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Distribution
Docket
ORB #3
NRR Reading
Local PDR
NRC PDR
HDenton
DEisenhut
RTedesco
WGammill
RVollmer
LShao
JMiller
BGrimes
Ippolito
SNorris
JHannon
Atty, OELD
OI&E (5)
BJones (8)
BScharf (10)
ACRS (16)
RDiggs
JWetmore
NSIC
TERA
RDiggs
CMiles

APRIL 17 1980

Docket Nos. 50-321
and 50-366

Mr. R. J. Kelly, Vice President
and General Manager of Power Generation
Georgia Power Company
P. O. Box 4545
Atlanta, Georgia 30302

Dear Mr. Kelly:

The Commission has issued the enclosed Amendment No. 73 to Facility Operating License No. DPR-57 and Amendment No. 14 to Facility Operating License No. NPF-5 for the Edwin I. Hatch Nuclear Plant, Units 1 and 2, respectively. The amendments consist of changes to the operating licenses and Technical Specifications in response to your applications dated October 18, 1978, June 5, 1979, November 8, 1979, and February 28, 1980.

The amendments (1) substitute equivalent terminology for computation of Average Power Range Monitor (APRM) Rod Block and Scram setpoints with revised surveillance requirements; (2) amends the license to correctly identify the co-owners of Hatch Nuclear Plant; and (3) permits modification of the APRM trip system by incorporating a Thermal Power Monitor for Hatch Nuclear Plant Unit 2 (previously accomplished on Unit 1 by Amendment 69 to DPR-57). Miscellaneous editorial changes were made in the Hatch Unit 2 TS to bring them into conformance with current GE STS.

Copies of the Safety Evaluation and Notice of Issuance are also enclosed.

Sincerely,

V Rooney
for Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Enclosures:
1. Amendment No. to DPR-57
2. Amendment No. to NPF-5
3. Safety Evaluation
4. Notice

cc w/encls: See next page

OFFICE	DOR:ORB#3	DOR:ORB#3	DOR:ORP	OELD	DOR:ORB#3	
SURNAME	SNorris	JHannon:ms	WPGammill		TAIppolito	
DATE	4/2/80	4/ /80	4/ /80	4/ /80	4/ /80	

cc:

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

8005130064

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

DOCKET NO. 50-321

EDWIN I. HATCH NUCLEAR PLANT UNIT NO. 1
AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 73
License No. DPR-57

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

   A. The applications for amendment by Georgia Power Company, et al., (the licensee) dated October 18, 1978 and November 8, 1979, 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 paragraphs 2, 2.A, 2.B and 2.C.(2) of Facility Operating License No. DPR-57 are hereby amended to read as follows:

2. Facility Operating License No. DPR-57 is hereby issued to the Georgia Power Company, the Oglethorpe Power Corporation, the Municipal Electric Authority of Georgia and the City of Dalton, Georgia to read as follows:

   A. This license applies to the Edwin I. Hatch Nuclear Plant Unit No. 1, a direct cycle boiling water reactor and associated equipment (the facility), owned by the Georgia Power Company, the Oglethorpe Power Corporation, The Municipal Electric Authority of Georgia and the City of Dalton, Georgia. The facility is located eleven miles north of Baxley in Appling County, Georgia, and is described in the 'Final Safety Analysis Report' as supplemented and amended (Amendments 9 through 46) and the Environmental Report as supplemented and amended (Supplement 1 and Amendment 1).

   B. Subject to the conditions and requirements incorporated herein, the Commission hereby licenses:

      (1) Pursuant to Section 104b of the Act and 10 CFR Part 50, 'Licensing of Production and Utilization Facilities,' Georgia Power Company to possess, use, and operate the facility at the designated location in Appling County, Georgia, in accordance with the procedures and limitations set forth in this license; and the Georgia Power Company, the Oglethorpe Power Corporation, The Municipal Electric Authority of Georgia and the City of Dalton, Georgia to possess the facility in accordance with the procedures and limitations set forth in this license;

   C. (2) Technical Specifications

      The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 73 , 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 the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 17, 1980

ATTACHMENT TO LICENSE AMENDMENT NO. 73

FACILITY OPERATING LICENSE NO. DPR-57

DOCKET NO. 50-321

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.

Remove	Insert
1.0-7	1.0-7
1.1-2	1.1-2
1.1-3	1.1-3
1.1-13	1.1-13
3.1-1	3.1-1
3.1-2	3.1-2
3.1-17/18	3.1-17
	3.1-18*
3.11-4	3.11-4

*Overleaf page provided for convenience only.

MM. Minimum Critical Power Ratio (MCPR) - Minimum Critical Power Ratio (MCPR) is the value of the critical power ratio associated with the most limiting assembly in the reactor core. Critical Power Ratio (CPR) is the ratio of that power in a fuel assembly, which is calculated to cause some point in the assembly to experience boiling transition, to the actual assembly operating power.

NN. Trip System - A trip system means an arrangement of instrument channel trip signals and auxiliary equipment required to initiate action to accomplish a protective function. A trip system may require one or more instrument channel trip signals related to one or more plant parameters in order to initiate trip system action. Initiation of protective action may require the tripping of a single trip system or the coincident tripping of two trip systems.

OO. Cumulative Downtime - The cumulative downtime for those safety components and systems whose downtime is limited to 7 consecutive days prior to requiring reactor shutdown shall be limited to any 7 days in a consecutive 30 day period.

PP. Fire Suppression Water System - A Fire Suppression Water System shall consist of: water storage tanks, pumps, and distribution piping with associated sectionalizing control or isolation valves. Such valves include yard hydrant curb valves, and the first valve ahead of the water flow alarm device on each sprinkler, hose stand pipe or spray system riser.

QQ. Channel Calibration - A Channel Calibration is the adjustment, as necessary, of the channel output such that it responds with the necessary range and accuracy to known values of the parameter which the channel monitors. The Channel Calibration shall encompass the entire channel including the sensor and alarm and/or trip functions, and shall include the Channel Functional Test. The Channel Calibration may be performed by any series of sequential, overlapping or total channel steps such that the entire channel is calibrated.

RR. Channel Functional Test - A Channel Functional Test shall be:

a. Analog channels - the injection of a simulated signal into the channel as close to the primary sensor as practicable to verify operability including alarm and/or trip functions.

b. Bistable Channels - the injection of a simulated signal into the channel sensor to verify operability including alarm and/or trip functions.

SS. Fraction of Limiting Power Density (FLPD) - the ratio of the linear heat generation rate (LHGR) existing at a given location to the design LHGR for the bundle type. Design LHGRs are 18.5 KW/ft for 7x7 bundles and 13.4 KW/ft for 8x8 bundles.

