July 13, 1984

DMB 016

Docket No. 50-366

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Mr. J. T. Beckham, Jr.
Vice President - Nuclear Generation
Georgia Power Company
P. O. Box 4545
Atlanta, Georgia 30302

Dear Mr. Beckham:

The Commission has issued the enclosed Amendment No. 39 to Facility Operating License No. NPF-5 for the Edwin I. Hatch Nuclear Plant, Unit No. 2. The amendment consists of changes to the Technical Specifications (TSs) in response to your requests on the subjects indicated below:

Initial Licensee Requests	Supplementary Submittals	Subjects
1. Request to Change TS-Reload 4 (NED-84-192) dated April 3, 1984		1a. MAPLHGHR for New Fuel Bundle
		1b. OLMCPR Increase
		1c. Hybrid I Control Rods
		1d. Fuel Loading Around SRM Detectors
2. Proposed TS Changes to Support ATTS Installation (NED-84-017) dated January 23, 1984		2. ATTS Installation
	2a. Response to Verbal Questions (NED-84-281) dated June 7, 1984	2a. Provides more detailed information on the design and performance characteristics.
Revision to Request for TS Changes to Support ATTS Installation (NED-84-017) April 3, 1984	2b. Revised Responses (NED-84-321) dated June 14, 1984	2b. Provides certain additional detailed information on the design and performance characteristics.
	2c. Additional Clarification (NED-84-326) dated June 15, 1984	2c. Provides clarification of one sentence in the June 14, 1984 letter.

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PDR ADOCK 05000366
P PDR

Initial Licensee Requests	Supplementary Submittals	Subjects
3. Proposed TS Changes for ARTS Improvements (NED-84-030) dated February 6, 1984	3a. Additional Information on ARTS (NED-84-186) dated April 3, 1984	3. ARTS Improvements
	3b. Confirmation of Telephone Conversation-ARTS (NED-84-336) dated June 25, 1984	3b. Provides information on single loop operation.
	3c. ARTS Improvements (NED-84-346) dated June 27, 1984	3c. Withdraws proprietory notation from Technical Specification pages.

The supplementary correspondence did not alter the substance of your request, but was provided as confirmatory documentation of our understanding.

This amendment would permit Unit 2 to accomplish the following objectives: (a) introduce a new fuel type, (b) change the Operating Limit Minimum Critical Power Ratio so that subsequent cycles could be carried out under 10 CFR 50.59, (c) use the new Hybrid I control rod assemblies, (d) insert four assemblies around the Source Range Monitor detectors, (e) install the Analog Transmitter Trip System (ATTS), and (f) implement the Average Power Range Monitor/Rod Block Monitor/Technical Specification (ARTS) Improvement Program.

Please note that, although your TS change request to implement ARTS incorporated both Units 1 and 2, our current action is only effective for Unit 2. A copy of the Safety Evaluation for the requested changes is also enclosed.

In connection with the ARTS portion of this action, we prepared an "Environmental Assessment and Final Finding of No Significant Impact" which was sent to you separately.

Notice of Issuance for the reload and ATTS portions of this action will be included in the Commission's next Monthly Notice. A separate Notice of Issuance is enclosed for the ARTS action.

Sincerely,

"ORIGINAL SIGNED BY:"
George W. Rivenbark, Acting Chief
Operating Reactors Branch #4
Division of Licensing

Enclosures:
1. Amendment No. 39 to NPF-5
2. Safety Evaluation
3. Notice

cc w/enclosures: See next page

			ORB#4:DL	ICSB:DSI	CPB:DSI
			*PKadambi;cf	FRosa*	LPhillips*
			7/6/84	7/11/84	7/10/84
ORB#4:DL	ORB#4:DL	~~[illegible]~~	AD:OR:DL	OELD	CSB:DSI
RIngram	GRivenbark;cf*	~~[illegible]~~	GLainas	Goddard*	WButler*
7/12/84	7/ 6/84	~~[illegible]~~84	7/ /84	7/ 9/84	7/11/84

*See previous white for concurrences.

Initial Licensee Requests	Supplementary Submittals	Subjects
3. Proposed TS Changes for ARTS Improvements (NED-84-030) dated February 6, 1984	3a. Additional Information on ARTS (NED-84-186) dated April 3, 1984	3. ARTS Improvements
	3b. Confirmation of Telephone Conversation-ARTS (NED-84-336) dated June 20, 1984	
	3c. ARTS Improvements dated June 27, 1984	

This amendment would permit Unit 2 to accomplish the following objectives: (a) introduce a new fuel type, (b) change the Operating Limit Minimum Critical Power Ratio so that subsequent cycles could be licensed under 10 CFR 50.59, (c) use the new Hybrid I control rod assemblies, (d) insert four assemblies around the Source Range Monitor detectors, (e) install the Analog Transmitter Trip System (ATTS), and (f) implement the Average Power Range Monitor/Rod Block Monitor/Technical Specification (ARTS) Improvement Program.

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Sincerely,

George W. Rivenbark, Acting Chief
Operating Reactors Branch #4
Division of Licensing

Enclosures:
1. Amendment No. to NPF-5
2. Safety Evaluation
3. Notice

cc w/enclosures:
See next page

ICSB:DSI
F.Rosa
7/11/84

CPB:DSI
L.Phillips
7/10/84

OK with further exploration of our correspondence
done 7/12/84

ORB#4:DL	ORB#4:DL	C-ORB#4:DL	AD:OR:DL	OELD	AEB/CSB:DSI
RIngram	PKadambi;c	GRivenbark	GLainas	Goddard	W. Butler
7/6/84	7/6/84	7/6/84	7/ /84	7/9/84	7/11/84

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555
July 13, 1984

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Docket File
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RIngram

DOCKET No. 50-366

MEMORANDUM FOR: Docketing and Service Branch
Office of the Secretary of the Commission

FROM: Office of Nuclear Reactor Regulation

SUBJECT: EDWIN I. HATCH NUCLEAR PLANT, UNIT NO. 2

One signed original of the *Federal Register* Notice identified below is enclosed for your transmittal to the Office of the Federal Register for publication. Additional conformed copies (6) of the Notice are enclosed for your use.

- [ ] Notice of Receipt of Application for Construction Permit(s) and Operating License(s).
- [ ] Notice of Receipt of Partial Application for Construction Permit(s) and Facility License(s): Time for Submission of Views on Antitrust Matters.
- [ ] Notice of Consideration of Issuance of Amendment to Facility Operating License.
- [ ] Notice of Receipt of Application for Facility License(s); Notice of Availability of Applicant's Environmental Report; and Notice of Consideration of Issuance of Facility License(s) and Notice of Opportunity for Hearing.
- [ ] Notice of Availability of NRC Draft/Final Environmental Statement.
- [ ] Notice of Limited Work Authorization.
- [ ] Notice of Availability of Safety Evaluation Report.
- [ ] Notice of Issuance of Construction Permit(s).
- [XX] Notice of Issuance of Facility Operating License(s) or Amendment(s).
- [ ] Order.
- [ ] Exemption.
- [ ] Notice of Granting of Relief.
- [ ] Other: ______________________________

Division of Licensing, ORB#4
Office of Nuclear Reactor Regulation

Enclosure:
As stated

OFFICE	ORB#4:DL						
SURNAME	RIngram;cf						
DATE	7/13/84						

OFFICIAL RECORD COPY

cc w/enclosure(s):

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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:

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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
3/4 3-13	3/4 3-13
3/4 3-14	3/4 3-14
3/4 3-15	3/4 3-15
3/4 3-16	3/4 3-16
3/4 3-17	3/4 3-17
3/4 3-18	3/4 3-18
3/4 3-19	3/4 3-19
3/4 3-20	3/4 3-20
3/4 3-21	3/4 3-21
3/4 3-22	3/4 3-22
3/4 3-23	3/4 3-23
3/4 3-26	3/4 3-26
3/4 3-27	3/4 3-27
3/4 3-28	3/4 3-28
3/4 3-29	3/4 3-29

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.

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.

