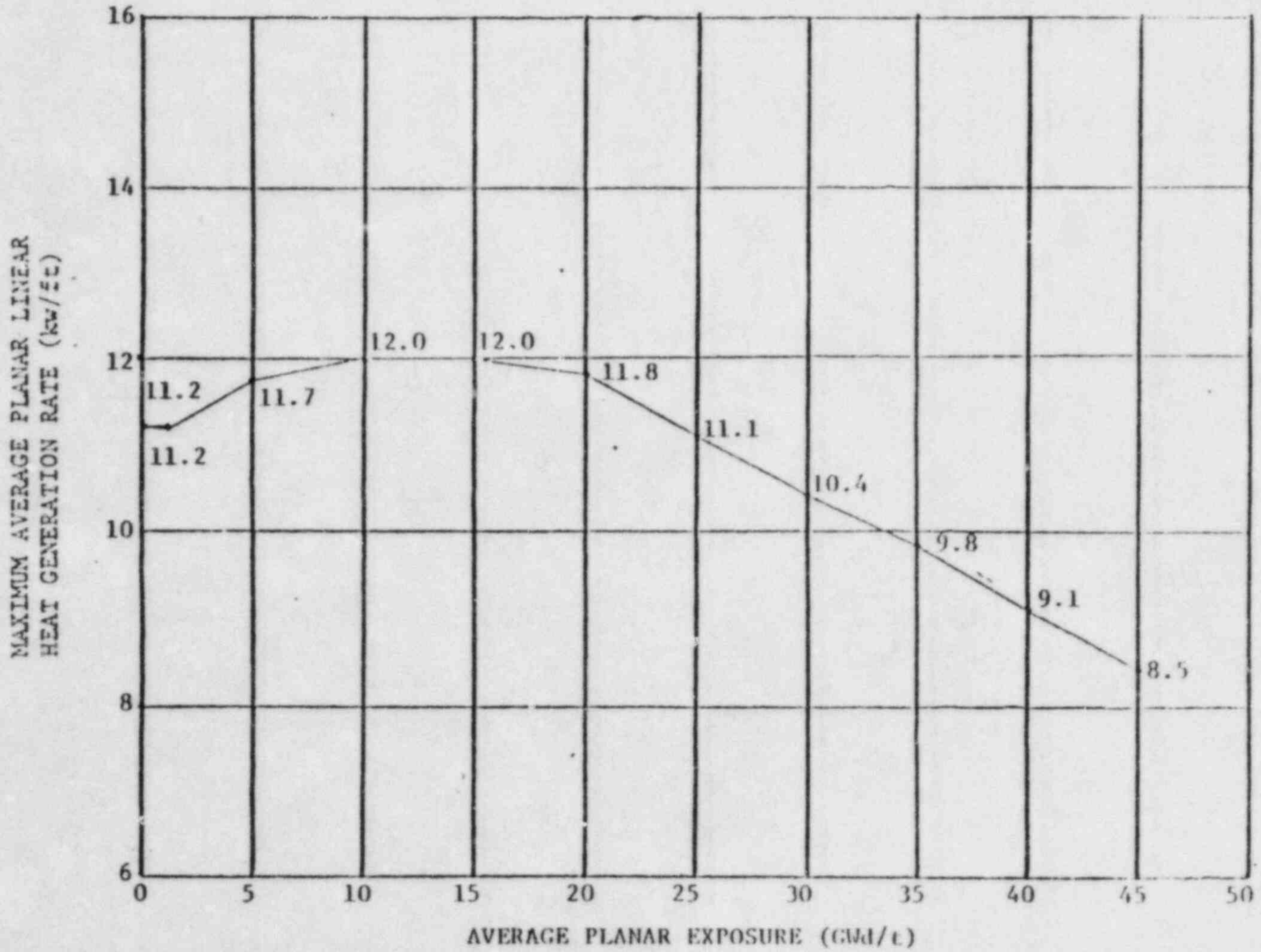


FUEL TYPE 8D8175(8DRL183)
MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION
RATE (MAPLHGR) VERSUS AVERAGE PLANAR EXPOSURE

FIGURE 3.2.1-1

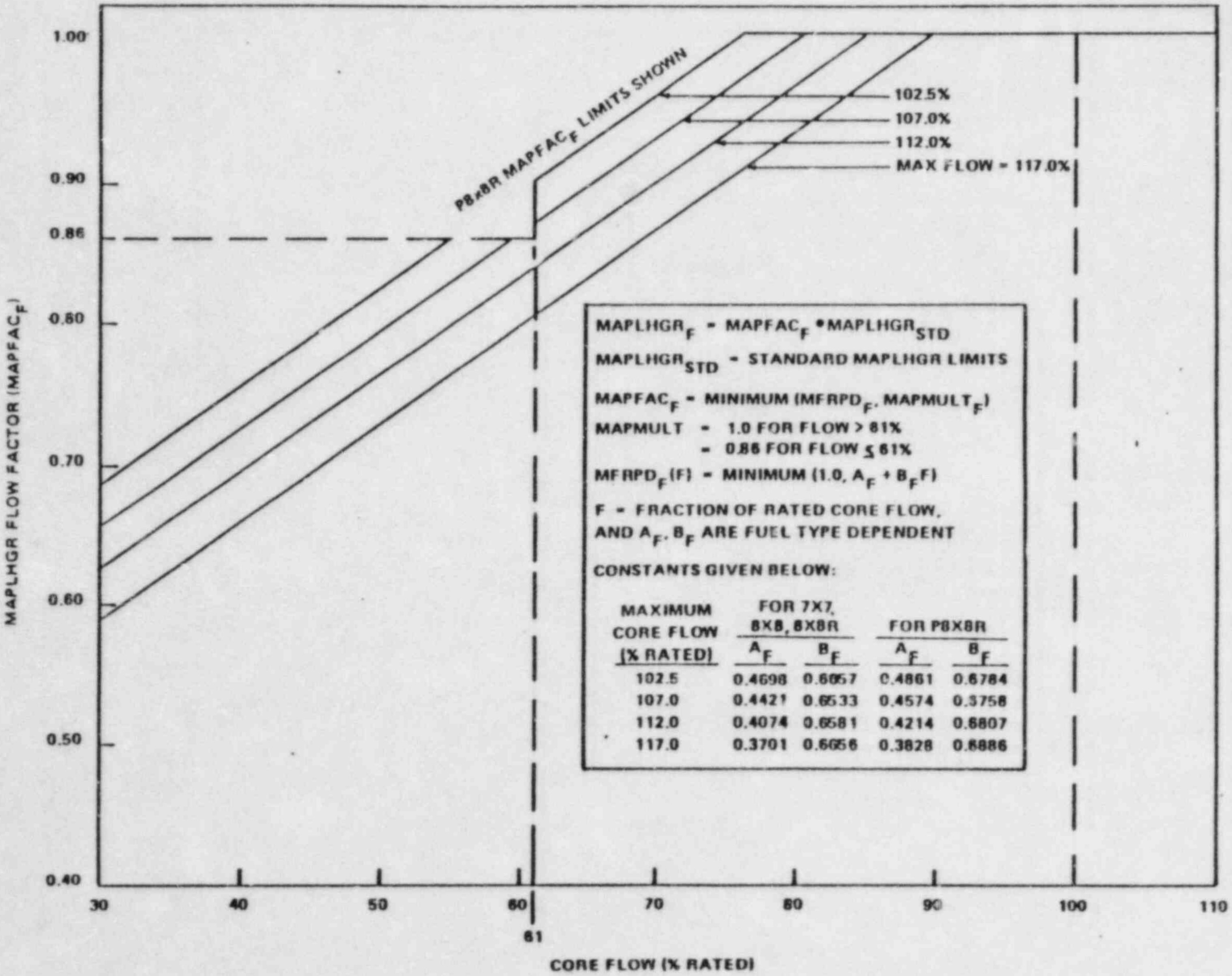


FUEL TYPE 8DRB265H
MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS AVERAGE PLANAR EXPOSURE
FIGURE 3.2.1-8



FUEL TYPE P8DRB284H
MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS AVERAGE PLANAR EXPOSURE

FIGURE 3.2.1-9



MAXIMUM CORE FLOW (% RATED)	FOR 7X7, 8X8, 8X8R A_F	FOR 7X7, 8X8, 8X8R B_F	FOR P8X8R A_F	FOR P8X8R B_F
102.5	0.4698	0.6057	0.4861	0.6784
107.0	0.4421	0.6533	0.4574	0.3758
112.0	0.4074	0.6581	0.4214	0.6807
117.0	0.3701	0.6656	0.3828	0.6886

FIGURE 3.2.1-10 $MAPFAC_p$

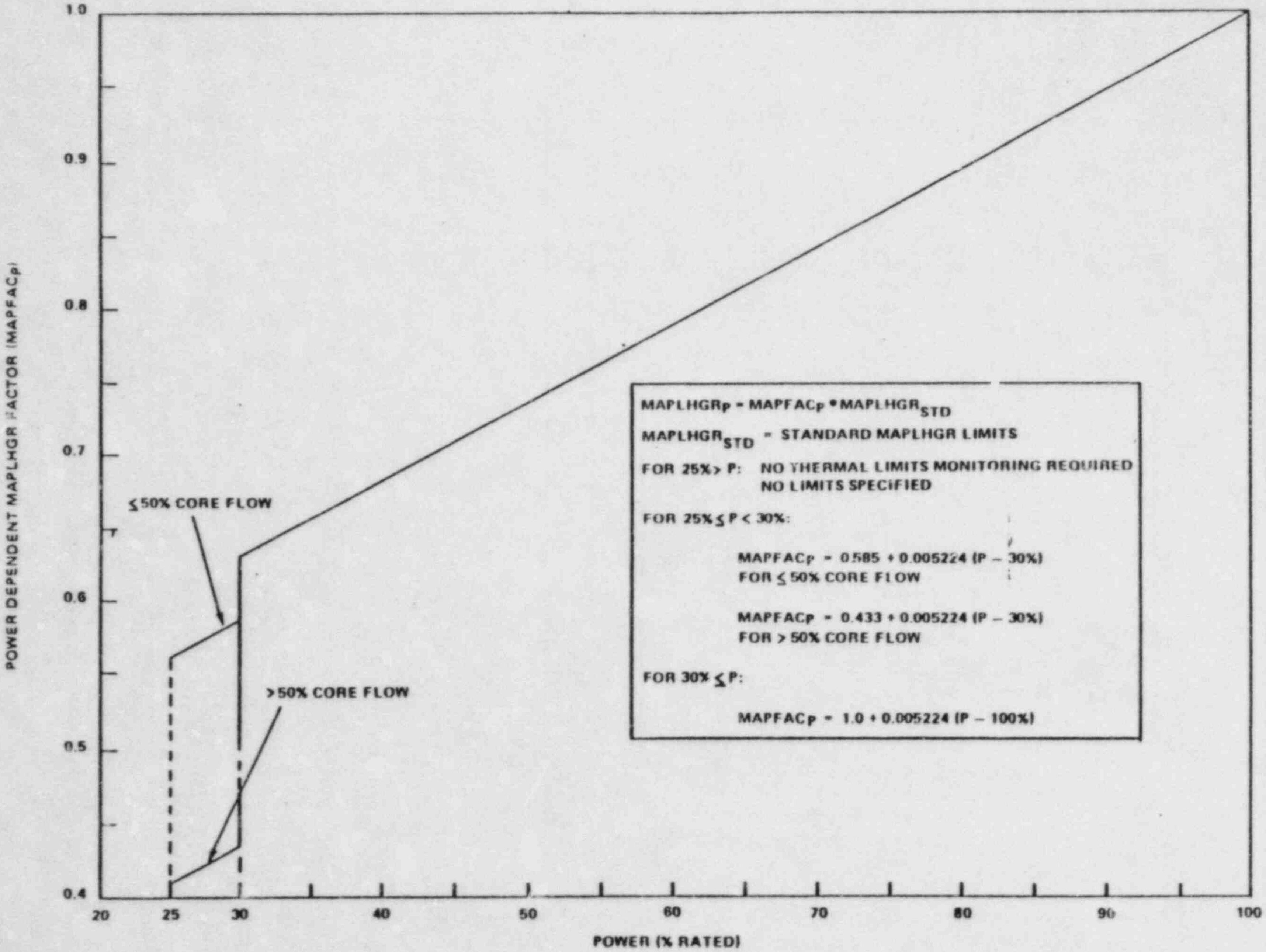


FIGURE 3.2.1-11 $MAPFAC_P$

3/4.2.2 APRM SETPOINTS

This section deleted.