TT. Core Maximum Fraction of Limiting Power Density (CMFLPD) - the CMFLPD is the highest value existing in the core of the FLPD.

Amendment No. ~~18~~, ~~30~~, 73

1.1.D. Reactor Water Level (Hot or Cold Shutdown Condition)

Whenever the reactor is in the Hot or Cold Shutdown Condition with irradiated fuel in the reactor vessel, the water level shall be > 378 inches above vessel invert when fuel is seated in the core.

2.1.A.1.c. APRM High High Flux Scram Trip Setting (Run Mode) (Continued)

$$S \leq 0.66\ W + 54\%$$

where:

S = Setting in percent of rated thermal power (2436 MWt)

W = Loop recirculation flow rate in percent of rated (rated loop recirculation flow rate equals 34.2×10^6 lb/hr)

In the event of operation with a core maximum fraction of limiting power density (CMFLPD) greater than the fraction of rated core thermal power $\left(\frac{\text{Core MW Thermal}}{2436}\right)$, the APRM gain shall be adjusted up to 95% of rated thermal power as follows:

$$\text{APRM Reading} \geq 100\% \times \text{CMFLPD}$$

Provided that the adjusted APRM reading does not exceed 100% of rated thermal power and the required gain adjustment increment does not exceed 10% of rated thermal power.

For no combination of loop recirculation flow rate and core thermal power shall the APRM flux scram trip setting be allowed to exceed 120% of rated thermal power.

Surveillance requirements for CMFLPD are given in Specification 4.1.B.

Amendment No. 27, 42, 52, 58, 69, 73

2.1.A.1.d. APRM Rod Block Trip Setting

The APRM rod block trip setting shall be:

$$S_{RB} \leq 0.66\ W + 42\%$$

where:

S_{RB} = Rod block setting in percent of rated thermal power (2436 MWt)

W = Loop recirculation flow rate in percent of rated (rated loop recirculation flow rate equals 34.2×10^6 lb/hr)

In the event of operation with a core maximum fraction of limiting power density (CMFLPD) greater than the fraction of rated core thermal power ($\frac{\text{Core MW Thermal}}{2436}$), the APRM gain shall be adjusted up to 95% of rated thermal power as follows:

$$\text{APRM Reading} \geq 100\% \times \text{CMFLPD}$$

Provided that the adjusted APRM reading does not exceed 100% of rated thermal power and the required gain adjustment increment does not exceed 10% of rated thermal power.

2. Reactor Water Low Level Scram Trip Setting (LL1)

Reactor water low level scram trip setting (LL1) shall be $\geq$12.5 inches (narrow range scale).

3. Turbine Stop Valve Closure Scram

Turbine stop valve closure scram trip setting shall be $\leq$ 10 percent valve closure from full open. This scram is only effective when turbine steam flow is above 30% of rated, as measured by turbine first stage pressure.

Amendment No. 19, 42, 58, 73

APRM Flux Scram Trip Settings (Run Mode) (Continued)

The APRM flow referenced scram trip setting at full recirculation flow is adjustable up to 120% of rated power. This reduced flow referenced trip setpoint will result in an earlier scram during slow thermal transients, such as the loss of 80°F feedwater heating event, than would result with the 120% fixed high 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 cycle.

The APRM fixed high 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 scram.

The APRM reading must be adjusted to ensure that the LHGR transient peak is not increased for any combination of CMFLPD and reactor core thermal power. The APRM reading is adjusted in accordance with the formula in Specification 2.1.A.1.c., when the CMFLPD is greater than the fraction of rated core thermal power.

Analyses of the limiting transients show that no scram adjustment is required to assure MCPR > 1.07 when the transient is initiated from the operating MCPR limit.

d. APRM Rod Block Trip Setting

Reactor power level may be varied by moving control rods or by varying the recirculation flow rate. The APRM system provides a control rod block to prevent rod withdrawal beyond a given point at constant recirculation flow rate, and thus to protect against the condition of a MCPR less than 1.07. This rod block trip setting, which is automatically varied with recirculation loop flow rate, prevents an increase in the reactor power level to excessive values due to control rod withdrawal. The flow variable trip setting provides substantial margin from fuel damage, assuming a steady-state operation at the trip setting, over the entire recirculation flow range. The margin to the Safety Limit increases as the flow decreases for the specified trip setting versus flow relationship; therefore, the worst case MCPR which would occur during a steady-state operation is at 108% of rated thermal power because of the APRM rod block trip setting. The actual power distribution in the core is established by specified control rod sequences and is monitored continuously by the in-core LPRM system. The APRM reading is adjusted to compensate for a CMFLPD greater than the fraction of rated core thermal power.

2. Reactor Water Low Level Scram Trip Setting (LLL)

The trip setting for low level scram is above the bottom of the separator skirt. This level is > 14 feet above the top of the active fuel. This level has been used in transient analyses dealing with coolant inventory decrease. The results reported in FSAR Section 14.3 show that a scram at this level adequately protects the fuel and the pressure barrier. The scram trip setting is approximately 33 inches below the normal operating range and is thus adequate to avoid spurious scrams.

3.1 REACTOR PROTECTION SYSTEM (RPS)

Applicability

The Limiting Conditions for Operation associated with the Reactor Protection System apply to the instrumentation and associated devices which initiate a reactor scram.

Objective

The objective of the Limiting Conditions for Operation is to assure the operability of the Reactor Protection System.

Specifications

A. Sources of a Trip Signal Which Initiate a Reactor Scram

The instrumentation requirements associated with each source of a scram signal shall be as given in Table 3.1-1.

The action to be taken if the number of operable channels is not met for both trip systems is also given in Table 3.1-1.

B. Core Maximum Fraction of Limiting Power Density (CMFLPD)

If at any time during operation it is determined by normal surveillance that the APRM readings are not in compliance with sections 2.1.A.1.c and 2.1.A.1.d, action will be initiated within 15 minutes to restore operation to within the prescribed limits. If CMFLPD is not reduced to comply with the above sections within 2 hours, the APRM readings shall be adjusted to comply with the existing CMFLPD according to Specifications 2.1.A.1.c and 2.1.A.1.d, or reduce thermal power to < 25% within the next 4 hours.

4.1 REACTOR PROTECTION SYSTEM (RPS)

Applicability

The Surveillance Requirements associated with the Reactor Protection System apply to the instrumentation and associated devices which initiate a reactor scram.

Objective

The objective of the Surveillance Requirements is to specify the type and frequency of surveillance to be applied to the protection instrumentation to assure operability.

Specifications

A. Test and Calibration Requirements for the RPS

RPS instrumentation systems and associated systems shall be functionally tested and calibrated as indicated in Table 4.1-1.