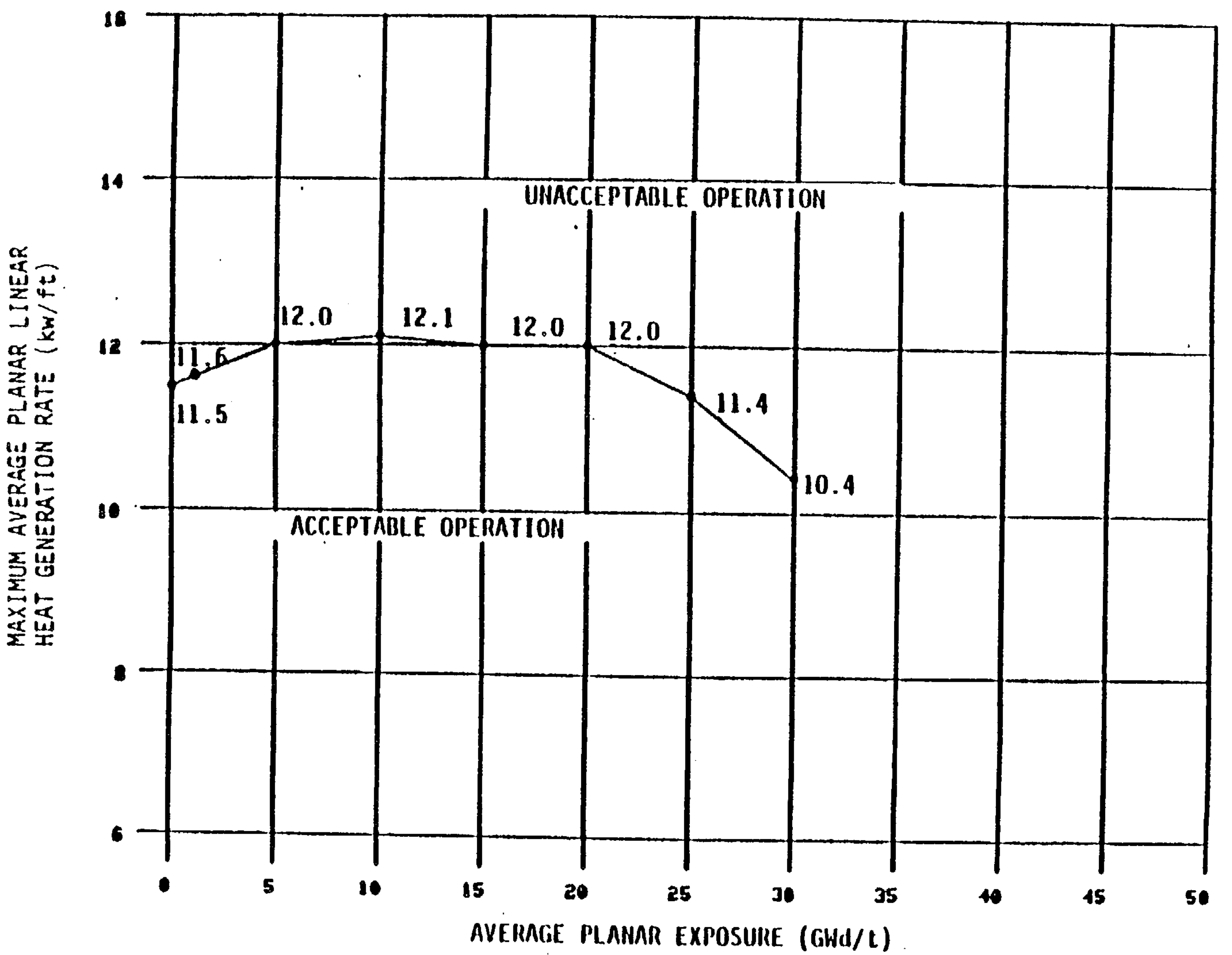
SURVEILLANCE REQUIREMENTS

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,

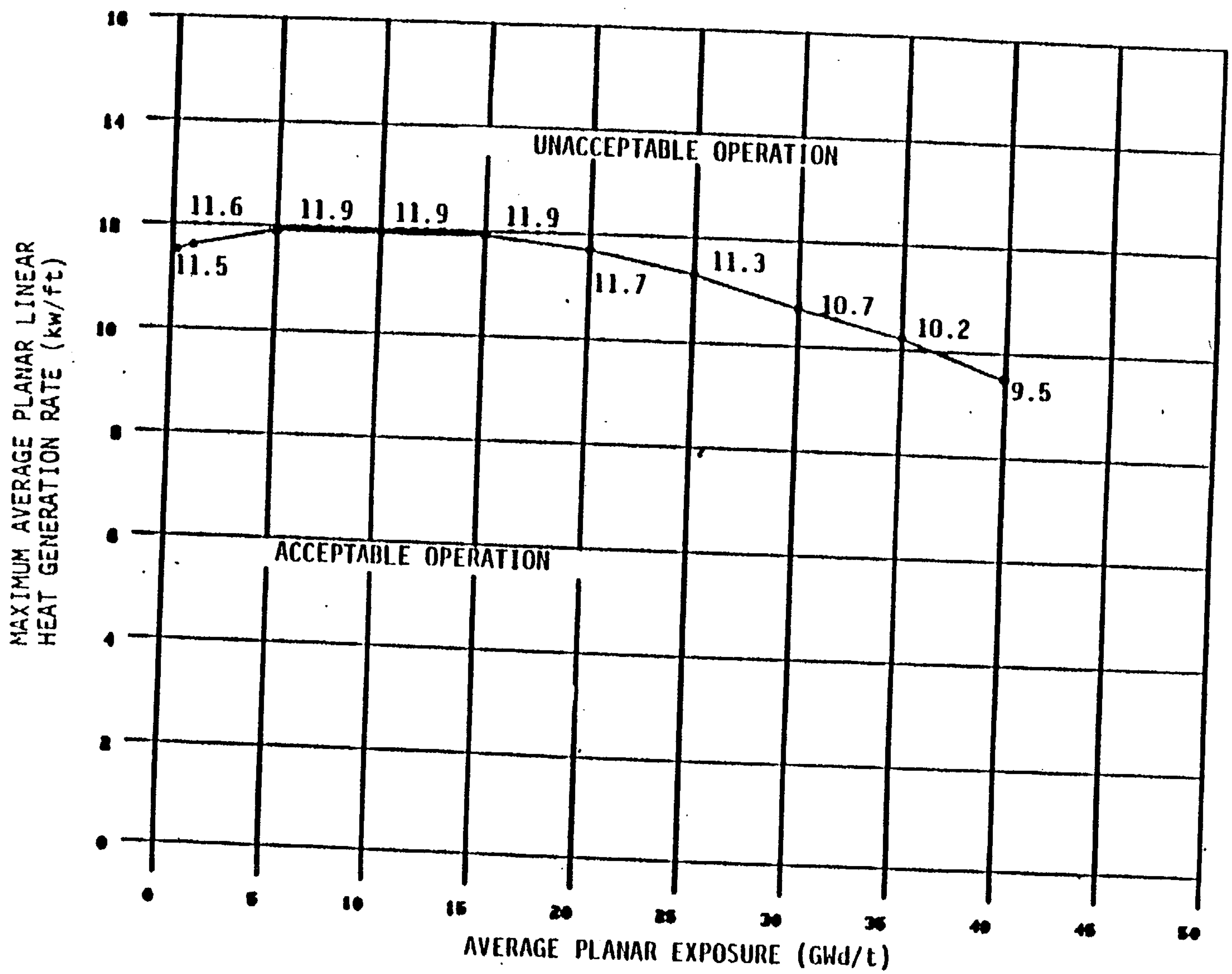
b. Whenever THERMAL POWER has been increased by at least 15% of RATED THERMAL POWER and steady state operating conditions have been established, and

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

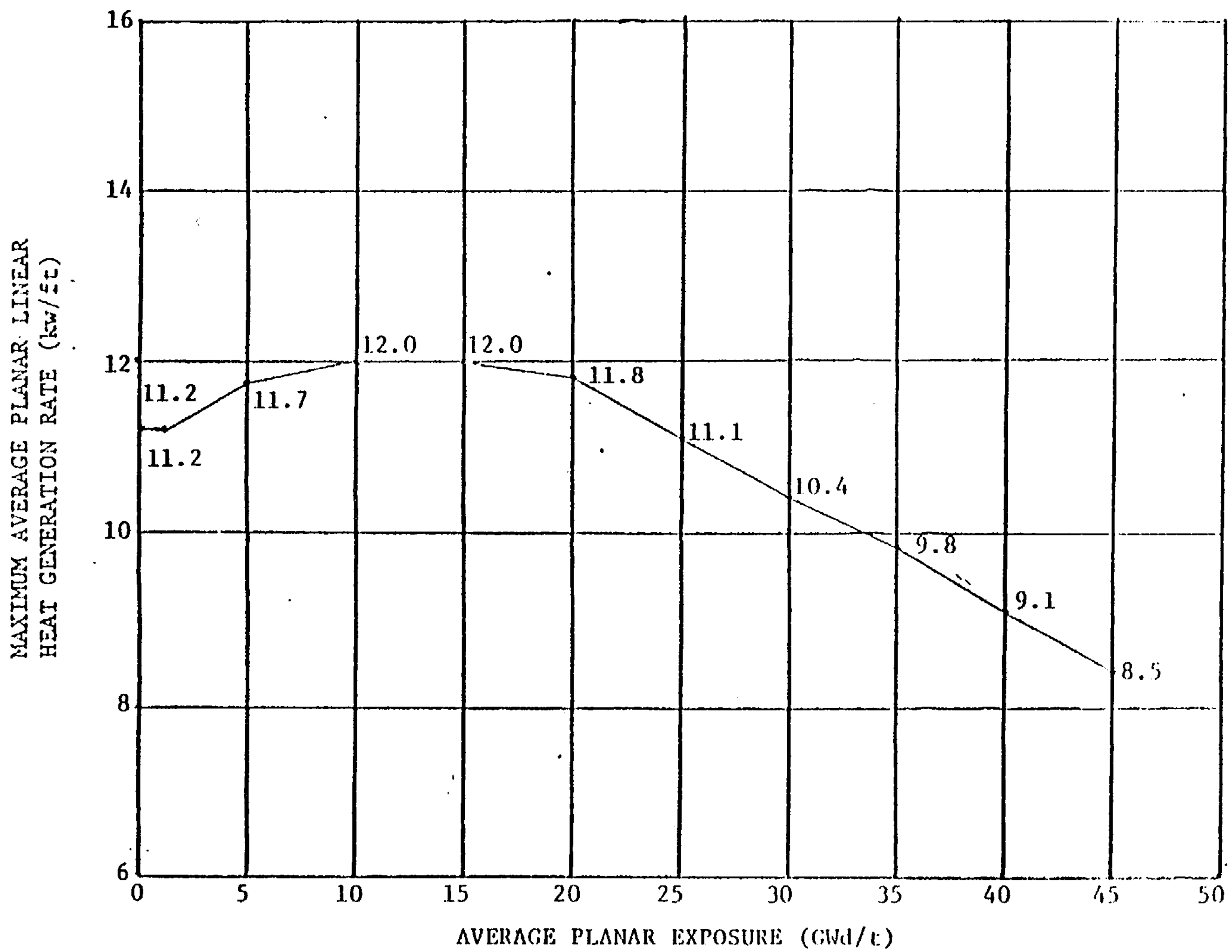


FUEL TYPE 8DIB175(8DRL183)
MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION
RATE (MAPLHGR) VERSUS AVERAGE PLANAR EXPOSURE

FIGURE 3.2.1-1

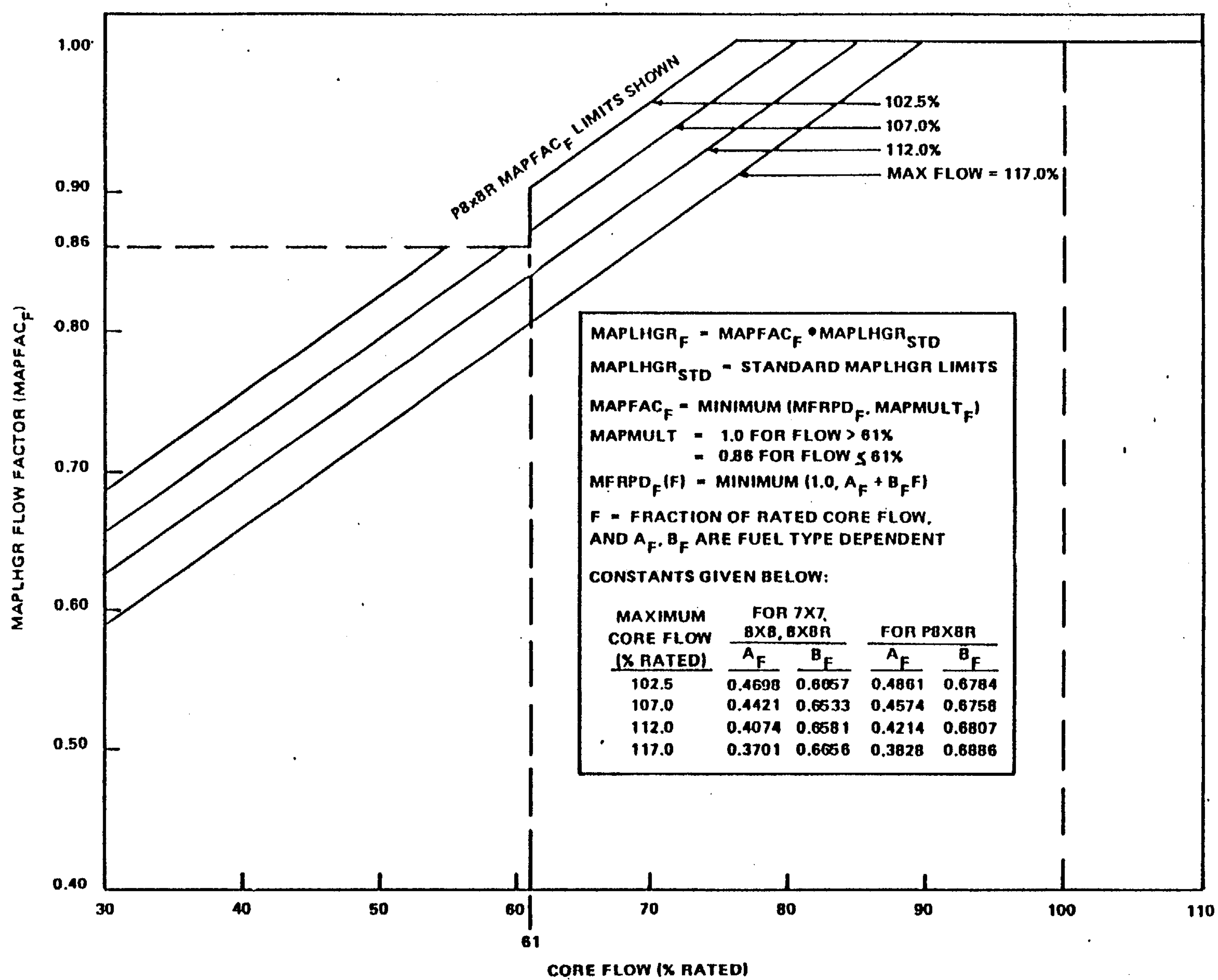


FUEL TYPE 8DRB265H
MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS AVERAGE PLANAR EXPOSURE
FIGURE 3.2.1-8



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		FOR P8X8R	
	A_F	B_F	A_F	B_F
102.5	0.4698	0.6057	0.4861	0.6784
107.0	0.4421	0.6533	0.4574	0.6758
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$_p$

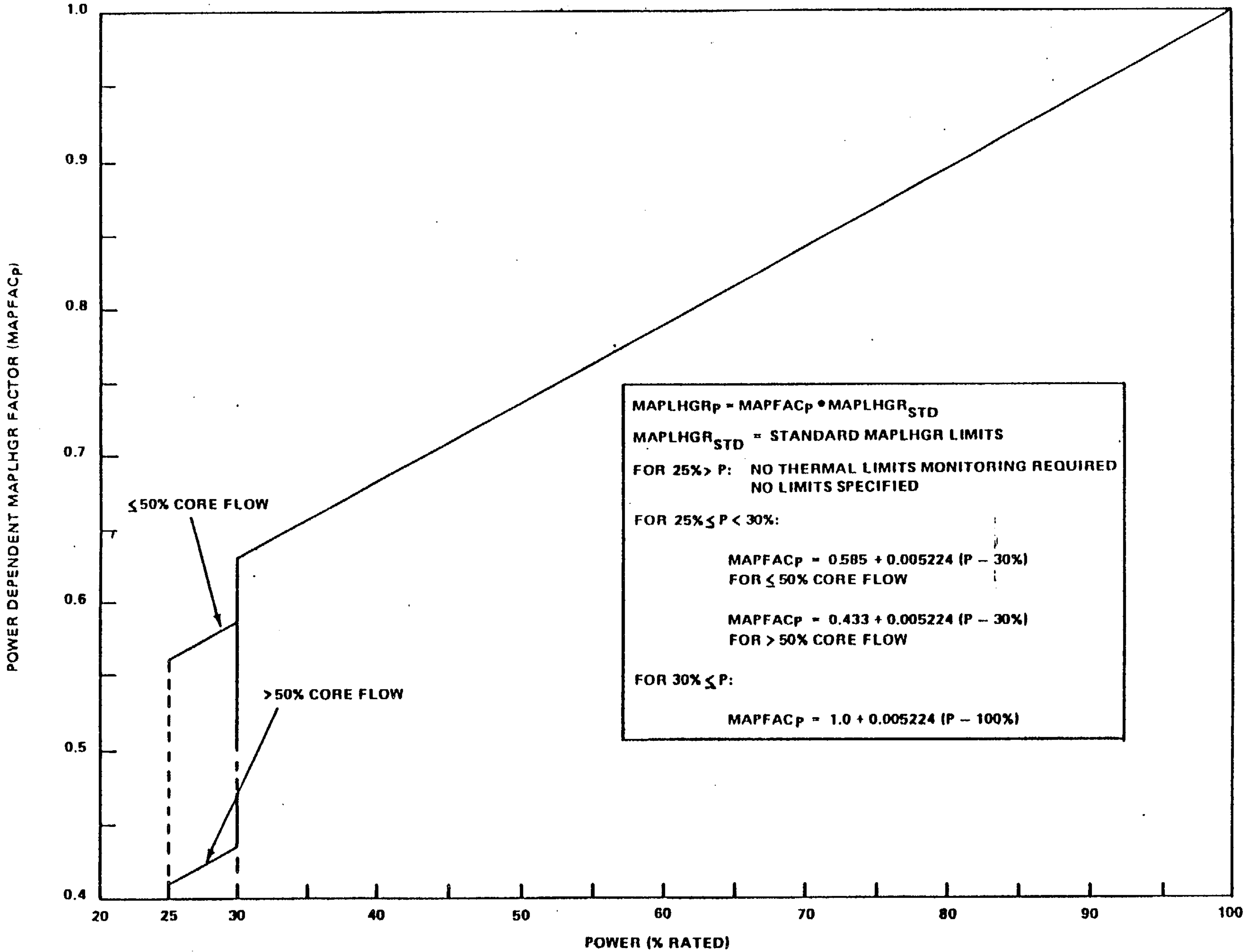


FIGURE 3.2.1-11 $MAPFAC_p$

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. $\tau = 1.0$ 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.

