DCD-016

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FEB 02 1983

Dockets Nos. 50-321
and 50-366

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

Dear Mr. Beckham:

The Commission has issued the enclosed Amendments Nos. 92 and 30 to Facility Operating Licenses Nos. DPR-57 and NPF-5, respectively for the Edwin I. Hatch Nuclear Plant Units Nos. 1 and 2. The amendments consist of changes to the Technical Specifications (TSs) in response to your applications dated March 10, 1982, and April 5, 1982, and your two applications dated January 3, 1983.

The amendments revise the TSs for both Hatch Unit No. 1 and Unit No. 2 to modify the numerical values of the reactor water levels and the setpoints for reactor water levels measured by the shroud water level instruments to make them consistent with a change in the zero reference level.

The amendments also revise the TSs for Hatch Unit No. 1 to: 1) establish an upper limit for the rod block monitor high flux trip setting; 2) increase the pressure setting of the safety relief valve tailpipe pressure switches; and 3) add a newly installed torus access hatch to the list of testable containment penetrations.

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

Sincerely,

"ORIGINAL SIGNED BY:"
George W. Rivenbark, Sr. Project Manager
Operating Reactors Branch #4
Division of Licensing

Enclosures:
1. Amendments Nos. 92 and 30
2. Safety Evaluation
3. Notice

cc w/enclosures: See next page

Concur Quest + FRN only

OFFICE	ORB#4:DL	ORB#4:DL	C-ORB#4:DL	AD:OR:DL	OELD		
SURNAME	RIngram	GRivenbark	JStolz	GLainas	Goddard		
DATE	1/28/83	1/28/83:cb	1/28/83	1/ /83	1/31/83		

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

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

DOCKET NO. 50-321

EDWIN I. HATCH NUCLEAR PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 92
License No. DPR-57

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

   A. The applications for amendment by Georgia Power Company, et al., (the licensee) dated March 10, 1982, April 5, 1982, and January 3, 1983, comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the applications, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Facility Operating License No. 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. 92 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. Within 90 days after the effective date of this amendment, or such later time as the Commission may specify, the Licensee shall satisfy any applicable requirement of P. L. 97-425 related to pursuing an agreement with the Secretary of Energy for the disposal of high-level radioactive waste and spent nuclear fuel.

4. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: February 2, 1983

ATTACHMENT TO LICENSE AMENDMENT NO. 92

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 a vertical line indicating the area of change.

Remove	Insert
3.2-11	3.2-11
3.2-16	3.2-16
3.2-22	3.2-22
3.2-61	3.2-61
3.6-9a	3.6-9a
3.7-21	3.7-21
3.7-22	3.7-22*

*Overleaf page provided for document completeness.

TABLE 3.2-5

INSTRUMENTATION WHICH INITIATES OR CONTROLS THE LPCI MODE OF RHR

Ref. No. (a)	Instrument	Trip Condition Nomenclature	Required Operable Channels Per Trip System (b)	Trip Setting	Remarks
1	Reactor Water Level (Yarway)				
		Low Low Low (LL3)	2	≥ - 146.5 inches	Initiates LPCI mode of RHR
2	Drywell	High	2	≤ 2 psig	Initiates LPCI mode of RHR
3	Reactor Pressure	High (Shutdown Cooling Mode)	1	≤ 135 psig	With primary containment isolation signal, closes RHR (LPCI) inboard motor operated injection valves
		Low	2	≤ 335 psig	Permissive to close Recirculation Discharge Valve and Bypass Val
		Low	2	≤ 500 psig	Permissive to open LPCI injection valves
4	Reactor Water Level (Shroud Level Indicator)		1	≥ - 203.5 inches	Acts as permissive to divert some LPCI flow to containment spray
5	LPCI Cross Connect Valve Open Annunciator	N/A	1	Valve not closed	Initites annunciator when valve is not closed

Table 3.2-7 (Continued)