3/4.2.3 MINIMUM CRITICAL POWER RATIO

LIMITING CONDITION FOR OPERATION

3.2.3 All MINIMUM CRITICAL POWER RATIOS (MCPRs), shall be equal to or greater than the MCPR operating limit (OLMCPR), which is a function of average scram time, core flow, and core power. For 25% ≤ Power < 30%, the OLMCPR is given in Figure 3.2.3-5. For Power ≥ 30%, the OLMCPR is the greater of either:

a. The applicable limit determined from Figure 3.2.3-4, or

b. The appropriate K_p given by Figure 3.2.3-5, multiplied by the appropriate limit from Figure 3.2.3-1, 3.2.3-2, or 3.2.3-3, where:

$$\tau = 0 \text{ or } \left[\frac{\tau_{ave} - \tau_B}{\tau_A - \tau_B}\right], \text{ whichever is greater,}$$

τ_A = 1.096 sec (Specification 3.1.3.3 scram time limit to notch 36),

$$\tau_B = 0.834 + 1.65 \left[\frac{N_1}{\sum_{i=1}^{n} N_i}\right]^{1/2} (0.059),$$

$$\tau_{ave} = \frac{\sum_{i=1}^{n} N_i \tau_i}{\sum_{i=1}^{n} N_i}$$

n = number of surveillance tests performed to date in cycle,

N_i = number of active control rods measured in the i^{th} surveillance test,

τ_i = average scram time to notch 36 of all rods measured in the i^{th} surveillance test, and

N_1 = total number of active rods measured in 4.1.3.2.a.

APPLICABILITY: CONDITION 1, when THERMAL POWER ≥ 25% RATED THERMAL POWER

ACTION:

With MCPR less than the applicable limit determined from Specification 3.2.3.a, or 3.2.3.b, initiate corrective action within 15 minutes and continue corrective action so that MCPR is equal to or greater than the applicable limit within 2 hours or reduce THERMAL POWER to less than or equal to 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.3 The MCPR limit at rated flow and rated power shall be determined for each type of fuel (8X8R, P8X8R, and 7X7) from Figures 3.2.3-1, 3.2.3-2, and 3.2.3-3 using:

a. τ = 1.0 prior to the initial scram time measurements for the cycle performed in accordance with Specification 4.1.3.2.a, or

b. τ as defined in Specification 3.2.3; the determination of the limit must be completed within 72 hours of the conclusion of each scram time surveillance test required by Specification 4.1.3.2.

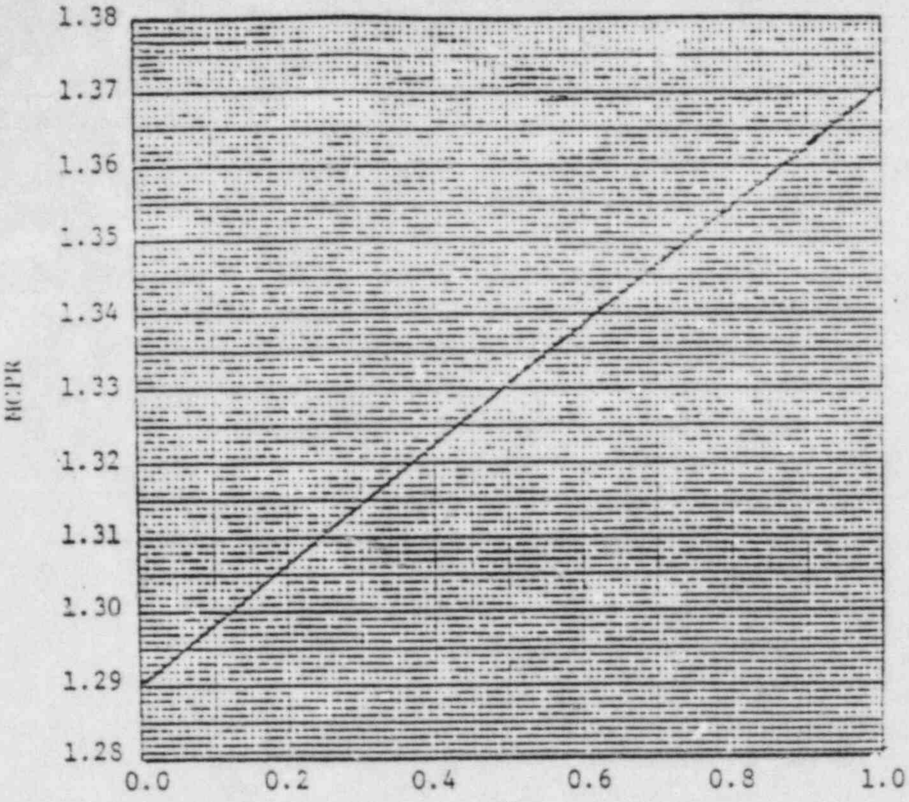
MCPR shall be determined to be equal to or greater than the applicable limit:

a. At least once per 24 hours,

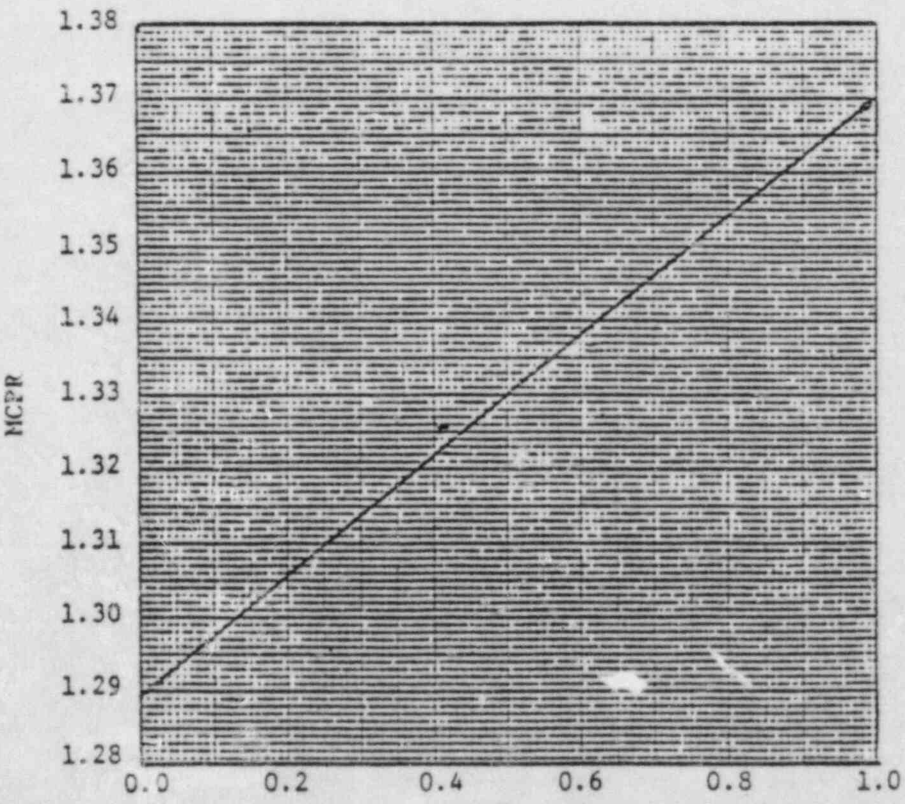
b. Whenever THERMA[illegible]ER has been increased by at least 15% of RATED THERMAL P[illegible] and steady state operating conditions have been established, and