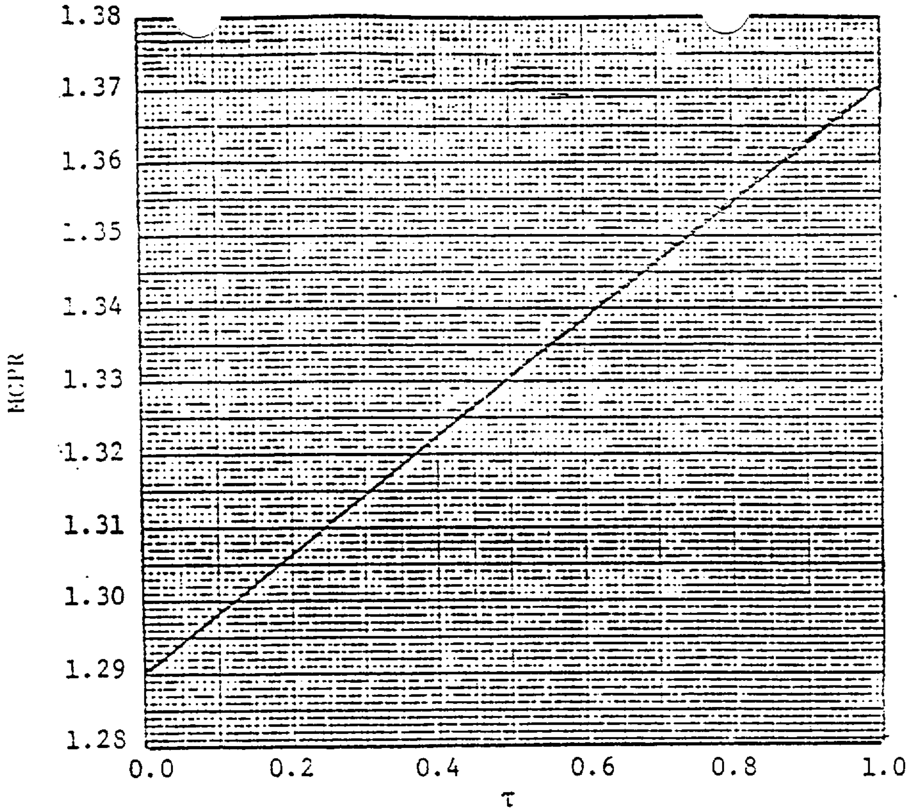
MCPR shall be determined to be equal to or greater than the applicable limit:

a. At least once per 24 hours,

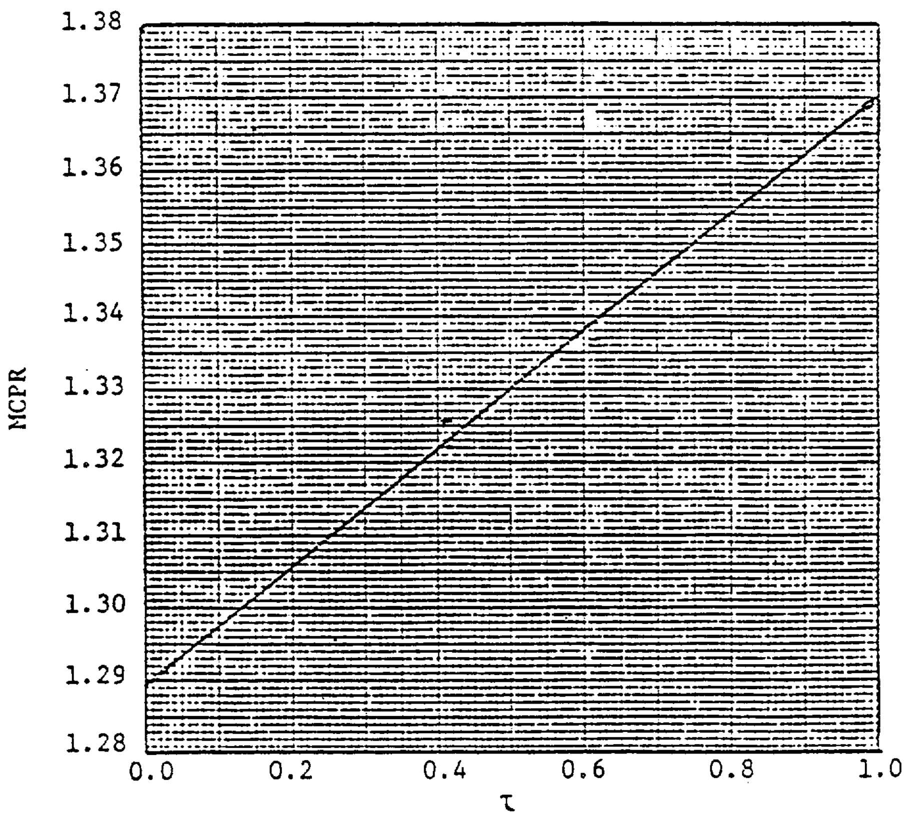
b. Whenever THERMAL POWER has been increased by at least 15% of RATED THERMAL POWER and steady state operating conditions have been established, and