When it is determined that a channel has failed in the unsafe condition, the other RPS channels that monitor the same variable shall be functionally tested immediately before the trip system containing the failure is tripped. The trip system containing the unsafe failure may be placed in the untripped condition during the period in which surveillance testing is being performed on the other RPS channels.

B. Core Maximum Fraction of Limiting Power Density (CMFLPD)

The CMFLPD shall be determined daily during reactor power operation $\geq$ 25% and the APRM readings shall be adjusted if necessary as required by Specifications 2.1.A.1.c and 2.1.A.1.d.

3.1.C RPS Response Time

The system response time from the opening of the sensor contact up to and including the opening of the trip actuator contacts shall not exceed 50 milliseconds.

4.1.A: Test and Calibration Requirements for the RPS (Continued)

Group C devices are active only during a given portion of the operational cycle. For example, the IRM is active during startup and inactive during full-power operation. Thus, the only test that is meaningful is the one performed just prior to shutdown or startup; i.e., the tests that are performed just prior to use of the instrument.

Calibration frequency of the instrument channel is divided into two categories: These are as follows:

i. Passive type indicating devices that can be compared with like units on a continuous reference.

ii. Vacuum tube or semiconductor devides and detectors that drift or lose sensitivity.

Experience with passive type instruments in generating stations and substations indicates that the specified calibrations are adequate. For those devices which employ amplifiers, etc., drift specifications call for drift to be less than 0.4%/month; i.e., in the period of a month a drift of .4% could occur and still provide for adequate margin. For the APRM system, drift of electronic apparatus is not the only consideration in determining a calibration frequency. Change in power distribution and loss of chamber sensitivity dictate a calibration every seven (7) days. Calibration on this frequency assures plant operation at or below thermal limits.

The sensitivity of LPRM detectors decreases with exposure to neutron flux at a slow and approximately constant rate. This is compensated for in the APRM system by calibrating twice a week using heat balance data and by calibrating individual LPRM's every 1000 effective full power hours using TIP traverse data.

B. Maximum Fraction of Limiting Power Density (MFLPD)

Since changes due to burnup are slow, and only a few control rods are moved daily, a daily check of the MFLPD is adequate. The determination of the MFLPD would establish whether or not adjustment of the APRM reading is required.

3.1.C. References

1. I. M. Jacobs, "Reliability of Engineered Safety Features as a Function of Testing Frequency," Nuclear Safety, Volume 9, No. 4, July-August, 1968, pp 303-312.

3.11.B. Linear Heat Generation Rate (LHGR)

This specification assures that the linear heat generation rate in any rod is less than the design linear heat generation if fuel pellet densification is postulated. The power spike penalty specified for 7 x 7 fuel is based on the analysis presented in Section 3.2.1 of Reference 4 and References 5 and 6, and assumes a linearly increasing variation in axial gaps between core bottom and top, and assures with a 95% confidence, that no more than one fuel rod exceeds the design linear heat generation rate due to power spiking. The LHGR as a function of core height shall be checked daily during reactor operation at ≥ 25% power to determine if fuel burnup, or control rod movement has caused changes in power distribution. For LHGR to be a limiting value below 25% rated thermal power, the ratio of peak LHGR to core average LHGR would have to be greater than 9.6, which is precluded by a considerable margin when employing any permissible control rod pattern.

C. Minimum Critical Power Ratio (MCPR)

The required operating limit MCPR as specified in Specification 3.11.C is derived from the established fuel cladding integrity Safety Limit MCPR of 1.07 and an analysis of abnormal operational transients presented in Reference 7.

Various transient events will reduce the MCPR below the operating MCPR. To assure that the fuel cladding integrity safety limit (MCPR of 1.07) is not violated during anticipated abnormal operational transients, the most limiting transients have been analyzed to determine which one results in the largest reduction in critical power ratio (Δ MCPR). Addition of the largest Δ MCPR to the safety limit MCPR gives the minimum operating limit MCPR to avoid violation of the safety limit should the most limiting transient occur. The type of transients evaluated were loss of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease.

The evaluation of a given transient begins with the system initial parameters shown in Table 6-2 of Reference 9 that are input to a GE core dynamic behavior transient computer program described in Reference 8. Also, the void reactivity coefficients that were input to the transient calculational procedure are based on a new method of calculation termed NEW 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 Reference 1. The principal result of this evaluation is the reduction in MCPR caused by the transient.

From BOC4 to EOC4, the most limiting transient for the 8 x 8R fuel is the loss of 100°F feedwater heating with a ΔCPR of 0.14. The most limiting event throughout cycle 4 for 8 x 8 and 7 x 7 fuel is the Rod Withdrawal Error (RWE) with a ΔCPR of 0.17 for 8 x 8 and 0.19 for 7 x 7. Therefore, the MCPR's specified in 3.11.C are based on loss of 100°F feedwater heating and the Rod Withdrawal Error.

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. 14
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 June 5, 1979, November 8, 1979, and February 28, 1980, 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 paragraphs 1.A, 1.F, 2, 2.A, 2.B and 2.C.(2) of Facility Operating License No. DPR-57 are hereby amended to read as follows:

1. A. The application for license filed by Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, And the City of Dalton, Georgia (the licensees) complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I and all required notifications to other agencies or bodies have been duly made;

   F. Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, and the City of Dalton, Georgia are financially qualified to engage in the activities authorized by this operating license in accordance with the rules and regulations of the Commission;

2. Facility Operating License No. NPF-5 is hereby issued to Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, and the City of Dalton, Georgia to read as follows:

   A. The license applies to the Edwin I. Hatch Nuclear Plant, Unit No. 2, a boiling water reactor and associated equipment (the facility) owned by Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, and the City of Dalton, Georgia. The facility is located in Appling County, Georgia, and is described in the Final Safety Analysis Report as supplemented and amended (Amendments 18 through 45) and the Environmental Report as supplemented and amended (Supplements 1 and 2 and Amendment 1).

   B. Subject to the conditions and requirements incorporated herein, the Commission hereby licenses Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, and the City of Dalton, Georgia:

   C. (2) Technical Specifications

      The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 14 , 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 the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 17, 1980

ATTACHMENT TO LICENSE AMENDMENT NO. 14

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.