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

MCPR LIMIT AT RATED FLOW AND RATED POWER



8X8R FUEL
FIGURE 3.2.3-1



P8X8R FUEL
FIGURE 3.2.3-2

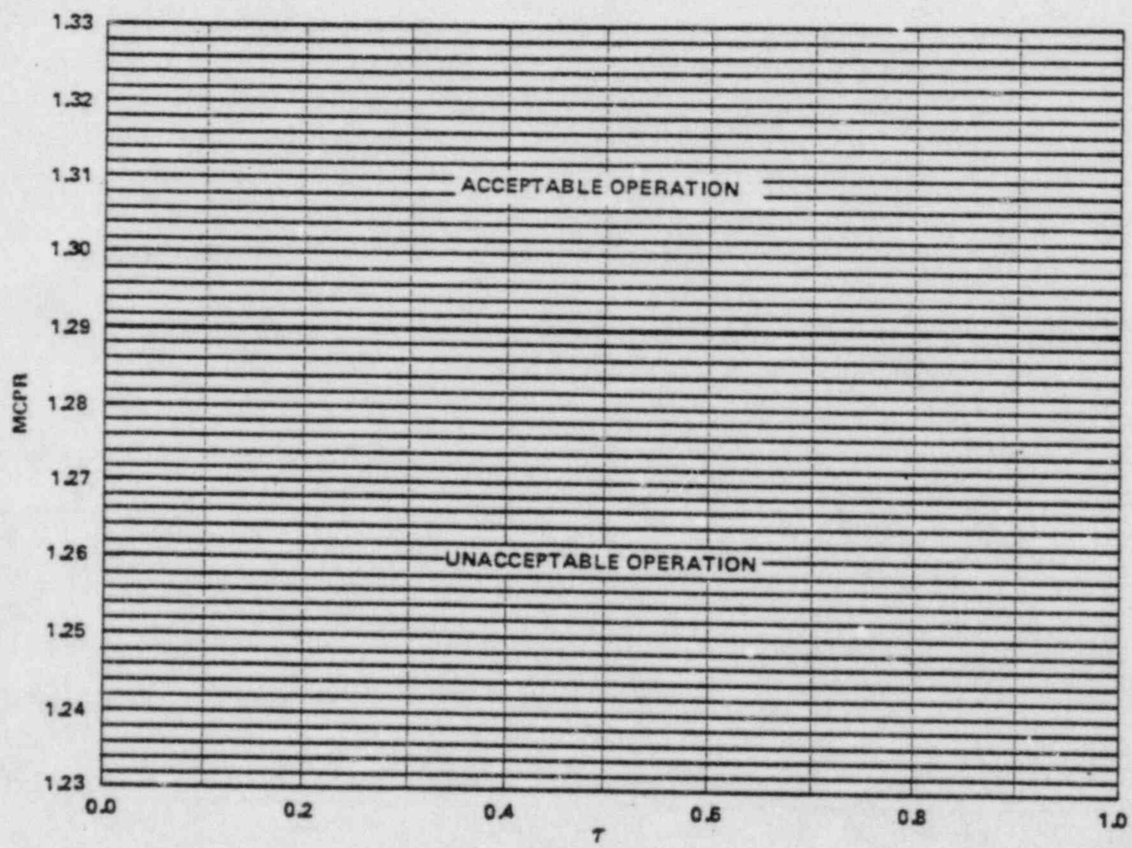


FIGURE 3.2.3-3
MCPR LIMIT FOR 7X7 FUEL
AT RATED FLOW AND RATED POWER

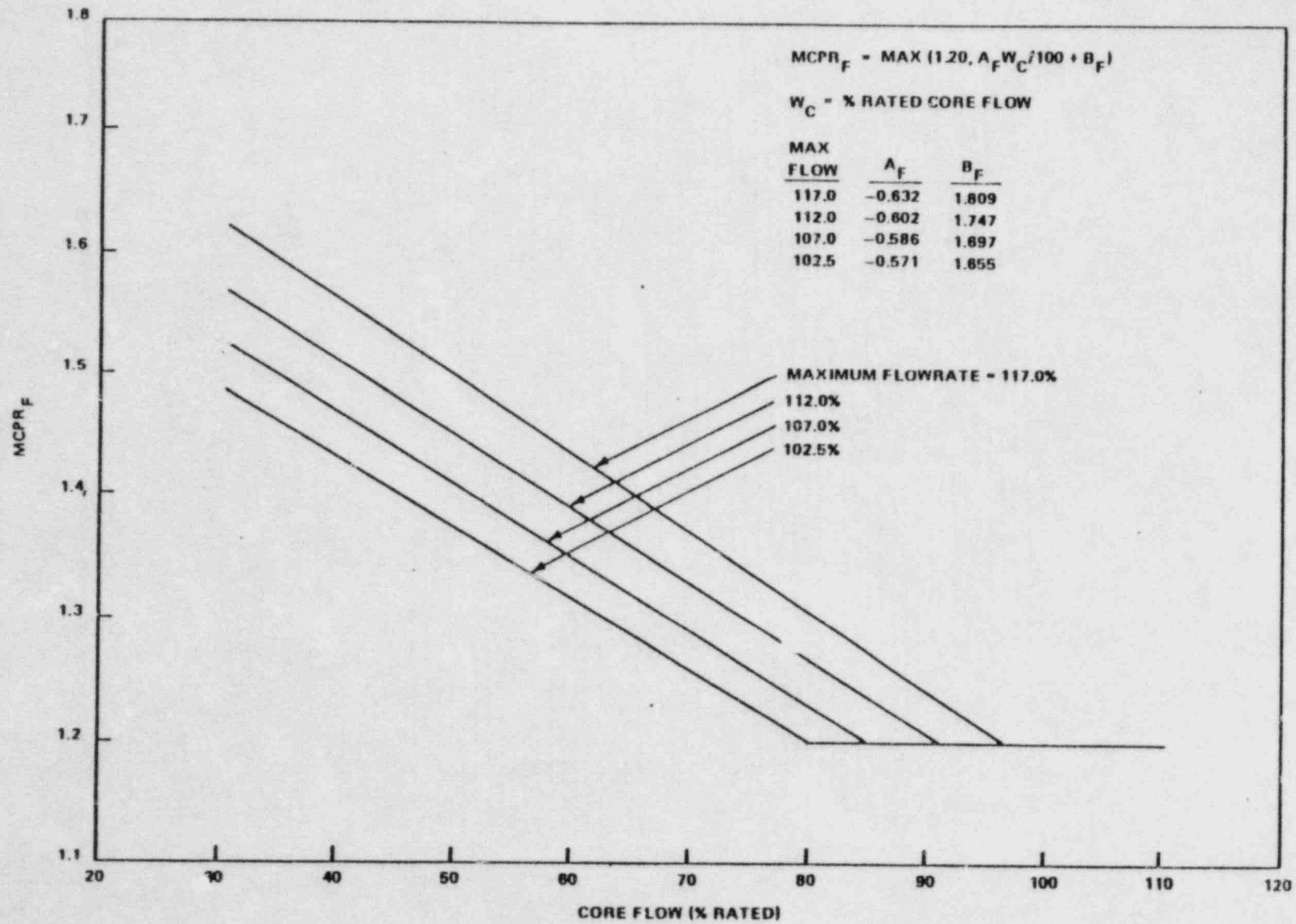


FIGURE 3.2.3-4 $MCPR_F$

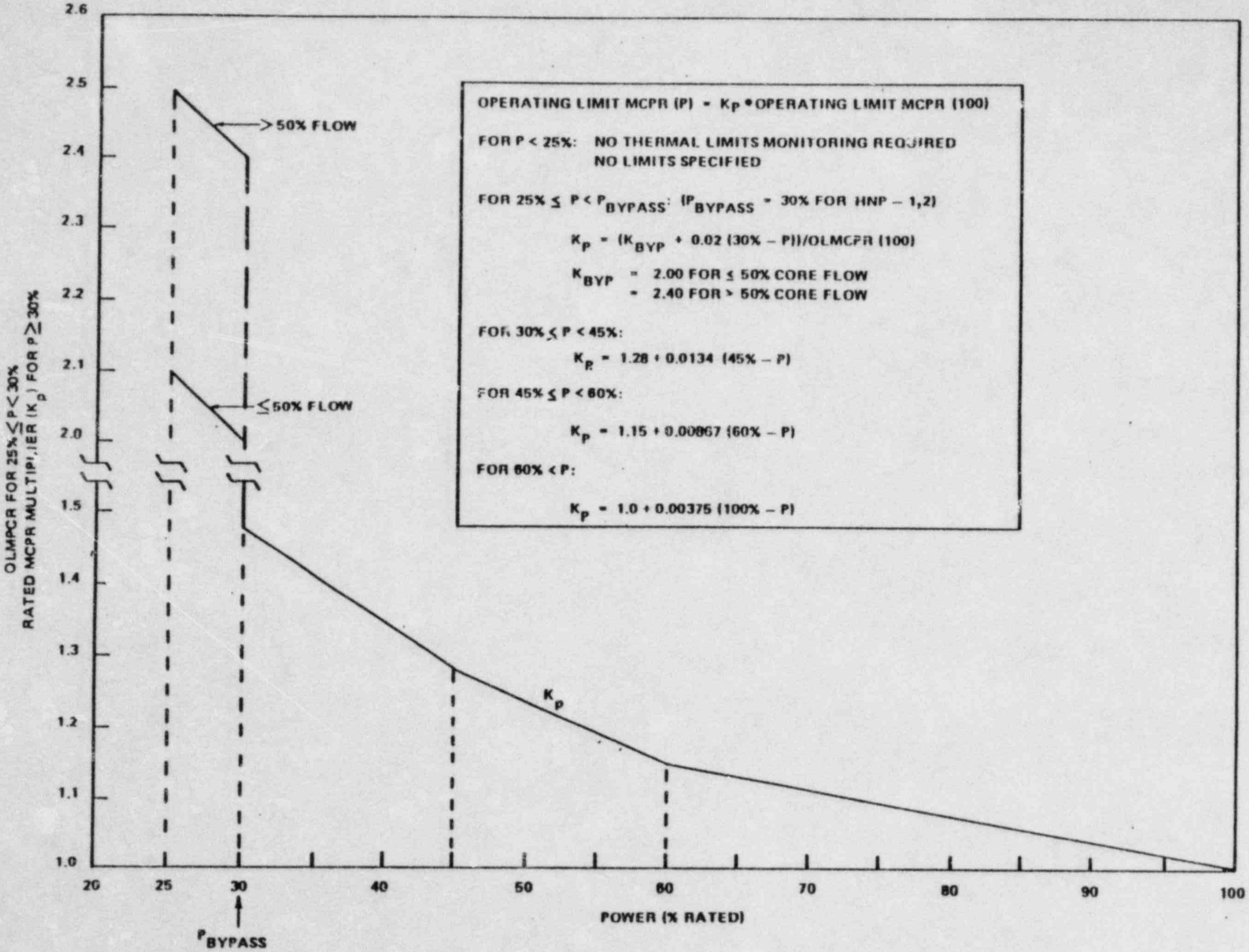


FIGURE 3.2.3-5 Kp

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1 As a minimum, the reactor protection system instrumentation channels shown in Table 3.3.1-1 shall be OPERABLE with the REACTOR PROTECTION SYSTEM RESPONSE TIME as shown in Table 3.3.1-2. Set points and interlocks are given in Table 2.2.1-1.

APPLICABILITY: As shown in Table 3.3.1-1.

ACTION:

a. With the requirements for the minimum number of OPERABLE channels not satisfied for one trip system, place at least one inoperable channel in the tripped condition within one hour.

b. With the requirements for the minimum number of OPERABLE channels not satisfied for both trip systems, place at least one inoperable channel in at least one trip system* in the tripped condition within one hour and take the ACTION required by Table 3.3.1-1.

c. The provisions of Specification 3.0.3 are not applicable in OPERATIONAL CONDITION 5.

SURVEILLANCE REQUIREMENTS

4.3.1.1 Each reactor protection system instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTION TEST and CHANNEL CALIBRATION operations during the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.1-1.

4.3.1.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed at least once per 18 months and shall include calibration of time delay relays and timers necessary for proper functioning of the trip system.

4.3.1.3 The REACTOR PROTECTION SYSTEM RESPONSE TIME of each reactor trip function of Table 3.3.1-2 shall be demonstrated to be within its limit at least once per 18 months. Each test shall include at least one logic train such that both logic trains are tested at least once per 36 months and one channel per function such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific reactor trip function.

*If both channels are inoperable in one trip system, select at least one inoperable channel in that trip system to place in the tripped conditions, except when this could cause the Trip Function to occur.

TABLE 3.3.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	APPLICABLE OPERATIONAL CONDITIONS	MINIMUM NUMBER OPERABLE CHANNELS PER TRIP SYSTEM(a)	ACTION
1. Intermediate Range Monitors: (2C51-K601, A, B, C, D, E, F, G, H)			
a. Neutron Flux - High	2(c), 5(b)	3	1
	3, 4	2	2
b. Inoperative	2,5(b)	3	1
	3,4	2	2
2. Average Power Range Monitor: (2C51-K605 A, B, C, D, E, F)			
a. Neutron Flux - Upscale, 15%	2, 5	2	1
b. Flow Referenced Simulated Thermal Power - Upscale	1	2	3
c. Fixed Neutron Flux - Upscale, 118%	1	2	3
d. Inoperative	1, 2, 5	2	4
e. Downscale	1	2	3
f. LPRM	1, 2, 5	(d)	NA
3. Reactor Vessel Steam Dome Pressure - High (2B21-N678 A, B, C, D)	1, 2(e)	2(j, 2B21-NC45 A, B, C, D)	5
4. Reactor Vessel Water Level - Low (Level 3) (2B21-N680 A, B, C, D)	1, 2	2(j, 2B21-N681 A, B, C, D)	5
5. Main Steam Line Isolation Valve - Closure (NA)	1(f)	4	3
6. Main Steam Line Radiation - High (2D11-K603 A, B, C, D)	1, 2(e)	2	6
7. Drywell Pressure - High (2C71-N650 A, B, C, D)	1, 2(g)	2	5

TABLE 4.3.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION(a)	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
1. Intermediate Range Monitors:				
a. Neutron Flux - High	D	S/U(b)(c)	R	2
	D	W	R	3, 4, 5
b. Inoperative	NA	W	NA	2, 3, 4, 5
2. Average Power Range Monitor:				
a. Neutron Flux - Upscale, 15%	S	S/U(b)(c), W(d)	S/U(b), W(d)	2
	S	W	W	5
b. Flow Referenced Simulated Thermal Power - Upscale	S	S/U(b), W	W(e)(f), SA	1
c. Fixed Neutron Flux - Upscale, 118%	S	S/U(b), W	W(e), SA	1
d. Inoperative	NA	W	NA	1, 2, 5
e. Downscale	NA	W	NA	1
f. LPRM	D	NA	(g)	1, 2, 5
3. Reactor Vessel Steam Dome Pressure - High	S	M	R	1, 2
4. Reactor Vessel Water Level - Low (Level 3)	S	M	R	1, 2
5. Main Steam Line Isolation Valve - Closure	NA	M	R(h)	1
6. Main Steam Line Radiation - High	D	W(i)	R(j)	1, 2
7. Drywell Pressure - High	S	M	R	1, 2
8. Scram Discharge Volume Water Level - High	NA	M	R(h)	1, 2, 5

TABLE 4.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
9. Turbine Stop Valve - Closure	NA	M	$R^{(h)}$	1
10. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	NA	M	R	1
11. Reactor Mode Switch in Shutdown Position	NA	R	NA	1, 2, 3, 4, 5
12. Manual Scram	NA	M	NA	1, 2, 3, 4, 5

a. Neutron detectors may be excluded from CHANNEL CALIBRATION.

b. Within 24 hours prior to startup, if not performed within the previous 7 days.

c. The APRM, IRM and SRM channels shall be compared for overlap during each startup, if not performed within the previous 7 days.

d. When changing from CONDITION 1 to CONDITION 2, perform the required surveillance within 12 hours after entering CONDITION 2.

e. This calibration shall consist of the adjustment of the APRM channel to conform to the power values calculated by a heat balance during CONDITION 1 when THERMAL POWER $\geq$ 25% of RATED THERMAL POWER. Adjust the APRM channel if the absolute difference $\geq$ 2%.

f. This calibration shall consist of the adjustment of the APRM flow referenced simulated thermal power channel to conform to a calibrated flow signal.

g. The LPRM's shall be calibrated at least once per 1000 effective full power hours (EFPH) using the TIP system.

h. Physical inspection and actuation of switches for instruments 2C11-N013A, B, C, D.

i. Instrument alignment using a standard current source.

j. Calibration using a standard radiation source.

TABLE 3.3.2-1

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	VALVE GROUPS OPERATED BY SIGNAL(a)	MINIMUM NUMBER OPERABLE CHANNELS PER TRIP SYSTEM(b)(c)	APPLICABLE OPERATIONAL CONDITION	ACTION
1. PRIMARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level				
1. Low (Level 3) (2B21-N680 A, B, C, D)	2, 6, 10, 11, 12	2	1, 2, 3	20
2. Low-Low (Level 2) (2B21-N682 A, B, C, D)	5, #, *	2	1, 2, 3	20
3. Low-Low-Low (Level 1) (2B21-N681 A, B, C, D)	1	2	1, 2, 3	20
b. Drywell Pressure - High (2C71-N650 A, B, C, D)	2, 6, 7, 10, 12, #, *	2	1, 2, 3	20
c. Main Steam Line				
1. Radiation - High (2D11-K603 A, B, C, D)	1, 12, #, (d)	2	1, 2, 3	21
2. Pressure - Low (2B21-N015 A, B, C, D)	1	2	1	22
3. Flow - High (2B21-N686 A, B, C, D) (2B21-N687 A, B, C, D) (2B21-N688 A, B, C, D) (2B21-N689 A, B, C, D)	1, #	2/line	1, 2, 3	21
d. Main Steam Line Tunnel Temperature - High (2B21-N623 A, B, C, D) (2B21-N624 A, B, C, D) (2B21-N625 A, B, C, D) (2B21-N626 A, B, C, D)	1	2/line(e)	1, 2, 3	21
e. Condenser Vacuum - Low (2B21-N056 A, B, C, D)	1	2	1, 2,(f), 3(f)	23
f. Turbine Building Area Temperature - High (2U61-R001, 2U61-R002, 2U61-R003, 2U61-R004)	1	2(e)	1, 2, 3	21