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

MCPR LIMIT AT RATED FLOW AND RATED POWER



8X8R FUEL
FIGURE 3.2.3-1



P8X8R FUEL
FIGURE 3.2.3-2

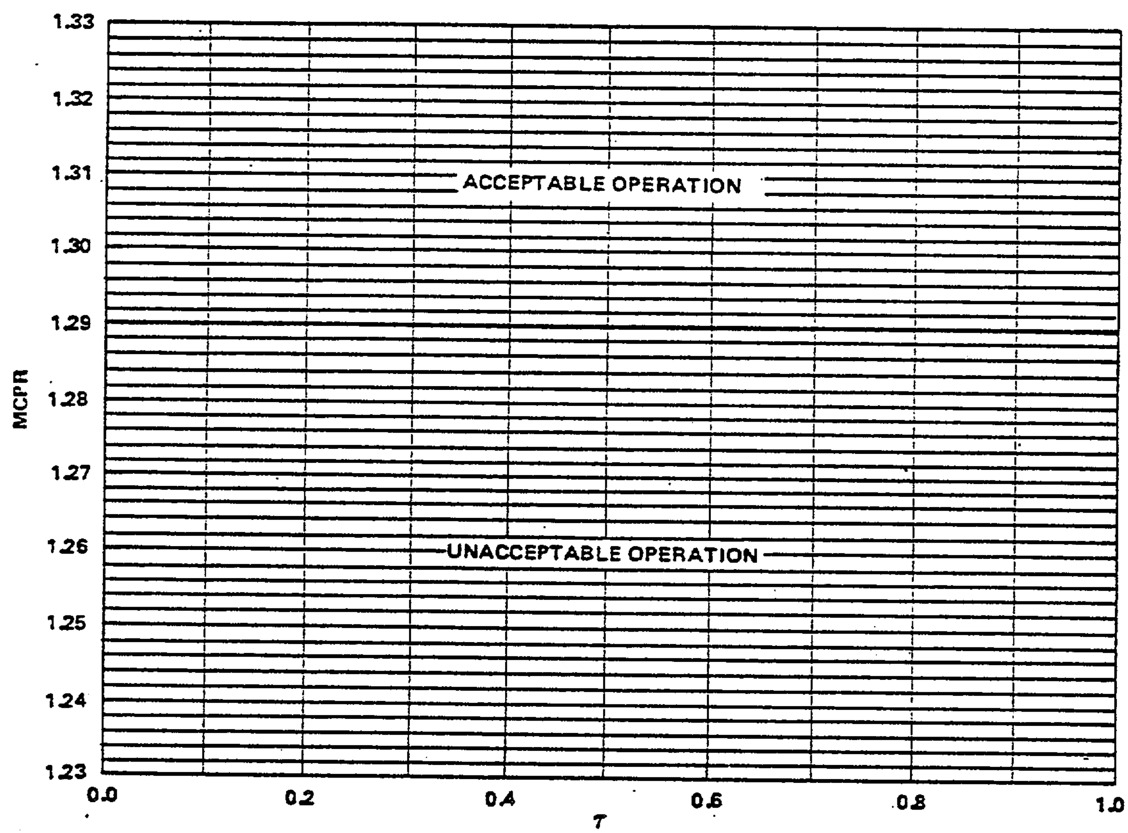


FIGURE 3.2.3-3
MCPR LIMIT FOR 7X7 FUEL
AT RATED FLOW AND RATED POWER

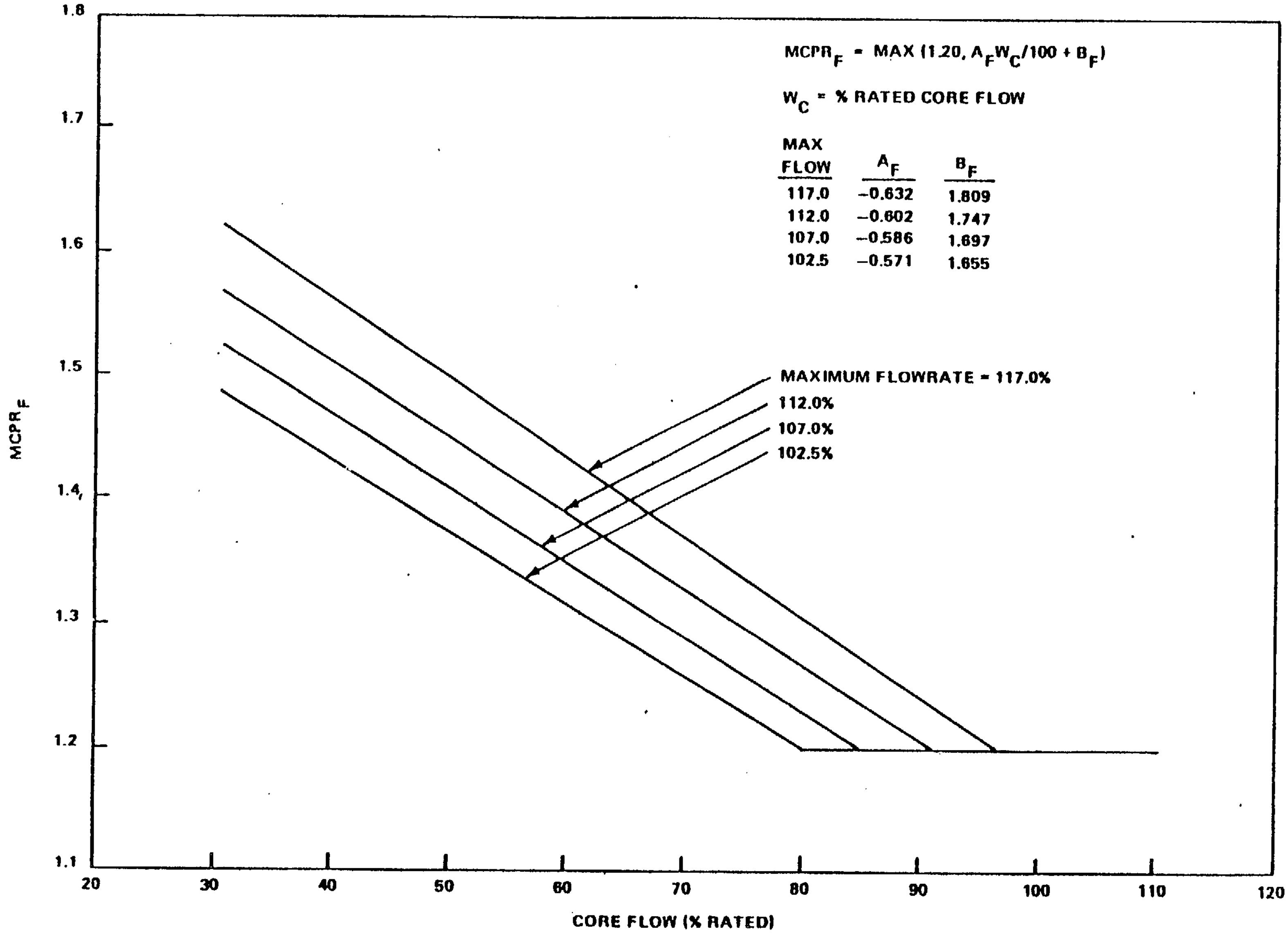


FIGURE 3.2.3-4 $MCPR_F$

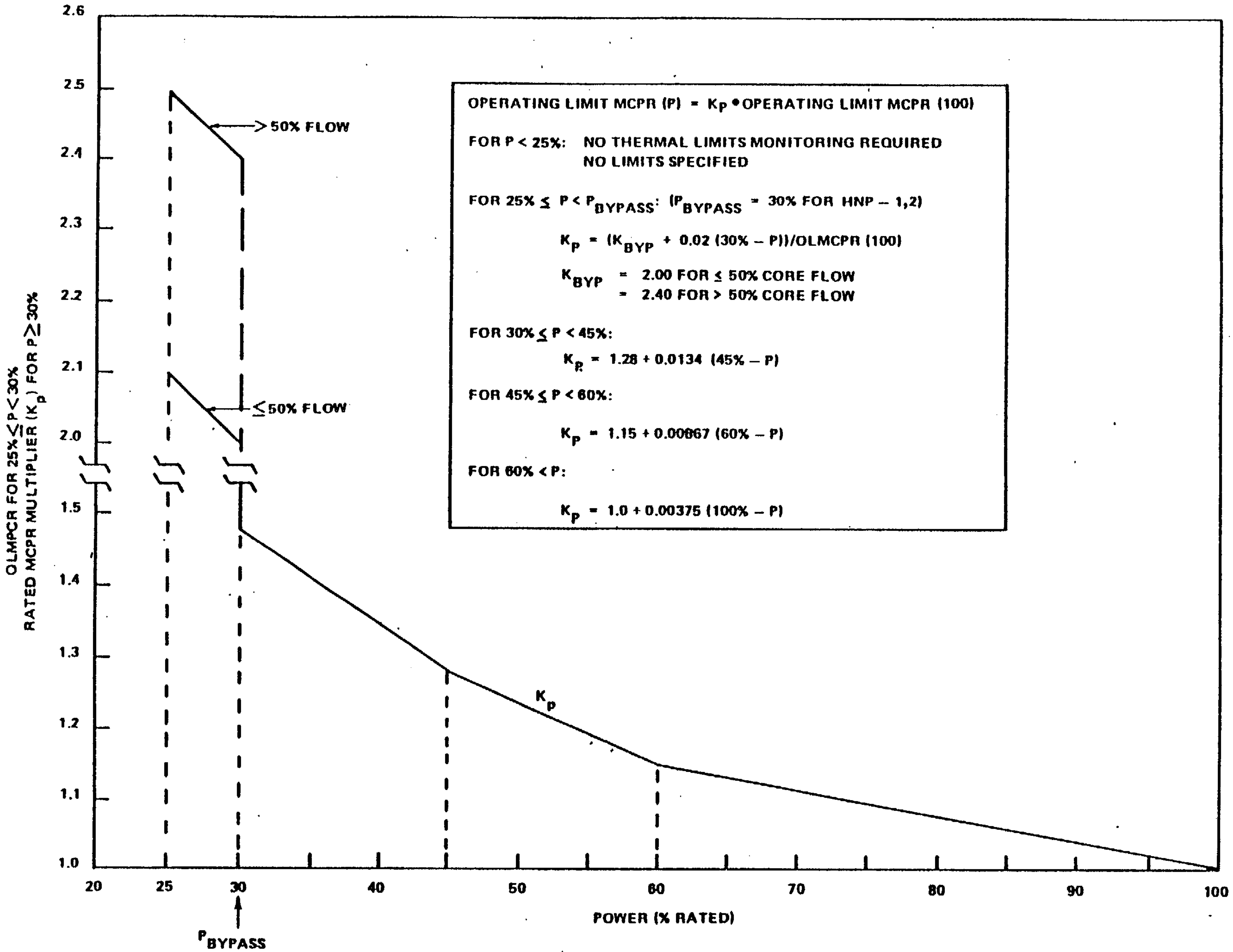


FIGURE 3.2.3-5 Kp

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-N045 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)	≥ -121.5 inches*	≥ -121.5 inches*
b. Drywell Pressure - High	≤ 1.85 psig	≤ 1.85 psig
c. Reactor Steam Dome Pressure - Low	≥ 422 psig**	≥ 422 psig**
d. Logic Power Monitor	NA	NA
2. LOW PRESSURE COOLANT INJECTION MODE OF RHR SYSTEM		
a. Drywell Pressure - High	≤ 1.85 psig	≤ 1.85 psig
b. Reactor Vessel Water Level - Low Low Low (Level 1)	≥ -121.5 inches*	≥ -121.5 inches*
c. Reactor Vessel Shroud Level (Level 0) - High	≥ -207 inches*	≥ -207 inches*
d. Reactor Steam Dome Pressure-Low	≥ 422 psig**	≥ 422 psig**
e. Reactor Steam Dome Pressure-Low	≥ 325 psig	≥ 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	$\leq$ 27
2. LOW PRESSURE COOLANT INJECTION MODE OF RHR SYSTEM	$\leq$ 40
3. HIGH PRESSURE COOLANT INJECTION SYSTEM	$\leq$ 30
4. AUTOMATIC DEPRESSURIZATION SYSTEM	NA
5. ARM LOW LOW SET SYSTEM	$\leq$ 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 ≤ 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	$\leq (0.58\ W + 50\%)^{(a)}$	$\leq (0.58\ W + 50\%)^{(a)}$
b. Inoperative	NA	NA
c. Downscale	$\geq$ 3/125 of full scale	$\geq$ 3/125 of full scale
d. Neutron Flux - High, 12%	$\leq$ 12/125 of full scale	$\leq$ 12/125 of full scale
2. ROD BLOCK MONITOR		
a. Upscale$^{(b)}$		
1) Low Trip Setpoint (LTSP)	$\leq$ 115.1/125 of full scale	$\leq$ 115.5/125 of full scale
2) Intermediate Trip Setpoint (ITSP)	$\leq$ 109.3/125 of full scale	$\leq$ 109.7/125 of full scale
3) High Trip Setpoint (HTSP)	$\leq$ 105.5/125 of full scale	$\leq$ 105.9/125 of full scale
b. Inoperative	NA	NA
c. Downscale	$\geq$ 94/125 of full scale	$\geq$ 93/125 of full scale
d. Power Range Setpoint$^{(c)}$		
1) Low Power Setpoint (LPSP)	$\leq$ 27% of RATED THERMAL POWER	$\leq$ 29% of RATED THERMAL POWER
2) Intermediate Power Setpoint (IPSP)	$\leq$ 62% of RATED THERMAL POWER	$\leq$ 64% of RATED THERMAL POWER
3) High Power Setpoint (HPSP)	$\leq$ 82% of RATED THERMAL POWER	$\leq$ 84% of RATED THERMAL POWER
e. RBM Bypass Time Delay $(td_2)^{(d)}$	$\leq$ 2.0 sec	$\leq$ 2.0 sec
3. SOURCE RANGE MONITORS		
a. Detector not full in	NA	NA
b. Upscale	$\leq 1 \times 10^5$ cps	$\leq 1 \times 10^5$ cps
c. Inoperative	NA	NA
d. Downscale	$\geq$ 3 cps	$\geq$ 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.

INSTRUMENTATION

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 COOLANT SYSTEM

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.

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.

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)

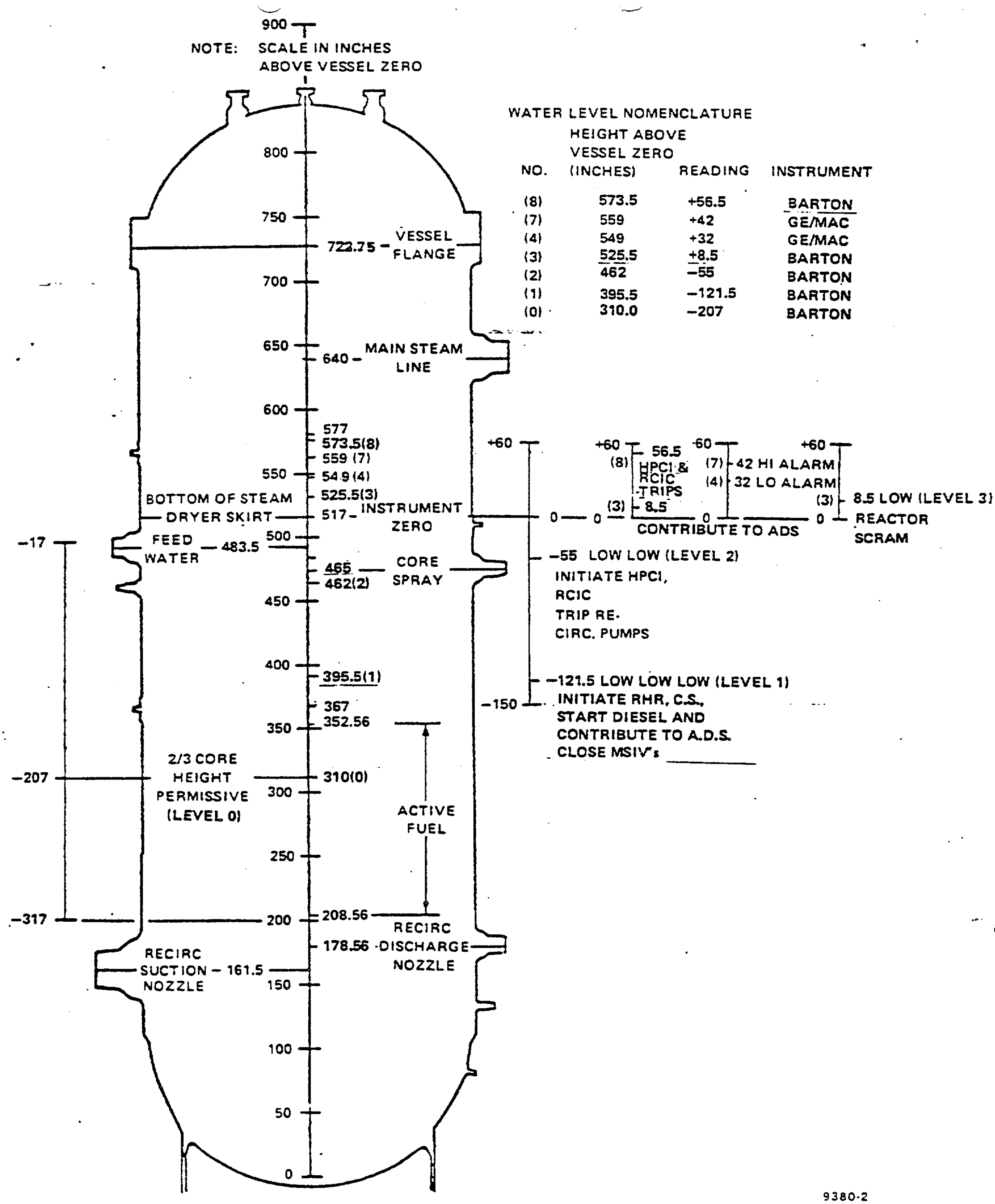
In the event that a portion of the fire detection instrumentation is inoperable, increasing the frequency of fire patrols in the affected areas is required to provide detection capability until the inoperable instrumentation is restored to OPERABILITY.

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 undervoltage 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)	573.5	+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

3/4.9 REFUELING OPERATIONS

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.

DESIGN FEATURES

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 $\leq$ 0.95 when flooded with unborated water. The k_{eff} of $\leq$ 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.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 39 TO FACILITY OPERATING LICENSE NO. NPF-5

GEORGIA POWER COMPANY
OGLETHORPE POWER CORPORATION
MUNICIPAL ELECTRIC AUTHORITY OF GEORGIA
CITY OF DALTON, GEORGIA

EDWIN I. HATCH NUCLEAR PLANT, UNIT NO. 2
DOCKET NO. 50-366

1.0 INTRODUCTION

The Georgia Power Company (GPC or the licensee) has requested certain changes to the Technical Specifications of the Hatch Nuclear Plant, Unit 2, in order to reload and operate the unit for Cycle 5 (Reload 4). The correspondence transmitting the requests and the subjects dealt with are as follows:

Initial Licensee Requests	Supplementary Submittals	Subjects
1. Request to Change TS-Reload 4 (NED-84-192) dated April 3, 1984		1a. MAPLHGHR for New Fuel Bundle
		1b. OLMCPR Increase
		1c. Hybrid I Control Rods
		1d. Fuel Loading Around SRM Detectors
2. Proposed TS Changes to Support ATTS Installation (NED-84-017) dated January 23, 1984	2a. Response to Verbal Questions (NED-84-281) dated June 7, 1984	2. ATTS Installation
Revision to Request for TS Changes to Support ATTS Installation (NED-84-017) April 3, 1984	2b. Revised Responses (NED-84-321) dated June 14, 1984	
	2c. Additional Clarification (NED-84-326) dated June 15, 1984	

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Initial Licensee Requests	Supplementary Submittals	Subjects
3. Proposed TS Changes for ARTS Improvements (NED-84-030) dated February 6, 1984	3a. Additional Information on ARTS (NED-84-186) dated April 3, 1984	3. ARTS Improvements
	3b. Confirmation of Telephone Conversation-ARTS (NED-84-336) dated June 20, 1984	
	3c. ARTS Improvements dated June 27, 1984	

A brief description of each subject follows.