Remove	Insert
I	I
II	II
1-1	1-1
1-2	1-2
1-3	1-3
1-4	1-4
1-5	1-6
2-3*	2-3*
2-4	2-4
2-5	2-5
2-6 (Deleted)	
B 2-9	B 2-9
B 2-10	B 2-10
B 2-11	B 2-11
B 2-12*	B2-12*
3/4 2-5	3/4 2-5
3/4 2-6*	3/4 2-6*
3/4 3-1*	3/4 3-1*
3/4 3-2	3/4 3-2
3/4 3-5*	3/4 3-5*
3/4 3-6	3/4 3-6
3/4 3-7	3/4 3-7
3/4 3-8	3/4 3-8
3/4 3-37*	3/4 3-37*
3/4 3-38	3/4 3-38
3/4 3-39*	3/4 3-39*
3/4 3-40	3/4 3-40
3/4 3-41	3/4 3-41
3/4 3-42*	3/4 3-42*
B 3/4 2-3	B 3/4 2-3
B 3/4 2-4*	B 3/4 2-4*

*Overleaf page provided for convenience only.

SECTION

1.0 DEFINITIONS

SECTION

DEFINITIONS (Continued) PAGE

The following terms are defined so that uniform interpretation of these specifications may be achieved. The defined terms appear in capitalized type and shall be applicable throughout these Technical Specifications.

ACTION

ACTIONS shall be those additional requirements specified as corollary statements to each specification and shall be part of the specifications.

AVERAGE PLANAR EXPOSURE

The AVERAGE PLANAR EXPOSURE shall be applicable to a specific planar height and is equal to the sum of the exposure of all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle.

AVERAGE PLANAR LINEAR HEAT GENERATION RATE

The AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) shall be applicable to a specific planar height and is equal to the sum of the LINEAR HEAT GENERATION RATES for all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle.

CHANNEL CALIBRATION

A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds with the necessary range and accuracy to known values of the parameter which the channel monitors. The CHANNEL CALIBRATION shall encompass the entire channel including the sensor and alarm and/or trip functions, and shall include the CHANNEL FUNCTIONAL TEST. The CHANNEL CALIBRATION may be performed by any series of sequential, overlapping or total channel steps such that the entire channel is calibrated.

CHANNEL CHECK

A CHANNEL CHECK shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrument channels measuring the same parameter.

CHANNEL FUNCTIONAL TEST

A CHANNEL FUNCTIONAL TEST shall be:

a. Analog channels - the injection of a simulated signal into the channel as close to the primary sensor as practicable to verify OPERABILITY including alarm and/or trip functions and channel failure trips.

b. Bistable channels - the injection of a simulated signal into the channel sensor to verify OPERABILITY including alarm and/or trip functions.

CORE ALTERATION

CORE ALTERATION shall be the addition, removal, relocation or movement of fuel, sources, incore instruments or reactivity controls within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of the movement of a component to a safe conservative position.

CORE MAXIMUM FRACTION OF LIMITING POWER DENSITY

The CORE MAXIMUM FRACTION OF LIMITING POWER DENSITY (CMFLPD) shall be the largest FLPD which exists in the core for a given operating condition.

CRITICAL POWER RATIO

The CRITICAL POWER RATIO (CPR) shall be the ratio of that power in the assembly which is calculated by application of the GEXL correlation to cause some point in the assembly to experience boiling transition, divided by the actual assembly operating power.

DOSE EQUIVALENT I-131

DOSE EQUIVALENT I-131 shall be that concentration of I-131, μCi/gram, which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites."

$\bar{E}$-AVERAGE DISINTEGRATION ENERGY

$\bar{E}$ shall be the average, weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling, of the sum of the average beta and gamma energies per disintegration, in MeV, for isotopes with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME

The EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ECCS actuation setpoint at the channel sensor until the ECCS equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays where applicable.

FRACTION OF LIMITING POWER DENSITY

The FRACTION OF LIMITING POWER DENSITY (FLPD) shall be the ratio of local LHGR for any specific location on a fuel rod divided by the design LHGR.

FREQUENCY NOTATION

The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.1.

IDENTIFIED LEAKAGE

IDENTIFIED LEAKAGE shall be:

a. Leakage into collection systems, such as pump seal or valve packing leaks, that is captured and conducted to a sump or collecting tank, or

b. Leakage into the containment atmosphere from sources that is both specifically located and known either not to interfere with the operation of the leakage detection systems or not to be PRESSURE BOUNDARY LEAKAGE.

ISOLATION SYSTEM RESPONSE TIME

The ISOLATION SYSTEM RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its isolation actuation setpoint at the channel sensor until the isolation valves travel to their required positions. Times shall include diesel generator starting and sequence loading delays where applicable.

LIMITING CONTROL ROD PATTERN

A LIMITING CONTROL ROD PATTERN shall be a pattern which results in the core being on a thermal hydraulic limit, i.e., operating on a limiting value for APLHGR, LHGR, or MCPR.

LINEAR HEAT GENERATION RATE

LINEAR HEAT GENERATION RATE (LHGR) shall be the power generation in an arbitrary length of fuel rod, usually one foot. It is the integral of the heat flux over the heat transfer area associated with the unit length.

LOGIC SYSTEM FUNCTIONAL TEST

A LOGIC SYSTEM FUNCTIONAL TEST shall be a test of all relays and contacts of a logic circuit, from sensor to activated device, to ensure that components are OPERABLE per design requirements.

MINIMUM CRITICAL POWER RATIO

The MINIMUM CRITICAL POWER RATIO (MCPR) shall be the smallest CPR which exists in the core.

OPERABLE - OPERABILITY

A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s). Implicit in this definition shall be the assumption that all necessary attendant instrumentation, controls, normal and emergency electrical power sources, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s).

OPERATIONAL CONDITION

An OPERATIONAL CONDITION shall be any one inclusive combination of mode switch position and average reactor coolant temperature as indicated in Table 1.2.

PHYSICS TESTS

PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and 1) described in Chapter 14.0 of the FSAR, 2) authorized under the provisions of 10 CFR 50.59, or 3) otherwise approved by the Commission.

PRESSURE BOUNDARY LEAKAGE

PRESSURE BOUNDARY LEAKAGE shall be leakage through a non-isolable fault in a reactor coolant system component body, pipe wall or vessel wall.

PRIMARY CONTAINMENT INTEGRITY

PRIMARY CONTAINMENT INTEGRITY shall exist when:

a. All penetrations required to be closed during accident conditions are either:

   1. Capable of being closed by an OPERABLE containment automatic isolation valve system, or

   2. Closed by at least one manual valve, blind flange, or deactivated automatic valve secured in its closed position, except as provided in Table 3.6.3-1 of Specification, 3.6.3.1.

b. All equipment hatches are closed and sealed.

c. Each containment air lock is OPERABLE pursuant to Specification 3.6.1.3.

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

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

RATED THERMAL POWER

RATED THERMAL POWER shall be a total reactor core heat transfer to the reactor coolant of 2436 MWT.