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	VALVE GROUPS OPERATED BY SIGNAL(a)	MINIMUM NUMBER OPERABLE CHANNELS PER TRIP SYSTEM(b)(c)	APPLICABLE OPERATIONAL CONDITION	ACTION
2. SECONDARY CONTAINMENT ISOLATION				
a. Reactor Building Exhaust Radiation - High (2D11-K609 A, B, C, D)	6, 10, 12, *	2	1,2,3,5 and**	24
b. Drywell Pressure - High (2C71-N650 A, B, C, D)	2, 6, 7, 10, 12, #, *	2	1, 2, 3	24
c. Reactor Vessel Water Level - Low Low (Level 2) (2B21-N682 A, B, C, D)	5, #, *	2	1, 2, 3	24
d. Refueling Floor Exhaust Radiation - High (2D11-K611 A, B, C, D)	6, 10, 12, #, *	2	1,2,3,5 and**	24
3. REACTOR WATER CLEANUP SYSTEM ISOLATION				
a. Δ Flow - High (2G31-N603 A, B)	5	1	1, 2, 3	25
b. Area Temperature - High (2G31-N662 A, D, E, H, J, M)	5	1	1, 2, 3	25
c. Area Ventilation Δ Temp. - High (2G31-N663 A, D, E, H, J, M; 2G31-N661 A, D, E, H, J, M; 2G31-N662 A, D, E, H, J, M)	5	1	1, 2, 3	25
d. SLCS Initiation (NA)	5(g)	NA	1, 2, 3	25
e. Reactor Vessel Water Level - Low Low (Level 2) (2B21-N682 A, B, C, D)	5, #, *	2	1, 2, 3	25

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	VALVE GROUPS OPERATED BY SIGNAL(a)	MINIMUM NUMBER OPERABLE CHANNELS PER TRIP SYSTEM(b)(c)	APPLICABLE OPERATIONAL CONDITION	ACTION
4. HIGH PRESSURE COOLANT INJECTION SYSTEM ISOLATION				
a. HPCI Steam Line Flow - High (2E41-N657 A,B)	3	1	1, 2, 3	26
b. HPCI Steam Supply Pressure - Low (2E41-N658 A,B,C,D)	3,8	2	1, 2, 3	26
c. HPCI Turbine Exhaust Diaphragm Pressure - High (2E41-N655 A,B,C,D)	3	2	1, 2, 3	26
d. HPCI Pipe Penetration Room Temperature - High (2E41-N671 A, B)	3	1	1, 2, 3	26
e. Suppression Pool Area Ambient Temperature-High (2E51-N666 C, D)	3	1	1, 2, 3	26
f. Suppression Pool Area Δ Temp.-High (2E51-N665 C, D; 2E51-N663 C, D; 2E51-N664 C, D)	3	1	1, 2, 3	26
g. Suppression Pool Area Temperature Timer Relays (2E41-M603 A, B)	3(i)	1	1, 2, 3	26
h. Emergency Area Cooler Temperature-High (2E41-N670 A, B)	3	1	1, 2, 3	26
i. Drywell Pressure-High (2E11-N694 C, D)	8	1	1, 2, 3	26
j. Logic Power Monitor (2E41-K1)	NA(h)	1	1, 2, 3	27

TABLE 3.3.2-1 (Continued)
ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	VALVE GROUPS OPERATED BY SIGNAL(a)	MINIMUM NUMBER OPERABLE CHANNELS PER TRIP SYSTEM(b)(c)	APPLICABLE OPERATIONAL CONDITION	ACTION
5. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION				
a. RCIC Steam Line Flow-High (2E51-N657 A,B)	4	1	1, 2, 3	26
b. RCIC Steam Supply Pressure - Low (2E51-N658 A, B, C, D)	4, 9	2	1, 2, 3	26
c. RCIC Turbine Exhaust Diaphragm Pressure - High (2E51-N685 A, B, C, D)	4	2	1, 2, 3	26
d. Emergency Area Cooler Temperature - High (2E51-N661 A, B)	4	1	1, 2, 3	26
e. Suppression Pool Area Ambient Temperature-High (2E51-N666 A, B)	4	1	1, 2, 3	26
f. Suppression Pool Area Δ T-High (2E51-N665 A, B; 2E51-N663 A,B; 2E51-N664 A,B)	4	1	1, 2, 3	26
g. Suppression Pool Area Temperature Timer Relays (2E51-M602 A, B)	4(i)	1	1, 2, 3	26
h. Drywell Pressure - High (2E11-N694 A, B)	9	1	1, 2, 3	26
i. Logic Power Monitor (2E51-K1)	NA(h)	1	1, 2, 3	27
6. SHUTDOWN COOLING SYSTEM ISOLATION				
a. Reactor Vessel Water Level-Low (Level 3)(2B21-N680 A, B, C, D)	6, 10, 11, 2 12	2	3, 4, 5	26
b. Reactor Steam Dome Pressure-High (2B31-N679 A, D)	11	1	1, 2, 3	28

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

ACTION

ACTION 20 - Be in at least HOT SHUTDOWN within 6 hours and in COLD SHUTDOWN within the next 30 hours.

ACTION 21 - Be in at least STARTUP with the main steam line isolation valves closed within 2 hours or be in at least HOT SHUTDOWN within 6 hours and in COLD SHUTDOWN within the next 30 hours.

ACTION 22 - Be in at least STARTUP within 2 hours.

ACTION 23 - Be in at least STARTUP with the Group 1 isolation valves closed within 2 hours or in at least HOT SHUTDOWN within 6 hours.

ACTION 24 - Establish SECONDARY CONTAINMENT INTEGRITY with the standby gas treatment system operating within one hour.

ACTION 25 - Isolate the reactor water cleanup system.

ACTION 26 - Close the affected system isolation valves and declare the affected system inoperable.

ACTION 27 - Verify power availability to the bus at least once per 12 hours or close the affected system isolation valves and declare the affected system inoperable.

ACTION 28 - Close the shutdown cooling supply and reactor vessel head spray isolation valves unless reactor steam dome pressure $\leq$ 145 psig.

NOTES

Actuates operation of the main control room environmental control system in the pressurization mode of operation.

* Actuates the standby gas treatment system.

** When handling irradiated fuel in the secondary containment.

a. See Specification 3.6.3, Table 3.6.3-1 for valves in each valve group.

b. A channel may be placed in an inoperable status for up to 2 hours for required surveillance without placing the trip system in the tripped condition provided at least one other OPERABLE channel in the same trip system is monitoring that parameter.

c. With a design providing only one channel per trip system, an inoperable channel need not be placed in the tripped condition where this would cause the Trip Function to occur. In these cases, the inoperable channel shall be restored to OPERABLE status within 2 hours or the ACTION required by Table 3.3.2-1 for that Trip Function shall be taken.

d. Trips the mechanical vacuum pumps.

e. A channel is OPERABLE if 2 of 4 instruments in that channel are OPERABLE.

f. May be bypassed with all turbine stop valves closed.

g. Closes only RWCU outlet isolation valve 2G31-F004.

h. Alarm only.

i. Adjustable up to 60 minutes.

TABLE 3.3.2-2

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. **PRIMARY CONTAINMENT ISOLATION**		
a. Reactor Vessel Water Level		
1. Low (Level 3)	≥ 8.5 inches*	≥ 8.5 inches*
2. Low Low (Level 2)	≥ -55 inches*	≥ -55 inches*
3. Low Low Low (Level 1)	≥ -121.5 inches*	≥ -121.5 inches*
b. Drywell Pressure - High	≤ 1.85 psig	≤ 1.85 psig
c. Main Steam Line		
1. Radiation - High	≤ 3 x full power background	≤ 3 x full power background
2. Pressure - Low	≥ 825 psig	≥ 825 psig
3. Flow - High	≤ 138% rated flow	≤ 138% rated flow
d. Main Steam Line Tunnel Temperature - High	≤ 194°F	≤ 194°F
e. Condenser Vacuum - Low	≥ 7" Hg vacuum	≥ 7" Hg vacuum
f. Turbine Building Area Temp.-High	≤ 200°F	≤ 200°F
2. **SECONDARY CONTAINMENT ISOLATION**		
a. Reactor Building Exhaust Radiation - High	≤ 60 mr/hr	≤ 60 mr/hr
b. Drywell Pressure - High	≤ 1.85 psig	≤ 1.85 psig
c. Reactor Vessel Water Level - Low Low (Level 2)	≥ -55 inches*	≥ -55 inches*
d. Refueling Floor Exhaust Radiation - High	≤ 20 mr/hr	≤ 20 mr/hr

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

TABLE 3.3.2-2 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
3. REACTOR WATER CLEANUP SYSTEM ISOLATION		
a. Δ Flow - High	≤ 79 gpm	≤ 79 gpm
b. Area Temperature-High	≤ 124°F	≤ 124°F
c. Area Ventilation Δ Temperature - High	≤ 67°F	≤ 67°F
d. SLCS Initiation	NA	NA
e. Reactor Vessel Water Level-Low Low (Level 2)	≥ -55 inches*	≥ -55 inches*
4. HIGH PRESSURE COOLANT INJECTION SYSTEM ISOLATION		
a. HPCI Steam Line Flow-High	≤ 307% of rated flow	≤ 307% of rated flow
b. HPCI Steam Supply Pressure - Low	≥ 100 psig	≥ 100 psig
c. HPCI Turbine Exhaust Diaphragm Pressure-High	≤ 20 psig	≤ 20 psig
d. HPCI Pipe Penetration Room Temperature - High	≤ 169°F	≤ 169°F
e. Suppression Pool Area Ambient Temperature-High	≤ 169°F	≤ 169°F
f. Suppression Pool Area ΔT - High	≤ 42.5°F	≤ 42.5°F
g. Suppression Pool Area Temperature Timer Relays	NA	NA
h. Emergency Area Cooler Temperature - High	≤ 169°F	≤ 169°F
i. Drywell Pressure - High	≤ 1.85 psig	≤ 1.85 psig
j. Logic Power Bus Monitors	NA	NA

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

TABLE 3.3.2-2 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
5. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION		
a. RCIC Steam Line Flow - High	≤ 312% of rated flow	≤ 312% of rated flow
b. RCIC Steam Supply Pressure - Low	≥ 60 psig	≥ 60 psig
c. RCIC Turbine Exhaust Diaphragm Pressure - High	≤ 20 psig	≤ 20 psig
d. Emergency Area Cooler Temperature-High	≤ 169°F	≤ 169°F
e. Suppression Pool Area Ambient Temperature High	≤ 169°F	≤ 169°F
f. Suppression Pool Area ΔT - High	≤ 42.5°F	≤ 42.5°F
g. Suppression Pool Area Temperature Timer Relays	NA	NA
h. Drywell Pressure - High	≤ 1.85 psig	≤ 1.85 psig
i. Logic Power Monitor	NA	NA
6. SHUTDOWN COOLING SYSTEM ISOLATION		
a. Reactor Vessel Water Level - Low (Level 3)	≥ 8.5 inches*	≥ 8.5 inches*
b. Reactor Steam Dome Pressure - High	≤ 145 psig	≤ 145 psig

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

TABLE 3.3.2-3

ISOLATION SYSTEM INSTRUMENTATION RESPONSE TIME

TRIP FUNCTION	RESPONSE TIME (Seconds)#
1. PRIMARY CONTAINMENT ISOLATION	
a. Reactor Vessel Water Level	
1. Low (Level 3)	≤ 13*
2. Low Low (Level 2)	≤ 13*
3. Low Low Low (Level 1), except MSIVs	≤ 13*
b. Drywell Pressure - High	≤ 13*
c. Main Steam Line	
1. Radiation - High***	≤ 1.0**
2. Pressure - Low	≤ 13*
3. Flow - High	≤ 1.0**
4. Reactor Vessel Water Level - Low Low Low (Level 1)	≤ 1.0**
d. Main Steam Line Tunnel Temperature - High	≤ 13*
e. Condenser Vacuum - Low	NA
f. Turbine Building Area Temperature - High	NA
2. SECONDARY CONTAINMENT ISOLATION	
a. Reactor Building Exhaust Radiation - High***	≤ 13*
b. Drywell Pressure - High	≤ 13*
c. Reactor Vessel Water Level - Low Low (Level 2)	≤ 13*
d. Refueling Floor Exhaust Radiation - High***	≤ 13*

*The isolation actuation instrumentation response time shall be measured and recorded as a part of the ISOLATION SYSTEM RESPONSE TIME. Response time specified is diesel generator start delay time assumed in accident analysis.

**Isolation actuation instrumentation response time.

***Radiation detectors are exempt from response time testing. Response time shall be measured from detector output or the input of the first electronic component in the channel.

#Times to be added to valve movement times shown in Tables 3.6.3-1, 3.6.5.2-1 and 3.9.5 2-1 to obtain ISOLATION SYSTEM RESPONSE TIME for each valve.