Ref. No. (a)	Instrument	Trip Condition Nomenclature	Required Operable Channels Per Trip System (b)	Trip Setting	Remarks
3	APRM	Downscale	2(e)	≥ 3/125 of full scale	Not required while performing low power physics test at atmospheric pressure during or after refueling at power levels not to exceed 5 MWt.
		12% Flux	2(e)	≤ 12/125 of full scale	This function is bypassed when the Mode Switch is placed in the RUN position.
		High Flux	2(e)	≤ 0.66W + 42%	W is the loop recirculation flow rate in percent of rated. Trip level setting is in percent of rated power. Not required while performing low power physics tests at atmospheric pressure during or after refueling at power levels not to exceed 5 MWt.
4	RBM	Inoperative	1(e)(f) (g)(h)	Not applicable	Inoperative trip produced by switch not in operate, circuit boards not in circuit, fails to null, less than required number of LPRM inputs for rod selected.
		Downscale	1(e)(f) (g)(h)	≥ 3/125 of full scale	
		High Flux	1(e)(f)	≤ 0.66W + 41% Not to exceed 107%	W is the loop recirculation flow rate in percent of rated. Trip level setting is in percent of rated thermal power.

TABLE 3.2-11

INSTRUMENTATION WHICH PROVIDES SURVEILLANCE INFORMATION

Ref. No. (a)	Instrument (b)	Required Operable Instrument Channels	Type and Range	Action	Remarks
1	Reactor Water Level (GE/MAC	1	Recorder	(c)	(d)
		2	Indicator 0 to 60"	(c)	(d)
2	Shroud Water Level	1	Recorder	(c)	(d)
		1	Indicator -317" to -17"	(c)	(d)
3	Reactor Pressure	1	Recorder	(c)	(d)
		2	Indicator 0 to 1200 psig	(c)	(d)
4	Drywell Pressure	2	Recorder -5 to +80 psig	(c)	(d)
5	Drywell Temperature	2	Recorder 0 to 500°F	(c)	(d)
6	Suppression Chamber Air Temperature	2	Recorder 0 to 500°F	(c)	(d)
7	Suppression Chamber Water Temperature	2	Recorder 0 to 250°F	(c)	(d)
8	Suppression Chamber Water Level	2	Indicator 0 to 300"	(c)	(d)
		2	Recorder 0 to 30"	(c)(e)	(d)
9	Suppression Chamber Pressure	2	Recorder -5 to +80 psig	(c)	(d)
10	Rod Position Information System (RPIS)	1	28 Volt Indicating Lights	(c)	(d)
11	Hydrogen and Oxygen Analyzer	1	Recorder 0 to 52	(c)	(d)
12	Post LOCA Radiation Monitoring System	1	Recorder	(c)	(d)
			Indicator 1 to 10^6 R/hr	(c)	(d)
13	Drywell/Suppression Chamber Differential Pressure	2	Recorder -0.5 to + 2.5 psid	(c)(e)	(d)
14	a) Safety/Relief Valve Position Primary Indicator	1	Pressure Switch 4-100 psig	(f)	
	b) Safety/Relief Valve Position Secondary Indicator	1	Temperature element 0-600°F	(f)	

3.2.E.3. Reactor Pressure Low (Continued)

jection valves. The valves do not open, however, until reactor pressure falls below the discharge head of LPCI.

4. Reactor Water Level (Shroud Level Indicator)

A reactor water level $\geq$ 203.5 inches below instrument zero is indicative that LPCI has made progress in reflooding the core. A simultaneous high drywell pressure trip indicates the need for containment cooling. The $\geq$ 203.5 inch setpoint acts as a permissive for manual diversion for some of the LPCI flow to containment spray.

5. LPCI Cross Connect Valve Open Annunciator

With the modified LPCI arrangement, the cross connect valve status was changed from normally open to normally closed. Inadvertent opening of this valve could negate the LPCI system injection when needed. The annunciator will alarm when the LPCI cross connect valve is not fully closed.

6. RHR (LPCI) Pump Discharge Pressure Interlocks

A pressure $\geq$ 100 psig on the RHR pump discharge indicates that the pump has started successfully. The setpoint provides a permissive signal to ADS which allows ADS initiation if other requirements are met.