REACTOR PROTECTION SYSTEM RESPONSE TIME

REACTOR PROTECTION SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until de-energization of the scram pilot valve solenoids.

REPORTABLE OCCURRENCE

A REPORTABLE OCCURRENCE shall be any of those conditions specified in Specification 6.9.1.8 and 6.9.1.9.

ROD DENSITY

ROD DENSITY shall be the number of control rod notches inserted as a fraction of the total number of control rod notches. All rods fully inserted is equivalent to 100% ROD DENSITY.

SECONDARY CONTAINMENT INTEGRITY

SECONDARY CONTAINMENT INTEGRITY shall exist when:

a. All secondary containment ventilation system automatic isolation dampers are OPERABLE or secured in the isolated position,

b. The Standby Gas Treatment System is OPERABLE pursuant to Specification 3.6.6.1.

c. At least one door in each access to the secondary containment is closed.

d. The sealing mechanism associated with each penetration (e.g., welds, bellows or O-rings) is OPERABLE.

SHUTDOWN MARGIN

SHUTDOWN MARGIN shall be the amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming all control rods are fully inserted except for the single control rod of highest reactivity worth which is assumed to be fully withdrawn and the reactor is in the shutdown condition; cold, i.e. 68°F; and xenon free.

STAGGERED TEST BASIS

STAGGERED TEST BASIS shall consist of:

a. A test schedule for n systems, subsystems, trains or other designated components obtained by dividing the specified test interval into n equal subintervals,

b. The testing of one system, subsystem, train or other designated components at the beginning of each subinterval.

THERMAL POWER

THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

UNIDENTIFIED LEAKAGE

UNIDENTIFIED LEAKAGE shall be all leakage which is not IDENTIFIED LEAKAGE.

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	≤ (0.66 W + 51%), with a maximum ≤ 113.5% of RATED THERMAL POWER	≤ (0.66 W + 54%), 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-N023 A,B,C,D)	≤ 1045 psig	≤ 1045 psig
4. Reactor Vessel Water Level - Low (2C21-N017 A,B,C,D)	≥ 12.5 inches above instrument zero*	≥ 12.5 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-N002A,B,C,D)	≤ 2 psig	≤ 2 psig

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

TABLE 2.2.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
8. Scram Discharge Volume Water Level - High (2C11-N013A,B,C,D)	≤ 57.15 gallons	≤ 57.15 gallons
9. Turbine Stop Valve - Closure (NA)	≤ 10% closed	≤ 10% closed
10. Turbine Control Valve Fast Closure, Trip Oil Pressure-Low (2C71-N005A,B,C,D)	≥ 600 psig	≥ 600 psig
11. Reactor Mode Switch in Shutdown Position (NA)	NA	NA
12. Manual Scram (NA)	NA	NA

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 1.06. 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. The flow referenced trip setpoint or APRM gain must be adjusted by the specified formula in Specification 3.2.2 in order to maintain these margins when the CMFLPD exceeds the FRTP.

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 30% of rated flow, 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

3/4.2.2 APRM SETPOINTS

LIMITING CONDITION FOR OPERATION

3.2.2 The APRM flow referenced simulated thermal power scram trip setpoint (S) and control rod block trip setpoint (S_{RB}) shall be established* according to the following relationships:

$$S \leq (0.66W + 51\%)$$

$$S_{RB} \leq (0.66W + 42\%)$$

where: S and S_{RB} are in percent of RATED THERMAL POWER, and
W = Loop recirculation flow in percent of rated flow.

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

ACTION:

With S or S_{RB} exceeding the allowable value, initiate corrective action within 15 minutes and continue corrective action so that S and S_{RB} are within the required limits* within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.2 The CMFLPD shall be determined and the APRM flow referenced simulated thermal power scram and control rod block trip setpoints or APRM readings adjusted, as required:

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 CMFLPD $\geq$ FRTP.

*With CORE MAXIMUM FRACTION OF LIMITING POWER DENSITY (CMFLPD) greater than the fraction of RATED THERMAL POWER (FRTP), $\frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$, up to 95% of RATED THERMAL POWER, rather than adjusting the APRM setpoints, the APRM gain may be adjusted such that APRM readings are greater than or equal to 100% times CMFLPD, provided that the adjusted APRM reading does not exceed 100% of RATED THERMAL POWER and the required gain adjustment increment does not exceed 10% of RATED THERMAL POWER.

3/4.2.3 MINIMUM CRITICAL POWER RATIO

LIMITING CONDITION FOR OPERATION

3.2.3 The MINIMUM CRITICAL POWER RATIO (MCPR), as a function of core flow, shall be equal to or greater than shown below x the K_f shown in Figure 3.2.3-1.

a. 1.30 up to 6,900 MWD/ton uranium, and

b. 1.34 from 6,900 MWD/ton uranium to the end of the first fuel cycle.

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

ACTION:

With MCPR less than the applicable limit determined from Figure 3.2.3-1, 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 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.3 MCPR shall be determined to be equal to or greater than the applicable limit determined from Figure 3.2.3-1:

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.

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-N023 A, B, C, D)	1, 2(e)	2(j, 2B21-N045-A, B, C, D)	5
4. Reactor Vessel Water Level - Low (2B21-N017 A, B, C, D)	1, 2	2(j, 2B21-N024-A, B and 2B21-N025-A, B)	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-N002 A, B, C, D)	1, 2(g)	2	5

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

ACTION 9 - In OPERATIONAL CONDITION 1 or 2, be in at least HOT SHUTDOWN within 6 hours.

In OPERATIONAL CONDITION 3 or 4, lock the reactor mode switch in the Shutdown position within one hour.

In OPERATIONAL CONDITION 5, suspend all operations involving CORE ALTERATIONS or positive reactivity changes and fully insert all insertable control rods within one hour.

TABLE NOTATIONS

a. 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 OPERABLE channel in the same trip system is monitoring that parameter.

b. The "shorting links" shall be removed from the RPS circuitry during CORE ALTERATIONS and shutdown margin demonstrations performed in accordance with Specification 3.10.3.

c. The IRM scrams are automatically bypassed when the reactor vessel mode switch is in the Run position and all APRM channels are OPERABLE and on scale.

d. An APRM channel is inoperable if there are less than 2 LPRM inputs per level or less than eleven LPRM inputs to an APRM channel.

e. These functions are not required to be OPERABLE when the reactor pressure vessel head is unbolted or removed.

f. This function is automatically bypassed when the reactor mode switch is in other than the Run position.

g. This function is not required to be OPERABLE when PRIMARY CONTAINMENT INTEGRITY is not required.