TABLE 3.3.2-3 (Continued)

ISOLATION SYSTEM INSTRUMENTATION RESPONSE TIME

TRIP FUNCTION	RESPONSE TIME (Seconds)#
3. REACTOR WATER CLEANUP SYSTEM ISOLATION	
a. Δ Flow - High	≤13*
b. Area Temperature - High.	≤13*
c. Area Ventilation Temperature ΔT - High	≤13*
d. SLCS Initiation	NA
e. Reactor Vessel Water Level-Low Low (Level 2)	≤13*
4. HIGH PRESSURE COOLANT INJECTION SYSTEM ISOLATION	
a. HPCI Steam Line Flow-High	3 ≤ Isolation Time ≤ 13*
b. HPCI Steam Supply Pressure - Low	≤13*
c. HPCI Turbine Exhaust Diaphragm Pressure - High	NA
d. HPCI Pipe Penetration Room Temperature - High	NA
e. Suppression Pool Area Ambient Temp. - High	NA
f. Suppression Pool Area ΔT - High	NA
g. Suppression Pool Area Temp. Timer Relays	NA
h. Emergency Area Cooler Temperature - High	NA.
i. Drywell Pressure - High	≤13*
j. Logic Power Monitor	NA
5. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION	
a. RCIC Steam Line Flow - High	3 ≤ Isolation Time ≤ 13*
b. RCIC Steam Supply Pressure - Low	NA
c. RCIC Turbine Exhaust Diaphragm Pressure - High	NA
d. Emergency Area Cooler Temperature - High	NA
e. Suppression Pool Area Ambient Temp. - High	NA
f. Suppression Pool Area ΔT - High	NA
g. Suppression Pool Area Temperature Timer Relays	NA
h. Drywell Pressure - High	≤13*
i. Logic Power Monitor	NA
6. SHUTDOWN COOLING SYSTEM ISOLATION	
a. Reactor Vessel Water Level - Low (Level 3)	NA
b. Reactor Steam Dome Pressure - High	NA

TABLE 4.3.2-1

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
1. PRIMARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level				
1. Low (Level 3)	S	M	R	1, 2, 3
2. Low Low (Level 2)	S	M	R	1, 2, 3
3. Low Low Low (Level 1)	S	M	R	1, 2, 3
b. Drywell Pressure - High	S	M	R	1, 2, 3
c. Main Steam Line				
1. Radiation - High	D	W(a)	R	1, 2, 3
2. Pressure - Low	NA	M	Q	1
3. Flow - High	S	M	R	1, 2, 3
d. Main Steam Line Tunnel Temperature - High	S	M	R	1, 2, 3
e. Condenser Vacuum - Low	NA	M	Q	1, 2#, 3#
f. Turbine Building Area Temp. - High	NA	M	R	1, 2, 3
2. SECONDARY CONTAINMENT ISOLATION				
a. Reactor Building Exhaust Radiation - High	D	M(a)	R	1, 2, 3, 5 and *
b. Drywell Pressure - High	S	M	R	1, 2, 3
c. Reactor Vessel Water Level - Low Low (Level 2)	S	M	R	1, 2, 3
d. Refueling Floor Exhaust Radiation - High	D	M(a)	Q	1, 2, 3, 5 and *

*When handling irradiated fuel in the secondary containment.
#May be bypassed with all turbine stop valves closed.
aInstrument alignment using a standard current source.

TABLE 4.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
3. REACTOR WATER CLEANUP SYSTEM ISOLATION				
a. Δ Flow - High	D	M	R	1, 2, 3
b. Area Temperature - High	S	M	R	1, 2, 3
c. Area Ventilation Δ Temperature - High	S	M	R	1, 2, 3
d. SLCS Initiation	NA	R	NA	1, 2, 3
e. Reactor Vessel Water Level - Low Low (Level 2)	S	M	R	1, 2, 3
4. HIGH PRESSURE COOLANT INJECTION SYSTEM ISOLATION				
a. HPCI Steam Line Flow-High	S	M	R	1, 2, 3
b. HPCI Steam Supply Pressure-Low	S	M	R	1, 2, 3
c. HPCI Turbine Exhaust Diaphragm Pressure - High	S	M	R	1, 2, 3
d. HPCI Pipe Penetration Room Temperature - High	S	M	R	1, 2, 3
e. Suppression Pool Area Ambient Temp. - High	S	M	R	1, 2, 3
f. Suppression Pool Area ΔT - High	S	M	R	1, 2, 3
g. Suppression Pool Area Temp. Timer Relays	NA	SA	R	1, 2, 3
h. Emergency Area Cooler Temp. - High	S	M	R	1, 2, 3
i. Drywell Pressure - High	S	M	R	1, 2, 3
j. Logic Power Monitor	NA	R	NA	1, 2, 3

TABLE 4.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
5. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION				
a. RCIC Steam Line Flow-High	S	M	R	1, 2, 3
b. RCIC Steam Supply Pressure-Low	S	M	R	1, 2, 3
c. RCIC Turbine Exhaust Diaphragm Pressure-High	S	M	R	1, 2, 3
d. Emergency Area Cooler Temperature - High	S	M	R	1, 2, 3
e. Suppression Pool Area Ambient Temperature-High	S	M	R	1, 2, 3
f. Suppression Pool Area ΔT - High	S	M	R	1, 2, 3
g. Suppression Pool Area Temp. Timer Relays	NA	SA	R	1, 2, 3
h. Drywell Pressure - High	S	M	R	1, 2, 3
i. Logic Power Monitor	NA	R	NA	1, 2, 3
6. SHUTDOWN COOLING SYSTEM ISOLATION				
a. Reactor Vessel Water Level - Low (Level 3)	S	M	R	3, 4, 5
b. Reactor Steam Dome Pressure - High	S	M	R	1, 2, 3

3/4.3.3 EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3 The emergency core cooling system (ECCS) actuation instrumentation shown in Table 3.3.3-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.3-2 and with EMERGENCY CORE COOLING SYSTEM RESPONSE TIME as shown in Table 3.3.3-3.

APPLICABILITY: As shown in Table 3.3.3-1.

ACTION:

a. With an ECCS actuation instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.3-2, declare the channel inoperable and place the inoperable channel in the tripped condition until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

b. With the requirements for the minimum number of OPERABLE channels not satisfied for one trip system, place the inoperable channel in the tripped condition or declare the associated ECCS inoperable within one hour.

c. With the requirements for the minimum number of OPERABLE channels not satisfied for both trip systems, declare the associated ECCS inoperable within one hour.

d. The provisions of Specification 3.0.3 are not applicable in OPERATIONAL CONDITION 5.

SURVEILLANCE REQUIREMENTS

4.3.3.1 Each ECCS actuation instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, and CHANNEL CALIBRATION operations during the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.3-1.

4.3.3.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed at least once per 18 months and shall include calibration of time delay relays and timers necessary for proper functioning of the trip system.

SURVEILLANCE REQUIREMENTS (Continued)

4.3.3.3 The ECCS RESPONSE TIME of each ECCS function shown in Table 3.3.3-3 shall be demonstrated to be within the limit at least once per 18 months. Each test shall include at least one logic train such that both logic trains are tested at least once per 36 months and one channel per function such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific ECCS function.

TABLE 3.3.3-1

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

TRIP FUNCTION	MINIMUM NUMBER OPERABLE CHANNELS PER TRIP SYSTEM	APPLICABLE OPERATIONAL CONDITIONS
1. CORE SPRAY SYSTEM		
a. Reactor Vessel Water Level - Low Low Low (Level 1) (2B21-N691A,B,C,D)	2	1,2,3,4,5
b. Drywell Pressure - High (2E11-N694 A,B,C,D)	2	1,2,3
c. Reactor Steam Dome Pressure - Low(Injection Permissive) (2B21-N690A,B,C,D)	2	1,2,3,4,5
d. Logic Power Monitor (2E21-K1A,B)	1/bus[(a)]	1,2,3,4,5
2. LOW PRESSURE COOLANT INJECTION MODE OF RHR SYSTEM		
a. Drywell Pressure - High (2E11-N694A,B,C,D)	2	1,2,3
b. Reactor Vessel Water Level - Low Low Low (Level 1) (2B21-N691A,B,C,D)	2	1,2,3,4*,5*
c. Reactor Vessel Shroud Level (Level 0) - High (Drywell Spray Permissive) (2B21-N685A, B)	1	1,2,3,4*,5*
d. Reactor Steam Dome Pressure - Low (Injection Permissive) (2B21-N690A,B,C,D)	2	1,2,3,4*,5*
e. Reactor Steam Dome Pressure - Low (Recirc. Discharge Valve Permissive) (2B21-N641B,C and 2B21-N690E,F)	2	1,2,3,4*,5*
f. RHR Pump Start - Time Delay Relay	1/pump	1,2,3,4*,5*
1) Pump A (2E11-K70A, 2E11-K125B)		
2) Pump B (2E11-K70B, 2E11-K125A)		
3) Pump C (2E11-K75B)		
4) Pump D (2E11-K75A, 2E11-K126)		
g. Logic Power Monitor (2E11-K1A,B)	1/bus[(a)]	1,2,3,4*,5*

* Not applicable when two core spray system subsystems are OPERABLE per Specification 3.5.3.1.

(a) Alarm only. When inoperable, verify power availability to the bus at least once per 12 hours or declare the system inoperable.

TABLE 3.3.3-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

TRIP FUNCTION	MINIMUM NUMBER OPERABLE CHANNELS PER TRIP SYSTEM	APPLICABLE OPERATIONAL CONDITIONS#
3. HIGH PRESSURE COOLANT INJECTION SYSTEM		
a. Reactor Vessel Water Level - Low Low (Level 2) (2B21-N692 A,B,C,D)	2	1, 2, 3
b. Drywell Pressure - High (2E11-N694 A,B,C,D)	2	1, 2, 3
c. Condensate Storage Tank Level-Low (2E41-N002, 2E41-N003)	2(b)(c)	1, 2, 3
d. Suppression Chamber Water Level-High (2E41-N662B,D)	2(b)(c)	1, 2, 3
e. Logic Power Monitor (2E41-K1)	1(a)	1, 2, 3
f. Reactor Vessel Water Level-High (Level 8) (2B21-N693 B,D)	2	1, 2, 3
4. AUTOMATIC DEPRESSURIZATION SYSTEM		
a. Drywell Pressure - High (Permissive) (2E11-N694A,B,C,D)	2	1, 2, 3
b. Reactor Vessel Water Level - Low Low Low (Level 1) (2B21-N691 A,B,C,D)	2	1, 2, 3
c. ADS Timer (2B21-K5A,B)	1	1, 2, 3
d. Reactor Vessel Water Level-Low (Level 3) (Permissive) (2B21-N695A,B)	1	1, 2, 3
e. Core Spray Pump Discharge Pressure - High (Permissive) (2E21-N655A,B; 2E21-N652A,B)	2	1, 2, 3
f. RHR (LPCI MODE) Pump Discharge Pressure - High (Permissive) (2E11-N655A,B,C,D; 2E11-N656A,B,C,D)	2/loop	1, 2, 3
g. Control Power Monitor (2B21-K1A,B)	1/bus(a)	1, 2, 3
5. LOW LOW SET S/RV SYSTEM		
a. Reactor Steam Dome Pressure - High (Permissive) (2B21-N620A,B,C,D)	2	1, 2, 3

(a) Alarm only. When inoperable, verify power availability to the bus at least once per 12 hours or declare the system inoperable.
(b) Provides signal to HPCI pump suction valves only.
(c) When either channel of the automatic transfer logic is inoperable, align HPCI pump suction to the suppression pool.
\# HPCI and ADS are not required to be OPERABLE with reactor steam dome pressure $\leq$ 150 psig.

TABLE 3.3.3-2

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. CORE SPRAY SYSTEM		
a. Reactor Vessel Water Level - Low Low Low (Level 1)	$\geq$ -121.5 inches*	$\geq$ -121.5 inches*
b. Drywell Pressure - High	$\leq$ 1.85 psig	$\leq$ 1.85 psig
c. Reactor Steam Dome Pressure - Low	$\geq$ 422 psig**	$\geq$ 422 psig**
d. Logic Power Monitor	NA	NA
2. LOW PRESSURE COOLANT INJECTION MODE OF RHR SYSTEM		
a. Drywell Pressure - High	$\leq$ 1.85 psig	$\leq$ 1.85 psig
b. Reactor Vessel Water Level - Low Low Low (Level 1)	$\geq$ -121.5 inches*	$\geq$ -121.5 inches*
c. Reactor Vessel Shroud Level (Level 0) - High	$\geq$ -207 inches*	$\geq$ -207 inches*
d. Reactor Steam Dome Pressure-Low	$\geq$ 422 psig**	$\geq$ 422 psig**
e. Reactor Steam Dome Pressure-Low	$\geq$ 325 psig	$\geq$ 325 psig
f. RHR Pump Start - Time Delay Relay		
1) Pumps A, B and D	10 ± 1 seconds	10 ± 1 seconds
2) Pump C	0.5 ± 0.5 seconds	0.5 ± 0.5 seconds
g. Logic Power Monitor	NA	NA