1.1 Addition of MAPLHGR Curve for New Fuel Bundle

This change is requested in connection with the core reloading of Unit 2 to allow for introduction of a new fuel type. The licensee proposes to add a curve of Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) vs. Planar Exposure for the fuel based on the Unit 2 Loss of Coolant Accident (LOCA) response.

1.2 Operating Limit MCPR Increase

This change would increase the Operating Limit Minimum Critical Power Ratio (OLMCPR) for the fuel used in Cycle 5 and subsequent cycles. Approval of this change would permit licensing of subsequent cycles under 10 CFR 50.59.

1.3 Use of Hybrid I Control Rods

This change would allow operation of Hatch 2 with the new Hybrid I control rod assemblies to take advantage of improvements expected from the design which has been reviewed and approved by the NRC staff.

1.4 Fuel Loading Around SRM Detectors

This change would ensure achievement of the required minimum count rate in the Source Range Monitor (SRM) detectors.

1.5 ATTS Installation

This change would permit operation with the newly installed Analog Transmitter Trip System (ATTS) which takes the place of the mechanical type digital sensor switches originally used in the Reactor Protection System (RPS) and the Emergency Core Cooling System (ECCS). The ATTS essentially replaces pressure, level and temperature digital switches with analog sensor/trip unit combinations which provide continual monitoring of critical parameters in addition to performing basic logic trip operations. GE developed ATTS to offset operating disadvantages of the digital sensor switches of the original safety system instrumentation, and to improve sensor intelligence, reliability and testing procedures. We have reviewed and found acceptable the ATTS on a generic basis with the provision that plant specific information would also have to be reviewed (Reference 1, Bibliography).

1.6 The ARTS Improvement Program

This set of changes is required to implement the Average Power Range Monitor (APRM)/Rod Block Monitor (RBM)/Technical Specification (ARTS) Improvement Program. The primary goals of the ARTS program are to:

1. Replace the APRM scram and rod block trip setdown requirements with more meaningful limits to reduce the need for manual setpoint adjustments and to allow more direct thermal limits administration;
2. Change the RBM hardware using up-to-date electronics;
3. Change the Local Power Range Monitor (LPRM) input assignments to improve response of the RBM;
4. Revise the trip logic and signal normalization procedures for the RBM;
5. Introduce new requirements for power and flow dependent MAPLHGR and MCPR limits; and
6. Revise the Technical Specifications to be consistent with the changes.

The extended load line limit analysis provides the basis for changing the slope of the flow bias algorithm and for the revised APRM rod block line.

2.0 DESCRIPTION AND EVALUATION

2.1 Addition of MAPLHGR Curve for New Fuel Bundle

Addition of a new fuel bundle (type P8DRB284H) requires that an additional curve of MAPLHGR as a function of burnup be added to the Technical Specifications. This curve was obtained by methods described in GESTAR II (Reference 2) which have been approved for use in obtaining MAPLHGR values for extended burnup. We conclude that the MAPLHGR values for the new fuel bundle are acceptable.

2.2 Operating Limit MCPR Increase

The full power, full flow OLMCPR is being revised to bound required values for Cycle 5 and beyond in order to permit licensing these cycles under 10 CFR 50.59. The proposed revisions would increase the Option B (T = 0) limits to 1.29 and the Option A limits to 1.37. The limits are currently 1.26 and 1.32, respectively, for type 8x8R fuel and 1.27 and 1.35 for P8x8R fuel. These changes are conservative and are therefore acceptable.

Cycle specific analyses using the approved methods of GESTAR II (Reference 2) will be performed to confirm that the limits bound the requirements for each cycle.

2.3 Use of Hybrid I Control Rods

The description of the control rod assemblies is being revised to permit the replacement of the standard control rod assemblies with the General Electric Hybrid I Control Rod (HICR) assemblies. The use of these control rods in BWRs has been reviewed and approved by the NRC staff (Safety Evaluation letter dated August 22, 1983, Reference 3), and we conclude that their use is acceptable in Hatch Unit 2.

The details of the design and materials will not be included in the revised Technical Specifications. Since descriptions of the standard blades exist in the FSAR and of the HICR blades in approved Topical Report NEDE-22290-A (Reference 4), and the safety design criteria which control rods must meet are contained in the FSAR and in other Technical Specifications, we conclude that this is acceptable.

2.4 Loading of Fuel Assemblies Around SRM Detectors

The Technical Specifications require that a count rate of 3 counts per second (CPS) be present in Source Range Monitor Channels when loading fuel into the core. A spiral loading technique is to be used for Reload 4. In order to achieve the 3 CPS count rate, it is necessary to load irradiated fuel around the SRM detectors. Present Technical Specifications permit the loading of two assemblies and the revised specifications would permit as many as four. Since the present outage has been longer than usual, two assemblies may not be sufficient to provide the required count rate.

Since 16 or more fuel assemblies are required to achieve criticality and the k-effective of an uncontrolled 2x2 array of maximum reactivity assemblies is less than 0.95, we conclude that no criticality problems exist with the proposed configurations. Further, the same assemblies that were present around the SRM detectors during unloading will be returned there. Since these configurations were sub-critical at that time, they will also be so when reloaded. Hence, loading of up to four assemblies around the SRM detectors is acceptable.

2.5 ATTS Installation

The ATTS, as stated above, is a replacement for the mechanical type digital sensor switches. The existing logic arrangement will not be affected. The ATTS and the trip relays provide the input intelligence for the plant process parameters to the system logics for the RPS and the ECCS, including the reactor core isolation cooling (RCIC) system. The proposed instrument modifications are intended to: 1) reduce primary sensor element drift; 2) reduce the frequency of setpoint drift occurrences; 3) provide indication for each primary sensor which will verify operability of the sensor; 4) reduce the time RPS logic must be in half scram condition to functionally test or calibrate a Safety Trip; 5) reduce the functional test and calibration frequency for the primary sensor and facilitate calibration of the primary sensor when the reactor is shutdown for refueling; 6) reduce the likelihood of instrument valving errors; 7) reduce the potential for instrument testing related scrams; and 8) replace devices that are required to mitigate a LOCA and high energy line break with environmentally qualified hardware.

The analog trip system hardware is used to process inputs into the ECCS, RPS, and RCIC. All of the trip unit card files and power supplies for ECCS and RCIC are contained within two sets of Division 1 and 2 cabinets. These devices operate with logic in the energize-to-actuate mode, using the 125 VAC station emergency battery for their power source. Similarly, the RPS contains its own power supply and trip unit card files within four separate independent cabinets, one for each RPS division. These devices operate with logic in deenergized-to-actuate mode using the 120 VAC power from RPS motor generator sets. Since the dual channel design (with two trip systems) of the RPS is not being altered, the operation of the trip system remains the same. The automatic and manual initiation and protective action of essential systems remain unchanged.

The service environments applicable to each item of hardware comprising the ATTS are specified in the Product Qualification Program Requirements Document (22A7011). The cabinet mounted equipment consists of the trip unit hardware, trip unit calibration module, card file, trip relay, voltage convertors, and miscellaneous hardware. The reactor building mounted equipment consists of the pressure and differential pressure transmitters which are mounted locally either on the structures or instrument racks; the sealed sensor differential pressure transmitters which are locally mounted on customer supplied supports in the reactor building, and the RTD temperature sensors which are also locally mounted on supports in the reactor building. The methods used to demonstrate the qualification program of the ATTS at Hatch Units 1 and 2 included type testing and/or analysis. In type testing, the equipment tested was aged and subjected to all applicable environmental influences to provide assurance that all such equipment would be able to perform the intended functions for the required minimum operating time. Qualification by analysis included the construction of valid mathematical models of the equipment to be qualified, verification of the mathematical models by test, and quantitative analysis of the mathematical models to demonstrate that the product performance characteristics met or exceeded the equipment design requirements.

Inductive or capacitive coupled electro-magnetic interaction (EMI) from radiated electromagnetic fields are limited only to near-fields because the distance from the interfering source is usually less than $\lambda/2\pi$, where λ is the wave-length of the interference signal. The following type of EMI susceptibility tests were conducted on the ATTS: 1) conducted EMI transients, 100 to 500 KHZ, 300 VAC peak-to-peak or ±5.0V (24 VDC); 2) conducted RFEMI, 0.5 to 100 MHZ, 5 V peak-to-peak; 3) radiated transient EMI fields, 100 to 500 KHZ, 300 VAC or ±5.0V (24 VDC); and 4) radiated RFEMI fields, 0.5 to 100 MHZ, 5 V peak-to-peak.

Only conducted tests were performed on the converter input leads and relay leads. Only radiated tests were done on the transmitter, RTD, and auxiliary analog output leads since there are no associated branch connections. The EMI tests with the transmitters showed that EMI requirements could only be met with the addition of an EMI filter on each transmitter lead. An EMI filter assembly was designed for use with the transmitters and has become part of the ATTS design. The filter assembly was qualified by analysis to the same environmental requirements as the transmitters. Therefore, we concluded that the ATTS is qualified for operation in its present EMI environment.

The licensee has stated that the wiring for the ATTS design conforms to the recommendations of Regulatory Guide 1.75 to the maximum practical extent. Divisional separation is maintained within the cabinet. Class IE/non-Class IE separation is carried through up to the trip relay. The annunicator trip relays provide separation between 1E and non-1E circuits (i.e., separation is via the contact to coil separation within the relay).

Within the cabinets, the minimum separation distance is 6 in. up to the relay. Within the relay, limitations exist related to the distance from the contact to the coil. This design prevents maintaining complete physical separation between the annunicator wiring and the class 1E wiring. This does not pose a problem because the annunicator circuitry is a low energy circuit. The annunicators interrogate contacts in the ATTS with a 140 V dc signal that is currently limited to a maximum of 1 ma by the annunicator input resistance.

In another area, the licensee indicated that there were non-class 1E loads powered from class 1E buses with a circuit breaker as the only separation device. This is an acceptable means of separation consistent with the original design basis of the plant. We examined the new hardware associated with the addition of the ATTS with respect to susceptibility to failures (i.e., voltage variations, hot shorts, open circuits) caused by non-class 1E loads. The licensee stated that the additional hardware associated with the ATTS is no more susceptible to these failures than the hardware it replaced. Therefore, no new failure modes have been introduced in the Hatch design, and the original licensing basis of the plant with respect to the application of isolation breakers has not been modified. We conclude that this portion of the design is enveloped by the original design basis of the plant and is acceptable.

Previous instruction manuals (4471-1 Rev. A) have contained a warning regarding operating at a low ATTS power supply voltage because if certain conditions exist (e.g., load length, wire diameter, temperature), a low supply voltage at the transmitter may cause it to operate improperly and a desired trip may not occur. Thus, we requested information to address a concern that an undervoltage condition could exist that would incapacitate the trip functions of all the effected ATTS units.

The licensee responded stating that the purpose of the maximum lead length requirement is to assure sufficient voltage out of the trip unit to drive the transmitter. Calculations by General Electric indicate that lead lengths as long as 3820 feet are acceptable using 16 gauge wire. The maximum length of cable used in the Hatch ATTS design is 1800 feet, utilizing 16 gauge wire.

In addition, the licensee stated that the RPS portion of the ATTS is supplied, as is the remainder of the RPS, from the RPS motor-generator (MG) set which has a class 1E electrical protection assembly (EPA) that is installed between each RPS bus and each power source. This protects each RPS bus against a sustained over/under voltage or underfrequency condition. Each EPA consists of a circuit breaker with a trip coil driven by logic circuitry that senses line voltage and frequency and trips the circuit breaker open on conditions of overvoltage, undervoltage or underfrequency. The system itself is a fail-safe system. Therefore, with a loss of power, all instruments go to their safety position.

The ECCS portion of the ATTS is powered off the plant batteries. The class 1E batteries are divisionalized and supplied by chargers that are powered off the emergency buses. The batteries are sized per FSAR Section 8.3.2.1.1.a for two hours continuous duty without the chargers. The power supply for the ECCS portion of ATTS is consistent with the original design basis of the plant. Undervoltage on ECCS portions of the ATTS is protected via the protective design features included in the battery and charger that provide power to ECCS. The minimum voltage that the batteries would ever supply based on the FSAR requirement is 105 VDC. The ATTS has voltage converters which operate from 105 to 140 VDC on the input and provide a nominal output of 25 VDC. We find this to be acceptable since the ATTS is designed to operate with a minimum voltage at 23.5 VDC.

The operability of the trip unit and auxiliary relays is verified by periodic functional testing using special test equipment supplied as part of the ATTS. Operability of the transmitters is verified by periodic comparison of the redundant indicators on the master trip units which monitor the same parameter. Gross transmitter failure is detected by special monitoring circuits. The licensee stated that the high/low gross failure setpoints are to be set at values of 35 ± 0.5 and .5 ± 0.5 respectively. These values are provided to indicate a short-circuit and open-circuit. Therefore, the setpoint values can be varied significantly outside the saturation range of the transmitter and still provide adequate protection.