7. RHR (LPCI) Pump Flow (Δp Switch) Low

A flow switch is provided downstream of each RHR pump to indicate the condition of each pump. To protect the pumps from overheating at low flow rates a minimum flow bypass line, which routes water from the pump discharge to the suppression chamber, is provided for each pair of pumps. A single motor-operated valve controls the condition of each bypass line. The minimum flow bypass valve automatically opens upon sensing low flow in the discharge lines from both pumps of the associated pump pair. The valve automatically closes whenever the flow from either of the associated main system pumps is above the low flow setting.

8. RHR (LPCI) Pump Start Timers

If normal AC power is available, four pumps automatically start without delay. If normal AC power is not available, one pump starts without delay as soon as power becomes available from the standby sources. The other three pumps start after a 10-second delay. The timer provides correct sequencing of the loads to the diesel generator.

4.6.H. Relief/Safety Valves (Continued)

5. Operability of Tail Pipe Pressure Switches

a. Functional Test:

2. At each scheduled outage greater than 72 hours during which entry is made into the primary containment, if not performed within the previous 31 days.

b. Calibration and verifying the setpoint to be 85,+15,-5 psig at least once per 18 months.

I. Jet Pumps

Whenever the reactor is in the Start & Hot Standby or Run Mode with both recirculating pumps operating, all jet pumps shall be operable. If it is determined that a jet pump is inoperable, an orderly shutdown shall be initiated and the reactor shall be in the Cold Shutdown Conditions within 24 hours.

I. Jet Pumps

Whenever both recirculating pumps are operating with the reactor in the Start & Hot Standby or Run Mode, jet pump operability shall be checked daily by verifying that the following conditions do not occur simultaneously.

1. The two recirculation loops have a flow imbalance of 15% or more when the pumps are operated at the same speed.

Table 3.7-2

Testable Penetrations with Double O-Ring Seals

Penetration No.	Penetration Description	Notes
X-1 A&B	Equipment Hatch	(1) (2) (4) (6)
X-2	Personnel Lock	(1) (4) (7) (8)
X-4	Head Access Hatch	(1) (2) (4) (6)
X-6	CRD Removal Hatch	(1) (2) (4) (6)
X-35-A through X35-E	TIP System	(1) (2) (4) (6)
X-43	Drywell Test	(1) (2) (4) (6)
X-200 A,B,&C	Suppression Chamber Access Manhole	(1) (2) (4) (6)
X-218 A&B	Construction Drain	(1) (2) (4) (6)

Table 3.7-3

Testable Penetrations with Testable Bellows

Penetration Number	Penetration Description	Notes
X-7A	Primary Steamline 'A'	(1) (2) (4) (6)
X-7B	Primary Steamline 'B'	(1) (2) (4) (6)
X-7C	Primary Steamline 'C'	(1) (2) (4) (6)
X-7D	Primary Steamline 'D'	(1) (2) (4) (6)
X-8	Steamline Condensate Drain	(1) (2) (4) (6)
X-9A	Feedwater Line 'A'	(1) (2) (4) (6)
X-9B	Feedwater Line 'B'	(1) (2) (4) (6)
X-10	Steam to RCIC Turbine	(1) (2) (4) (6)
X-11	Steam Line to HPCI Turbine	(1) (2) (4) (6)
X-12	RHRS Shutdown Cooling Suction	(1) (2) (4) (6)
X-13A	RHR LPCI to Reactor	(1) (2) (4) (6)
X-13B	RHR LPCI to Reactor	(1) (2) (4) (6)
X-14	Reactor Water Cleanup Line	(1) (2) (4) (6)
X-16A	Core Spray to Reactor	(1) (2) (4) (6)
X-16B	Core Spray to Reactor	(1) (2) (4) (6)
X-17	RPV Head Spray	(1) (2) (4) (6)
X-201A through X-201H	Drywell Suppression Chamber to Vent Line	(1) (2) (4) (6)

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

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

DOCKET NO. 50-366

EDWIN I. HATCH NUCLEAR PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 30
License No. NPF-5

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

   A. The application for amendment by Georgia Power Company, et al., (the licensee) dated March 10, 1982, 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;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

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

Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 30 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. Within 90 days after the effective date of this amendment, or such later time as the Commission may specify, the Licensee shall satisfy any applicable requirement of P. L. 97-425 related to pursuing an agreement with the Secretary of Energy for the disposal of high-level radioactive waste and spent nuclear fuel.

4. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: February 2, 1983

ATTACHMENT TO LICENSE AMENDMENT NO. 30

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 a vertical line indicating the area of change. The overleaf pages are provided to maintain document completeness.

Remove	Insert
3/4 3-28	3/4 3-28
B3/4 3-6	B3/4 3-6

TABLE 3.3.3-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

TRIP FUNCTION	MINIMUM NUMBER OPERABLE CHANNELS PER TRIP SYSTEM	APPLICABLE OPERATIONAL CONDITIONS #
3. HIGH PRESSURE COOLANT INJECTION SYSTEM		
a. Reactor Vessel Water Level - Low Low (2B21-N031A,B,C,D)	2	1, 2, 3
b. Drywell Pressure - High (2E11-N011 A,B,C,D)	2	1, 2, 3
c. Condensate Storage Tank Level-Low(2E41-N002, 2E41-N003)	2[b][c]	1, 2, 3
d. Suppression Chamber Water Level-High (2E41-N015A,B)	2[b][c]	1, 2, 3
e. Logic Power Monitor (2E41-K1)	1[a]	1, 2, 3
f. Reactor Vessel Water Level-High (2B21-N017 B,D)	2	1, 2, 3
4. AUTOMATIC DEPRESSURIZATION SYSTEM		
a. Drywell Pressure - High (2E11-N011A,B,C,D)	2	1, 2, 3
b. Reactor Vessel Water Level - Low Low Low (2B21-N031 A,B,C,D)	2	1, 2, 3
c. ADS Timer (2B21-K5A,B)	1	1, 2, 3
d. Reactor Vessel Water Level-Low (Permissive)(2B21-N042A,B)	1	1, 2, 3
e. Core Spray Pump Discharge Pressure - High (Permissive) (2E21-N008A,B; 2E21-N009A,B)	2	1, 2, 3
f. RHR (LPCI MODE) Pump Discharge Pressure - High (Permissive) (2E11-N016A,B,C,D; 2E11-N020A,B,C,D)	2/loop	1, 2, 3
g. Control Power Monitor (2B21-K1A,B)	1/bus[a]	1, 2, 3

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

(b) Provides signal to HPCI pump suction valves only.

(c) When either channel of the automatic transfer logic is inoperable, align HPCI pump suction to the suppression pool.

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

TABLE 3.3.3-2

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. CORE SPRAY SYSTEM		
a. Reactor Vessel Water Level - Low Low Low	$\geq$-146.5 inches*	$\geq$-146.5 inches*
b. Drywell Pressure - High	$\leq$2 psig	$\leq$2 psig
c. Reactor Steam Dome Pressure - Low	$\leq$500 psig	$\leq$500 psig
d. Logic Power Monitor	NA	NA
2. LOW PRESSURE COOLANT INJECTION MODE OF RHR SYSTEM		
a. Drywell Pressure - High	$\leq$2 psig	$\leq$2 psig
b. Reactor Vessel Water Level - Low Low Low	$\geq$-146.5 inches*	$\geq$-146.5 inches*
c. Reactor Vessel Shroud Level - High	$\geq$-203.5 inches*	$\geq$-203.5 inches*
d. Reactor Steam Dome Pressure-Low	$\leq$500 psig	$\leq$500 psig
e. Reactor Steam Dome Pressure-Low	$\leq$335 psig	$\leq$335 psig
f. RHR Pump Start - Time Delay Relay		
1) Pumps A, B and D	10 $\pm$ 1 seconds	10 $\pm$ 1 seconds
2) Pump C	0.5 $\pm$ 0.5 seconds	0.5 $\pm$ 0.5 seconds
g. Logic Power Monitor	NA	NA