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

**This trip function shall be less than or equal to 500 psig.

TABLE 3.3.3-2 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
3. HIGH PRESSURE COOLANT INJECTION SYSTEM		
a. Reactor Vessel Water Level - Low Low (Level 2)	$\geq$ -55 inches*	$\geq$ -55 inches*
b. Drywell Pressure-High	$\leq$ 1.85 psig	$\leq$ 1.85 psig
c. Condensate Storage Tank Level - Low	$\geq$ 0 inches**	$\geq$ 0 inches**
d. Suppression Chamber Water Level - High	$\leq$ 33.2 inches	$\leq$ 33.2 inches
e. Logic Power Monitor	NA	NA
f. Reactor Vessel Water Level-High (Level 8)*	$\leq$ 56.5 inches	$\leq$ 56.5 inches
4. AUTOMATIC DEPRESSURIZATION SYSTEM		
a. Drywell Pressure-High	$\leq$ 1.85 psig	$\leq$ 1.85 psig
b. Reactor Vessel Water Level - Low Low Low (Level 1)	$\geq$ -121.5 inches*	$\geq$ -121.5 inches*
c. ADS Timer	$\leq$ 120 seconds	$\leq$ 120 seconds
d. Reactor Vessel Water Level-Low (Level 3)	$\geq$ 8.5 inches*	$\geq$ 8.5 inches*
e. Core Spray Pump Discharge Pressure - High	$\geq$ 130 psig	$\geq$ 130 psig
f. RHR (LPCI MODE) Pump Discharge Pressure - High	$\geq$ 105 psig	$\geq$ 105 psig
g. Control Power Monitor	NA	NA
5. LOW LOW SET S/RV SYSTEM		
a. Reactor Steam Dome Pressure - High	$\leq$ 1054 psig	$\leq$ 1054 psig

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

** Equivalent to 10,000 gallons of water in the CST.

TABLE 3.3.3-3

EMERGENCY CORE COOLING SYSTEM RESPONSE TIMES

ECCS	RESPONSE TIME (Seconds)
1. CORE SPRAY SYSTEM	≤ 27
2. LOW PRESSURE COOLANT INJECTION MODE OF RHR SYSTEM	≤ 40
3. HIGH PRESSURE COOLANT INJECTION SYSTEM	≤ 30
4. AUTOMATIC DEPRESSURIZATION SYSTEM	NA
5. ARM LOW LOW SET SYSTEM	≤ 1

TABLE 4.3.3-1

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
1. CORE SPRAY SYSTEM				
a. Reactor Vessel Water Level - Low Low Low (Level 1)	S	M	R	1, 2, 3, 4, 5
b. Drywell Pressure - High	S	M	R	1, 2, 3
c. Reactor Steam Dome Pressure - Low	S	M	R	1, 2, 3, 4, 5
d. Logic Power Monitor	NA	R	NA	1, 2, 3, 4, 5
2. LOW PRESSURE COOLANT INJECTION MODE OF RHR SYSTEM				
a. Drywell Pressure - High	S	M	R	1, 2, 3
b. Reactor Vessel Water Level - Low Low Low (Level 1)	S	M	R	1, 2, 3, 4*, 5*
c. Reactor Vessel Shroud Level (Level 0) - High	S	M	R	1, 2, 3, 4*, 5*
d. Reactor Steam Dome Pressure - Low	S	M	R	1, 2, 3, 4*, 5*
e. Reactor Steam Dome Pressure - Low	S	M	R	1, 2, 3, 4*, 5*
f. RHR Pump Start-Time Delay Relay	NA	NA	R	1, 2, 3, 4*, 5*
g. Logic Power Monitor	NA	R	NA	1, 2, 3, 4*, 5*

*Not applicable when two core spray system subsystems are OPERABLE per Specification 3.5.3.1.

TABLE 4.3.3-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED#
3. HIGH PRESSURE COOLANT INJECTION SYSTEM				
a. Reactor Vessel Water Level - Low Low (Level 2)	S	M	R	1, 2, 3
b. Drywell Pressure-High	S	M	R	1, 2, 3
c. Condensate Storage Tank Level - Low	NA	M	Q	1, 2, 3
d. Suppression Chamber Water Level - High	S	M	R	1, 2, 3
e. Logic Power Monitor	NA	R	NA	1, 2, 3
f. Reactor Vessel Water Level-High (Level 8)	S	M	R	1, 2, 3
4. AUTOMATIC DEPRESSURIZATION SYSTEM				
a. Drywell Pressure-High	S	M	R	1, 2, 3
b. Reactor Vessel Water Level - Low Low Low (Level 1)	S	M	R	1, 2, 3
c. ADS Timer	NA	NA	R	1, 2, 3
d. Reactor Vessel Water Level - Low (Level 3)	S	M	R	1, 2, 3
e. Core Spray Pump Discharge Pressure - High	S	M	R	1, 2, 3
f. RHR (LPCI MODE) Pump Discharge Pressure - High	S	M	R	1, 2, 3
g. Control Power Monitor	NA	R	NA	1, 2, 3
5. LOW LOW SET S/RV SYSTEM				
a. Reactor Steam Dome Pressure - High	S	M	R	1, 2, 3

HPCI and ADS are not required to be OPERABLE with reactor steam dome pressure $\leq$ 150 psig.

3/4.3.4 REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.4 The reactor core isolation cooling (RCIC) system actuation instrumentation shown in Table 3.3.4-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.4-2.

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

ACTION:

a. With a RCIC system actuation instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.4-2, declare the channel inoperable and place the inoperable channel in the tripped condition until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

b. With the requirements for the minimum number of OPERABLE channels not satisfied for one trip system, place the inoperable channel in the tripped condition or declare the RCIC system inoperable within one hour.

c. With the requirements for the minimum number of OPERABLE channels not satisfied for both trip systems, declare the RCIC system inoperable within one hour.

SURVEILLANCE REQUIREMENTS

4.3.4.1 Each RCIC system actuation instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION at the frequencies shown in Table 4.3.4-1.

4.3.4.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed at least once per 18 months and shall include calibration of time delay relays and timers necessary for proper functioning of the trip system.

TABLE 3.3.4-1

REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION

FUNCTIONAL UNITS	MINIMUM NUMBER OF OPERABLE CHANNELS PER TRIP SYSTEM
a. Reactor Vessel Water Level - Low Low (Level 2) (2B21-N692 A, B, C, D)	2
b. Condensate Storage Tank Water Level - Low (2E51-N060, 2E51-N061)	2(a)
c. Suppression Pool Water Level-High (2E51-N062A, B)	2(a)

(a) Provides Signal to RCIC Pump Suction Valves Only

TABLE 3.3.4-2

REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION

FUNCTIONAL UNITS	TRIP SETPOINT	ALLOWABLE VALUE
a. Reactor Vessel Water Level - Low Low (Level 2)	$\geq$ -55 inches*	$\geq$ -55 inches*
b. Condensate Storage Tank Level - Low	$\geq$ 0 inches**	$\geq$ 0 inches**
c. Suppression Pool Water Level-High	$\leq$ 151 inches	$\leq$ 151 inches

*See Bases Figure B 3/4 3-1

** This corresponds to a level of 131'-0" above mean sea level.

TABLE 4.3.4-1

REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNITS	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
a. Reactor Vessel Water Level-Low Low (Level 2)	S	M	R
b. Condensate Storage Tank Level- Low	NA	M	Q
c. Suppression Pool Water Level- High	NA	M	Q

TABLE 3.3.5-1 (Continued)

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION

NOTE

a. When THERMAL POWER exceeds the preset power level of the RWM and RSCS and when the limiting condition defined in section 3.1.4.3 exists.

b. This function is bypassed if detector is reading > 100 cps or the IRM channels are on range 3 or higher.

c. This function is bypassed when the associated IRM channels are on range 8 or higher.

d. A total of 6 IRM instruments must be OPERABLE.

e. This function is bypassed when the IRM channels are on range 1.

f. With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.11.1 or 3.9.11.2.

TABLE 3.3.5-2

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. APRM		
a. Flow Referenced Simulated Thermal Power - Upscale	≤ (0.58 W + 50%)[a]	≤ (0.58 W + 50%)[a]
b. Inoperative	NA	NA
c. Downscale	≥ 3/125 of full scale	≥ 3/125 of full scale
d. Neutron Flux - High, 12%	≤ 12/125 of full scale	≤ 12/125 of full scale
2. ROD BLOCK MONITOR		
a. Upscale[b]		
1) Low Trip Setpoint (LTSP)	≤ 115.1/125 of full scale	≤ 115.5/125 of full scale
2) Intermediate Trip Setpoint (ITSP)	≤ 109.3/125 of full scale	≤ 109.7/125 of full scale
3) High Trip Setpoint (HTSP)	≤ 105.5/125 of full scale	≤ 105.9/125 of full scale
b. Inoperative	NA	NA
c. Downscale	≥ 94/125 of full scale	≥ 93/125 of full scale
d. Power Range Setpoint[c]		
1) Low Power Setpoint (LPSP)	≤ 27% of RATED THERMAL POWER	≤ 29% of RATED THERMAL POWER
2) Intermediate Power Setpoint (IPSP)	≤ 62% of RATED THERMAL POWER	≤ 64% of RATED THERMAL POWER
3) High Power Setpoint (HPSP)	≤ 82% of RATED THERMAL POWER	≤ 84% of RATED THERMAL POWER
e. RBM Bypass Time Delay (td_2)[d]	≤ 2.0 sec	≤ 2.0 sec
3. SOURCE RANGE MONITORS		
a. Detector not full in	NA	NA
b. Upscale	≤ 1×10^5 cps	≤ 1×10^5 cps
c. Inoperative	NA	NA
d. Downscale	≥ 3 cps	≥ 3 cps

TABLE 3.3.5-2 (Continued)

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
4. INTERMEDIATE RANGE MONITORS		
a. Detector not full in	NA	NA
b. Upscale	≤ 108/125 of full scale	≤ 108/125 of full scale
c. Inoperative	NA	NA
d. Downscale	≥ 5/125 of full scale	≥ 5/125 of full scale
5. SCRAM DISCHARGE VOLUME		
a. Water Level-High	≤ 36.2 gallons	≤ 36.2 gallons

NOTES:

a. W = Loop recirculation flow in percent of rated flow.

b. There are three upscale trip levels. Only one is applicable over a specified operating core thermal power range. All RBM trips are automatically bypassed below the low power setpoint. The upscale LTSP is applied between the low power setpoint and the intermediate power setpoint. The upscale ITSP is applied between the intermediate power setpoint and the high power setpoint. The HTSP is applied above the high power setpoint.

c. Power range setpoints control enforcement of appropriate upscale trips over the proper core thermal power ranges. The power signal to the RBM is provided by the APRM.

d. RBM bypass time delay is set low enough to assure minimum rod movement while upscale trips are bypassed.

TABLE 4.3.5-1

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION(a)	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
1. APRM				
a. Flow Referenced Simulated Thermal Power- Upscale	NA	S/U(b),M	R	1
b. Inoperative	NA	S/U(b),M	NA	1, 2, 5
c. Downscale	NA	S/U(b),M	R	1
d. Neutron Flux - High, 12%	NA	S/U(b),M	R	2, 5
2. ROD BLOCK MONITOR				
a. Upscale	NA	S/U(b), M	R	1(d)
b. Inoperative	NA	S/U(b), M	NA	1(d)
c. Downscale	NA	S/U(b), M	R	1(d)
3. SOURCE RANGE MONITORS				
a. Detector not full in	NA	S/U(b),W	NA	2, 5
b. Upscale	NA	S/U(b),W	R	2, 5
c. Inoperative	NA	S/U(b),W	NA	2, 5
d. Downscale	NA	S/U(b),W	R	2, 5
4. INTERMEDIATE RANGE MONITORS				
a. Detector not full in	NA	S/U(b),W(c)	NA	2, 5
b. Upscale	NA	S/U(b),W(c)	R	2, 5
c. Inoperative	NA	S/U(b),W(c)	NA	2, 5
d. Downscale	NA	S/U(b),W(c)	R	2, 5
5. SCRAM DISCHARGE VOLUME				
a. Water Level-High	NA	Q	R	1, 2, 5(e)

TABLE 4.3.5-1 (Continued)

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

NOTES:

a. Neutron detectors may be excluded from CHANNEL CALIBRATION.

b. Within 24 hours prior to startup, if not performed within the previous 7 days.

c. When changing from CONDITION 1 to CONDITION 2, perform the required surveillance within 12 hours after entering CONDITION 2.

d. When THERMAL POWER exceeds the preset power level of the RWM and RSCS. The additional surveillance defined in Specification 4.1.4.3 will be required when the Limiting Condition defined in Specification 3.1.4.3 exists.

e. With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.11.1 or 3.9.11.2.

POST-ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.6.4 The post-accident monitoring instrumentation channels shown in Table 3.3.6.4-1 shall be OPERABLE.

APPLICABILITY: CONDITIONS 1 and 2.

ACTION:

a. With one or more of the above required post-accident monitoring channels inoperable, either restore the inoperable channel(s) to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours.

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.6.4 Each of the above required post-accident monitoring instrumentation channels shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3.6.4-1.