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

i. These functions are bypassed when turbine first stage pressure is <250* psig, equivalent to THERMAL POWER less than 30% of RATED THERMAL POWER.

j. Also trips reactor coolant system recirculation pump MG sets.

k. Also trips reactor coolant system recirculation pump motors.

* Initial setpoint. Final setpoint to be determined during startup testing.

TABLE 3.3.1-2

REACTOR PROTECTION SYSTEM RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME (Seconds)
1. Intermediate Range Monitors:	
a. Neutron Flux - High*	NA
b. Inoperative	NA
2. Average Power Range Monitor:*	
a. Neutron Flux - Upscale, 15%	NA
b. Flow Referenced Simulated Thermal Power - Upscale	≤ 0.09**
c. Fixed Neutron Flux - Upscale, 118%	≤ 0.09
d. Inoperative	NA
d. Inoperative	NA
e. Downscale	NA
f. LPRM	NA
3. Reactor Vessel Steam Dome Pressure - High	≤ 0.55
4. Reactor Vessel Water Level - Low	≤ 1.05
5. Main Steam Line Isolation Valve -Closure	≤ 0.06
6. Main Steam Line Radiation - High	NA
7. Drywell Pressure - High	NA
8. Scram Discharge Volume Water Level - High	NA
9. Turbine Stop Valve - Closure	≤ 0.06
10. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	≤ 0.08#
11. Reactor Mode Switch in Shutdown Position	NA
12. Manual Scram	NA

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

**Not including simulated thermal power time constant.

#Measured from start of turbine control valve fast closure.

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, 119%	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	NA	M	Q	1, 2
4. Reactor Vessel Water Level - Low	D	M	Q	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	NA	M	Q	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

FUNCTIONAL 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%. Any APRM channel gain adjustment made in compliance with Specification 3.2.2 shall not be included in determining the absolute difference.

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 APRM'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.

i. Instrument alignment using a standard current source.

j. Calibration using a standard radiation source.

3/4.3.5 CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.5 The control rod withdrawal block instrumentation shown in Table 3.3.5-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.5-2.

APPLICABILITY: As shown in Table 3.3.5-1.

ACTION:

a. With a control rod withdrawal block instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.5-2, declare the channel inoperable 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 any trip function, place that trip function in the tripped condition within one hour.

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

SURVEILLANCE REQUIREMENTS

4.3.5 Each of the above required control rod withdrawal block instrumentation channels 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 4.3.5-1.

TABLE 3.3.5-1

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION

TRIP FUNCTION	MINIMUM NUMBER OF OPERABLE CHANNELS PER TRIP FUNCTION	APPLICABLE OPERATIONAL CONDITIONS
1. APRM (2C51-K605 A, B, C, D, E, F)		
a. Flow Referenced Simulated Thermal Power - Upscale	4	1
b. Inoperative	4	1, 2, 5
c. Downscale	4	1
d. Neutron Flux - High, 12%	4	2, 5
2. ROD BLOCK MONITOR (2C51-K605 RBM A and B)		
a. Upscale	1	1[a]
b. Inoperative	1	1[a]
c. Downscale	1	1[a]
3. SOURCE RANGE MONITORS (2C51-K600 A, B, C, D)		
a. Detector not full in[b]	3	2
	2	5
b. Upscale[c]	3	2
	2	5
c. Inoperative[c]	3	2
	2	5
d. Downscale[b]	3	2
	2	5
4. INTERMEDIATE RANGE MONITORS[d] (2C51-K601 A, B, C, D, E, F, G, H)		
a. Detector not full in[e]	6	2, 5
b. Upscale	6	2, 5
c. Inoperative[e]	6	2, 5
d. Downscale[e]	6	2
5. SCRAM DISCHARGE VOLUME (2C11-N013E)		
a. Water Level-High	1	1, 2, 5[f]

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.

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	$\leq$ (0.66 W + 42%)*	$\leq$ (0.66 W + 42%)*
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	$\leq$ (0.66W + 41%)	$\leq$ (0.66 W + 41%)
b. Inoperative	NA	NA
c. Downscale	$\geq$ 3/125 of full scale	$\geq$ 3/125 of full scale
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
4. INTERMEDIATE RANGE MONITORS		
a. Detector not full in	NA	NA
b. Upscale	$\leq$ 108/125 of full scale	$\leq$ 108/125 of full scale
c. Inoperative	NA	NA
d. Downscale	$\geq$ 5/125 of full scale	$\geq$ 5/125 of full scale
5. SCRAM DISCHARGE VOLUME		
a. Water Level-High	$\leq$ 36.2 gallons	$\leq$ 36.2 gallons

* The Average Power Range Monitor rod block function is varied as a function of recirculation loop flow (W). The trip setting of this function must be maintained in accordance with Specification 3.2.2.

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.

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

3/4.2.2 APRM SETPOINTS

The fuel cladding integrity Safety Limits of Specification 2.1 were based on a power distribution which would yield the design LHGR at RATED THERMAL POWER. The scram setting and rod block functions of the APRM instruments or APRM readings must be adjusted to ensure that the MCPR does not become less than 1.0 in the degraded situation. The scram settings and rod block settings or APRM readings are adjusted in accordance with the formula in this specification when the combination of THERMAL POWER and CMFLPD indicates a higher peaked power distribution to ensure that an LHGR transient would not be increased in the degraded condition.

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 1.06, 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 is not exceeded during any anticipated abnormal operational transient, the most limiting transients have been analyzed to determine which result 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.

The limiting transient which determines the required steady state MCPR limit is the load rejection trip with failure of the turbine bypass. This transient yields the largest Δ MCPR. When added to the Safety Limit MCPR of 1.06 the required minimum operating limit MCPR of Specification 3.2.3 is obtained.

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 K_f factor is to define operating limits at other than rated flow conditions. At less than 100% of rated flow the required MCPR is the product of the operating limit MCPR and the K_f factor. Specifically, the K_f factor provides the required thermal margin to protect against a flow increase transient. The most limiting transient initiated from less than rated flow conditions is the recirculation pump speed up caused by a motor-generator speed control failure.

For operation in the automatic flow control mode, the K_f factors assure that the operating limit MCPR of Specification 3.2.3 will not be violated should the most limiting transient occur at less than rated flow. In the manual flow control mode, the K_f factors assure that the Safety Limit MCPR will not be violated should the most limiting transient occur at less than rated flow.

The K_f factor values shown in Figure 3.2.3-1 were developed generically and are applicable to all BWR/2, BWR/3 and BWR/4 reactors. The K_f factors were derived using the flow control line corresponding to RATED THERMAL POWER at rated core flow.