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

TABLE 3.3.3-2

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. CORE SPRAY SYSTEM		
a. Reactor Vessel Water Level - Low Low Low	≥-146.5 inches*	≥-146.5 inches*
b. Drywell Pressure - High	≤2 psig	≤2 psig
c. Reactor Steam Dome Pressure - Low	≤500 psig	≤500 psig
d. Logic Power Monitor	NA	NA
2. LOW PRESSURE COOLANT INJECTION MODE OF RHR SYSTEM		
a. Drywell Pressure - High	≤2 psig	≤2 psig
b. Reactor Vessel Water Level - Low Low Low	≥-146.5 inches*	≥-146.5 inches*
c. Reactor Vessel Shroud Level - High	≥-203.5 inches*	≥-203.5 inches*
d. Reactor Steam Dome Pressure-Low	≤500 psig	≤500 psig
e. Reactor Steam Dome Pressure-Low	≤335 psig	≤335 psig
f. RHR Pump Start - Time Delay Relay		
1) Pumps A, B and D	10 ± 1 seconds	10 ± 1 seconds
2) Pump C	0.5 ± 0.5 seconds	0.5 ± 0.5 seconds
g. Logic Power Monitor	NA	NA

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

MONITORING INSTRUMENTATION (Continued)

FIRE DETECTION INSTRUMENTATION (Continued)

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

3/4.3.7 TURBINE OVERSPEED PROTECTION SYSTEM

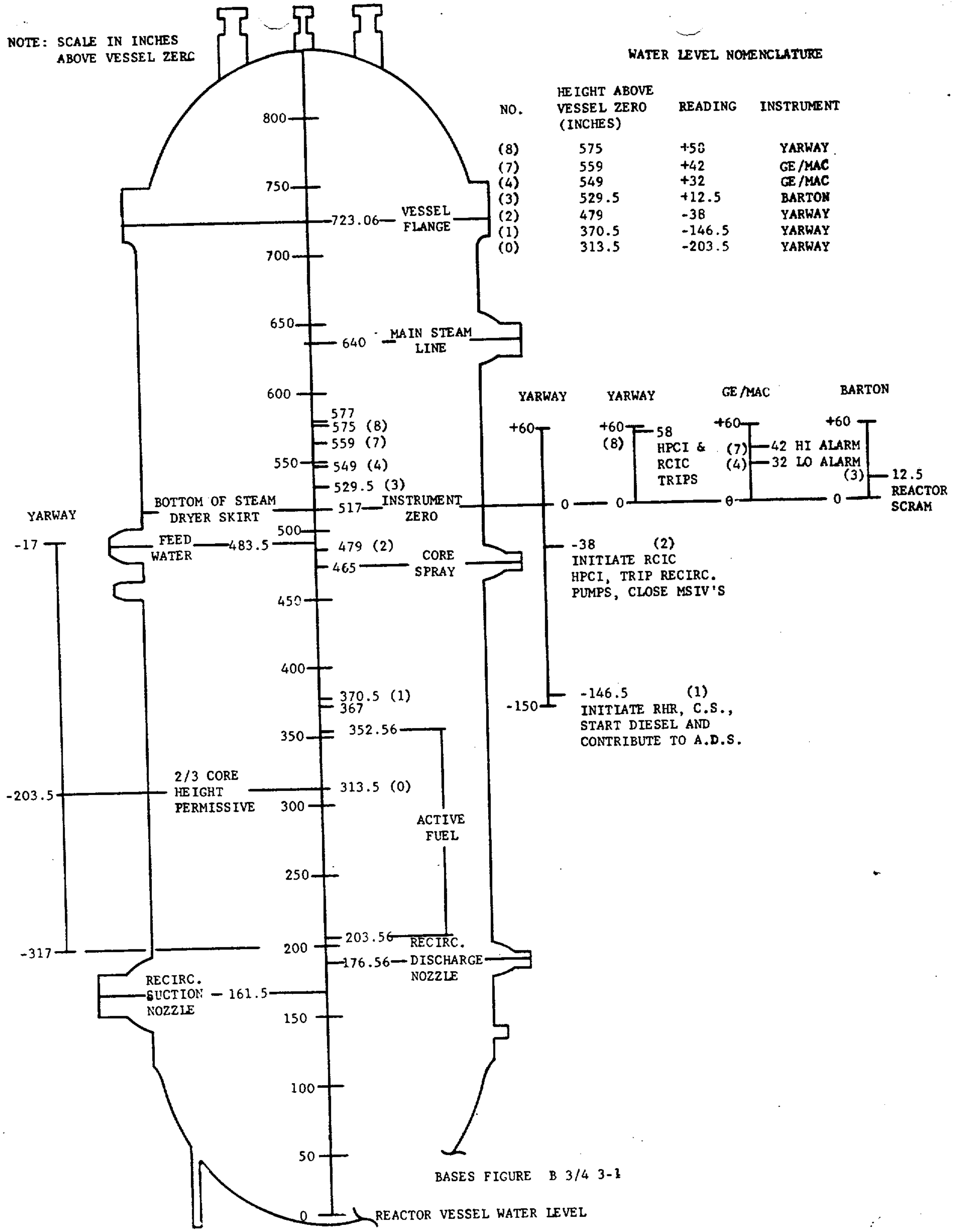
This specification is provided to ensure that the turbine overspeed protection system instrumentation and the turbine speed control valves are OPERABLE and will protect the turbine from excessive overspeed. Protection from turbine excessive overspeed is required since excessive overspeed of the turbine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