TABLE 3.3.6.4-1

POST-ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE
1. Reactor Vessel Pressure (2B21-R623 A, B)	2
2. Reactor Vessel Shroud Water Level (2B21-R610, 2B21-R615)	2
3. Suppression Chamber Water Level (2T48-R622 A. B)	2
4. Suppression Chamber Water Temperature (2T47-R626, 2T47-R627)	2
5. Suppression Chamber Pressure (2T48-R608, 2T48-R609)	2
6. Drywell Pressure (2T48-R608, 2T48-R609)	2
7. Drywell Temperature (2T47-R626, 2T47-R627)	2
8. Post-LOCA Gamma Radiation (2D11-K622 A, B, C, D)	2
9. Drywell H_2 -O_2 Analyzer (2P33-R601 A, B)	2
10.a)Safety/Relief Valve Position Primary Indicator (2B21-N301 A-H and K-M)	*
b)Safety/Relief Valve Position Secondary Indicator (2B21-N004 A-H and K-M)	*

*If either the primary or secondary indication is inoperable, the torus temperature will be monitored at least once per shift to observe any unexplained temperature increases which might be indicative of an open SRV. With both the primary and secondary monitoring channels of an SRV inoperable, either verify that the S/RV is closed through monitoring the backup low low set logic position indicators (2B21-N302 A-H and K-M) at least once per shift or restore sufficient inoperable channels such that no more than one SRV has both primary and secondary channels inoperable within 7 days or be in at least hot shutdown within the next 12 hours.

TABLE 4.4.6.1.3-1

REACTOR VESSEL MATERIAL IRRADIATION SURVEILLANCE SCHEDULE

SPECIMEN	REMOVAL INTERVAL
1.	10 years
2.	30 years
3.	Reserve

REACTOR STEAM DOME

LIMITING CONDITION FOR OPERATION

3.4.6.2 The pressure in the reactor steam dome shall be less than 1054 psig.

APPLICABILITY: CONDITION 1* and 2*.

ACTION:

With the reactor steam dome pressure exceeding 1054 psig, reduce the pressure to less than 1054 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 1054 psig at least once per 12 hours.

* Not applicable during anticipated transients.

CONTAINMENT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

3. Verifying a system flow rate of 4000 +0, -1000 cfm during system operation when tested in accordance with ANSI N510-1975.

c. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 1, July 1976, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 1, July 1976.

d. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is < 6 inches Water Gauge while operating the filter train at a flow rate of 4000 +0, -1000 cfm.

2. Verifying that the filter train starts and isolation dampers open on each of the following test signals:

a. Drywell pressure-high,

b. High radiation on the;

1) Refueling floor,

2) Reactor building.

c. Reactor Vessel Water Level-Low Low (Level 2).

3. Verifying that the heaters dissipate 18.5 + 1.5 KW when tested in accordance with ANSI N510-1975.

SURVEILLANCE REQUIREMENTS (Continued)

e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove $\geq$ 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 4000 +0, -1000 cfm.

f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove $\geq$ 99% of a halogenated hydrocarbon regrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 4000 + 0, -1000 cfm.

4.6.6.1.2 Each Hatch-Unit 1 standby gas treatment subsystem shall be demonstrated OPERABLE per Hatch-Unit 1 Technical Specifications.

3/4.9.2 INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.9.2 At least 2 source range monitor* (SRM) channels shall be OPERABLE and inserted to the normal operating level:

a. Each with continuous visual indication in the control room,

b. At least one with an audible alarm in the control room,

c. One of the SRM detectors located in the quadrant where CORE ALTERATIONS are being performed and the other SRM detector located in an adjacent quadrant, and

d. The "shorting links" removed from the RPS circuity during CORE ALTERATIONS and shutdown margin demonstrations.

APPLICABILITY: CONDITION 5.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS** or positive reactivity changes and actuate the manual scram. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.2 Each of the above required SRM channels shall be demonstrated OPERABLE by:

a. At least once per 12 hours;

1. Performance of a CHANNEL CHECK,

2. Verifying the detectors are inserted to the normal operating level,

3. During CORE ALTERATIONS, verifying that the detector of an OPERABLE SRM channel is located in the core quadrant where CORE ALTERATIONS are being performed and one is located in the adjacent quadrant.

*The use of special movable detectors during CORE ALTERATIONS in place of the normal SRM nuclear detectors is permissible as long as these special detectors are connected to the normal SRM circuits.

**Except movement of SRM or special movable detectors.

SURVEILLANCE REQUIREMENTS CONTINUED

b. Performance of a CHANNEL FUNCTIONAL TEST:

1. Within 24 hours prior to the start of CORE ALTERATIONS, and

2. At least once per 7 days.

c. Verify that the channel count rate is at least 3 cps at least once per 12 hours during CORE ALTERATIONS, and at least once per 24 hours, except:

1. The 3 cps is not required during core alterations involving only fuel unloading provided the SRMs were confirmed to read at least 3 cps initially and were checked for neutron response.

2. The 3 cps is not required initially on a full core reload. Prior to the reload, up to four fuel assemblies will be loaded into their previous core positions next to each of the 4 SRMs to obtain the required count rate.

d. Verifying that the RPS circuitry "shorting links" have been removed and that the RPS circuitry is in a non-coincidence trip mode within 8 hours prior to starting CORE ALTERATIONS or shutdown margin demonstrations.

CONTROL RODS (Continued)

than has been analyzed even though control rods with inoperable accumulators may still be inserted with normal drive water pressure. Operability of the accumulator ensures that there is a means available to insert the control rods even under the most unfavorable depressurization of the reactors.

Control rod coupling integrity is required to ensure compliance with the analysis of the rod drop accident in the FSAR. The overtravel position feature provides the only positive means of determining that a rod is properly coupled and therefore this check must be performed prior to achieving criticality after each refueling. The subsequent check is performed as a backup to the initial demonstration.

In order to ensure that the control rod patterns can be followed and therefore that other parameters are within their limits, the control rod position indication system must be OPERABLE.

The control rod housing support restricts the outward movement of a control rod to less than (3) inches in the event of a housing failure. The amount of rod reactivity which could be added by this small amount of rod withdrawal is less than a normal withdrawal increment and will not contribute to any damage to the primary coolant system. The support is not required when there is no pressure to act as a driving force to rapidly eject a drive housing.

The required surveillance intervals are adequate to determine that the rods are OPERABLE and not so frequent as to cause excessive wear on the system components.

3/4.1.4 CONTROL ROD PROGRAM CONTROLS

Control rod withdrawal and insertion sequences are established to assure that the maximum insequence individual control rod or control rod segments which are withdrawn at any time during the fuel cycle could not be worth enough to cause the peak fuel enthalpy for any postulated control rod accident to exceed 280 cal/gm. The specified sequences are characterized by homogeneous, scattered patterns of control rod withdrawal. When THERMAL POWER is $\geq$ 20% of RATED THERMAL POWER, there is no possible rod worth which, if dropped at the design rate of the velocity limiter, could result in a peak enthalpy of 280 cal/gm. Thus, requiring the RSCS and RWM to be OPERABLE below 20% of RATED THERMAL POWER provides adequate control.

CONTROL ROD PROGRAM CONTROLS (Continued)

The RSCS and RWM provide automatic supervision to assure that out-of-sequence rods will not be withdrawn or inserted.

The analysis of the rod drop accident is presented in Section 15.1.38 of the FSAR and the techniques of the analysis are presented in a topical report, Reference 1, and two supplements, References 2 and 3.

The RBM is designed to automatically prevent fuel damage in the event of erroneous rod withdrawal from locations of high power density during high power operation. The RBM is only required to be operable when the Limiting Condition defined in Specification 3.1.4.3 exists. Two channels are provided. Tripping one of the channels will block erroneous rod withdrawal soon enough to prevent fuel damage. This system backs up the written sequence used by the operator for withdrawal of control rods. Further discussion of the RBM system and power dependent setpoints may be found in NEDC-30474-P (Ref. 4).

3/4.1.5 STANDBY LIQUID CONTROL SYSTEM

The standby liquid control system provides a backup capability for maintaining the reactor subcritical in the event that insufficient rods are inserted in the core when a scram is called for. The volume of the poison solution and weight percent of poison material in solution is based on being able to bring the reactor to the subcritical condition as the plant cools to ambient condition. The temperature requirement is necessary to keep the sodium pentaborate in solution. Checking the volume and temperature once each 24 hours assures that the solution is available for use.

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.

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 water is added; thus, a check on the temperature and volume once each 24 hours assures that the solution is available for use.

STANDBY LIQUID CONTROL SYSTEM (Continued)

Replacement of the explosive charges in the valves at regular intervals will assure that these valves will not fail because of deterioration of the charges.

1. C. J. Paone, R. C. Stirn and J. A. Woodley, "Rod Drop Accident Analysis for Large BWRs," GE Topical Report NEDO-10527, March 1972.

2. C. J. Paone, R. C. Stirn and R. M. Yound, Supplement 1 to NEDO-10527, July 1972.

3. J. A. Haum, C. J. Paone and R. C. Stirn, Addendum 2, "Exposed Cores," Supplement 2 to NEDO-10527, January 1973.

4. "Average Power Range Monitor, Rod Block Monitor and Technical Specification Improvement (ARTS) Program for Edwin I. Hatch Nuclear Plant, Units 1 and 2," NEDC-30474-P, December 1983.

BASES

The specifications of this section assure that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the 2200°F limit specified in the Final Acceptance Criteria (FAC) issued in June 1971 considering the postulated effects of fuel pellet densification. These specifications also assure that fuel design margins are maintained during abnormal transients.

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

This specification assures that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the limit specified in 10 CFR 50, Appendix K.

The peak cladding temperature (PCT) following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is dependent only secondarily on the rod-to-rod power distribution within an assembly. The peak clad temperature is calculated assuming an LHGR for the highest powered rod which is equal to or less than the design LHGR corrected for densification. This LHGR times 1.02 is used in the heatup code along with the exposure dependent steady state gap conductance and rod-to-rod local peaking factor. The Technical Specification APLHGR is this LHGR of the highest powered rod divided by its local peaking factor. The limiting value for APLHGR is shown in the figures for in Technical Specification 3/4.2.1.

The calculational procedure used to establish the APLHGR shown in the figures in Technical Specification 3/4.2.1 is based on a loss-of-coolant accident anaysis. The analysis was performed using General Electric (GE) calculational models which are consistent with the requirements of Appendix K to 10 CFR 50. A complete discussion of each code employed in the analysis is presented in Reference 1. Differences in this analysis compared to previous analyses performed with Reference 1 are: (1) the analysis assumes a fuel assembly planar power consistent with 102% of the MAPLHGR shown in the figures in Technical Specification 3/4.2.1; (2) fission product decay is computed assuming an energy release rate of 200 MEV/fission; (3) pool boiling is assumed after nucleate boiling is lost during the flow stagnation period; and (4) the effects of core spray entrainment and counter-current flow limitation as described in Reference 2, are included in the reflooding calculations.

A flow dependent correction factor incorporated into Figure 3.2.1-9 is applied to the rated conditions APLHR to assure that the 2200°F PCT limit is complied with during a LOCA initiated from less than rated core flow. In addition, other power and flow dependent corrections given in Figures 3.2.1-10 and 3.2.1-11 are applied to the rated conditions to assure that the fuel thermal-mechanical design criteria are preserved during abnormal transients initiated from off-rated conditions.

A list of the significant plant input parameters to the loss-of-coolant accident analysis is presented in bases Table B 3.2.1-1. Further discussion of the APLHGR limits is given in Reference 4.

Bases Table B 3.2.1-1

SIGNIFICANT INPUT PARAMETERS TO THE

LOSS-OF-COOLANT ACCIDENT ANALYSIS

FOR HATCH-UNIT 2

Plant Parameters:

Core Thermal Power .... 2531 Mwt which corresponds to 105% of license core power*

Vessel Steam Output .... 10.96×10^6 lbm/h which corresponds to 105% of rated steam flow

Vessel Steam Dome Pressure .... 1055 psia

Design Basis Recirculation Line Break Area For:

a. Large Breaks .... 4.0, 2.4, 2.0, 2.1 and 1.0 ft^2

b. Small Breaks .... 1.0, 0.9, 0.4 and 0.07 ft^2

Fuel Parameters:

FUEL TYPE	FUEL BUNDLE GEOMETRY	PEAK TECHNICAL SPECIFICATION LINEAR HEAT GENERATION RATE (kw/ft)	DESIGN AXIAL PEAKING FACTOR	INITIAL MINIMUM CRITICAL POWER RATIO
Initial Core	8 x 8	13.4	1.4	1.18

A more detailed list of input to each model and its source is presented in Section II of Reference 1 and subsection 6.3.3 of the FSAR.

*This power level meets the Appendix K requirement of 102%. The core heatup calculation assumes a bundle power consistent with operation of the highest powered rod at 102% of its Technical Specification linear heat generation rate limit.

3/4.2.2 APRM SETPOINTS

This section deleted.

3/4.2.3 MINIMUM CRITICAL POWER RATIO

The required operating limit MCPRs at steady state operating conditions as specified in Specification 3.2.3 are derived from the established fuel cladding integrity Safety Limit MCPR of 1.07, and an analysis of abnormal operational transients. For any abnormal operating transient analysis evaluation with the initial condition of the reactor being at the steady state operating limit, it is required that the resulting MCPR does not decrease below the Safety Limit MCPR at any time during the transient assuming instrument trip setting as given in Specification 2.2.1.

To assure that the fuel cladding integrity Safety Limits are not exceeded during any anticipated abnormal operational transient, the most limiting transients have been analyzed to determine which results in the largest reduction in CRITICAL POWER RATIO (CPR). The type of transients evaluated were loss of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease.

