August 17, 1993

Docket Nos. 50-321
and 50-366

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Docket File
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G.Hill(4)
L.Cunningham
C.Gimes
ACRS(10)
OPA
OC/LFMB
OGC 15B18

Mr. J. T. Beckham, Jr.
Senior Vice President -
Nuclear Operations
Georgia Power Company
P. O. Box 1295
Birmingham, Alabama 35201

Dear Mr. Beckham:

SUBJECT: ISSUANCE OF AMENDMENTS - EDWIN I. HATCH NUCLEAR PLANT, UNITS 1 AND 2 (TAC NOS. M84786 AND M84787)

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 188 to Facility Operating License DPR-57 and Amendment No. 127 to Facility Operating License NPF-5 for the Edwin I. Hatch Nuclear Plant, Units 1 and 2. The amendments consist of changes to the Technical Specifications (TS) in response to your application dated October 19, 1992, as supplemented May 3 and July 27, 1993.

The amendments delete the main steam isolation valve closure, the reactor scram, and the control room pressurization functions of the main steam line radiation monitors.

A copy of the related Safety Evaluation is also enclosed. A Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

/s/
Kahtan N. Jabbour, Project Manager
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No.188 to DPR-57
2. Amendment No.127 to NPF-5
3. Safety Evaluation

cc w/enclosures:
See next page

*SEE PREVIOUS CONCURRENCE

OFFICE	PDII-3/LA	PDII-3/PM	OGC*	RPB/BC	PDII-3/D
NAME	L. BERRY	K. JABBOUR		L. CUNNINGHAM	D. MATTHEWS
DATE	8/5/93	8/5/93	7/8/93	8/12/93	8/12/93

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Senior Vice President -
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DATE	8/5/93	8/5/93	7/8/93	8/12/93	8/12/93

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

August 17, 1993

Docket Nos. 50-321
and 50-366

Mr. J. T. Beckham, Jr.
Senior Vice President -
Nuclear Operations
Georgia Power Company
P. O. Box 1295
Birmingham, Alabama 35201

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SUBJECT: ISSUANCE OF AMENDMENTS - EDWIN I. HATCH NUCLEAR PLANT, UNITS 1 AND 2 (TAC NOS. M84786 AND M84787)

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Kahtan N. Jabbour

Kahtan N. Jabbour, Project Manager
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 188 to DPR-57
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3. Safety Evaluation

cc w/enclosures:
See next page

Mr. J. T. Beckham, Jr.
Georgia Power Company

Edwin I. Hatch Nuclear Plant

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Chairman
Appling County Commissioners
County Courthouse
Baxley, Georgia 31513

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

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 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 188
License No. DPR-57

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

A. The application for amendment to the Edwin I. Hatch Nuclear Plant, Unit 1 (the facility) Facility Operating License No. DPR-57 filed by the Georgia Power Company, acting for itself, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, and City of Dalton, Georgia (the licensees), dated October 19, 1992, as supplemented May 3 and July 27, 1993, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as set forth in 10 CFR Chapter I;

B. The facility will operate in conformity with the application, 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 set forth in 10 CFR Chapter I;

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 hereby amended by page changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-57 is hereby amended to read as follows:

   Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 188 , 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 and shall be implemented no later than 60 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert Martin

for David B. Matthews, Director
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification
Changes

Date of Issuance: August 17, 1993

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

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 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 127
License No. NPF-5

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

   A. The application for amendment to the Edwin I. Hatch Nuclear Plant, Unit 2 (the facility) Facility Operating License No. NPF-5 filed by the Georgia Power Company, acting for itself, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, and City of Dalton, Georgia (the licensees), dated October 19, 1992, as supplemented May 3 and July 27, 1993, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, 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 set forth in 10 CFR Chapter I;

   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 hereby amended by page changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-5 is hereby amended to read as follows:

   Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 127 , 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 and shall be implemented no later than 60 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert Martin
David B. Matthews, Director
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification
Changes

Date of Issuance: August 17, 1993

ATTACHMENT TO LICENSE AMENDMENT NO. 188

FACILITY OPERATING LICENSE NO. DPR-57

DOCKET NO. 50-321

AND

TO LICENSE AMENDMENT NO. 127

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 areas of change.

	Remove Pages	Insert Pages
Unit 1	3.1-5	3.1-5
	3.1-6	3.1-6
	3.1-6a	3.1-6a
	3.1-8	3.1-8
	3.1-13	3.1-13
	3.2-3	3.2-3
	3.2-4	3.2-4
	3.2-19	3.2-19
	3.2-51	3.2-51
	3.2-66	3.2-66
	3.7-19	3.7-19
	3.12-4	3.12-4
Unit 2	2-4	2-4
	B 2-11	B 2-11
	3/4 3-2	3/4 3-2
	3/4 3-4	3/4 3-4
	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 3-11	3/4 3-11
	3/4 3-15	3/4 3-15
	3/4 3-15a	3/4 3-15a
	3/4 3-16	3/4 3-16
	3/4 7-8	3/4 7-8
	3/4 3-58a	3/4 3-58a
	3/4 3-58b	3/4 3-58b
	3/4 3-58c	3/4 3-58c
	3/4 3-58d	3/4 3-58d

Table 3.1-1 (Cont'd)