In addition to the ATTS modification as discussed above, the licensee provided information regarding several proposed Technical Specification revisions. The purposes of these proposed revisions are to utilize the benefits of the ATTS addition, prevent unnecessary plant transients by using less conservation setpoints or delete certain isolation, actuation and permissive sensors. These revisions are as follows.

1. Reactor Water Low Low (Level 2) Trip Setpoint Modifications

The proposed Technical Specification trip setpoint/allowable value for the reactor vessel water level 2 signal is $\geq$-55 in. Reactor vessel water level 2 is for the initiation of high-pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) and the recirculation pump trip. The proposed analytical limit of -58 in. was selected by the licensee to provide the best flexibility and protective margin for the plant. The ECCS calculations are insensitive to the variation in HPCI actuation water level so that a lower water level for level 2 has no significant effect on the ECCS system performance. In addition, this proposed change will have no effect on the (MAPLHGR) limit. The requirements of 10 CFR 100 will still be met.

2. Deletion of High Drywell Pressure Signal for Residual Heat Removal, RPV Head Spray Valves, and Reactor Water Cleanup System Isolation

High drywell pressure has been used as a signal to isolate reactor water cleanup (RWCU) and the shutdown cooling mode of RHR. Small steam leaks in the drywell can cause a high drywell signal which would prohibit an acceptable normal shutdown procedure by preventing operation of the RHR and RWCU systems during the shutdown cooling mode. To resolve this operational concern, the high drywell pressure signal would be deleted from the isolation logic for the RHR shutdown cooling suction and discharge valves, as well as the reactor pressure vessel (RPV) head spray isolation valves and RWCU isolation valves.

The use of high drywell pressure as an isolation signal has little effect in preventing coolant losses due to an RHR or RWCU pipe break inside the drywell since the inboard isolation valves are located as close as possible to the drywell wall. Such pipe breaks do not present a site boundary dose problem since the leaked fluid and associated radioactivity are completely retained within the primary containment boundary. The high drywell pressure signal for the RPV head spray isolation valves will also be deleted. Since the RPV head spray valves are used as part of the shutdown cooling procedures, this change is consistent with above mentioned proposed RHR (shutdown cooling mode) system modification.

This change does not affect the Appendix K calculation results presented in the FSAR. Therefore, the requirements of 10 CFR 100 will still be met. In addition, this modification has been implemented and accepted by the staff on other BWR/4s.

3. Lowered Water Level Trip Setpoint for Isolation of Reactor Water Cleanup System and Secondary Containment, and Starting of Standby Gas Treatment System (SGTS)

Reactor scram from normal power levels (above 50 percent of rated) usually results in a reactor vessel water level transient due to void collapse that causes isolation of the RWCU system at reactor water level 3.

The result is typically the dropping of the cleanup filter cake, added radwaste processing, loss of ability to remove water from the reactor vessel immediately after scram, and other undesirable operational problems. These results adversely affect plant availability and operability. By lowering the isolation setpoint to reactor water level 2, these problems may be resolved without any adverse safety impact. The lowering of the level trip for isolation of RWCU from reactor water level 3 to reactor water level 2 will not have any adverse effect on plant transient and accident analyses. For any reactor pressure coolant boundary line breaks inside the primary containment, the LOCA design basis accident (DBA) analysis shows that the ECCS is capable of mitigating all break sizes including and up to the recirculation line break. For a RWCU line break outside the primary containment, the break detection is provided by the high differential temperature rather than by water level variation.

By lowering the SGTS actuation and secondary containment isolation from reactor water level 3 to reactor water level 2, a potential for spurious trips is reduced. The ECCS analysis design basis assumes that the SGTS will initiate at the same time as the ECCS which initiates at reactor water level 2.

This modification has been implemented and accepted by the staff on other BWR/4s. The requirements of 10 CFR 100 will still be met.

4. <u>Deletion of Ambient Temperature Loops in Leak Detection System</u>

Typically, the leak detection system uses ambient and differential temperatures to detect the small high-temperature leaks. In the earlier design, the ambient temperature trip was provided by an independent temperature element and trip device, and the differential temperature trip was provided by two independent temperature elements and a Δ T trip device. By using the ATTS, the ambient temperature trip may be obtained from one leg of the differential temperature trip. With this arrangement, the sensitivity of leak detection may be changed slightly, dependent on heating, ventilation, and air-conditioning design; but it will not defeat the intended function of the system. This arrangement is suitable for the small rooms containing leak detection temperature monitoring as part of the isolation logic because only large rooms, such as the turbine building, need the spatial location of sensors to adequately protect the room against leaks. This scheme allows the deletion of several unnecessary temperature loops in the RWCU system.

The RWCU temperature and differential temperature sensors sense the temperature in the two pump rooms and the heat exchanger room. Each room has a redundant set of temperature instrumentation that provides input to the RWCU isolation logic. By using the hot leg of the differential temperature sensor for the high ambient trip, several devices may be deleted without any loss of protective function.

For these modifications, single-failure criteria will be maintained.

The proposed Technical Specification revisions will reference the trip unit loop from which the ambient temperature trip is taken in place of the existing ambient temperature trip instrument. Included in these proposed changes are new surveillance frequencies which correspond with the surveillance requirements of the ATTS.

5. Deletion of Drywell Pressure Sensors E11-N011A, B, C, D

The original design of Plant Hatch has the high drywell pressure signals for the ECCS coming from eight sensing devices. For example, E11-N011A, B, C, D (existing MPL numbers) provide signals to RHR, core spray, and HPCI; E11-N010A, B, C, D (existing MPL numbers) provide signals to ADS. This configuration is inconsistent with the inputs for the reactor water levels 1 and 2 trips which are provided by only four sensing devices, namely B21-N031A, B, C, D (existing MPL numbers). To make drywell pressure sensor configuration consistent with that for the water levels 1 and 2 sensors, drywell pressure sensors E11-N010A, B, C, D may be used to provide signals for all four systems of the ECCS and still maintain single-failure criteria. Plant safety margin is not being reduced since the level of redundancy to serve a trip function is maintained.

This change deletes instruments E11-N011A, B, C, D and transfers their associated trip function to instruments E11-N010A, B, C, D. Since these instruments (E11-N010A, B, C, D) are being incorporated into the ATTS modification, the instrument number was changed to E11-N694A, B, C, D.

It is proposed that the surveillance frequencies be modified to those of instruments E11-N010A, B, C, D. It should be noted that the surveillance frequencies of both E11-N010A, B, C, D, and E11-N011A, B, C, D are the same in the existing Unit 2 Technical Specifications.

6. Trip Setpoint/Allowable Value Setpoint Modifications

The instruments to be incorporated into the ATTS possess less drift and greater accuracy than the existing instruments in use at Plant Hatch. Therefore, new calculations were performed to determine the setpoint value for each instrument. The Plant Hatch analytical limits were used (were applicable) to develop the allowable values and trip setpoints. The values that are proposed to be inserted into the Technical Specifications are the calculated allowable values. The setpoints used at Plant Hatch will take into consideration instrument drift and will be developed from the allowable values. The proposed Technical Specification revisions include modification of the trip setpoint/allowable values for the following instruments:

- o Main steamline flow-high (B21-N686A, B, C, D through B21-N689A, B, C, D)
- o Main steamline tunnel temperature-high (B21-N623A, B, C, D through B21-N626A, B, C, D)

- o Reactor vessel steam dome pressure-high (B21-N678A, B, C, D)
- o Reactor vessel water level-level 3 (B21-N680A, B, C, D)
- o Reactor vessel steam dome pressure-low(B31-N679A, D)
- o Reactor vessel water level-level 1 (B21-N681A, B, C, D)
- o Drywell pressure-high (C71-N650A, B, C, D)
- o RWCU room ambient temperature-high (G31-N622A, D, E, H, J, M)
- o RWCU area differential temperature-high (G31-N663A, D, E, H, J, M) (G31-N661A, D, E, H, J, M)

7. Reactor Vessel Water Level - High (Level 8) Trip Instrumentation Modifications

After HPCI and RCIC have activated, this trip function prevents the water level in the reactor vessel from reaching the height of the main steam outlet. Its intended protective function is accomplished by tripping the HPCI steam turbine enclosing the HPCI and RCIC steam supply valves when the water level in the reactor vessel reaches the level 8 setting. The trip function is to protect the HPCI and RCIC steam turbine system from potential damage.

This trip function is currently assigned to B21-N017A, B, C, D which controls the RPS reactor vessel water level 3 instrumentation. To separate the RPS and ECCS functions, the ATTS design assigns the reactor vessel water level 8 trip function to ECCS instrumentation. The functions of the level 8 trip remain the same.

The analytical limit for this function is 59.5 in. The licensee stated that the trip setpoint/allowable value of $\leq$ 56.5 in. was developed using the criteria of Regulatory Guide 1.105, and the designated trip setpoint for the plant will take into consideration setpoint drift.

8. Elimination of the Reactor Pressure Permissive to the Bypass of the MSIV Closure Signal Due to Low Condenser Vacuum

The licensee proposed to delete the reactor steam dome pressure permissive which prevents the group 1 isolation valves signal from being bypassed on a low condenser vacuum isolation.

The manual bypass is provided to facilitate the following operations:

A. The bypass allows cold shutdown testing of the main steamline isolation logic and allows stroking the MSIVs open and closed for maintenance even though there is no condenser vacuum.

B. The bypass allows the MSIVs to be opened so seal steam and ejector steam can be available at the turbine and condensor, thereby allowing restart of the reactor from a hot pressurized condition. Attempting to establish condenser vacuum without seal steam from the hot condition by the mechanical vacuum pump may damage the turbine shaft seals.

Thus, the manual bypass of the MSIV closure is performed only when the reactor is not operating at full power. In addition, the manual bypass, which is annunciated in the control room, has the following three permissive conditions:

A. When the keylocked manual switches located on the back cabinets housing the MSIV logic are in the bypass position. One keylocked switch is in each isolation logic string.

B. When the turbine stop valves are less than 90 percent full open, the four independent contacts of the turbine stop valve position switch sensor relay of the RPS will trip.

C. When the reactor is below 1045 psig.

Of these three permissives on the manual bypass of the MSIV closure, only the reactor pressure permissive (item C above) does not have a safety function. This is also the only permissive that is proposed for deletion. With the setpoint at 1045 psig, which is the same setpoint for the high reactor pressure scram, the manual bypass can be performed at any operating reactor pressure provided the other two permissives are cleared. When the manual bypass is activated, plant protection is provided by those two other permissives by the normal scram and isolation signals, e.g., turbine stop valve position, low reactor water level, high steam flow, high steam tunnel temperature, and turbine building temperature, and by the annunciators in the control room. Eliminating the reactor pressure permissive does not affect the existing plant protection in any way.

We have reviewed the acceptability of these proposed Technical Specification revisions. We questioned the licensee regarding the basis for the analytical limits used in the safety analysis. The licensee stated that the analytical limits are the values used as inputs to the safety analysis in the FSAR. For Hatch, the analytical limits were selected to prevent violation of the applicable safety limits. For example, the analytical limit for the level 1 reactor water level trip satisfies the peak cladding temperature of 2200°F in the Hatch Appendix K LOCA analyses. Unless otherwise noted (revisions (1) and (3) as discussed on pages 7 and 8 of this evaluation), the analytical limits used in the setpoint calculations were the original analytical limits used in the Hatch Safety Analysis. For the analytical limits that were revised, the licensee stated that, in no case with these new limits do the FSAR analyzed transients or accidents exceed the safety limits which are specified in the Hatch Technical Specifications. The conservatisms in the Hatch design basis computer codes were not used in place of the analytical limit for the starting value of the calculations.

The allowable value was obtained by either adding or subtracting (whichever was conservative) the loop accuracy from the analytical limit. Loop accuracy was determined by utilizing the square root of the sum of the squares of the transmitter accuracy, trip unit accuracy and calibration accuracy. These accuracies are treated as independent variables between the analytical limit and allowable value. The trip setpoint was calculated by adding or subtracting (whichever was used to obtain the allowable value) the loop drift and the leave alone range from the allowable value.

Each of these terms is a function of other parameters; for instance, the transmitter accuracy reflects transmitter performance with regard to the transmitter basic reference accuracy, transmitter temperature specifications, power supply specifications and static pressure specifications. The licensee stated that these parameters envelope the Hatch Unit 2 design requirements. Drift of the trip units will be monitored on a monthly basis and drift of the transmitters will be monitored on an operating cycle basis using plant procedures. The licensee intends to evaluate the performance of the ATTS against the manufacturer's specifications and, if necessary, propose modifications to the surveillance frequencies specified in the Technical Specifications.

The transmitter and trip unit drifts are treated as independent variables between the allowable value and trip setpoint. The total loop accuracy and the total loop drift (dependent variables) are directly added to obtain the trip setpoint. Setpoint drift is the only value that is extrapolated in the licensee's setpoint methodology. In many cases, the manufacturer's specifications only provide drift values for 6 or 12 month intervals. These values were extrapolated linearly to provide 18 and 24 month drift values for use in the Hatch setpoint calculations.