For the manual flow control mode, the K_f factors were calculated such that for the maximum flow rate, as limited by the pump scoop tube set point, and the corresponding THERMAL POWER along the rated 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 rated flow control line corresponding to different core flows. The ratio of the MCPR calculated at a given point of core flow, divided by the operating limit MCPR, determines the K_f.

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

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 73 TO FACILITY OPERATING LICENSE NO. DPR-57

AND AMENDMENT NO. 14 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 NOS. 1 & 2
DOCKET NOS. 50-321 AND 50-366

A. Average Power Range Monitor (APRM) Rod Block and Scram Setpoints

I. INTRODUCTION

By letter dated October 18, 1978, Georgia Power Company (licensee) requested an amendment to the Technical Specifications appended to Operating License No. DPR-57 for the Edwin I. Hatch Nuclear Plant Unit No. 1. The proposed amendment would (1) substitute equivalent terminology for computation of Average Power Range Monitor (APRM) Rod Block and Scram setpoints and (2) revise associated surveillance requirements.

By letter dated June 5, 1979, the licensee requested a similar amendment to the Technical Specifications appended to Operating License No. NPF-5 for the Edwin I. Hatch Nuclear Plant Unit No. 2. This amendment was requested to provide for consistent procedural usage between both units.

II. EVALUATION

(1) APRM Rod Block/Scram Setpoints

The current Technical Specifications require adjustment of the APRM rod block and scram setpoints in the event of operation with the maximum total peaking factor (MTPF) exceeding the design total peaking factor (DTPF). Under such conditions the setpoint is adjusted by the ratio DTPF/MTPF to ensure that the fuel cladding integrity limits are not exceeded during anticipated operational transients. The Hatch 1 core currently contains three types of fuel (7x7, 8x8 and 8x8R) each with different DTPFs. Therefore, three formulas are required to ensure compliance with the stated specification.

The licensee proposed to substitute the equivalent expression FRP/CMFLPD for DTPF/MTPF, where:

FRP is the fraction of rated power
CMFLPD is the Core Maximum Fraction of limiting power density.

Using this terminology, a single formula with a unique solution is obtained.

In the Unit 2 core, the assemblies are all the same length and contain the same number of rods. The principle reason for making the change for Unit 2 is to provide for consistent procedural usage between units.

The staff has previously reviewed and approved (Reference 1) the proposed terminology as being equivalent. The equivalency is explained as follows:

The DTPF can be expressed as the design linear heat generation rate divided by the plant rated thermal power per unit length of fuel rod. In a similar manner the MTPF can be expressed as the maximum linear heat generation rate divided by the plant operating power per unit length of fuel rod. The CMFLPD is defined as the highest value of the ratio of the linear heat generation rate existing at a given location to the design linear heat generation rate for the bundle type. From these definitions it is easily determined that the ratio DTPF/MTPF is the ratio of the design linear heat generation rate to the maximum linear heat generation rates times the fraction of rated thermal power, or 1/CMFLPD times FRP. Thus FRP/CMFLPD and DTPF/MTPF are equivalent.

Currently, APRM rod block and trip settings are adjusted through multiplication by the ratio of DTPF/MTPF. Such a reduction in set points is required in the event of operation with MTPF>DTPF. Instead of multiplying the APRM set points by FRP/CMFLPD the same result can be achieved by multiplying the APRM reading by CMFLPD/FRP to get a gain-adjusted APRM reading. If the reactor is operating in a steady state mode the APRM reading (before gain adjustment) is equal to FRP. Therefore by adjusting the gain until the APRM reading is equal to CMFLPD, the APRM reading has effectively been multiplied by CMFLPD/FRP as required.

(2) APRM Indicated Power Level

During normal operation the actual core peaking factors can exceed the design values. This variance is accommodated by reducing the APRM setpoints to retain the same desired "margin to trip." The "margin to trip" can be reduced either by lowering the trip set-point or by increasing the APRM indicated power level. From the viewpoint of equipment performance, these two methods are equivalent.

From the viewpoint of reactor operator performance, causing a power monitor to indicate a level that is different from the true power level introduces an additional source of possible operator confusion.

IEEE Standard 279 Section 4.20 specifies the design principle that indications that could be confusing to the reactor operator should be minimized. Because the APRM setpoints are flow-biased, recalibration involves re-adjusting several parameters with external test equipment. Our review of the calibration procedure and discussion with operating personnel indicates that re-calibration requires about one hour for each of the six APRM channels. The design does not lend itself to a simple re-calibration procedure.

The present APRM instrumentation channels do have gain adjustments to adjust the channel response toward a more accurate indication with respect to true power as determined by a heat balance. Routine use of the gain adjustment to cause the channels to read higher than actual is not consistent with the general design principles of IEEE 279, and could lead to operation in a non-conservative manner.

When full power and temperature equilibrium are attained, setpoint re-adjustments are very infrequent. During a reactor startup and approach to full power, the APRM setpoints may need to be re-adjusted several times. To require a lengthy six-hour calibration procedure several times during approach to full power is not reasonable nor in the best interest of safety.

We have determined that use of the APRM gain adjustments to maintain an adequate "margin to trip" during full power operation is not a justifiable deviation from the general principles.

However, within certain limitations, use of APRM gain adjustment can be allowed during an approach to full power. The limitations appropriate for allowing the gain to be used to maintain the APRM "margin to trip" are as follows: (1) gain adjustment should be used only when the reactor is less than 95% of rated power; (2) the magnitude of such adjustments should be less than 10% of rated power not to exceed 100% full power indicated; (3) any intentional inaccuracy of the APRM channels should be made obvious to the reactor operations staff at all times. Appropriate log entries are to be made for each such gain adjustment. Each affected APRM indicator should be marked in an obvious manner to identify the offset between true power and the power level indicated by the APRM instrumentation.

(3) Surveillance Requirements

The licensee proposed that the CMFLPD be determined daily during reactor power operation equal to or greater than 25%. The daily frequency for determining the CMFLPD is identical with the current specification for determining MTPF. However, the revised surveillance requirement would be applicable only above a specified power level. We have reviewed the licensee's submittal and determined that a power level of 25% above which the CMFLPD would be determined is acceptable. This is acceptable because below 25% power the ratio of the peak LHGR to core average LHGR would have to exceed 12 (for 8x8 fuel) in order for LGHR to be at limiting value. Such large peaking factors are highly improbable because permissible rod patterns preclude such large peaking factors. Based on the above and current licensing practices as set forth in Ref. 2, the licensee's proposal is acceptable.

III. CONCLUSIONS

We have determined that deviation from the prescribed APRM calibration procedures during full power operations is not consistent with the general principles of IEEE 279. However, during reactor startup and approach to full power, the APRM channel gain adjustments may be used, within limitations, to change the "margin to trip " The limitations involved are enumerated above.

To summarize, in the event of operation between 25% and 95% power with CMFLPD greater than FRP, the APRM gain shall be adjusted such that the APRM reading >100% times CMFLPD. This change does not involve a reduction in margin to the trip point and prescribes a more direct use of limits monitoring data from the plant process computer. In addition adjusting the APRM gain is much easier than changing the APRM trip setting, so that there is less chance for human error.

For operation above 95% power, with CMFLPD greater than FRP, APRM rod block and trip settings will continue to be adjusted by multiplication by the ratio of FRP/CMFLPD. The proposed Technical Specifications have been modified to be consistent with the above restraints. The necessary revisions were discussed with and agreed to by the licensee.

B. Oglethorpe Power Corporation Name Change

I. INTRODUCTION

By letter dated November 8, 1979 the licensee proposed an amendment to Operating Licenses DPR-57 and NPF-5, consisting of a name change in the operating licenses.

II. EVALUATION

On November 7, 1978, Oglethorpe Electric Membership Corporation changed its name to the Oglethorpe Power Corporation (OPC). Because OPC is a co-owner of Hatch Plant Units 1 and 2, it is necessary to amend the licenses to correctly identify the co-owners.

III. CONCLUSION

The change of name of a co-owner is an administrative action that is appropriate at this time. Therefore the licenses will be modified to correctly identify the co-owners in this amendment.

C. Hatch Unit 2 Thermal Power Monitor

I. INTRODUCTION

By letter dated February 28, 1980 the licensee proposed a change to the Plant Hatch Unit 2 Technical Specifications to modify the Average Power Range Monitor (APRM) high-high flux scram trip logic. The proposed amendment would replace the present APRM trip logic with circuitry to condition the APRM flux through a first order low pass filter that has a 6 second RC time constant. This circuit represents the fuel dynamics which will approximate the reactor thermal power during a transient or steady state condition. The intent of this modification is to avoid spurious scrams caused by momentary anomalous neutron flux spikes. This modification was previously licensed on Plant Hatch Unit No. 1 by Amendment No. 69 to Facility Operating License No. DPR-57.

II. EVALUATION

(1) Thermal Power Monitor

The Thermal Power Monitor (TPM), also called an APRM Simulated Thermal Power (STP) Trip in some documents, is a modification to the APRM trip system. The modified system generates two trips: a trip with a flow biased setpoint and a second trip with a setpoint fixed at 120% power. The flow biased setpoint is unchanged from that presently in the Technical Specifications. However, the TPM conditions the APRM output to apply a time constant of about six seconds, which is less than but comparable to the fuel thermal time constants (seven to ten seconds). Thus, the signal compared to the flow biased setpoint is a conservative simulation of fuel rod heat flux. This feature allows the plant operator to avoid spurious trips due to minor neutron flux overshoots when maneuvering the reactor.

The signal which is compared to the fixed 120% power setpoint is not modified. Thus, there is always a "fast scram" at 120% power in addition to the "heat flux" scram which may be below 120% power, depending on flow.

(2) Effect of TPM on Safety Analyses

Since all the transient analyses are done assuming full design flow, the TPM has no effect because the 120% "fast" trip is identical to the original system at full power. Any effect due to the TPM must be on analyses which are initiated from low flow conditions.

GE has addressed the analysis of the various transients initiated from low flow conditions on pp. 5-8 of Reference 3. The generic analyses described there show that only the idle recirculation pump startup, recirculation flow controller failure (increasing), feedwater controller failure (max demand), and rod withdrawal error can become more severe at low flow conditions. This is the basis for the flow-dependent multiplier (K_f) in every GE plant's Technical Specifications.

The analyses supporting the K_f factor did not take credit for the flow biasing, but instead conservatively assumed the trip to occur at 120% power. Therefore, the analyses supporting the flow-dependent multiplier (K_f) remain bounding.

Similarly, the various accident analyses which involve a neutron flux induced trip (e.g., rod drop accident) assume the trip to occur at 120% power regardless of initial power or flow conditions. Therefore, the validity of the accident analyses is also unaffected by the introduction of the TPM.

III. ENVIRONMENTAL CONSIDERATION

We have determined that the amendments do not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendments involve an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement, negative declaration, or environmental impact appraisal need not be prepared in connection with the issuance of the amendments.

IV. CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) because the amendments do not involve a significant increase in the probability or consequences of accidents previously considered and do not involve a significant decrease in a safety margin, the amendments do not involve a significant hazards consideration, (2) there is reasonable

assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

V. REFERENCES

(1) Safety Evaluation by NRR Supporting Amendment No. 35 to DPR-33, January 10, 1978, Docket No. 50-259
(2) NUREG-0123 Rev. 1, Standard Technical Specifications for General Electric Boiling Water Reactors, April 1, 1978
(3) General Electric Boiling Water Reactor Generic Reload Application, NEDE-24011-P-A, May 1979

Dated: April 17, 1980

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NOS. 50-321 AND 50-366

GEORGIA POWER COMPANY, ET AL.
NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment Nos. 73 and 14 to Facility Operating License Nos. DPR-57 and NPF-5 issued to Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, and City of Dalton, Georgia, which revised Technical Specifications for operation of the Edwin I. Hatch Nuclear Plant, Unit Nos. 1 and 2 (the facility) located in Appling County, Georgia. The amendments are effective as of the date of issuance.

The amendments (1) substitute equivalent terminology for computation of Average Power Range Monitor (APRM) Rod Block and Scram setpoints with revised surveillance requirements; (2) amend the license to correctly identify the co-owners of Hatch Nuclear Plant; and (3) permit modification of the APRM trip system by incorporating a Thermal Power Monitor for Hatch Nuclear Plant Unit 2 (previously accomplished on Unit 1 by Amendment 69 to DPR-57). Miscellaneous editorial changes were made in the Hatch Unit 2 TS to bring them into conformance with current General Electric Standard Technical Specifications.

The applications for the amendments comply 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 amendments. Prior public notice of these amendments was not required since the amendments do not involve a significant hazards consideration.

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

For further details with respect to this action, see (1) the applications for amendments dated October 18, 1978, June 5, 1979, November 8, 1979, and February 28, 1980, (2) Amendment Nos. 73 and 14 to License Nos. DPR-57 and 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, NW., Washington, D. C. and at the Appling County Public Library Parker Street, 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 Operating Reactors.

Dated at Bethesda, Maryland, this 17th day of April 1980.

FOR THE NUCLEAR REGULATORY COMMISSION

Vernon L. Rooney, Acting Chief
Operating Reactors Branch #3
Division of Operating Reactors