3/4.3.8 DEGRADED STATION VOLTAGE PROTECTION INSTRUMENTATION

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

WATER LEVEL NOMENCLATURE

NO.	HEIGHT ABOVE VESSEL ZERO (INCHES)	READING	INSTRUMENT
(8)	575	+58	YARWAY
(7)	559	+42	GE/MAC
(4)	549	+32	GE/MAC
(3)	529.5	+12.5	BARTON
(2)	479	-38	YARWAY
(1)	370.5	-146.5	YARWAY
(0)	313.5	-203.5	YARWAY

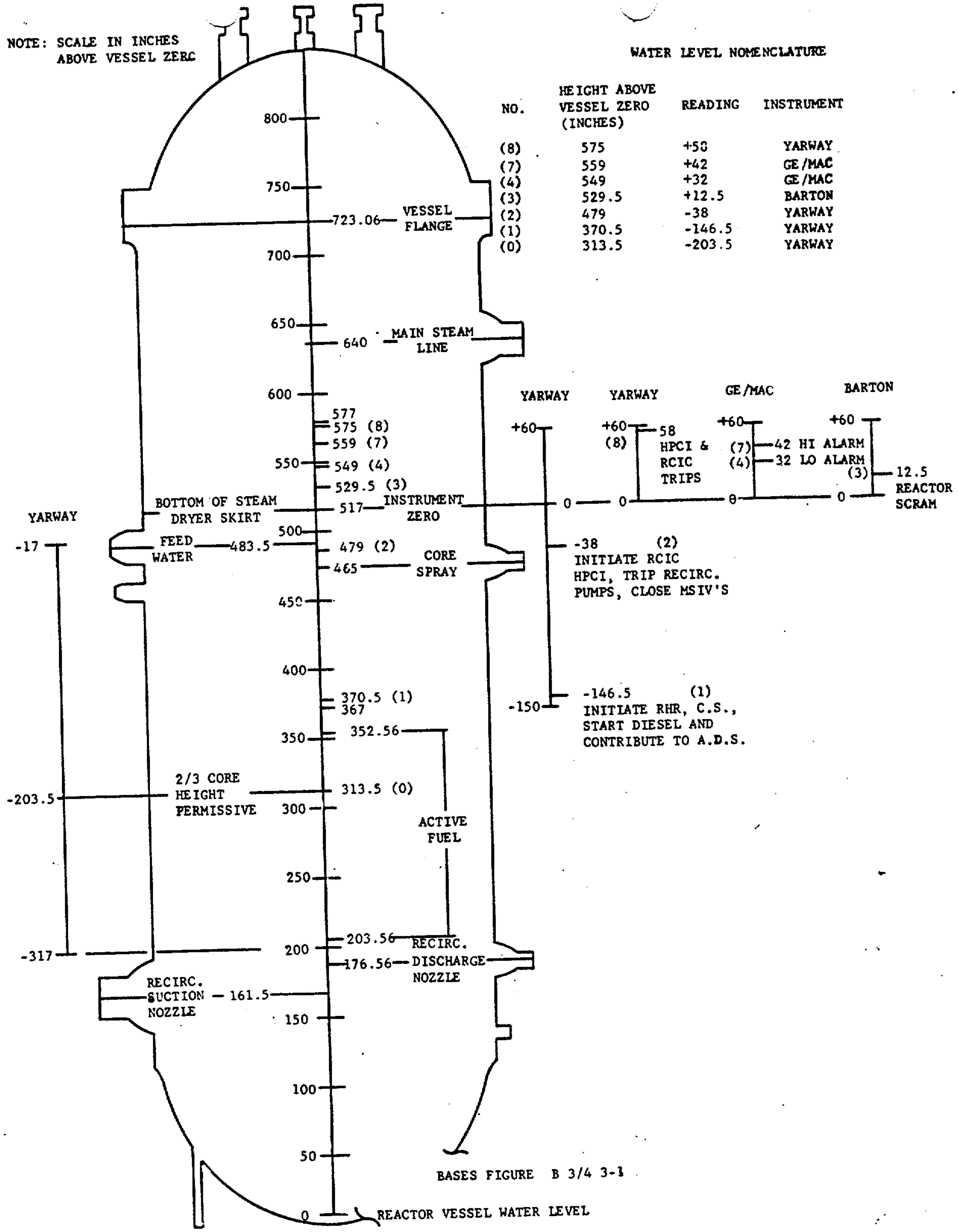


BASES FIGURE B 3/4 3-1

REACTOR VESSEL WATER LEVEL

WATER LEVEL NOMENCLATURE

NO.	HEIGHT ABOVE VESSEL ZERO (INCHES)	READING	INSTRUMENT
(8)	575	+58	YARWAY
(7)	559	+42	GE/MAC
(4)	549	+32	GE/MAC
(3)	529.5	+12.5	BARTON
(2)	479	-38	YARWAY
(1)	370.5	-146.5	YARWAY
(0)	313.5	-203.5	YARWAY



BASES FIGURE B 3/4 3-1

REACTOR VESSEL WATER LEVEL

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 92 TO FACILITY OPERATING LICENSE NO. DPR-57

AND AMENDMENT NO. 30 TO FACILITY OPERATING LICENSE NO. NPF-5

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

EDWIN I. HATCH NUCLEAR PLANT, UNITS NOS. 1 & 2
DOCKETS NOS. 50-321 AND 50-366

By letter dated March 10, 1982, Georgia Power Company (GPC or the licensee) proposed revisions to the Technical Specifications (TSs) for both Hatch Unit No. 1 and Unit No. 2 to change the numerical values of the reactor water levels and the setpoints for reactor water levels measured by the shroud water level instruments to make them consistent with a change in the zero reference level. The change is required in order to make the TSs consistent with previous modification of the reactor water level instruments to utilize a common zero reference level pursuant to NUREG-0737, Item II.K.3.27. In