MINIMUM CRITICAL POWER RATIO (Continued)

The evaluation of a given transient begins with the system initial parameters shown in FSAR Table 15.1-6 that are input to a GE-core dynamic behavior transient computer program described in NEDO-10802[3]. Also, the void reactivity coefficients that were input to the transient calculational procedure are based on a new method of calculation termed NEV which provides a better agreement between the calculated and plant instrument power distributions. The outputs of this program along with the initial MCPR form the input for further analyses of the thermally limiting bundle with the single channel transient thermal hydraulic SCAT code described in NEDO-20566[1]. The principal result of this evaluation is the reduction in MCPR caused by the transient.

The purpose of the $MCPR_f$, and the K_p of Figures 3.2.3-4 and 3.2.3-5, respectively is to define operating limits at other than rated core flow and power conditions. At less than 100% of rated flow and power, the required MCPR is the larger value of the $MCPR_f$ and $MCPR_p$ at the existing core flow and power state. The $MCPR_f$s are established to protect the core from inadvertent core flow increases such that the 99.9% MCPR limit requirement can be assured.

The $MCPR_f$s were calculated such that for the maximum core flow rate and the corresponding THERMAL POWER along the 105% of rated steam flow control line, the limiting bundle's relative power was adjusted until the MCPR was slightly above the Safety Limit. Using this relative bundle power, the MCPRs were calculated at different points along the 105% of rated steam flow control line corresponding to different core flows. The calculated MCPR at a given point of core flow is defined as $MCPR_f$.

The core power dependent MCPR operating limit MCPR is the power rated flow MCPR operating limit multiplied by the K_p factor given in Figure 3.2.3-5.

The K_ps are established to protect the core from transients other than core flow increases, including the localized event such as rod withdrawal error. The K_ps were determined based upon the most limiting transient at the given core power level. For further information on MCPR operating limits for off-rated conditions, reference NEDC-30474-P. [4]

MINIMUM CRITICAL POWER RATIO (Continued)

At THERMAL POWER levels less than or equal to 25% 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 indicated that the resulting MCPR value is in excess of requirements by a considerable margin. With this low void content, any inadvertent core flow increase would only place operation in a more conservative mode relative to MCPR. During initial startup testing of the plant, an MCPR evaluation will be made at 25% of RATED THERMAL POWER 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 above 25% 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 LHGR specification assures that the linear heat generation rate in any rod is less than the design linear heat generation even if fuel pellet densification is postulated.

References:

1. General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10 CFR 50, Appendix K, NEDO-20566 (Draft), August 1974.

2. General Electric Refill Reflood Calculation (Supplement to SAFE Code Description) transmitted to USAEC by letter, G. L. Gyorey to V. Stello, Jr., dated December 20, 1974.

3. R. B. Linford, Analytical Methods of Plant Transient Evaluations for the GE BWR, February 1973 (NEDO-10802).

4. "Average Power Range Monitor, Rod Block Monitor and Technical Specification Improvement (ARTS) Program for Edwin I. Hatch Nuclear Plant, Units 1 and 2," NEDC-30474-P, December 1983.

MONITORING INSTRUMENTATION (Continued)

FIRE DETECTION INSTRUMENTATION (Continued)

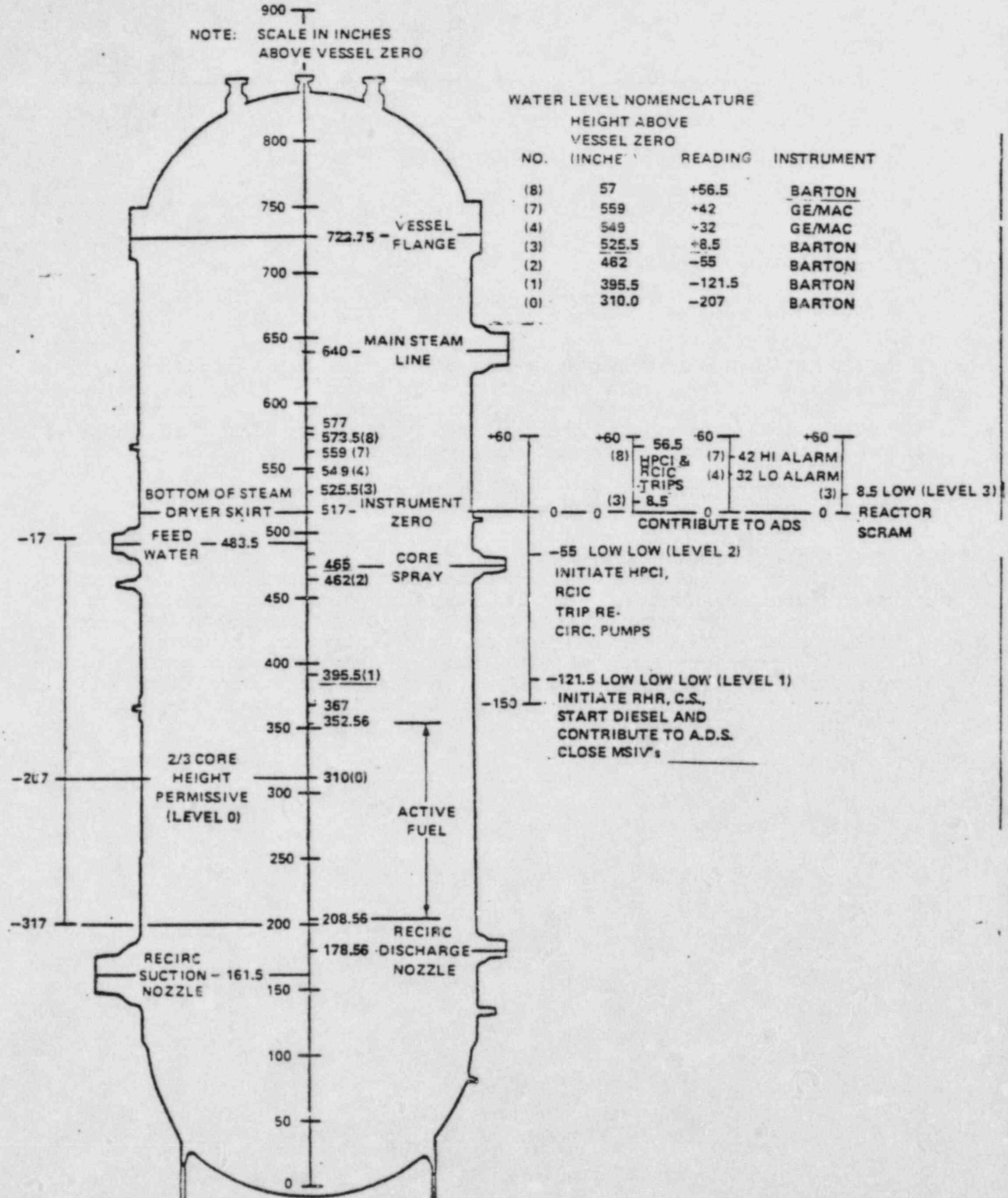
In the event that a portion of the fire detection instrumentation is inoperable, increasing the frequency of fire patrols in the affected areas is required to provide detection capability until the inoperable instrumentation is restored to OPERABILITY.

3/4.3.7 TURBINE OVERSPEED PROTECTION SYSTEM

This specification is provided to ensure that the turbine overspeed protection system instrumentation and the turbine speed control valves are OPERABLE and will protect the turbine from excessive overspeed. Protection from turbine excessive overspeed is required since excessive overspeed of the turbine could generate potentially damaging missiles which could impact and damage safety-related components, equipment or structures.

3/4.3.8 DEGRADED STATION VOLTAGE PROTECTION INSTRUMENTATION

The undervoltate relays shall automatically initite the disconnection of offsite power sources whenever the voltage setpoint and time delay limits have been exceeded. This action shall provide voltage protection for the emergency power systems by preventing sustained degraded voltage conditions due to the offsite power source and interaction between the offsite and onsite emergency power systems. The undervoltage relays have a time delay characteristic that provides protection against both a loss of voltage and degraded voltage condition and thus minimizes the effect of short duration disturbances without exceeding the maximum time delay, including margin, that is assumed in the FSAR accident analyses.



WATER LEVEL NOMENCLATURE

NO.	HEIGHT ABOVE VESSEL ZERO (INCHES)	READING	INSTRUMENT
(8)	57	+56.5	BARTON
(7)	559	+42	GE/MAC
(4)	549	+32	GE/MAC
(3)	525.5	+8.5	BARTON
(2)	462	−55	BARTON
(1)	395.5	−121.5	BARTON
(0)	310.0	−207	BARTON

BASES FIGURE B 3/4 3-1

REACTOR VESSEL WATER LEVELS

BASES

3/4.9.1 REACTOR MODE SWITCH

Locking the OPERABLE reactor mode switch in the refuel position ensures that the restrictions on rod withdrawal and refueling platform movement during the refueling operations are properly activated. These conditions reinforce the refueling procedures and reduce the probability of inadvertent criticality, damage the reactor internals or fuel assemblies, and exposure of personnel to excessive radioactivity.

3/4.9.2 INSTRUMENTATION

The OPERABILITY of at least two source range monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core. During the unloading, it is not necessary to maintain 3 cps because core alterations will involve only reactivity removal and will not result in criticality. The loading of up to four bundles around the SRMs before attaining the 3 cps is permissible because these bundles were in subcritical configuration when they were removed and therefore will remain subcritical when placed back in the previous positions.

3/4.9.3 CONTROL ROD POSITION

The requirement that all control rods be inserted during CORE ALTERATIONS ensures that fuel will not be loaded into a cell without a control rod and prevents two positive reactivity changes from occurring simultaneously.

3/4.9.4 DECAY TIME

The minimum requirement for reactor subcriticality prior to fuel movement ensures that sufficient time has elapsed to allow the radioactive decay of the short lived fission products. This decay time is consistent with the assumptions used in the accident analyses.

3/4.9.5 SECONDARY CONTAINMENT

Secondary containment is designed to minimize any ground level release of radioactive material which may result from an accident. The reactor building provides secondary containment during normal operation when the drywell is sealed and in service. When the reactor is shutdown or during refueling, the drywell may be open and the reactor building then becomes the primary containment. The refueling floor is maintained under the secondary containment integrity of Hatch-Unit 1.

Establishing and maintaining a vacuum in the building with the standby gas treatment system once per 18 months, along with the surveillance of the doors, hatches and dampers, is adequate to ensure that there are no violations of the integrity of the secondary containment. Only one closed damper in each penetration line is required to maintain the integrity of the secondary containment.

BASES

3/4.9.6 COMMUNICATIONS

The requirement for communications capability ensures that refueling station personnel can be promptly informed of significant changes in the facility status or core reactivity condition during movement of fuel within the reactor pressure vessel.

3/4.9.7 CRANE AND HOIST OPERABILITY

The OPERABILITY requirements of the cranes and hoists used for movement of fuel assemblies ensures that: (1) each has sufficient load capacity to lift a fuel element, and (2) the core internals and pressure vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations.

3/4.9.8 CRANE TRAVEL-SPENT FUEL STORAGE POOL

The restriction on movement of loads in excess of the nominal weight of a fuel element over irradiated fuel assemblies ensures that no more than the contents of one fuel assembly will be ruptured in the event of a fuel handling accident. This assumption is consistent with the activity release assumed in the accident analyses.

3/4.9.9 and 3/4.9.10 WATER LEVEL-REACTOR VESSEL AND WATER LEVEL-SPENT FUEL STORAGE POOL

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. This minimum water depth is consistent with the assumptions of the accident analysis.

3/4.9.11 CONTROL ROD REMOVAL

This specification ensures that maintenance or repair of control rods or control rod drives will be performed under conditions that limit the probability of inadvertent criticality. The requirements for simultaneous removal of more than one control rod are more stringent since the SHUTDOWN MARGIN specification provides for the core to remain subcritical with only one control rod fully withdrawn.

CONTROL ROD ASSEMBLIES

5.3.2 The reactor core shall contain 137 cruciform-shaped control rod assemblies.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The reactor coolant system is designed and shall be maintained:

a. In accordance with the code requirements specified in Section 5.2 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of 1250 psig, and

c. For a temperature of 575°F

VOLUME

5.4.2 The total water and steam volume of the reactor vessel and recirculation system is approximately 17,050 cubic feet at a nominal T_{ave} of 540°F.

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown on Figure 5.1.1-1.

5.6 FUEL STORAGE

CRITICALITY

5.6.1 The new and spent fuel storage racks are designed and shall be maintained with sufficient center-to-center distance between fuel assemblies placed in the storage racks to ensure a k_{eff} equivalent to ≤ 0.95 when flooded with unborated water. The k_{eff} of ≤ 0.95 includes conservative allowances for uncertainties.

DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 185 feet.

CAPACITY

5.6.3 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 2845 fuel assemblies.

FUEL STORAGE

5.6.4 Fuel in the Spent Fuel Pool shall have a maximum fuel loading of 15.2 grams of Uranium-235 per axial centimeter.

5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT

5.7.1 The components identified in Table 5.7.1-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.7.1-1.