August 28, 1991

Docket Nos. 50-321
and 50-366

Distribution
See next page

Mr. W. G. Hairston, III
Senior Vice President -
Nuclear Operations
Georgia Power Company
P.O. Box 1295
Birmingham, Alabama 35201

Dear Mr. Hairston:

SUBJECT: ISSUANCE OF AMENDMENT NO. 172 TO FACILITY OPERATING LICENSE DPR-57 AND AMENDMENT NO. 112 TO FACILITY OPERATING LICENSE NPF-5 - EDWIN I. HATCH NUCLEAR PLANT, UNITS 1 AND 2 (TACs 80592/80593)

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 172 to Facility Operating License No. DPR-57 and Amendment No. 112 to Facility Operating License No. NPF-5 for the Edwin I. Hatch Nuclear Plant, Units 1 and 2. The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated June 13, 1991.

The amendments 1) revise Unit 1 TS 3.10.D and Unit 2 TS 3/4.9.10 and their associated bases to require at least 21 feet of water above irradiated fuel assemblies seated in the spent fuel pool (SFP) fuel storage racks, and revise Unit 1 TS 4.10.D to require surveillance of the SFP water level every 7 days consistent with the Unit 2 TSs and the BWR-4 Standard TSs; 2) revise Unit 1 TS Tables 3.2-11 and 4.2-11 to require that the post-LOCA radiation monitors be calibrated at least once every 18 months; and 3) correct administrative errors in Unit 2 TS Tables 3.3.2-1 and 3.8.2.6-1.

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

Sincerely,
ORIGINAL SIGNED BY: F. RINALDI for/

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

9109120051 910828
PDR ADOCK 05000321
P PDR

Enclosures:
1. Amendment No. 172 to DPR-57
2. Amendment No. 112 to NPF-5
3. Safety Evaluation

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See next page

LA:PDII3	PM:PDII3	SPLB	SICB	PRPB
RIngram	KJabbour:cw	CMcCracken	SNewberry	LCunningham
7/1/91	7/15/91 for	7/16/91	8/12/91	8/5/91
OGC	D:PDII3			
	DMatthews			
8/21/91	8/28/91			

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

August 28, 1991

Docket Nos. 50-321
and 50-366

Mr. W. G. Hairston, III
Senior Vice President -
Nuclear Operations
Georgia Power Company
P.O. Box 1295
Birmingham, Alabama 35201

Dear Mr. Hairston:

SUBJECT: ISSUANCE OF AMENDMENT NO. 172 TO FACILITY OPERATING LICENSE DPR-57 AND AMENDMENT NO. 112 TO FACILITY OPERATING LICENSE NPF-5 - EDWIN I. HATCH NUCLEAR PLANT, UNITS 1 AND 2 (TACs 80592/80593)

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 172 to Facility Operating License No. DPR-57 and Amendment No. 112 to Facility Operating License No. NPF-5 for the Edwin I. Hatch Nuclear Plant, Units 1 and 2. The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated June 13, 1991.

The amendments 1) revise Unit 1 TS 3.10.D and Unit 2 TS 3/4.9.10 and their associated bases to require at least 21 feet of water above irradiated fuel assemblies seated in the spent fuel pool (SFP) fuel storage racks, and revise Unit 1 TS 4.10.D to require surveillance of the SFP water level every 7 days consistent with the Unit 2 TSs and the BWR-4 Standard TSs; 2) revise Unit 1 TS Tables 3.2-11 and 4.2-11 to require that the post-LOCA radiation monitors be calibrated at least once every 18 months; and 3) correct administrative errors in Unit 2 TS Tables 3.3.2-1 and 3.8.2.6-1.

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

Sincerely,

Frances Rinaldi, for

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

Enclosures:
1. Amendment No. 172 to DPR-57
2. Amendment No. 112 to NPF-5
3. Safety Evaluation

cc w/enclosures:
See next page

Mr. W. G. Hairston, III
Georgia Power Company

Edwin I. Hatch Nuclear Plant,
Units Nos. 1 and 2

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DATED: August 28, 1991

AMENDMENT NO. 172 TO FACILITY OPERATING LICENSE DPR-57 - Edwin I. Hatch Nuclear Plant, Unit 1

AMENDMENT NO. 112 TO FACILITY OPERATING LICENSE NPF-5 - Edwin I. Hatch Nuclear Plant, Unit 2

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S. Newberry	8-H-3
L. Cunningham	10-D-24

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. 50-321

EDWIN I. HATCH NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 172
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 Georgia Power Company, acting for itself, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, and City of Dalton, Georgia (the licensees) dated June 13, 1991, 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;

   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 license 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.

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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. 172 , 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 within 30 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

DBMatthews

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 28, 1991

ATTACHMENT TO LICENSE AMENDMENT NO.172

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 areas of change.

Remove Pages	Insert Pages
3.2-23	3.2-23
3.2-48	3.2-48
3.10-2	3.10-2
3.10-7	3.10-7

NOTES FOR TABLE 3.2-11

a. The column entitled "Ref. No." is only for convenience so that a one-to-one relationship can be established between items in Table 3.2-11 and items in Table 4.2-11.

b. Limiting Conditions for Operation for the Neutron Monitoring System are listed in Table 3.2-7.

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

Continued operation is permissible for seven days from and after the date that one of these parameters is not indicated in the control room. Surveillance of local panels will be substituted for indication in the control room during the seven days.

d. Drywell and Suppression Chamber Pressure are each recorded on the same recorders. Each output channel has its own recorder.

Drywell and Suppression Chamber air temperature and suppression chamber water temperature are all recorded on the same recorders. Each output channel has its own recorder. Each recorder takes input from several temperature elements.

Hydrogen and Oxygen are indicated on one recorder. The recorder has two pens, one pen for each parameter.

Each channel of the post LOCA radiation monitoring system includes two detectors; one located in the drywell and the other in the suppression chamber. Each detector feeds a signal to a separate log count rate meter. The meter output goes to a two pen recorder. One high radiation level alarm is provided per channel and annunciation of alarm is provided in the control room.

High Range Drywell Pressure and High Range Drywell Radiation are recorded on the same recorders. Each output channel has its own recorder.

e. In the event that all indications of this parameter is disabled and such indication cannot be restored in six (6) hours, an orderly shutdown shall be initiated and the reactor shall be in a Hot Shutdown condition in six (6) hours and a Cold Shutdown condition in the following eighteen (18) hours.

f. If either the primary or secondary indication is inoperable, the torus temperature will be monitored at least once per shift to observe any unexplained temperature increase which might be indicative of an open SRV. With both the primary and secondary monitoring channels of two or more SRVs inoperable either restore sufficient inoperable channels such that no more than one SRV has both primary and secondary channels inoperable within 7 days or be in at least hot shutdown within the next 12 hours.

Table 4.2-11

Check and Calibration Minimum Frequency for Instrumentation Which Provides Surveillance Information

Ref. No. (a)	Instrument	Instrument Check Minimum Frequency (b)	Instrument Calibration Minimum Frequency (c)
1	Reactor Vessel Water Level	Each shift	Once/operating cycle (f)
2	Shroud Water Level	Each shift	Once/operating cycle (f)
3	Reactor Pressure	Each shift	Once/operating cycle (f)
4	Drywell Pressure	Each shift	Every 6 months
5	Drywell Temperature	Each shift	Every 6 months
6	Suppression Chamber Air Temperature	Each shift	Every 6 months
7	Suppression Chamber Water Temperature	Each shift	Every 6 months
8	Suppression Chamber Water Level	Each shift	Every 6 months
9	Suppression Chamber Pressure	Each shift	Every 6 months
10	Rod Position Information System (RPIS)	Each shift	N/A
11	Hydrogen and Oxygen Analyzer	Monthly	Every 3 months
12	Post LOCA Radiation	Each shift	Every 18 months
13	a) Safety/Relief Valve Position Primary Indicator	Monthly	Every 18 months
	b) Safety/Relief Valve Position Secondary Indicator	Monthly	Every 18 months

3.10.C. Core Monitoring During Core Alterations

1. During normal core alterations, two SRMs shall be operable; one in the core quadrant where fuel or control rods are being moved and one in an adjacent quadrant, except as specified in 2 and 3 below.

   For an SRM to be considered operable, it shall be inserted to the normal operating level and shall have a minimum of 3 cps with all rods capable of normal insertion fully inserted.

2. Prior to spiral unloading the SRMs shall be proven operable as stated above, however, during spiral unloading the count rate may drop below 3 cps.

3. Prior to spiral reload, up to four (4) fuel assemblies will be loaded into core positions next to each of the 4 SRMs to obtain the required 3 cps. These assemblies may be any which have been shown to meet the criteria for storage in the spent fuel pool. Until these assemblies have been loaded, the 3 cps requirement is not necessary.

D. Spent Fuel Pool Water Level

At least 21 feet of water shall be maintained over the top of irradiated fuel assemblies seated in the spent fuel storage racks.

E. Control Rod Drive Maintenance

1. Requirements for Withdrawal of 1 or 2 Control Rods

   A maximum of two control rods separated by at least two control cells in all directions may be withdrawn or removed from the core for the purpose of performing control rod drive maintenance provided that:

   a. The Mode Switch is locked in the REFUEL position. The refueling interlock which prevents more than one control rod from being withdrawn may be bypassed for one of the control rods on which maintenance is being

4.10.C. Core Monitoring During Core Alterations

Prior to making normal alterations to the core the SRMs shall be functionally tested and checked for neutron response. Thereafter, while required to be operable, the SRMs will be checked daily for response.

Use of special movable, dunking type detectors during initial fuel loading and major core alterations in place of normal detectors is permissible as long as the detector is connected to the normal SRM circuit.

Prior to spiral unloading or reloading the SRMs shall be functionally tested. Prior to spiral unloading the SRMs should also be checked for neutron response.

D. Spent Fuel Pool Water Level

The water level in the spent fuel pool shall be determined to be at least its minimum required depth at least once per 7 days.

E. Control Rod Drive Maintenance

1. Requirements for Withdrawal of 1 or 2 Control Rods

   a. This surveillance requirement is the same as given in 4.10.A.

3.10.A.2. Fuel Grapple Hoist Load Setting Interlocks

Fuel handling is normally conducted with the fuel grapple hoist. The total load on this hoist when the interlock is required consists of the weight of the fuel grapple and the fuel assembly. This total is approximately 1500 lbs. in comparison to the load setting of 485 ± 30 lbs.

3. Auxiliary Hoists Load Setting Interlock

Provisions have also been made to allow fuel handling with either of the three auxiliary hoists and still maintain the refueling interlocks. The 485 ± 30 lb load setting of these hoists is adequate to trip the interlock when a fuel bundle is being handled.

B. Fuel Loading

To minimize the possibility of loading fuel into a cell containing no control rod, it is required that all control rods are fully inserted when fuel is being loaded into the reactor core. This requirement assures that during refueling the refueling interlocks, as designed, will prevent inadvertent criticality.

C. Core Monitoring During Core Alterations

The SRMs are provided to monitor the core during periods of Unit shutdown and to guide the operator during refueling operations and Unit startup. Requiring two operable SRMs in or adjacent to any core quadrant where fuel or control rods are being moved assures adequate monitoring of that quadrant during such alterations. The requirements of 3 counts per second provides assurance that neutron flux is being monitored.

During spiral unloading, it is not necessary to maintain 3 cps because core alterations will involve only reactivity removal and will not result in criticality.

The loading of up to four fuel bundles around the SRMs before attaining the 3 cps is permissible because these bundles form a subcritical configuration.

D. Spent Fuel Pool Water Level

The minimum water level in the spent fuel pool shall be maintained at least 21 feet above the top of the upper tie plates of the irradiated fuel assemblies seated in the spent fuel pool racks. This minimum level ensures removal of at least 98.6% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. This water depth is consistent with the assumptions for the fuel handling accident analysis outlined in Regulatory Guide 1.25 and the requirements in Standard Review Plan 15.7.4 for radiological releases resulting from that accident.

E. Control Rod Drive Maintenance

During certain periods, it is desirable to perform maintenance on two control rod drives at the same time.

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 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 112
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 Georgia Power Company, acting for itself, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, and City of Dalton, Georgia (the licensees) dated June 13, 1991, 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;

   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 license 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. 112, 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 within 30 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

DBMatthews

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 28, 1991

ATTACHMENT TO LICENSE AMENDMENT NO.112

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. The corresponding overleaf pages are also provided to maintain document completeness.

Remove Pages	Insert Pages
3/4 3-13	3/4 3-13
3/4 3-14*	3/4 3-14
3/4 8-23	3/4 8-23
3/4 8-24*	3/4 8-24
3/4 9-15	3/4 9-15
3/4 9-16*	3/4 9-16
B 3/4 9-2	B 3/4 9-2

*Overleaf pages

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

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

TABLE 3.3.2-1 (Continued)
ISOLATION ACTUATION INSTRUMENTATION

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

TABLE 3.3.2-1 (Continued)
ISOLATION ACTUATION INSTRUMENTATION

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

TABLE 3.8.2.6-1 (Continued)

PRIMARY CONTAINMENT PENETRATION CONDUCTOR
OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER AND LOCATION*	SYSTEM/COMPONENT POWERED
13. 600 VAC, MCB, MO 2R24-S011, FR 18C	DRYWELL RETURN AIR FAN 2T47-C001A
14. 600 VAC, MCB, MO 2R24-S013, FR 3B	DRYWELL COOLING UNIT 2T47-B010A
15. 600 VAC, MCB, MO 2R24-S014, FR 8A	DRYWELL COOLING UNIT 2T47-B010B
g. Type 7:	
1. 208 VAC, MCB, MO 2R24-S013, FR 11D	DRYWELL CHEMICAL DRAIN SUMP PUMP 2G11-C101
2. 208 VAC, MCB, MO 2R24-S012, FR 23C	DRYWELL RETURN AIR FAN 2T47-C002B
3. 208 VAC, MCB, MO 2R24-S011, FR 22C	DRYWELL RETURN AIR FAN 2T47-C002A

*MCB - molded case circuit breaker
MO - magnetic only
TM - thermal magnetic

3/4.8.2 ONSITE POWER DISTRIBUTION SYSTEMS

ELECTRIC POWER MONITORING FOR REACTOR PROTECTION SYSTEM

LIMITING CONDITION FOR OPERATION

3.8.2.7 The power monitoring system for a RPS MG set or the Alternate Source shall be OPERABLE if in service.

APPLICABILITY: At all times.

ACTION:

With the power monitoring system for a RPS MG set or the Alternate Source inoperable, restore the inoperable power monitoring system to OPERABLE status within 30 minutes or remove the RPS MG set or Alternate Source associated with the inoperable power monitoring system from service.

One channel of a power monitoring system may be inoperable, as necessary for test or maintenance, not to exceed 8 hours per month.

SURVEILLANCE REQUIREMENTS

4.8.2.7 The above specified RPS power monitoring system instrumentation shall be determined OPERABLE:

a. At least once per 6 months by performing a FUNCTIONAL TEST; and

b. At least once per operating cycle by demonstrating the OPERABILITY of over-voltage, under-voltage and under-frequency protective instrumentation by performance of a CHANNEL CALIBRATION including simulated automatic actuation of the protective relays, tripping logic and output circuit breakers and verifying the following setpoints.

    1. Over-voltage $\leq$ 132 VAC,

    2. Under-voltage $\geq$ 108 VAC, with time delay relay set to zero*, and

    3. Under-frequency $\geq$ 57 Hz.

*Pending NRC approval of different value.

ELECTRIC POWER MONITORING FOR REACTOR PROTECTION SYSTEM

LIMITING CONDITION FOR OPERATION

3.8.2.7 The power monitoring system for a RPS MG set or the Alternate Source shall be OPERABLE if in service.

APPLICABILITY: At all times.

ACTION:

With the power monitoring system for a RPS MG set or the Alternate Source inoperable, restore the inoperable power monitoring system to OPERABLE status within 30 minutes or remove the RPS MG set or Alternate Source associated with the inoperable power monitoring system from service.

One channel of a power monitoring system may be inoperable, as necessary for test or maintenance, not to exceed 8 hours per month.

SURVEILLANCE REQUIREMENTS

4.8.2.7 The above specified RPS power monitoring system instrumentation shall be determined OPERABLE:

a. At least once per 6 months by performing a FUNCTIONAL TEST; and

b. At least once per operating cycle by demonstrating the OPERABILITY of over-voltage, under-voltage and under-frequency protective instrumentation by performance of a CHANNEL CALIBRATION including simulated automatic actuation of the protective relays, tripping logic and output circuit breakers and verifying the following setpoints.

   1. Over-voltage $\leq$ 132 VAC,

   2. Under-voltage $\geq$ 108 VAC, with time delay relay set to zero*, and

   3. Under-frequency $\geq$ 57 Hz.

*Pending NRC approval of different value.

3/4.9.10 WATER LEVEL - SPENT FUEL STORAGE POOL

LIMITING CONDITION FOR OPERATION

3.9.10 At least 21 feet of water shall be maintained over the top of irradiated fuel assemblies seated in the spent fuel storage pool racks.

APPLICABILITY: Whenever irradiated fuel assemblies are in the spent fuel storage pool.

ACTION:

With the requirements of the above specification not satisfied, suspend all movement of fuel assemblies and crane operations with loads in the spent fuel storage pool area after placing the load in a safe condition. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.10 The water level in the spent fuel storage pool shall be determined to be at least its minimum required depth at least once per 7 days.

3/4.9.11 CONTROL ROD REMOVAL

SINGLE CONTROL ROD REMOVAL

LIMITING CONDITION FOR OPERATION

3.9.11.1 One control rod and/or the associated control rod drive mechanism may be removed from the reactor pressure vessel provided that at least the following requirements are satisfied until the control rod and associated control rod drive mechanism are reinstalled and the control rod is fully inserted in the core.

a. The reactor mode switch is OPERABLE and locked in the Refuel position per Specification 3.9.1.

b. The source range monitors (SRM) are OPERABLE per Specification 3.9.2.

c. The SHUTDOWN MARGIN requirements of Specification 3.1.1 are satisfied, except that the control rod selected to be removed;

   1. May be assumed to be the highest worth control rod required to be assumed to be fully withdrawn by the SHUTDOWN MARGIN test, and

   2. Need not be assumed to be immovable or untrippable.

d. All other control rods in a five-by-five array centered on the control rod being removed are fully inserted and electrically disarmed.

e. All other control rods are either fully inserted or have the surrounding four fuel assemblies removed.

APPLICABILITY: CONDITION 5.

ACTION:

With the requirements of the above specification not satisfied, suspend removal of the control rod and/or associated control rod drive mechanism from the reactor pressure vessel and initiate action to satisfy the above requirements. The provisions of Specification 3.0.3 are not applicable.

3/4.9.11 CONTROL ROD REMOVAL

SINGLE CONTROL ROD REMOVAL

LIMITING CONDITION FOR OPERATION

3.9.11.1 One control rod and/or the associated control rod drive mechanism may be removed from the reactor pressure vessel provided that at least the following requirements are satisfied until the control rod and associated control rod drive mechanism are reinstalled and the control rod is fully inserted in the core.

a. The reactor mode switch is OPERABLE and locked in the Refuel position per Specification 3.9.1.

b. The source range monitors (SRM) are OPERABLE per Specification 3.9.2.

c. The SHUTDOWN MARGIN requirements of Specification 3.1.1 are satisfied, except that the control rod selected to be removed;

   1. May be assumed to be the highest worth control rod required to be assumed to be fully withdrawn by the SHUTDOWN MARGIN test, and

   2. Need not be assumed to be immovable or untrippable.

d. All other control rods in a five-by-five array centered on the control rod being removed are fully inserted and electrically disarmed.

e. All other control rods are either fully inserted or have the surrounding four fuel assemblies removed.

APPLICABILITY: CONDITION 5.

ACTION:

With the requirements of the above specification not satisfied, suspend removal of the control rod and/or associated control rod drive mechanism from the reactor pressure vessel and initiate action to satisfy the above requirements. The provisions of Specification 3.0.3 are not applicable.

3/4.9.6 COMMUNICATIONS

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

3/4.9.7 CRANE AND HOIST OPERABILITY

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

3/4.9.8 CRANE TRAVEL-SPENT FUEL STORAGE POOL

The restriction on movement of loads in excess of the nominal weight of a fuel element over irradiated fuel assemblies ensures that no more than the contents of one fuel assembly will be ruptured in the event of a fuel handling accident. This assumption is consistent with the activity release assumed in the accident analyses. All fuel loaded into the Edwin I. Hatch Nuclear Plant spent fuel pool shall have an uncontrolled lattice K_∞ less than or equal to the limit for high density fuel racks described in the "General Electric Standard Application for Reactor Fuel" (GESTAR II), NEDE-24011-P-A-8. Alternatively, fuel not described in GESTAR II shall have been analyzed with another NRC approved methodology to ensure conformity to the FSAR design basis for fuel in the spent fuel racks.

3/4.9.9 WATER LEVEL-REACTOR VESSEL

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

3/4.9.10 WATER LEVEL-SPENT FUEL STORAGE POOL

The minimum water level in the spent fuel pool shall be maintained at least 21 feet above the top of the upper tie plates of the irradiated fuel assemblies seated in the spent fuel pool racks. This minimum level ensures removal of at least 98.6% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. This water depth is consistent with the assumptions for the fuel handling accident analysis outlined in Regulatory Guide 1.25 and the requirements in Standard Review Plan 15.7.4 for radiological releases resulting from that accident.

3/4.9.11 CONTROL ROD REMOVAL

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

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 172 TO FACILITY OPERATING LICENSE DPR-57

AND AMENDMENT NO. 112 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 June 13, 1991, Georgia Power Company, et al. (the licensee) proposed the following changes to the Edwin I. Hatch Nuclear Plant, Units 1 and 2, Technical Specifications (TSs). Proposed Change 1 would revise (a) Unit 1 TS 3.10.D, Unit 2 TS 3/4.9.10 and their associated bases to require at least 21 feet of water above irradiated fuel assemblies seated in the spent fuel pool (SFP) fuel storage racks; and (b) Unit 1 TS 4.10.D to require surveillance of the SFP water level every 7 days consistent with Unit 2 TSs and the BWR-4 Standard TSs. Proposed Change 2 would revise Unit 1 TS Tables 3.2-11 and 4.2-11 to require that the post-LOCA (Loss of Coolant Accident) radiation monitors be calibrated at least once every 18 months. Proposed Change 3 would correct administrative errors in Unit 2 TS Tables 3.3.2-1 and 3.8.2.6-1.

2.0 EVALUATION

a. Proposed Change 1

The licensee stated that the as-built configuration of the Hatch SFPs does not support the current TS requirement of at least 23 feet of water above the irradiated fuel assemblies seated in the SFP high density storage racks, when water depth is measured from the top of the stored assembly upper tie plate to the low level SFP alarm. With these reference points, the Hatch SFPs cannot support the 23 foot requirement but can support only 21 feet. In actuality, there has been no physical change to the Hatch SFPs but rather there has been a recognition that water depth should be measured from the top of the upper tie plate to the low level SFP alarm. The licensee has evaluated the relevant fuel handling accident analyses using 21 feet from the top of the fuel assembly upper tie plate to the low level SFP alarm. The licensee's calculated thyroid exposures at the exclusion area boundary (EAB) and low population zone (LPZ) were 0.358 and 0.370 rem, respectively, for 21 feet. In its dose calculations, the licensee assumed (1) 0.34% of the total radioactive iodine in the failed fuel rods is released at the time of the accident, and (2) the decontamination factor (DF) of 71.6 with the pool water depth of 21 feet. The licensee calculated the DF using the methodology provided in a reference in Standard Review Plan (SRP) Section 15.7.4 and the staff finds it to be acceptable. However, item (1) above is not acceptable to the staff because a technical basis was not provided for deviating from Regulatory Position c.1.d in SRP Section 15.7.4 which delineates the total radioactive iodine released from the failed fuel rods at the time of the accident to be 10%.

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In NUREG-0411, "Safety Evaluation Report related to Operation of Edwin I. Hatch Nuclear Plant, Unit No. 2," dated June 1978, the staff's calculated offsite radiation doses to the thyroid at EAB and LPZ were 29 rem each using the pool water DF of 100 (23 feet water depth) and 10% radioiodine gap release. With a pool water DF of 71.6 (21 feet water depth) and the same radioiodine gap release, the resulting offsite thyroid would be 41 rem. The whole-body doses due to the noble gases are not affected by the pool water depth.

The NRC staff finds that the calculated offsite doses (41 rem) at EAB and LPZ due to a fuel handling accident at the Hatch station are well within 10 CFR Part 100 dose reference values and meet the dose guidelines (75 rem) provided in SRP Section 15.7.4.

The licensee also stated that changing the Unit 1 surveillance requirement for determining the SFP water level from daily to weekly will not significantly reduce the margin of safety because the radiological consequences of a fuel handling accident in the SFP will not exceed the acceptance limit for that event when the pool level is as low as its low level alarm.

Based on its review of the licensee's submittal, the NRC staff agrees with the licensee's conclusions and finds that Proposed Change 1 will have no adverse impact on safety and will not pose an undue risk to public health and safety. Therefore, it is acceptable.

b. Proposed Change 2

The licensee stated that the post-LOCA radiation monitors installed at Hatch Units 1 and 2 are considered a passive portion of radiation control because they do not actively handle or control radiation hazards but only provide alarm capability. Hatch Unit 1 TSs currently require that these monitors be calibrated at least once every 6 months while identical monitors at Unit 2 are required to be calibrated once every 18 months. Furthermore, the licensee stated that field data of Hatch Unit 1 monitors were reviewed over a 36-month period and the monitors were found not to drift outside their acceptance range.

Based on its review of the licensee's submittal, the NRC staff agrees with the licensee's conclusions and finds that Proposed Change 2 will have no adverse impact on safety and will not pose an undue risk to public health and safety. Therefore, it is acceptable.

c. Proposed Change 3

The licensee stated that this change would correct administrative errors in Unit 2 TS Tables 3.3.2-1 and 3.8.2.6-1. A footnote was inadvertently moved from Item 4.g of Table 3.3.2-1 to Item 4.h and two Items, f.16 and f.17, of Table 3.8.2.6-1 were inadvertently retained in Amendment No. 109 issued March 18, 1991.

Based on its review of the licensee's submittal, the NRC staff agrees with the licensee's conclusions and finds that Proposed Change 3 will have no adverse impact on safety and will not pose an undue risk to public health and safety. Therefore it is 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 requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and change surveillance requirements. 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 (56 FR 29276). 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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Date: August 28, 1991