that modification, the zero water level references for the instruments measuring the reactor shroud water level were changed from a level near the bottom head of the reactor vessel to that level near the bottom of the steam drier skirt that is used as the common zero level reference by all of the other reactor water level instruments. The changes to the TSs involve only numerical changes. The actual levels at which protective functions or permissive signals are initiated are unchanged. On the basis of the above discussion, we conclude that the proposed change is acceptable.

By letters, as noted below, GPC proposed revisions to the Hatch Unit No. 1 TSs as follows:

1. Letter dated April 5, 1982, proposed to establish an upper limit for the rod block monitor high flux trip setting.

2. Letter dated January 3, 1983, proposed to increase the pressure setting of the Safety Relief Valve (SRV) tailpipe pressure switches.

3. Letter dated January 3, 1983, proposed to add a newly installed torus access hatch to the list of testable penetrations.

Our evaluation of these proposals and our conclusions are discussed below.

Power Limit for Rod Block Monitor Trip

The current Hatch Unit No. 1 TS Table 3.2-7 requires a high flux trip setting for the rod block monitor at a percentage of actual thermal power that is equal to 0.66 W + 41% where W is the loop recirculation flow. The current TS does not specify an upper limit for this trip setting. The licensee has requested in Attachment 3 to their letter dated April 5, 1982, that an upper limit of 107% rated power be placed on this rod block monitor trip setting.