An additional variable called the leave-alone band was added (treated as a dependent variable) between the allowable value and trip setpoint. This band is set at ± 0.25 percent of the trip unit range and allows a range of values that the trip unit may vary. A setpoint adjustment is not required when the trip unit setting is within this ±0.25 percent range. If the trip unit is out of the range from the setpoint on a monthly calibration functional test, the operator resets the trip unit trip setpoint within the 0.25 percent range. Currently, if the trip unit is outside the ± 0.60% (sum of leave alone range + trip unit drift), a deficiency report will be generated internally at Hatch by the licensee.

The calibration accuracy leads to the only possible component of error caused by a man-machine interface. To counter this error, the licensee has installed a requirement that calibration be performed with instruments of 0.25 percent or better accuracy. This value was assumed in the setpoint calculations.

The trip setpoint milliamp value is read directly from the calibration unit. The calibration unit locks in the trip setpoint value and presents a digital display. During channel calibration, the readings are taken with a digital voltmeter. Sufficient stability of these readouts is presented such that the human ability to read the display presents insignificant errors in the overall results of the setpoint calculations.

We questioned the licensee regarding the effects of a harsh environment on the resulting setpoint. The licensee stated that the two areas explicitly considered in the harsh environment effects were radiation and temperature compensation. These were considered as independent effects. The reasoning that they are independent effects is that temperature peaks relatively early in a LOCA event while significant radiation integrated doses occur later. As a result of a GE evaluation for Barton transmitters, it was determined that radiation effects were not a significant effect in the setpoint calculations. Therefore, the setpoint calculations did not explicitly consider radiation as a parameter. An evaluation was performed which allowed exclusion of the radiation effect also for those trip functions where Rosemount transmitters are to be installed. Humidity was not an explicit parameter in the setpoint calculations. The testing program for the transmitters included exposure to a steam environment during the DBE/post-DBE testing phases. Therefore, the effects of humidity are accounted for in the temperature compensation factor. In addition, post-accident harsh environment pressure effects on the ATTS accuracies was also evaluated. This evaluation has shown that this environmental factor has a negligible effect on setpoint drift or instrument error.

The final consideration of environmental effects on setpoints is presently an ongoing study which is being performed by the utilities as a part of the equipment qualification program. The findings of the staff review of this study will be factored into the setpoint methodology for the Hatch Plant.

We have reviewed the acceptability of the proposed Technical Specification revisions and have concluded that the proposed Technical Specification revisions permit the operation of the facility in a manner that is consistent with the licensing basis and accident analysis. Therefore, we find that, with the provisions of the generic review noted above, the Technical Specification revisions related to the ATTS are acceptable.

In conclusion, we have previously reviewed (Reference 1) the use of the ATTS and found that, provided certain interface requirements were satisfied, the system is acceptable. Based on our review of the documentation submitted by the licensee, we conclude that the modifications proposed satisfy the constraints of our prior approval and also satisfy the requirements of the applicable General Design Criterion and Regulatory Guides. In addition, based on the data submitted, we conclude that:

1) The reliability, accuracy, and response time of the replacement instrumentation are better than that of the existing instrumentation.

2) The separation criteria of the original plant design is unchanged or improved in some areas. Separation is provided by locating equipment on separate racks and panels and by running cable in separated cable trays or conduits. The power supply used for an instrument channel is dependent on that channel's divisional assignment.

3) No new single failure events have been created. Therefore, no single failure will result in any action not previously evaluated in the FSAR.

4) All new equipment has been tested or analyzed to assure that the design environmental conditions and the design basis seismic requirements are met.

5) Means are provided to test the trip units periodically by injecting a signal and observing the trip output. Operability of the analog loop is verified by instrument checks.

6) Proposed Technical Specification revisions permit the operation of the facility in a manner that is consistent with the licensing basis and accident analysis for Hatch 2.

Therefore, we conclude that the modifications of the RPS, ECCS and RCIC as discussed above are acceptable. It is further concluded that the applicable, revised Technical Specification pages are acceptable.

2.6 The ARTS Improvement Program

Each of the components of the improvement program is discussed below:

2.6.1 Extended Load Line Limit Analysis

The effect on transient analysis and core stability were examined for the extended load line limit operation which permits higher powers for low flow conditions by changing the slope of the APRM rod block line. The effect is to allow operation at 100 percent power for greater than 87 percent flow and to increase the permitted power at 40 percent flow by about 5 percent to 63.2 percent.

2.6.1.1 Transient and Accident Analyses

The transient and accident analyses described in the evaluation of the ARTS program below have all assumed operation with the extended load line limit. The changes in core behavior caused by the extended operating range have thus been accounted for in the revised analyses.

2.6.1.2 Thermal-Hydraulic Stability

The results of the thermal-hydraulic analysis (NEDO-30260, Reference 5) show that the core has the smallest stability margin for the power/flow map at the point where the extrapolated rod block line intercepts the natural circulation line and the corresponding maximum decay ratios are 0.91 and 0.93 for Unit 1 and Unit 2, respectively. In addition, the licensee has committed that operating procedures for both Units, prior to Cycle 4 startup for Unit 2, will implement the recommendations of General Electric Company Service Information Letter (SIL) #380 regarding precautionary monitoring of local and average power instrumentation to avoid unstable operation at low flow and will provide for insertion of control rods to or below the 80% rod line in event of pump trip leading to single loop operation. Since (1) the calculated maximum decay ratios are less than that of some of the operating plants (for example, Peach Bottom Units 2 and 3 have the maximum decay ratio of 0.98), (2) there will be added margin to the core stability because the Technical Specifications prohibit natural circulation as a normal operating mode, and (3) since the licensee is implementing operating procedures to assure thermal-hydraulic stability while operating at low flow or with a single recirculation loop in service, we have concluded that the thermal-hydraulic stability results are acceptable for extension of the load line limits for Unit 1, Cycle 7 and Unit 2, Cycle 4 operations.

2.6.2 APRM System Improvements

Each APRM channel consists of a number of LPRMs which are chosen in such a way that the channel output is proportional to core power. The APRM signals are compared to a fixed scram trip (at 120% full power) and to a flow biased rod withdrawal block trip. In addition, the APRM signals are passed through a filter having a time constant of approximately six seconds to form the simulated thermal power monitor (STPM). The STPM output is then compared to a flow biased scram trip.

Current Hatch Technical Specifications require that the flow biased APRM setpoints be lowered (set down) if the core maximum fraction of limiting power density (CMFLPD) exceeds the fraction of rated power (FRP). This may be accomplished by increasing the APRM channel gain.

If CMFLPD exceeds FRP and the core power is raised to its full value, the operating limit value for MAPLHGR or MCPR would be exceeded and the assumptions used in the plant transient analyses violated.

In the proposed APRM system, the setdown requirement would be removed. It would be replaced by power and flow dependent MAPLHGR and MCPR limits. Analyses have been performed to obtain the multipliers to be applied to the full power values of MAPLHGR and MCPR in order to prevent violation of safety criteria during transients and accidents. The LOCA and limiting transients were reanalyzed.

2.6.2.1 Loss-of-Coolant Accident

Previous analyses of the LOCA at less than rated flow have assumed operation under the proposed flow bias APRM rod block line (0.58 W + 50) with the APRM setdown in effect. The analyses showed that no revision of the MAPLHGR limits was required for low flow. However, if the setdown factor (FRP/CMFLPD) is not applied, the evaluation shows that a factor of 0.86 must be applied to the MAPLHGR operating limits when core flow is below 61 percent of rated flow. Accordingly, a Technical Specification curve including this factor, along with others, as described below, is constructed for the Hatch Plants. Because the LOCA analyses were performed with previously used and approved methods, we find them to be acceptable.

2.6.2.2 Transients

In order to restore safety margins which might be reduced when the APRM setdown is removed, the limiting transients were reanalyzed assuming the absence of this feature. The analyses assumed operation within the proposed extended power/flow domain with flows up to 105 percent of rated flow. Analyses of the transient events were made as a function of initial power and flow and the results used to determine multipliers to be applied to full power-full flow values of MCPR and MAPLHGR. The power dependence was most sensitive at full flow and the feedwater controller failure was the transient showing the largest sensitivity. This event was then used to construct a curve of MCPR multiplier, K_p, and MAPLHGR multiplier, $MAPFAC_p$, as a function of core power. Conservative curves were drawn in order to bound future cycles.

Flow dependence of MCPR and MAPLHGR was determined from analyses of flow runout events in which the core flow is ramped rapidly upward to the maximum value permitted by the setting of the recirculation pump scoop. The flow multipliers, K_f and $MAPFAC_f$, are thus a function of the initial flow and the maximum flow and a family of curves is drawn. The multipliers are chosen so that a flow runout to the maximum flow will not result in a violation of MCPR or LHGR safety limits. The $MAPFAC_f$ curves are combined with the results of the LOCA analysis described above and the combined family of curves is used in the Technical Specifications. For inclusion in the Technical Specifications, the K_f curve family is transposed to a $MCPR_f$ family by assuming a value of 1.2 for the full flow MCPR. This is the lowest value that may be used for the Hatch Units (constrained by the ECCS analyses).

The discussion immediately above applies to the power range from 30 to 100 percent of full power. Below 30 percent of full power the turbine stop and control valve scrams are bypassed and the analyses do not apply. Below 25 percent of full power no MCPR and MAPLHGR limits are defined. In the interval between 25 and 30 percent of full power, flow dependent effects are taken into account by having two power dependent curves - one for flows greater than 50 percent of rated and one for lower flows. Analyses are then performed to obtain limiting MCPR and MAPLHGR values in these domains.

Approved methods were used to perform the analyses described above except for those used for the loss of feedwater heater event. For that event the trend analysis was performed by a code approved for other purposes. However this event is not limiting and safety analyses for the event are done by approved methods. We find this acceptable.

We conclude that deletion of the APRM setdown requirement is acceptable when it is replaced by the power and flow dependent operating limits described above.

2.6.3 Rod Block Monitor System Improvements

The Rod Block Monitor (RBM) System is used to prevent violation of fuel thermal-hydraulic limits in the event of inadvertent continuous withdrawal of a control rod. When a rod is selected for withdrawal, the surrounding LPRM strings are selected. Their response to the withdrawal is monitored, and a withdrawal block is initiated by the RBM if that response exceeds certain limits. These limits are selected so that no violation of fuel limits occurs. The RBM has two independent channels either of which will initiate a rod block if tripped.

The proposed Rod Block Monitor improvements include:

1. Re-ordering of the assignment of LPRM detectors to the two RBM channels in order to increase instrument sensitivity and provide more uniformity of response between the two channels.

2. Changing the baseline normalization of the RBM from an APRM channel to a fixed signal in order to reduce the number of unnecessary rod blocks.

3. Replacing the flow-biased trip setpoints with fixed power-dependent trip setpoints, and

4. Elimination of the resettable trips in order to make operation simpler.

In addition, the electronics hardware has been updated to increase the reliability of operation.

The change in LPRM assignments is described in the licensee submittal and a comparison of the RBM channel responses to those of the current design made. The revised design shows similar responses for the two channels each of which has a response similar to that of the most responsive channel in the current design.

A block diagram of the revised RBM system is presented and a discussion of the electronics changes given in Appendix A to NEDC-30474-P (Reference 6). We conclude that sufficient information is given in the report to permit the conclusion that the proposed revisions to the RBM system design are acceptable. The electronics changes are discussed separately below.

2.6.3.1 Reanalysis of Rod Withdrawal Error Event

The revisions of the RBM system necessitate the re-evaluation of the Rod Withdrawal Error Event. The present deterministic, bounding, cycle specific analysis is replaced with a statistical analysis valid for application to all Hatch cores using GE fuel up to type P8x8R inclusive. A data base calculated from actual plant operating states was created which covers the spectrum of plant sizes and power densities. The data base construction began with the selection of operating states at near full power which had low MCPRs and/or high MAPLHGRs in bundles near deeply inserted control rods. The rod configurations were then adjusted to bring the MCPR values to approximately 1.20. Thirty-nine such configurations were chosen. In order to investigate power and flow dependence, the rod configuration in 26 of the above cases was held constant, the flow was reduced to 40 percent of rated and xenon allowed to equilibrate. Finally, for the 26 cases, the flow was held constant at 40 percent and the rod pattern altered to yield 40 percent power with no xenon. For each of the 91 cases described above 100 rod withdrawal error (RWE) analyses were performed assuming a random distribution of starting points for the error rod (and thus initial MCPR values, $MCPR_I$,) and random failures of the LPRMs which provide inputs to the Rod Block Monitor. All cases which did not result in a rod block were rejected from the data base unless the rod started from the fully inserted position. A 15 percent random failure rate was assigned to each LPRM. Experience has shown this value to be high.

The Rod Block Monitor response was generated as a function of error rod position for each RWE. The currently used and approved methods were employed in the analyses. The results were tabulated as error rod position vs assumed Rod Block Monitor setting. These results were then transformed into values of normalized MCPR change ($\Delta MCPR/MCPR_I$) and the mean and standard deviation of the distribution for each set of 100 RWE analyses were determined for each RBM setting. These data were then combined to obtain a mean and standard deviation for the entire data base at each power/flow state for each RBM channel at each assumed RBM setting.

A plot of the required initial MCPR value ($MCPR_I$) as a function of Rod Block Monitor trip setting is constructed. The required value of $MCPR_I$ is that which assures that 95 percent of the rod withdrawal errors which are initiated from it do not violate the MCPR safety limit (1.07) with a 95 percent confidence level.

The final step is the selection of suitable setpoints for the Rod Block Monitor. These are chosen so that the rod withdrawal event is not limiting. At any power level the required operating limit MCPR for this event is not greater than that required for other transients as described in Section 2.6.2 above. A value of 1.20 at full power/full flow is dictated by ECCS considerations. In keeping with the three trip settings of the present system, the power range from 25 percent to full power is divided into three intervals with a constant setpoint in each interval. For Hatch the intervals are 30-65, 65-85, and 85-100 percent of full power. The analytic setpoints for the intervals are respectively, 118, 112, and 108 percent of the reference signal.

The effect of the absence of LPRM strings for certain rods near the periphery of the core has been analyzed and it was shown that the setpoints described above are adequate to mitigate the consequences of the rod withdrawal error on the periphery of the core.

A downscale trip at about 94 percent of the reference signal also inhibits rod withdrawal.

The analyses described above assumed unfiltered LPRM signal inputs to the RBM. However provision is made in the instrument for a filter having a time constant of up to 0.55 seconds. Use of such a filter would necessitate the reduction of the setpoints given above by an amount which depends on the time constant chosen. Analyses were performed to determine the required adjustments and the applicable values are given in NEDC-30474-P (Reference 5). If anything other than no filtering is chosen, the maximum time constant is recommended. In addition, a delay occurs between the time when the input signal reaches the setpoint and the imposition of the rod block. A value of 2.0 seconds was assumed for this delay and no greater value may be permitted. This value is incorporated into the Technical Specifications.

In order to confirm the use of a 15 percent failure probability in the statistical analysis, a sensitivity study was performed in which failure rates up to 30 percent were assumed. Increasing the failure rate to the higher value had a negligible effect on the results.

The Rod Block Monitor is currently required to be operable when core power is greater than some low power setpoint (25-30 percent of full power). Additional surveillance is required if the core has a "limiting control rod pattern" - defined to be a pattern which causes the core to be at the operating limit on MCPR, APLHGR or LHGR. Strictly speaking however, the RBM is only required if the complete withdrawal of any single rod in the core would violate safety limits. Analyses have been performed using the data base described above to obtain operating limit MCPR values above which no rod withdrawal error could lead to violation of the limits. Two values are defined - one for power levels greater than 90 percent full power and one for levels from 25 to 90 percent full power. If the plant is operating at or below these limits, it is on a "limiting control rod pattern" and the RBM is required to be operable. It may be bypassed when operating above these limits.

2.6.3.2 Electrical Instrumentation and Control

The RBM system is designed to automatically detect and block control rod withdrawal that could violate Technical Specification safety limits during a single control rod withdrawal error (RWE) transient. It is assumed that the core is operated in compliance with plant Technical Specifications before the RWE event. There are two RBM channels, either of which can initiate a rod block (i.e., prevent control rod withdrawal). The RBM channels are powered from the Reactor Protection System (RPS) buses (RBM channel A is powered from RPS bus A, and RBM channel B is powered from RPS bus B). Although the RBM system is not safety related, separation is provided between the RBM channels to allow for single failures, and to allow one channel to be bypassed if necessary. RBM channel bypass is accomplished via a single three position bypass switch such that only one RBM channel can be bypassed at a time. Both RBM channels are operable when the switch is placed in the center (normal) position. Both local and remote indication of an RBM channel bypass are provided via indicator lights. The licensee has stated that implementation of the ARTS program will not compromise the redundancy provided between the RBM channels, and that isolation will be maintained between the RBM system and safety related circuits. The RBM output functions (i.e., recorders located on the reactor operator's console, local meters, trip units, and the on-line computer) will remain unchanged, although in some cases the signals used for these functions have been modified. The hardware changes involved in the ARTS modification include new model printed circuit (PC) cards, relays, relay sockets, mounting hardware, and wiring.

Upon selecting a control rod for movement, each RBM channel automatically computes the average of all assigned (and unbypassed) local power range monitor channels. The average signal is then filtered (to reduce signal noise), delayed (to allow the signal to reach its maximum/equilibrium value), and then amplified to read the same as a fixed reference signal. This process (referred to as the RBM null sequence) is reinitiated each time a new rod is selected for movement. Control rod motion is blocked during the null sequence. Each RBM channel then compares the calibrated (nulled) signal to an automatically selected preset rod block alarm/trip level (one of three power biased upscale trip levels is selected dependent upon the current reactor power level). The trip level is selected based on the magnitude of a reference APRM. If the local neutron flux level increases to the upscale trip level, further control rod withdrawal is blocked, thus limiting the change (increase) in local power. Thus, the ARTS modification to the RBM trip logic replaces the standard RBM flow biased (recirculation flow) trip feature with power (neutron flux level) biased trips. This modification will be implemented by changes to PC card electronics (averaging cards, null sequence cards, RBM setpoint cards, and quad trip cards).

It should be noted that an adjustable time delay (t_{d2} 1 to 50 seconds ± 0.5 seconds) has been added to delay the calibrated (nulled) average local neutron flux signal to the RBM trip logic. The purpose of this delay is to allow minimum rod movements despite abnormally high signal noise not removed by filtering. This delay is typically set at a value of 1 to 2 seconds. The design of the control rod drive system is for a normal speed of 3 inches per second ± 0.6 inches per second. The licensee has analyses that show the delay is short enough to limit rod movement well below that which could cause a thermal limits violation. However, if this time delay is set above the minimum value, it is considered a bypass of the associated RBM channel since the analyses did not consider time delays in excess of the minimum value. The licensee has stated that testing and calibration of the time delay will be performed at each refueling as part of the RBM system calibration procedure. The licensee's supporting document, Reference 6, indicates that setting of t_{d2} above the minimum could be used as a means for bypassing the RBM. We have taken exception to this provision, and the licensee has stated that RBM channel bypass will be affected using only the RBM bypass switch. We consider this to be acceptable procedure.

Other RBM trip functions include too few LPRM inputs (either inoperative or bypassed), downscale (RBM signal abnormally low), and instrument inoperative (e.g., calibrate-operate switch not in the operate position and RBM equipment interlocks such as module removed and failure to null to the reference signal). The licensee has stated that the response time and accuracy (including setpoint drift) of the new RBM circuitry either equals or exceeds that of the existing design. All rod blocks are alarmed. The upscale rod block alarm can only be reset by activating a reset switch or selecting another rod for movement. Locally mounted color coded status lights are provided to indicate the type of rod block (upscale-amber, instrument inspective and downscale-white).

The RBM system is required to be operable whenever a limiting rod pattern exists. A limiting rod pattern exists when any control rod in the core would result in violation of the safety limit MCPR if it were fully withdrawn. During operation with a limiting rod pattern, both RBM channels should be operable. If only one RBM channel is operable, an instrument functional test of the operable (unbypassed) channel must be performed prior to withdrawal of any control rods. If the inoperable channel is not restored within 24 hours, then all control rod withdrawal shall be blocked. If both RBM channels are inoperable, then all control rod withdrawal shall be blocked within an hour. We find the Hatch Technical Specification requirements for RBM system operability and the associated LCOs to be acceptable. It should be noted that the operators are responsible for determining whether a limiting rod pattern exists (and therefore, for determining RBM system operability requirements) prior to control rod withdrawal in accordance with plant operating procedures. We have found this to be acceptable. The APRM and RBM instrument surveillance requirements (i.e., instrument functional tests and calibrations) have not changed as a result of implementation of the ARTS improvement program.

2.6.4 Technical Specification Changes

Implementation of the hardware changes and revised analyses described above requires changes in the Hatch Nuclear Plant, Units 1 and 2, Technical Specifications. These changes are discussed below:

2.6.4.1 APRM Specification Changes

The requirement for the setdown of the trip setpoint is deleted from the specification and the setdown factor (Fraction of Rated Power divided by Core Maximum Fraction of Limiting Power Density) is removed from the equation for the trip setpoint. The slope and intercept of the APRM flow biased rod block line and of the APRM/STPM flow biased scram are altered to permit operation within the domain defined by the extended load line limit analysis.

2.6.4.2 Rod Block Monitor Technical Specifications

The RBM flow biased trip equation is replaced by power dependent setpoint definitions and incorporate RBM filter and time delay setpoints. Current operability requirements are replaced by the new ones including the revised definition of the limiting control rod pattern.

2.6.4.3 Thermal-Hydraulic Operating Limit Specifications

The following changes are required in the Power Distribution Limit Specifications:

1. A curve of MCPR multiplier, K_p, as a function of power must be added.

2. The K_f family of curves must be replaced with curves of $MCPR_f$ as a function of flow.

3. The MCPR Technical Specification must be altered to define the manner in which the two curves are combined with the full power, full flow value of the operating limit MCPR to obtain the power/flow dependent limit.

4. Power and flow dependent multiplier factors ($MAPFAC_p$ and $MAPFAC_f$) must be added and the MAPLHGR Technical Specification must be altered to define the manner in which the two curves are combined with the full power/full flow MAPLHGR curves to obtain the power and flow dependent MAPLHGR limits.

5. The bases for the various Technical Specifications must be modified to account for the altered Technical Specifications.

2.6.5 Conclusions

Based on our review, which is described above, we conclude that the proposed ARTS Improvement Program is acceptable for use in Hatch Nuclear Plant, Units 1 and 2. We further conclude that the supporting document, NEDC-30474-P (Reference 6), may be used as a reference to describe the program and its analyses for Hatch 1 and 2, and to describe the methods used in applications of this program to other reactors. This conclusion is based on the following:

1. The analysis methods used for the safety analyses presented in the report are those which have been previously used and approved for reload safety analyses.

2. The revised operating limits and procedures do not result in reductions to safety margins relative to current values. In general, margins are increased.

3. The revised operating procedures are simpler to follow which tends to increase operating safety.

4. Implementation of this design complies with the requirements of Section 7.7 (Control Systems) of the Standard Review Plan (NUREG-0800), and therefore, is acceptable. The separation provided between redundant RBM channels and the isolation provided between the RBM system and safety related circuits have not been compromised as a result of the ARTS modification.

3.0 ENVIRONMENTAL CONSIDERATION

The reload and ATTS portions of this amendment involve changes in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. We have determined that they involve no significant increase in the amounts, and no significant change in the types of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that these items involve no significant hazards consideration and there has been no public comment on such finding. Accordingly, these items of the amendment meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of these items of the amendment.

An Environmental Assessment and Final Finding of No Significant Impact has been issued for the ARTS portion of the amendment.

4.0 CONCLUSION

We have concluded, based on the considerations discussed above, that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: July 13, 1984

The following NRC personnel have contributed to this Safety Evaluation: Jerry Mauck, Marty Virgilio, Rick Kendall, Walter Brooks, and S. Sun.

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U.S. NUCLEAR REGULATORY COMMISSION

GEORGIA POWER COMPANY
OGLETHORPE POWER CORPORATION
MUNICIPAL ELECTRIC AUTHORITY OF GEORGIA
CITY OF DALTON, GEORGIA

DOCKET NO. 50-366

NOTICE OF ISSUANCE OF AMENDMENT TO

FACILITY OPERATING LICENSE

The U.S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 39 to Facility Operating License No. NPF-5, issued to Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, and City of Dalton, Georgia (the licensees), which revised the Technical Specifications for operation of the Edwin I. Hatch Nuclear Plant, Unit No. 2, (the facility) located in Appling County, Georgia. The amendment was effective as of the date of its issuance.

This amendment revised the Technical Specifications to implement the Average Power Range Monitor/Rod Block Monitor/Technical Specification (ARTS) Improvement Program. This amendment relates to Unit 2 only. The remaining request on Unit 1 will be acted upon at a later date. This amendment also made other revisions to the TSs which are being separately noticed.

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

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Notice of Consideration of Issuance of Amendment and Opportunity for Prior Hearing in connection with this action was published in the FEDERAL REGISTER on May 16, 1984, 48 FR 20769. No request for a hearing or petition for leave to intervene was filed following this notice. Subsequent to this notice, the licensees submitted correspondence dated June 20 and 27, 1984. This correspondence did not alter the substance of the licensees' request, but was provided as confirmatory documentation of our understanding.

Also, in connection with this action, the Commission prepared an Environmental Assessment and Final Finding of No Significant Impact which was published in the FEDERAL REGISTER on July 12, 1984 (49 FR 28487).

For further details with respect to this action, see (1) the application for amendment dated February 6, 1984, as supplemented April 3, 1984, June 20 and 27, 1984, (2) Amendment No. 39 to License No. NPF-5, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C. 20555, and at the Appling County Public Library, 301 City Hall Drive, Baxley, Georgia. A copy of items (2) and (3) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland, this 13th day of July 1984.

FOR THE NUCLEAR REGULATORY COMMISSION

George W. Rivenbark

George W. Rivenbark, Acting Chief
Operating Reactors Branch #4
Division of Licensing