Scram Number (a)	Source of Scram Trip Signal	Operable Channels Required Per Trip System (b)	Scram Trip Setting	Source of Scram Signal is Required to be Operable Except as Indicated Below
8	APRM Downscale	2	≥3/125 of full scale	The APRM downscale trip is active only when the Mode Switch is in RUN. The APRM downscale trip is automatically bypassed when the IRM instrumentation is operable and not tripped.
	15% Flux	2	≤15/125 of full scale Tech Spec 2.1.A.1.b.	The APRM 15% Scram is automatically bypassed when the Mode Switch is in the RUN position.
9	(Deleted)			
10	Main Steam Line Isolation Valve Closure	4	≤10% valve closure from full open Tech Spec 2.1.A.5.	Automatically bypassed when the Mode Switch is not in the RUN position. The design permits closure of any two lines without a scram being initiated.
11	Turbine Control Valve Fast Closure	2	Within 30 milliseconds of the start of control valve fast closure Tech Spec 2.1.A.4.	Automatically bypassed when turbine steam flow is below that corresponding to 30% of rated thermal power as measured by turbine first stage pressure.

Table 3.1-1 (Cont'd)

Scram Number (a)	Source of Scram Trip Signal	Operable Channels Required Per Trip System (b)	Scram Trip Setting	Source of Scram Signal is Required to be Operable Except as Indicated Below
12	Turbine Stop Valve Closure	4	≤10% valve closure from full open Tech Spec 2.1.A.3.	Automatically bypassed when turbine steam flow is below that corresponding to 30% of rated thermal power as measured by turbine first stage pressure.

Notes for Table 3.1-1

a. The column entitled "Scram Number" is for convenience so that a one-to-one relationship can be established between items in Table 3.1-1 and items in Table 4.1-1.

b.1. There shall be two operable or tripped trip systems for each potential scram signal. If the number of operable channels cannot be met for one of the trip systems, the inoperable channel(s) or the associated trip system shall be tripped.

b.2. One instrument channel may be inoperable for up to 6 hours to perform required surveillances prior to entering other applicable actions, provided at least one operable channel in the same trip system is monitoring that parameter.

Notes for Table 3.1-1 (cont)

For SCRAMS 1 thru 7 and 8 APRM 15% Flux, if the number of operable channels is not met for both trip systems; initiate insertion of all control rods capable of being moved by control rod drive pressure and complete their insertion within four (4) hours.

For SCRAM 8 (APRM High Trips, Inoperative, and Downscale), if the number of operable channels is not met for both trip systems; initiate insertion of all control rods capable of being moved by control rod drive pressure and complete their insertion within four hours or reduce power to the IRM range and go to the START & HOT STANDBY position of the Mode Switch within eight hours.

For SCRAM 10, if the number of operable channels is not met for both trip systems; reduce turbine load and close main steam line isolation valves within eight hours or initiate insertion of all control rods capable of being moved by control rod drive pressure and complete their insertion within four hours.

For SCRAMS 11 and 12, if the number of operable channels is not met for both trip systems, reduce reactor power to 25% of rated thermal power or less within eight hours.

Table 4.1-1 (Cont.)

Scram Number (a)	Source of Scram Trip Signal	Group (b)	Instrument Check Minimum Frequency	Instrument Functional Test Minimum Frequency (c)	Instrument Calibration Minimum Frequency
9	(Deleted)				
10	Main Steam Line Isolation Valve Closure	A	NA	Every 3 months	(h)
11	Turbine Control Valve Fast Closure	A	NA	Every 3 months (j)	Once/Operating Cycle (k)
12	Turbine Stop Valve Closure	A	NA	Every 3 months	(h)
	RPS Channel Switch	A	NA	Once/Operating Cycle	Not Applicable
	Turbine First Stage Pressure Permissive	A	NA	Every 3 months	Every 6 months

a. The column entitled "Scram Number" is for convenience so that a one-to-one relationship can be established between items in Table 4.1-1 and items in Table 3.1-1.

b. The definition for each of the four groups is as follows:

Group A. On-off sensors that provide a scram trip signal.
Group B. Analog devices coupled with bi-stable trips that provide a scram trip signal.
Group C. Devices which only serve a useful function during some restricted mode of operation, such as startup or shutdown, or for which the only practical test is one that can be performed at shutdown.
Group D. Analog transmitters and trip units that provide a scram trip function.

c. Functional tests are not required when the systems are not required to be operable or are tripped. However, if functional tests are missed, they shall be performed prior to returning the systems to an operable status.

d. Calibrations are not required when the systems are not required to be operable or are tripped. However, if calibrations are missed, they shall be performed prior to returning the system to an operable status.

e. This instrumentation is exempted from the instrument functional test definition. This instrument functional test will consist of injecting a simulated electrical signal into the measurement channels.

f. Deleted

g. The water level in the reactor will be perturbed and the corresponding level indicator changes will be monitored. This perturbation test will be performed every 3 months after completion of the functional test program.

h. Physical inspection and actuation of these position switches will be performed once per operating cycle.

i. (Deleted)

j. Measure time interval from EHC pressure switch actuation to RPS relay K14 de-energization.

3.1.A.8.b. Inoperative

An APRM is inoperable if there are less than two LPRM inputs per level or there are less than 11 LPRM inputs to the APRM channel.

c. Downscale

The APRM downscale is active only when the Mode Switch is in the RUN position. The APRM downscale trip is automatically bypassed when the IRM instrumentation is operable and not tripped. Because of the APRM downscale limit of $\geq$3/125 of full scale when in the Run Mode and high level limit of $\leq$15/125 of full scale when the Start & Hot Standby Mode, the transition between the Start & Hot Standby and Run Modes must be made with the APRM instrumentation indicating between 3/125 and 15/125 of full scale or a control rod scram will occur. In addition, the IRM system must be indicating below the High High Flux setting (120/125 of full scale) or a scram will occur when in the Start & Hot Standby Mode. For normal operating conditions, these limits provide assurance of overlap between the IRM system and APRM system so that there are no "gaps" in the power level indications (i.e., the power level is continuously monitored from beginning of startup to full power and from full power to shutdown). When power is being reduced, if a transfer to the Start & Hot Standby Mode is made and the IRM's have not been fully inserted (a maloperational but not impossible condition) a control rod block immediately occurs so that reactivity insertion by control rod withdrawal cannot occur.

d. 15% Flux

The bases for the APRM 15% Flux Scram Trip Setting is discussed in the bases for Specification 2.1.A.1.b.

9. (Deleted)

10. Main Steam Line Isolation Valve Closure

The bases for the Main Steam Line Isolation Valve Closure Scram Trip Setting is discussed in the bases for Specification 2.1.A.5.

11. Turbine Control Valve Fast Closure

The bases for the Turbine Control Valve Fast Closure Scram Trip Setting is discussed in the bases for Specification 2.1.A.4.

12. Turbine Stop Valve Closure

The bases for the Turbine Stop Valve Closure Scram Trip Setting is discussed in the bases for Specification 2.1.A.3.

Table 3.2-1 (Cont.)

Ref. No. (a)	Instrument	Trip Condition Nomenclature	Required Operable Channels per Trip System (b)	Trip Setting	Action to be taken if number of channels is not met for both trip systems (c)	Remarks (d)
4	Main Steam Line Radiation	High	2	≤3 times normal full power back-ground(e)	Initiate closure of reactor water sample valves.	Initiates closure of reactor water sample valves B31-F019 and B31-F020.
5	Main Steam Line Pressure	Low	2	≥825 psig	Initiate an orderly load reduction and close MSIVs within 8 hours.	Initiates Group 1 isolation. Only required in RUN mode, therefore activated when Mode Switch is in RUN position.
6	Main Steam Line Flow	High	2	≤138% rated flow (≤115 psid)	Initiate an orderly load reduction and close MSIVs within 8 hours.	Initiates Group 1 isolation.
7	Main Steam Line Tunnel Temperature	High	2	≤194°F	Initiate an orderly load reduction and close MSIVs within 8 hours.	Initiates Group 1 isolation.
8	Reactor Water Cleanup System Differential Flow	High	1	20-80 gpm	Isolate reactor water cleanup system.	(f)
9	Reactor Water Cleanup Area Temperature	High	2	≤150°F	Isolate reactor water cleanup system.	
10	Reactor Water Cleanup Area Ventilation Differential Temperature	High	2	≤67°F	Isolate reactor water cleanup system.	
11	Condenser Vacuum	Low	2	≥7" Hg. vacuum	Initiate an orderly load reduction and close MSIVs within 8 hrs.	Initiate Group 1 Isolation
12	Drywell Radiation	High	1	≤138 R/HR.	Close the affected isolation valves within 24 hours or be in Hot Shutdown within the next 6 hours and in Cold Shutdown within the next 30 hours.	Isolates containment purge and vent valves.

Notes for Table 3.2-1

a. The column entitled "Ref. No." is only for convenience so that a one-to-one relationship can be established between lines in Table 3.2-1 and items in Table 4.2-1.

b.1. Primary containment integrity shall be maintained at all times prior to withdrawing control rods for the purpose of going critical, when the reactor is critical, or when the reactor water temperature is above 212°F and fuel is in the reactor vessel except while performing low-power physics tests at atmospheric pressure at power levels not to exceed 5 MWt, or performing an inservice vessel hydrostatic or leakage test.

When primary containment integrity is required, there shall be two operable or tripped trip systems for each function.

When performing inservice hydrostatic or leakage testing on the reactor vessel with the reactor coolant temperature above 212°F, reactor vessel water level instrumentation associated with the low low (Level 2) trip requires two operable or tripped channels. The drywell pressure trip is not required because primary containment integrity is not required.

b.2. One instrument channel may be inoperable for up to 6 hours to perform required surveillances prior to entering other applicable actions.

c.1. With the number of operable channels less than required by the Minimum Operable Channels per Trip System requirement for one trip system, either

1. place the inoperable channel(s) in the tripped condition* within 12 hours

   OR

2. take the action required by Table 3.2-1.

*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 3.2-1 for that Trip Function shall be taken.

c.2. One instrument channel may be inoperable for up to 6 hours to perform required surveillances prior to entering other applicable actions.

d. The valves associated with each Group isolation are given in Table 3.7-1.

e. Prior to the hydrogen injection system startup and with reactor power greater than 20% rated power, the normal full power radiation trip/alarm setpoints may be changed based on calculated expected radiation levels during hydrogen injection system operation. Associated trip/alarm setpoints may be adjusted during injection based on either calculations or measurements of actual radiation levels resulting from hydrogen injection. Following a reactor startup, a background radiation level will be determined and the associated trip/alarm setpoints adjusted within a 72-hour period. The radiation level shall be determined and associated trip/alarm setpoints shall be set within 24 hours of re-establishing normal radiation levels after a reduction in, or a completion of, hydrogen injection and prior to establishing reactor power levels below 20% of rated power.

f. The high differential flow signal to the RWCU isolation valves may be bypassed for up to 2 hours during periods of system restoration, maintenance, or testing.

3.2.A.2. Reactor Vessel Steam Dome Pressure (Shutdown Cooling Mode) Low Permissive

This setpoint is chosen to preserve the pressure integrity of the RHR system under conditions of increasing reactor pressure (startup). The RHR suction valves from the reactor (shutdown cooling mode) would be closed when the 145 psig setpoint is reached. This function protects against RHR system pipe breaks during the shutdown cooling mode of operation. Additionally, at reactor pressures below this setpoint the primary containment isolation signals are permitted to close the inboard motor operated injection valve (LPCI mode).

3. Drywell Pressure High

The Bases for Drywell Pressure High are discussed in the Bases for Specification 3.1.A.5. Pressure above the trip setting starts the SGTS and initiates primary and secondary containment isolation.

4. Main Steam Line Radiation High

Radiation monitors in the main steam line tunnel have been provided to detect gross fuel failure as in the control rod drop accident. This instrumentation causes isolation of the reactor water sample valves. With the established setting of approximately three times normal full power background, fission product release is limited so that 10 CFR 100 guidelines are not exceeded for this accident.

5. Main Steam Line Pressure Low

The Bases for Main Steam Line Pressure Low are discussed in the Bases for Specification 2.1.A.6.

6. Main Steam Line Flow High

Venturis are provided in the main steam lines as a means of measuring steam flow and also limiting the loss of mass inventory from the vessel during a steam line break accident. In addition to monitoring steam flow, instrumentation is provided which initiates Group 1 isolation. The primary function of the instrumentation is to detect a break in the main steam line. For the worst case accident, a main steam line break outside the drywell, the trip setting of 115 psid, corresponding to 138% of rated steam flow, in conjunction with the flow limiters and main steam isolation valve closure, limits the mass inventory loss such that fuel is not uncovered. Fuel temperatures remain approximately 1000°F and release of radioactivity to the environs is well below 10 CFR 100 guidelines. Ref. Section 14.6.5 of the FSAR.

7. Main Steam Line Tunnel Temperature High

Temperature monitoring instrumentation is provided in the main steam line tunnel to detect leaks in this area. Trips are provided on this instrumentation and when exceeded cause a Group 1 isolation. Its setting is low enough to detect leaks of the order of five to 10 gpm; thus, it is capable of covering the entire spectrum of breaks. For large breaks, it is a backup to high steam flow instrumentation discussed above, and for small breaks.

Table 3.2-8 (cont.)

Ref. No. (a)	Instrument	Trip Condition Nomencla-ture	Required Operable Channels per Trip System (b)	Trip Setting	Action to be taken if there are not two operable or tripped trip systems	Remarks
5.	Main Steam Line Radiation Monitor	Hi	2	≤3 times normal full power background (e)	Isolate the mechanical vacuum pump and the gland seal condenser exhauster	One trip per trip logic system will isolate the mechanical vacuum pump and the gland seal condenser exhauster.

a. The column entitled "Ref. No." is only for convenience so that a one-to-one relationship can be established between items in Table 3.2-8 and items in Table 4.2-8.

b.1. Whenever the systems are required to be operable, there shall be two operable or tripped trip systems. If this cannot be met, the indicated action shall be taken.

b.2. One instrument channel may be inoperable for up to 6 hours to perform required surveillances prior to entering other applicable actions.

c. In the event that both off-gas post treatment radiation monitors become inoperable, the reactor shall be placed in the Cold Shutdown within 24 hours unless one monitor is sooner made operable, or adequate alternative monitoring facilities are available.

d. From and after the date that one of the two off-gas post treatment radiation monitors is made or found to be inoperable, continued reactor power operation is permissible during the next fourteen days (the allowable repair time), provided that the inoperable monitor is tripped.

e. Prior to the hydrogen injection system startup and with reactor power greater than 20% rated power, the normal full power radiation trip/alarm setpoints may be changed based on calculated expected radiation levels during hydrogen injection system operation. Associated trip/alarm setpoints may be adjusted during injection based on either calculations or measurements of actual radiation levels resulting from hydrogen injection. Following a reactor startup, a background radiation level will be determined and the associated trip/alarm setpoints adjusted within a 72-hour period. The radiation level shall be determined and associated trip/alarm setpoints shall be set within 24 hours of re-establishing normal radiation levels after a reduction in, or a completion of, hydrogen injection and prior to establishing reactor power levels below 20% of rated power.

occurs with each monitor indicating HI HI HI, one monitor HI HI HI and the other downscale, or with both monitors downscale. The HI HI HI setpoint corresponds to the instantaneous release limit.

2. Refueling Floor Exhaust Vent Radiation Monitors

Four radiation monitors are provided which initiate isolation of the secondary containment and operation of the standby gas treatment system. The instrument channels monitor the radiation from the refueling area ventilation exhaust ducts.

Two instrument channels with two radiation monitors in each channel are arranged in a two upscale (either channel) trip logic. Trip settings for the monitors in the refueling floor exhaust ventilation ducts are based upon initiating normal ventilation isolation and standby gas treatment system operation so that none of the activity released during the refueling accident leaves the reactor building via the normal ventilation path but rather all the activity is processed by the standby gas treatment system.

3. Reactor Building Exhaust Vent Radiation Monitors

Four radiation monitors are provided which initiate secondary containment isolation, primary containment purge and vent valves isolation and standby gas treatment system actuation. The instrument channels monitor the radiation from the reactor building lower level ventilation exhaust duct.

Two instrument channels with two radiation detectors in each channel are arranged in a two upscale (either channel) trip logic. The trip settings are based on limiting the release of radioactivity via the normal ventilation path and rerouting this activity to be processed through the standby gas treatment system.

4. Control Room Intake Radiation Monitors

Two radiation monitors are provided to initiate pressurization of the main control room and recirculation of control room air through filters. The instrument channels monitor radiation from the control room ventilation intake duct.

Two instrument channels are arranged in one upscale, two downscale trip logic. The trip settings are based on limiting the radioactivity from entering the control room from outside.

5. Main Steam Line Radiation Monitors

The four Main Steam Line radiation monitors initiate isolation of the mechanical vacuum pump and the gland seal exhauster condenser. The instrument channels monitor the radiation in the main steam line tunnel. The purpose of automatically isolating the mechanical vacuum pump line is to provide timely protection against the release of radioactive materials from the main condenser. Upon receipt of main steam line high radiation signals, the primary containment and reactor vessel isolation control system initiates closure of the mechanical vacuum pump line valve. This isolation precludes or limits the release of fission product radioactivity which, upon fuel failure would be transported from

Table 3.7-1

Primary Containment Isolation Valves Which Receive a Primary Containment Isolation Valve Signal

These notes refer to the lower case letters in parentheses on the previous page.

NOTES:

a. Key: O = Open; C = Closed; SC = Stays closed; GC = Goes closed

b. Isolation Groupings are as follows:

GROUP 1: The valves in Group 1 are actuated by any one of the following conditions:

1. Reactor vessel water level Low Low Low (Level 1)
2. Main steam line radiation high*
3. Main steam line flow high
4. Main steam line tunnel temperature high
5. Main steam line pressure low
6. Condenser vacuum low
7. Turbine building temperature at the steam lines high

GROUP 2: The valves in group 2 are actuated by any one of the following conditions:

1. Reactor vessel water level low (Level 3)
2. Drywell pressure high

GROUP 3: Isolation valves in the high pressure coolant injection (HPCI) system are actuated by any one of the following conditions:

1. HPCI steam line flow high
2. High temperature in the vicinity of the HPCI steam line
3. HPCI steam supply pressure low
4. HPCI turbine exhaust diaphragm pressure
5. Torus room differential temperature high

GROUP 4: Primary Containment Isolation valves in the reactor core isolation cooling (RCIC) system are actuated by any one of the following conditions:

1. RCIC steam line flow high
2. High temperature in the vicinity of the RCIC steam line
3. RCIC steam line pressure low
4. RCIC turbine exhaust diaphragm pressure high
5. Torus room differential temperature high

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* Initiates closure of B31-F019 and B31-F020 only.

3.12. MAIN CONTROL ROOM ENVIRONMENTAL SYSTEM

The control room air treatment system is designed to filter the control room atmosphere for intake air and/or for recirculation during pressurization conditions.

A. Ventilation System Operability Requirements

The control room air treatment system operates on emergency power and is designed to filter the control room atmosphere for intake air and or recirculation air during control room pressurization conditions. The control room air treatment system is designed to automatically start upon receipt of an initiation signal and to align the system dampers to provide for pressurization of the control room.

Pressurization will be initiated upon receipt of any one of the following signals: High radiation at control room intake, LOCA signal from Unit 1 or 2, main steam line high flow from Unit 1 or 2, or refueling floor high radiation from Unit 1 or 2. In this mode the normal control room exhaust fan is stopped and outside air is taken in through one of the charcoal filters to pressurize the control room with respect to the surrounding turbine building. High efficiency particulate air (HEPA) filters are installed before the charcoal adsorbers to prevent clogging of the iodine adsorbers. The charcoal adsorbers are installed to reduce the potential intake of radioiodine to the control room. Bypass leakage for the charcoal adsorbers and particulate removal efficiency for HEPA filters are determined by halogenated hydrocarbon and DOP, respectively. The laboratory carbon sample test results indicate a radioactive methyl iodide removal efficiency for expected accident conditions. Operation of the fans significantly different from the design flow will change the removal efficiency of the HEPA filters and charcoal adsorbers. If the performances are as specified, the calculated doses would be less than the allowable levels stated in Criterion 19 of the General Design Criteria for Nuclear Power Plants, Appendix A to 10 CFR Part 50.

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% - 0.58ΔW)** with a maximum ≤ 113.5% of RATED THERMAL POWER	≤ (0.58 W + 62% - 0.58ΔW)** 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)	≥ 0 inches above instrument zero*	≥ 0 inches above instrument zero*
5. Main Steam Line Isolation Valve - Closure (NA)	≤ 10% closed	≤ 10% closed
6. (Deleted)		
7. Drywell Pressure - High (2C71-N650A,B,C,D)	≤ 1.92 psig	≤ 1.92 psig

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

** W = Total loop recirculation flow rate in percent of rated. Rated loop recirculation flow is equal to 34.2 MLB/hr.

ΔW = Maximum measured difference between two-loop and single-loop drive flow for the same core flow in percent of rated recirculation flow for single-loop operation. The value is zero for two-loop operation.

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, 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. (Deleted)

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	5
4. Reactor Vessel Water Level - Low (Level 3) (2B21-N680 A, B, C, D)	1, 2	2	5
5. Main Steam Line Isolation Valve - Closure (NA)	1[f]	4	3
6. (Deleted)			
7. Drywell Pressure - High (2C71-N650 A, B, C, D)	1, 2[g]	2	5

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

ACTION

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

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

ACTION 2 - Lock the reactor mode switch in the Shutdown position within one hour.

ACTION 3 - Be in at least STARTUP within 2 hours.

ACTION 4 - In OPERATIONAL CONDITION 1 or 2, be in at least HOT SHUTDOWN within 6 hours.

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

ACTION 5 - Be in at least HOT SHUTDOWN within 6 hours.

ACTION 6 - (Deleted)

ACTION 7 - Initiate a reduction in THERMAL POWER within 15 minutes and be at less than 30% of RATED THERMAL POWER within 2 hours.

ACTION 8 - In OPERATIONAL CONDITION 1 or 2, be in at least HOT SHUTDOWN within 6 hours.

In OPERATIONAL CONDITION 3 or 4, immediately and at least once per 12 hours verify that all control rods are fully inserted.

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 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 1 hour.

In OPERATIONAL CONDITION 5, suspend all operations involving CORE ALTERATIONS or positive reactivity changes and fully insert all insertable control rods within 1 hour.

TABLE NOTATIONS

a. Deleted.

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 11 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 $\leq$250* psig, equivalent to THERMAL POWER less than 30% of RATED THERMAL POWER.

j. (Deleted)

*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. (Deleted)	
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 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], Q	W[e][f], SA	1
c. Fixed Neutron Flux - Upscale, 118%	S	S/U[b], Q	W[e], SA	1
d. Inoperative	NA	Q	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	Q	R	1, 2
4. Reactor Vessel Water Level - Low (Level 3)	S	Q	R	1, 2
5. Main Steam Line Isolation Valve - Closure	NA	Q	R	1
6. (Deleted)				
7. Drywell Pressure - High	S	Q	R	1, 2
8. Scram Discharge Volume Water Level - High	NA	Q	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	Q	R[h]	1
10. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	NA	Q	R	1
11. Reactor Mode Switch in Shutdown Position	NA	R	NA	1, 2, 3, 4, 5
12. Manual Scram	NA	W	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.

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)	12, (d)(m)	2	1, 2, 3,(k)	30
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-P001, 2U61-P002, 2U61-P003, 2U61-P004)	1	2(e)	1, 2, 3	21
g. Drywell Radiation - High (2D11-K621 A, B)	(j)	1	1, 2, 3	29

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.

ACTION 29 - Either close the affected isolation valves within 24 hours or be in HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the next 30 hours.

ACTION 30 - Trip and isolate the mechanical vacuum pump and isolate the reactor water sample valves.

NOTES

* Actuates the standby gas treatment system.

** When handling irradiated fuel in the secondary containment.

*** When performing inservice hydrostatic or leak testing with the reactor coolant temperature above 212° F.

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

b. Deleted.

TABLE 3.3.2-1 (Continued)

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.

j. Isolates containment purge and vent valves.

k. Prior to the hydrogen injection system startup and with reactor power greater than 20% rated power, the normal full power radiation trip/alarm setpoints may be changed based on calculated expected radiation levels during hydrogen injection system operation. Associated trip/alarm setpoints may be adjusted during injection based on either calculations or measurements of actual radiation levels resulting from hydrogen injection. Following a reactor startup, a background radiation level will be determined and the associated trip/alarm setpoints adjusted within a 72-hour period. The radiation level shall be determined and associated trip/alarm setpoints shall be set within 24 hours of re-establishing normal radiation levels after a reduction in, or a completion of, hydrogen injection and prior to establishing reactor power levels below 20% rated power.

l. The high differential flow isolation signal to the RWCU isolation valves may be bypassed for up to 2 hours during periods of system restoration, maintenance or testing.

m. Isolates reactor water sample valves 2B31-F019 and 2B31-F020. These are Group 1 valves.

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)	≥ 0 inches*	≥ 0 inches*
2. Low Low (Level 2)	≥ -47 inches*	≥ -47 inches*
3. Low Low Low (Level 1)	≥ -113 inches*	≥ -113 inches*
b. Drywell Pressure - High	≤ 1.92 psig	≤ 1.92 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
g. Drywell Radiation - High	≤ 138 R/hr	≤ 138 R/hr
2. SECONDARY CONTAINMENT ISOLATION		
a. Reactor Building Exhaust Radiation - High	≤ 60 mr/hr	≤ 60 mr/hr
b. Drywell Pressure - High	≤ 1.92 psig	≤ 1.92 psig
c. Reactor Vessel Water Level - Low Low (Level 2)	≥ -47 inches*	≥ -47 inches*
d. Refueling Floor Exhaust Radiation - High	≤ 20 mr/hr	≤ 20 mr/hr

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

**Prior to the hydrogen injection system startup and with reactor power greater than 20% rated power, the normal full power radiation trip/alarm setpoints may be changed based on calculated expected radiation levels during hydrogen injection system operation. Associated trip/alarm setpoints may be adjusted during injection based on either calculations or measurements of actual radiation levels resulting from hydrogen injection. Following a reactor startup, a background radiation level will be determined and the associated trip/alarm setpoints adjusted within a 72-hour period. The radiation level shall be determined and associated trip/alarm setpoints shall be set within 24 hours of re-establishing normal radiation levels after a reduction in, or a completion of, hydrogen injection and prior to establishing reactor power levels below 20% of rated power.

SURVEILLANCE REQUIREMENTS (Continued)

3. Verifying that on each of the below pressurization mode actuation test signals, the system automatically switches to the pressurization mode of operation and maintains the main control room at a positive pressure of ≥ 0.1-in. W.G. relative to the adjacent turbine building during system operation at a flow rate ≤ 400 cfm.

   a) Reactor vessel water level - low low low

   b) Drywell pressure - high

   c) Refueling floor area radiation - high

   d) (Deleted)

   e) Main steam line flow - high

   f) Control room intake monitors radiation - high

f. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove ≥ 99 percent of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 2500 cfm ± 10 percent.

g. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove ≥ 99 percent of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 2500 cfm ± 10 percent.

TABLE 3.3.6.7-1 (SHEET 1 OF 2)

MCRECS ACTUATION INSTRUMENTATION

TRIP FUNCTION	MINIMUM NUMBER OPERABLE CHANNELS PER TRIP SYSTEM(a)(b)	APPLICABLE OPERATIONAL CONDITION	ACTION
1. Reactor Vessel Water Level - Low Low Low (Level 1) (c) 2B21-N691 A, B, C, D	2	1, 2, 3	52
2. Drywell Pressure - High (c) 2E11-N694 A, B, C, D	2	1, 2, 3	52
3. (Deleted)			
4. Main Steam Line Flow - High (c) 2B21-N686 A, B, C, D 2B21-N687 A, B, C, D 2B21-N688 A, B, C, D 2B21-N689 A, B, C, D	2/line	1, 2, 3	53
5. Refueling Floor Area Radiation - High (c) 2D21-K002 A, D	1	1, 2, 3, 5, *	54
6. Control Room Air Inlet Radiation - High (c) 1Z41-R615 A, B	1	1, 2, 3, 5, *	54

TABLE 3.3.6.7-1 (SHEET 2 OF 2)

MCRECS ACTUATION INSTRUMENTATION

ACTION

ACTION 52 - Take the ACTION required by Specification 3.3.3.

ACTION 53 - Take the ACTION required by Specification 3.3.2.

ACTION 54 -

a. With one of the required radiation monitors inoperable, restore the monitor to OPERABLE status within 7 days or, within the next 6 hours, initiate and maintain operation of the MCRECS in the pressurization mode of operation.

b. With no radiation monitors OPERABLE, within 1 hour initiate and maintain operation of the MCRECS in the pressurization mode of operation.

c. The provisions of Specification 3.0.4 are not applicable.

NOTES

* When handling irradiated fuel in secondary containment.

a. (Deleted)

b. 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 12 hours or the ACTION required by Table 3.3.6.7-1 for that Trip Function shall be taken.

c. Actuates the MCRECS in the control room pressurization mode.

d. (Deleted)

e. (Deleted)

TABLE 3.3.6.7-2

MCRECS ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. Reactor Vessel Water Level - Low Low Low (Level 1)	≥ -113 inches	≥ - 113 inches
2. Drywell Pressure - High	≤ 1.92 psig	≤ 1.92 psig
3. (Deleted)		
4. Main Steam Line Flow - High	≤ 138% rated flow	≤ 138% rated flow
5. Refueling Floor Area Radiation - High	≤ 20 mr/hour	≤ 20 mr/hour
6. Control Room Air Inlet Radiation - High	≤ 1 mr/hour	≤ 1 mr/hour

TABLE 4.3.6.7-1

MCRECS ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
1. Reactor Vessel Water Level - Low Low Low (Level 1)	S	Q	R	1, 2, 3
2. Drywell Pressure - High	S	Q	R	1, 2, 3
3. (Deleted)				
4. Main Steam Line Flow - High	S	Q	R	1, 2, 3
5. Refueling Floor Area Radiation - High	S	Q[a]	Q	1, 2, 3, 5 *
6. Control Room Air Inlet Radiation - High	NA	Q[a]	R	1, 2, 3, 5, *

* When handling irradiated fuel in the secondary containment.

a. Instrument alignment using a standard current source.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 188 TO FACILITY OPERATING LICENSE DPR-57

AND AMENDMENT NO. 127 TO FACILITY OPERATING LICENSE NPF-5

GEORGIA POWER COMPANY, ET AL.

EDWIN I. HATCH NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-321 AND 50-366

1.0 INTRODUCTION

By letter dated October 19, 1992, as supplemented May 3 and July 27, 1993, Georgia Power Company, et al. (GPC or the licensee), proposed license amendments to change the Technical Specifications (TS) for the Edwin I. Hatch Nuclear Plant, Units 1 and 2. The proposed changes would remove the main steam line radiation monitor (MSLRM) reactor scram and group isolation functions. The licensee referenced the BWR Owners Group (BWROG) topical report NEDO-31400, "Safety Evaluation for Eliminating the BWR MSIV Closure Function of the MSLRM." Furthermore, the licensee's request proposed revisions to the hydrogen water chemistry footnotes and changes to an action statement concerning the offgas post treatment monitors. The May 3 and July 27, 1993, letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

2.0 EVALUATION

The licensee referenced BWROG topical report NEDO-31400 in support of its request to eliminate the MSLRM scram and group isolation functions. In the topical report, GE analyzes a control rod drop accident (CRDA) where the main steam line high radiation isolation is eliminated. The resulting radiological exposures are small fractions of 10 CFR 100 limits. The topical report received NRC approval in a Safety Evaluation Report dated May 15, 1991. To ensure that the assumptions of NEDO-31400 are bounding for Hatch Units 1 and 2, GPC requested that GE perform an analysis specific to the plant. The dose rates for Hatch Units 1 and 2 that result from the elimination of the scram and MSIV isolation functions are also small fractions of 10 CFR 100 limits. The accident doses calculated by GE in support of this amendment request utilized a methodology, effective dose equivalent, which is not presently approved by the staff for application to accident analyses. It is presently approved only for use for 10 CFR Part 20 applications. Therefore, any future calculations of doses should be based upon the methodology in Regulatory Guide 1.3.

The licensee's evaluation, which provide the basis for the proposed revisions, notes that two parameters of Hatch Unit 2 CRDA analysis are not bounded by

NEDO-31400 parameters. The unbounded parameters are the fraction of damaged fuel that melts, and the iodine washout/plateout fraction.

The fraction of melted fuel assumed in the NEDO-31400 report is less than that assumed in the CRDA analysis contained in the Hatch Final Safety Analysis Report (FSAR). However, the number of fuel rods failed and the power level per rod assumed in the NEDO-31400 report are sufficiently more conservative than the Hatch FSAR values such that the non-conservatism in the fraction of melted fuel is offset.

The fraction of iodine assumed to plate out in the NEDO-31400 analysis is greater than the Hatch FSAR value by a factor of 5. Therefore, the fraction of condenser activity that remains airborne is less in the NEDO-31400 analysis than in the Hatch analysis. This non-conservatism, however, is offset by the higher condenser leak rate assumed (factor of 2) and a higher value of Chi/Q (factor of 8).

In addition to verification that input assumptions used in the CRDA analysis were applicable to a specific plant, the staff's SER on NEDO-31400 also required the following:

1. The licensee must provide reasonable assurance that increased levels of radioactivity in the main steam lines will be controlled expeditiously to limit both occupational doses and environmental releases; and,

2. The MSLRM and offgas radiation monitor alarm setpoint must be set at 1.5 times the nominal N-16 background dose rate at the monitor locations, and the licensee must promptly sample the reactor coolant to determine possible contamination levels if the setpoint of either monitor is exceeded.

The licensee indicated that both annunciator response procedures (ARPs) and abnormal operating procedures (AOPs) are entered upon receipt of MSLRM alarms which are set at 1.5 times background including N-16. These procedures provide guidance for checking secondary containment conditions and notifying plant personnel. AOP 34AB-OPS-062, "Closure of MSIVs on High Radiation" will be revised such that the entry condition is receipt of a high radiation alarm rather than MSIV closure on high radiation. This procedure will include a requirement to sample the reactor coolant upon receipt of a high radiation alarm. If area radiation monitors exceed specified levels, entry into the secondary containment control section of the emergency operating procedures is required. Radiological procedures are in place which provide guidance to plant personnel for exiting high radiation areas upon indication of unexpectedly high area radiation monitor readings. Finally, offgas pretreatment monitors which alarm at 1.5 times background including N-16 will be monitored, by procedure, upon receipt of a high MSLRM alarm. Performance of a prompt offsite dose assessment is required through entry into a separate AOP.

Based upon the incorporation of the above staff position, the staff concludes that, upon implementation of the revised procedures described in the licensee's submittal, the requirements of the staff's generic SER on NEDO-31400 are satisfied for Hatch Units 1 and 2.

The licensee's request also included elimination of the main steam line (MSL) drain valves from the isolation logic. Although not addressed specifically in the GE analysis, the request is considered to be insignificant because the exhaust from the drain path that discharges to the main condenser is minimal when compared to the MSIV's, and both paths are processed by the offgas system. This change is, therefore, acceptable.

Furthermore, the licensee's request included the removal of the MSLRM high radiation trip from the main control room environmental (MCREC) system. This change was not addressed in the generic (NEDO-31400) analysis. An important function of the pressurization mode of the MCREC system is to protect the main control room operators during a design basis accident (DBA) such as CRDA. The staff considers this change acceptable, however, because the pressurization mode is also initiated at the main control room air intake on a high radiation signal. Redundant air intake radiation monitors are provided which are not susceptible to a loss of function due to a single failure. Also, the abnormal operating procedure (AOP) will have instructions to manually activate the pressurization mode of the MCREC system if an MSL high radiation signal is confirmed.

Moreover, the licensee requested deletion of a requirement for chemistry personnel to adjust the MSLRM alarm setpoint within the 24 hour period prior to the start of hydrogen injection when reactor power is greater than 20% of rated thermal power. This is acceptable because the consequences of CRDA are insignificant above 20% power and the existing requirement remains unchanged below 20% power. The proposed amendment would also require that within 72 hours after a reactor startup chemistry personnel must make final adjustments to trip setpoints. This provides a reasonable amount of time for establishment of a steady state background radiation level and adjustment of setpoints.

In addition, the licensee's request proposed changes to an action statement concerning offgas post treatment monitors. The revision is due to the installation of new offgas post treatment radiation monitors. The new monitors do not have a "downscale" switch position. The new action statement will specify, without mention of a "downscale" switch position, that the monitors should be tripped if one of the two monitors are inoperable. Since the existing statement which directs that the monitors be tripped in the "downscale" position is no longer applicable, the staff agrees that it should be replaced with the revised action statement.

The staff has reviewed the proposed TS changes for Hatch, Units 1 and 2, which involve the elimination of the MSIV closure, the reactor scram, and the control room pressurization functions of the MSLRM. The request also included removal of the MSL drain valves from the isolation logic, revisions to the hydrogen water chemistry footnotes and an action statement concerning the offgas post treatment monitors.

Based on above evaluation, the staff finds that the proposed TS revisions have no adverse impact on safety, and do not pose an undue risk to public health and safety, and are, therefore, acceptable.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Georgia State official was notified of the proposed issuance of the amendments. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendments 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 the amendments involve no significant hazards consideration, and there has been no public comment on such finding (58 FR 19482 dated April 14, 1993). Accordingly, the amendments 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 the amendments.

5.0 CONCLUSION

The Commission has 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, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

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