We have reviewed the proposal and found that the fuel thermal margins are not reduced by this change, none of the present rod block or reactor setpoints are affected and that the 107% power limit provides additional protection to the fuel for an above rated core flow condition. We conclude for the reasons stated above that this change is acceptable.

Safety Relief Valve Tailpipe Pressure Switch Setting

By letter dated January 3, 1983, GPC proposed a change to Section 4.6.H of the TSs. The change would increase the setpoint of the tailpipe pressure switch of each main steam SRV from 30 ± 5 psig to 85, +15, -5 psig.

The licensee requested this change because it had been determined that the setpoint was not high enough to provide correct information on the "open" or "closed" status of the SRV to a reactor operator following a loss-of-coolant accident (LOCA). The revised setpoint accounts properly for an increase in primary containment pressure to the design value of 65 psig, combined with a static head pressure in the tailpipe of 10 psig caused by possible elevated water level in the torus. The setpoint also includes an allowance of 10 psi for inaccuracies and margin to prevent spurious SRV "open" indications during a LOCA. On the basis of the above, we conclude that the 85 psig setpoint can provide more reliable information to the operator during a LOCA and is acceptable.

New Torus Access Hatch

The Hatch Unit No. 1 facility is currently in a torus modification/ refueling outage. During this outage, a new torus access hatch was added to containment. The licensee's letter of January 3, 1983, proposed a revision to TS Table 3.7-2, "Testable Penetrations with Double O-Ring Seals". This revision would identify the new containment penetration as being subject to local leak rate testing as required by Appendix J to 10 CFR Part 50.

The new torus access hatch has been designated penetration number X-200C, suppression chamber access manhole. This containment penetration has been equipped with double O-ring seals so it can be locally (Type B) leak tested in accordance with Appendix J to 10 CFR Part 50. Table 3.7-2 of the Hatch TSs has been modified to include penetration X-200C. This table identifies those containment penetrations that will be locally (Type B) leak tested.

The probability of occurrence and the consequences of an accident or malfunction of equipment important to safety will not increase due to this change because the new torus hatch is constructed and sealed to meet the requirements of 10 CFR 50.55a as specified in ASME Code Case N236.

Modification to TS Table 3.7-2 is appropriate to identify the penetrations that will be locally leak tested in accordance with Appendix J. Therefore, we conclude that the proposed change is acceptable.

Environmental Consideration

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

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendments do not involve a significant increase in the probability or consequences of an accident previously evaluated, do not create the possibility of an accident of a type different from any evaluated previously, and do not involve a significant reduction in a margin of safety, the amendments do not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Dated: February 2, 1983

The following NRC personnel have contributed to this Safety Evaluation: George W. Rivenbark, Charles C. Graves and Douglas Pickett.

7590-01

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKETS NOS. 50-321 AND 50-366

GEORGIA POWER COMPANY, ET AL.

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendments Nos. 92 and 30 to Facility Operating Licenses Nos. DPR-57 and NPF-5, issued to Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, and City of Dalton, Georgia, which revised Technical Specifications (TSs) for operation of the Edwin I. Hatch Nuclear Plant, Units Nos. 1 and 2 (the facility) located in Appling County, Georgia. The amendments are effective as of the date of issuance.

The amendments revise the TSs for both Hatch Unit No. 1 and Unit No. 2 to modify the numerical values of the reactor water levels and the setpoints for reactor water levels measured by the shroud water level instruments to make them consistent with a change in the zero reference level. The amendments also revise the TSs for Hatch Unit No. 1 to: 1) establish an upper limit for the rod block monitor high flux trip setting; 2) increase the pressure setting of the safety relief valve tailpipe pressure switches; and 3) add a newly installed torus access hatch to the list of testable containment penetrations.

The applications for the amendments comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendments. Prior public notice of these amendments was not required since the amendments do not involve a significant hazards consideration.

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

For further details with respect to this action, see (1) the applications for amendments dated March 10, 1982, April 5, 1982, and January 3, 1983 (two applications), (2) Amendments Nos. 92 and 30 to Licenses Nos. DPR-57 and NPF-5, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the Appling County Public Library, 301 City Hall Drive, Baxley, Georgia 31513. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland, this 2nd day of February 1983.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing