August 30, 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. 173 TO FACILITY OPERATING LICENSE DPR-57 AND AMENDMENT NO. 113 TO FACILITY OPERATING LICENSE NPF-5 - EDWIN I. HATCH NUCLEAR PLANT, UNITS 1 AND 2 (TACs 77822/77823)

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 173 to Facility Operating License No. DPR-57 and Amendment No. 113 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 October 9, 1990.

The amendments revise the TSs to reduce the trip setpoint/allowable value for the low water level scram and isolation functions approximately 10 inches.

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,

/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. 173 to DPR-57
2. Amendment No. 113 to NPF-5
3. Safety Evaluation

cc w/enclosures:
See next page

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

August 30, 1991

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

Enclosures:
1. Amendment No. 173 to DPR-57
2. Amendment No. 113 to NPF-5
3. Safety Evaluation

cc w/enclosures:
See next page

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DATED: August 30, 1991

AMENDMENT NO.173TO FACILITY OPERATING LICENSE DPR-57 - Edwin I. Hatch Nuclear Plant, Unit 1

AMENDMENT NO.113TO FACILITY OPERATING LICENSE NPF-5 - Edwin I. Hatch Nuclear Plant, Unit 2

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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 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 173
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 October 9, 1990, 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. 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. 173, 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 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Kahtn N. Jabbour 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 30, 1991

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. 113
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 October 9, 1990, 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. 113, 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 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Kalta N. Talbom 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 30, 1991

ATTACHMENT TO LICENSE AMENDMENT NO. 173

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
1.1-3	1.1-3
1.1-13	1.1-13
Figure 2.1-1	Figure 2.1-1
3.1-4	3.1-4
3.2-2	3.2-2
3.2-10	3.2-10
3.2-50	3.2-50
3.2-58	3.2-58

2.1.A.1.d. APRM Rod Block Trip Setting
This section deleted.

2.1.A.2. Reactor Vessel Water Low Level Scram Trip Setting (Level 3)

Reactor vessel water low level scram trip setting (Level 3) shall be ≥ 0.0 inches (narrow range scale).

3. Turbine Stop Valve Closure Scram

Turbine stop valve closure scram trip setting shall be ≤ 10 percent valve closure from full open. This scram is only effective when turbine steam flow is above that corresponding to 30% of rated core thermal power, as measured by turbine first stage pressure.

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

The APRM flow referenced simulated thermal power monitor scram trip setting at full recirculation flow is adjustable up to 117% of rated power for two-recirculation loop and single-recirculation loop operations. This reduced flow referenced trip setpoint will result in an earlier scram during slow thermal transients, such as the loss of 100°F feedwater heating event, than would result with the 120% fixed high neutron flux scram trip. The lower flow referenced scram setpoint therefore decreases the severity (ΔCPR) of a slow thermal transient and allows lower Operating Limits if such a transient is the limiting abnormal operational transient during a certain exposure interval in the cycle.

The APRM fixed high-high neutron flux scram trip, adjustable up to 120% of rated power for two-recirculation loop and single-recirculation loop operations, does not incorporate the time constant, but responds directly to instantaneous neutron flux. This scram setpoint scrams the reactor during fast power increase transients if credit is not taken for a direct (position) scram, and also serves to scram the reactor if credit is not taken for the flow referenced scram.

2. Reactor Vessel Water Low Level Scram Trip Setting (Level 3)

The trip setting for low level scram is above the bottom of the separator skirt, Figure 2.1-1. This level is approximately 14 feet above the top of the active fuel. This level has been used in transient analyses dealing with coolant inventory decrease. The results reported in FSAR Section 14.3 show that a scram at this level adequately protects the fuel and the pressure barrier. The designated scram trip setting is at least 22 inches below the bottom of the normal operating range and is thus adequate to avoid spurious scrams.

WATER LEVEL NOMENCLATURE

NO.	HEIGHT ABOVE VESSEL ZERO (INCHES)	READING	INSTRUMENT
(8)	573.5	+56.5	BARTON/ROSEMOUNT
(7)	559	+42	GE/MAC
(4)	549	+32	GE/MAC
(3)	517	0.0	BARTON/ROSEMOUNT
(2)	470	- 47	BARTON/ROSEMOUNT
(1)	404	- 113	BARTON/ROSEMOUNT
(0)	315	- 202	BARTON/ROSEMOUNT

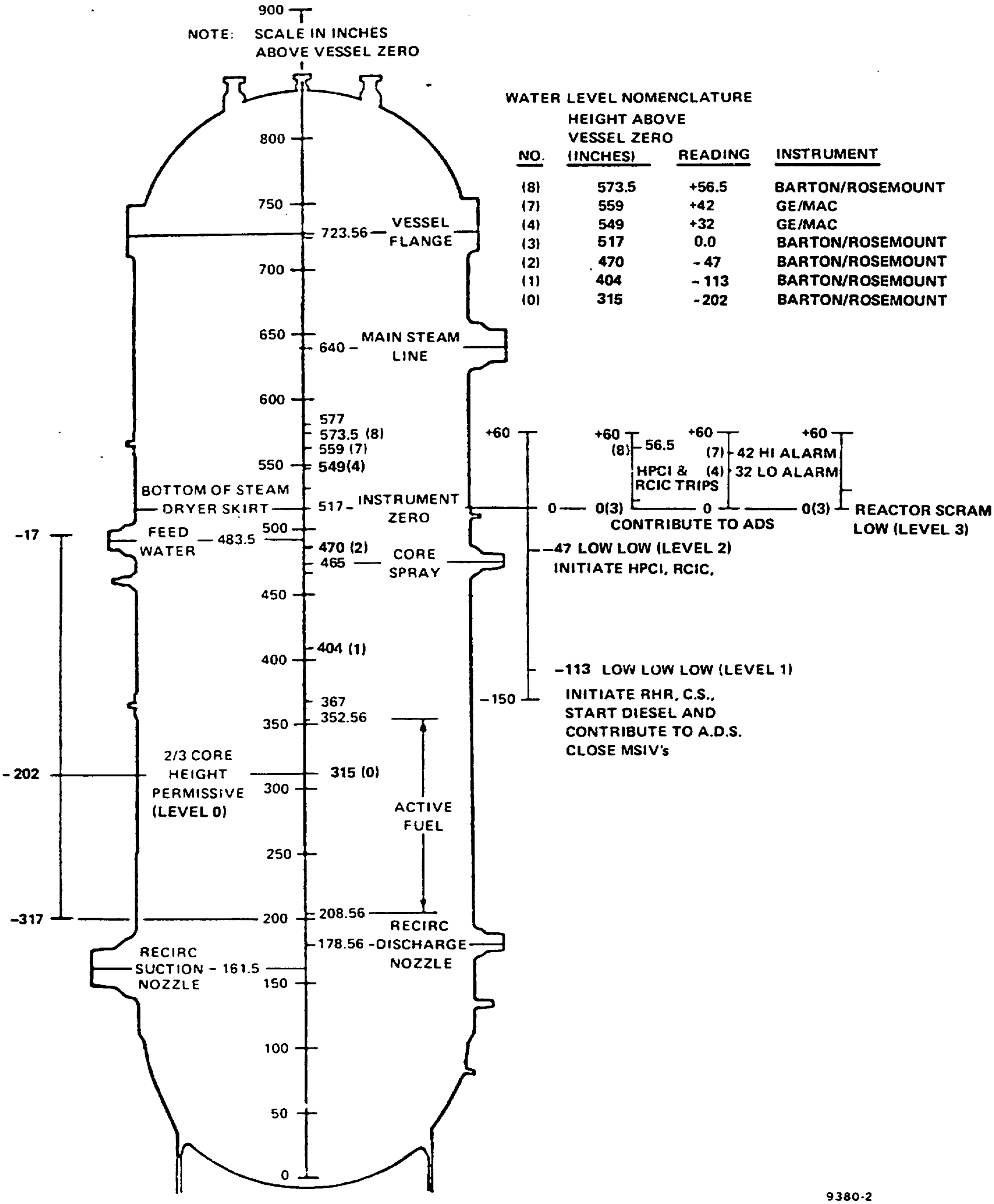


FIGURE 2.1-1
REACTOR VESSEL WATER LEVEL

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
5	High Drywell Pressure	2	≤ 1.92 psig	Not required to be operable when primary containment integrity is not required.
6	Reactor Vessel Water Level (Low) (Level 3)	2	≥ 0.0 inches	
7	Scram Discharge Volume High High Level			Permissible to bypass (initiates control rod block) in order to reset RPS when the Mode Switch is in the REFUEL or SHUTDOWN position.
	a. Float Switches	2	≤ 71 gallons	
	b. Thermal Level Sensors	2	≤ 71 gallons	
8	APRM Flow Referenced Simulated Thermal Power Monitor	2	S ≤ 0.58W+62% - 0.58 ΔW (Not to exceed 117%) Tech Spec 2.1.A.1.c(1)	See Specification 2.1.A.1.c(1) for definitions of W and ΔW.
	Fixed High High Neutron Flux	2	S ≤ 120% Power Tech Spec 2.1.A.1.c(2)	
	Inoperative	2	Not Applicable	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.

Table 3.2-1

INSTRUMENTATION WHICH INITIATES REACTOR VESSEL AND PRIMARY CONTAINMENT ISOLATION

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)
1	Reactor Vessel Water Level	Low (Level 3) Narrow Range	2	$\geq$ 0.0 inches	Initiate an orderly shutdown and achieve the Cold Shutdown Condition within 24 hours or isolate the shutdown cooling system.	Initiates Group 2 & 6 isolation.
		Low Low (Level 2)	2	$\geq$-47 inches	Initiate an orderly shutdown and achieve the Cold Shutdown Condition within 24 hours.	Starts the SGTS, initiates Group 5 isolation, and initiates secondary containment isolation.
		Low Low Low (Level 1)	2	$\geq$-113 inches	Initiate an orderly shutdown and achieve the Cold Shutdown Condition within 24 hours.	Initiates Group 1 isolation.
2	Reactor Vessel Steam Dome Pressure (Shutdown Cooling Mode)	Low Permissive	1	$\leq$145 psig	Isolate shutdown cooling.	Isolates the shutdown cooling suction valves of the RHR system.
3	Drywell Pressure	High	2	$\leq$1.92 psig	Initiate an orderly shutdown and achieve the Cold Shutdown Condition within 24 hours.	Starts the standby gas treatment system, initiates Group 2 isolation and secondary containment isolation.

Table 3.2-4

INSTRUMENTATION WHICH INITIATES OR CONTROLS ADS

Ref. No. (a)	Instrument	Trip Condition Nomenclature	Required Operable Channels per Trip System (b)	Trip Setting	Remarks
1.	Reactor Vessel Water Level	Low (Level 3)	1	≥0.0 inches	Confirms low level, ADS permissive
	Reactor Vessel Water Level	Low Low Low (Level 1)	2	≥-113 inches	Permissive signal to ADS timer
2.	Drywell Pressure	High	2	≤1.92 psig	Permissive signal to ADS timer
3.	RHR Pump Discharge Pressure	High	2	≥112 psig	Permissive signal to ADS timer
4.	CS Pump Discharge Pressure	High	2	≥137 psig	Permissive signal to ADS timer
5.	Auto Depressurization Low Water Level Timer		2	≤13 minutes	Bypasses high drywell pressure permissive upon sustained Level 1
6.	Auto Depressurization Timer		1	120 ± 12 seconds	With Level 3 and Level 1 and high drywell pressure and CS or RHR pump at pressure, timing sequence begins. If the ADS timer is not reset it will initiate ADS.
7.	Automatic Blowdown Control Power Failure Monitor		1	Not applicable	Monitors availability of power to logic system

a. The column entitled "Ref. No." is only for convenience so that a one-to-one relationship can be established between items in Table 3.2-4 and items in Table 4.2-4.

b. Whenever any CCCS subsystem is required to be operable by Section 3.5, there shall be two operable trip systems. If the required number of operable channels cannot be met for one of the trip systems, that system shall be repaired or the reactor shall be placed in the Cold Shutdown Condition within 24 hours after this trip systems is made or found to be inoperable.

3.2. PROTECTION INSTRUMENTATION

In addition to the Reactor Protection System (RPS) instrumentation which initiates a reactor scram, protective instrumentation has been provided which initiates action to mitigate the consequences of accidents which are beyond the operators ability to control, or terminates operator errors before they result in serious consequences. This set of Specifications provides the limiting conditions for operation of the instrumentation:
(a) which initiates reactor vessel and primary containment isolation,
(b) which initiates or controls the core and containment cooling systems,
(c) which initiates control rod blocks, (d) which initiates protective action,
(e) which monitors leakage into the drywell and (f) which provides surveillance information. The objectives of these specifications are (i) to assure the effectiveness of the protective instrumentation when required by preserving its capability to tolerate a single failure of any component of such systems even during periods when portions of such systems are out of service for maintenance, and (ii) to prescribe the trip settings required to assure adequate performance. When necessary, one channel may be made inoperable for brief intervals to conduct required functional tests and calibrations.

A. Instrumentation Which Initiates Reactor Vessel and Primary Containment Isolation (Table 3.2-1)

Isolation valves are installed in those lines which penetrate the primary containment and must be isolated during a loss of coolant accident so that the radiation dose limits are not exceeded during an accident condition. Actuation of these valves is initiated by protective instrumentation shown in Table 3.2-1 which senses the conditions for which isolation is required. Such instrumentation must be available whenever primary containment integrity is required. The objective is to isolate the primary containment so that the guidelines of 10 CFR 100 are not exceeded during an accident. The events when isolation is required are discussed in Appendix G of the FSAR. The instrumentation which initiates primary system isolation is connected in a dual bus arrangement.

1. Reactor Vessel Water Level

a. Reactor Vessel Water Level Low (Level 3) (Narrow Range)

The reactor water level instrumentation is set to trip when reactor water level is approximately 14 feet above the top of the active fuel. This level is referred to as Level 3 in the Technical Specifications and corresponds to a reading of 0.0 inches on the Narrow Range scale. This trip initiates Group 2 and 6 isolation but does not trip the recirculation pumps.

b. Reactor Vessel Water Level Low Low (Level 2)

The reactor water level instrumentation is set to trip when reactor water level is approximately 9 feet above the top of the active fuel. This level is referred to as Level 2 in the Technical Specifications and corresponds to a reading of -47 inches.
This trip initiates Group 5 isolation, starts the standby gas treatment system, and initiates secondary containment isolation.

D. Instrumentation Which Initiates or Controls ADS (Table 3.2-4)

The ADS is a backup system to HPCI. In the event of failure by HPCI to maintain reactor water level, ADS will initiate depressurization of the reactor in time for LPCI and CS to adequately cool the core. Four signals are required to initiate ADS: Low water level, confirmed low water level, high drywell pressure, and either a RHR or Core Spray pump available. The simultaneous presence of these four signals will initiate a 120 second timer which will depressurize the reactor if not reset.

1. Reactor Vessel Water Level

a. Reactor Vessel Water Level Low (Level 3)

The second reactor vessel low water level initiation setting (+0.0 inches) is selected to confirm that water level in the vessel is in fact low, thus providing protection against inadvertent depressurization in the event of an instrument line (water level) failure. Such a failure could produce a simultaneous high drywell pressure. A confirmed low level is one of four signals required to initiate ADS.

b. Reactor Vessel Water Level Low Low Low (Level 1)

The reactor vessel low water level setting of -113 inches is selected to provide a permissive signal to open the relief valve and depressurize the reactor vessel in time to allow adequate cooling of the fuel by the core spray and LPCI systems following a LOCA in which the other make up systems (RCIC and HPCI) fail to maintain vessel water level. This signal is one of four required to initiate ADS.

2. Drywell Pressure High

A primary containment high pressure of $\geq$ 2 psig indicates that a breach of the nuclear system process barrier has occurred inside the drywell. The signal is one of four required to initiate the ADS.

ATTACHMENT TO LICENSE AMENDMENT NO. 113

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
2-4	2-4
3/4 3-16	3/4 3-16
3/4 3-18	3/4 3-18
3/4 3-29	3/4 3-29
B 3/4 3-6	B 3/4 3-6

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)	$\leq$ 120/125 divisions of full scale	$\leq$ 120/125 divisions of full scale
2. Average Power Range Monitor: (2C51-K605 A,B,C,D,E,F)		
a. Neutron Flux-Upscale, 15%	$\leq$ 15/125 divisions of full scale	$\leq$ 20/125 divisions of full scale
b. Flow Referenced Simulated Thermal Power-Upscale	$\leq$ (0.58 W + 59% - 0.58ΔW)** with a maximum $\leq$ 113.5% of RATED THERMAL POWER	$\leq$ (0.58 W + 62% - 0.58ΔW)** with a maximum $\leq$ 115.5% of RATED THERMAL POWER
c. Fixed Neutron Flux-Upscale, 118%	$\leq$ 118% of RATED THERMAL POWER	$\leq$ 120% of RATED THERMAL POWER
3. Reactor Vessel Steam Dome Pressure - High (2B21-N678 A,B,C,D)	$\leq$ 1054 psig	$\leq$ 1054 psig
4. Reactor Vessel Water Level - Low (Level 3) (2B21-N680 A,B,C,D)	$\geq$ 0 inches above instrument zero*	$\geq$ 0 inches above instrument zero*
5. Main Steam Line Isolation Valve - Closure (NA)	$\leq$ 10% closed	$\leq$ 10% closed
6. Main Steam Line Radiation - High (2D11-K603A,B,C,D)	$\leq$ 3 x full-power background***	$\leq$ 3 x full-power background***
7. Drywell Pressure - High (2C71-N650A,B,C,D)	$\leq$ 1.92 psig	$\leq$ 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.

***Within 24 hours prior to the planned start of the hydrogen injection test with the reactor power at greater than 20% rated power, the normal full-power radiation background level and associated trip setpoints may be changed based on a calculated value of the radiation level expected during the test. The background radiation level and associated trip setpoints may be adjusted during the test based on either calculations or measurements of actual radiation levels resulting from hydrogen injection. The background radiation level shall be determined and associated trip setpoints shall be set within 24 hours of re-establishing normal radiation levels after completion of hydrogen injection and prior to establishing reactor power levels below 20% rated power.

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.

**Within 24 hours prior to the planned start of the hydrogen injection test with the reactor power at greater than 20% rated power, the normal full power radiation background level and associated trip setpoints may be changed based on a calculated value of the radiation level expected during the test. The background radiation level and associated trip setpoints may be adjusted during the test based on either calculations or measurements of actual radiation levels resulting from hydrogen injection. The background radiation level shall be determined and associated trip setpoints shall be set within 24 hours of re-establishing normal radiation levels after completion of hydrogen injection and prior to establishing reactor power levels below 20% rated power.

TABLE 3.3.2-2 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
5. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION		
a. RCIC Steam Line Flow - High	≤ 307% of rated flow	≤ 307% of rated flow
b. RCIC Steam Supply Pressure - Low	≥ 60 psig	≥ 60 psig
c. RCIC Turbine Exhaust Diaphragm Pressure - High	≤ 20 psig	≤ 20 psig
d. Emergency Area Cooler Temperature-High	≤ 169°F	≤ 169°F
e. Suppression Pool Area Ambient Temperature High	≤ 169°F	≤ 169°F
f. Suppression Pool Area ΔT - High	≤ 42°F	≤ 42°F
g. Suppression Pool Area Temperature Timer Relays	NA	NA
h. Drywell Pressure - High	≤ 1.92 psig	≤ 1.92 psig
i. Logic Power Monitor	NA	NA
6. SHUTDOWN COOLING SYSTEM ISOLATION		
a. Reactor Vessel Water Level - Low (Level 3)	≥ 0 inches*	≥ 0 inches*
b. Reactor Steam Dome Pressure - High	≤ 145 psig	≤ 145 psig

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

TABLE 3.3.3-2 (Continued)

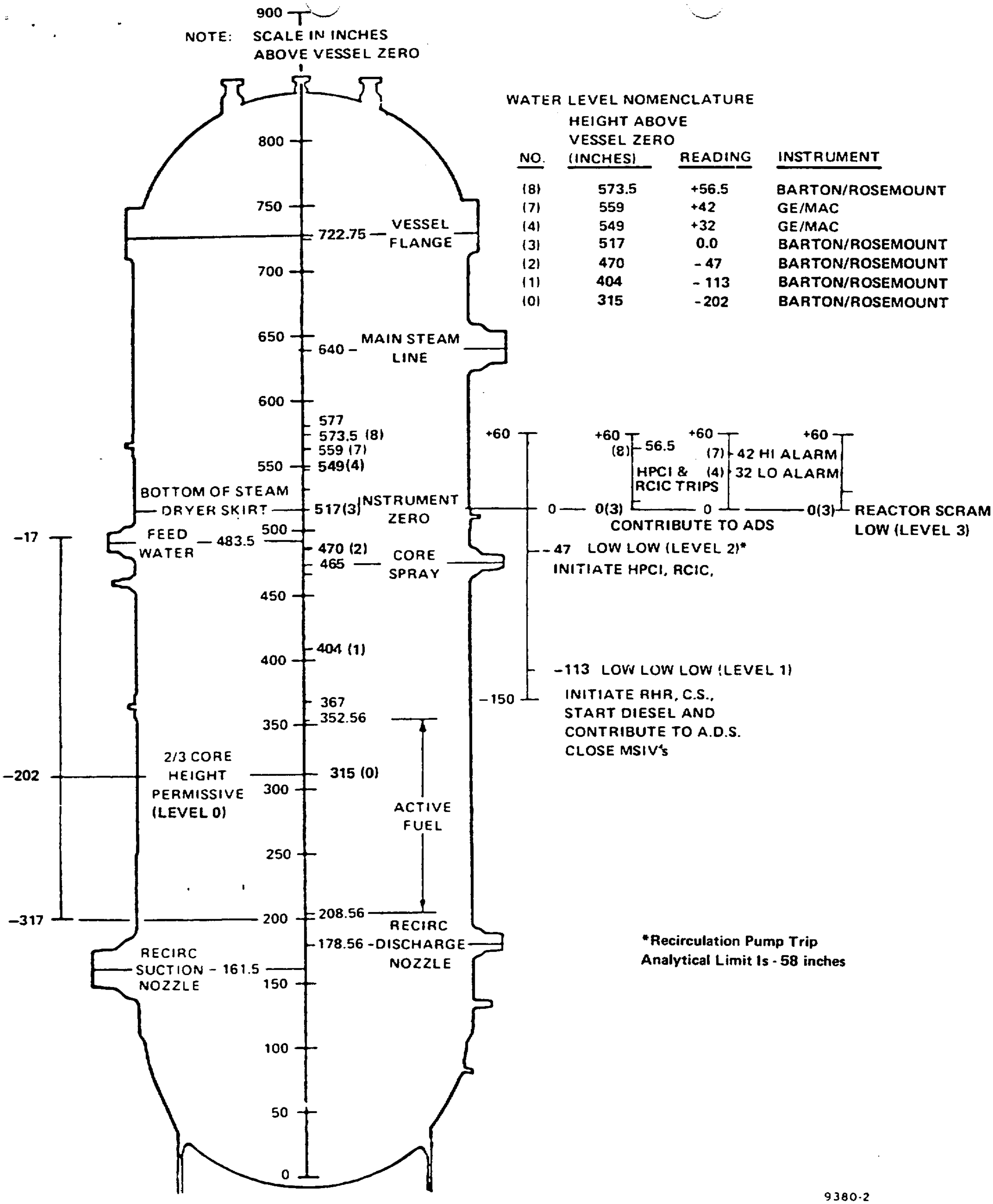
EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
3. HIGH PRESSURE COOLANT INJECTION SYSTEM		
a. Reactor Vessel Water Level - Low Low (Level 2)	≥ -47 inches*	≥ -47 inches*
b. Drywell Pressure-High	≤ 1.92 psig	≤ 1.92 psig
c. Condensate Storage Tank Level - Low	≥ 0 inches**	≥ 0 inches**
d. Suppression Chamber Water Level - High	≤ 154.2 inches***	≤ 154.2 inches***
e. Logic Power Monitor	NA	NA
f. Reactor Vessel Water Level-High (Level 8)*	≤ 56.5 inches	≤ 56.5 inches
4. AUTOMATIC DEPRESSURIZATION SYSTEM		
a. Drywell Pressure-High	≤ 1.92 psig	≤ 1.92 psig
b. Reactor Vessel Water Level - Low Low Low (Level 1)	≥ -113 inches*	≥ -113 inches*
c. ADS Timer	≤ 120 seconds	≤ 120 seconds
d. ADS Low Water Level Actuation Timer	≤ 13 minutes	≤ 13 minutes
e. Reactor Vessel Water Level - Low (Level 3)	≥ 0 inches*	≥ 0 inches*
f. Core Spray Pump Discharge Pressure - High	≥ 137 psig	≥ 137 psig
g. RHR (LPCI MODE) Pump Discharge Pressure - High	≥ 112 psig	≥ 112 psig
h. Control Power Monitor	NA	NA
5. LOW LOW SET S/RV SYSTEM		
a. Reactor Steam Dome Pressure - High	≤ 1054 psig	≤ 1054 psig

* See Bases Figure B 3/4 3-1.
** Equivalent to 10,000 gallons of water in the CST.
*** Measured above torus invert.

WATER LEVEL NOMENCLATURE

NO.	HEIGHT ABOVE VESSEL ZERO (INCHES)	READING	INSTRUMENT
(8)	573.5	+56.5	BARTON/ROSEMOUNT
(7)	559	+42	GE/MAC
(4)	549	+32	GE/MAC
(3)	517	0.0	BARTON/ROSEMOUNT
(2)	470	- 47	BARTON/ROSEMOUNT
(1)	404	- 113	BARTON/ROSEMOUNT
(0)	315	-202	BARTON/ROSEMOUNT



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

RELATED TO AMENDMENT NO. 173 TO FACILITY OPERATING LICENSE DPR-57

AND AMENDMENT NO. 113 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 9, 1990, Georgia Power Company, et al. (the licensee) submitted a request for changes to the Edwin I. Hatch Nuclear Plant, Units 1 and 2, Technical Specifications (TS). The requested changes would reduce the scram water level setpoint (Level 3) approximately 10 inches below the current setpoint. Specifically, the setpoints associated with fuel cladding integrity, reactor protection system instrumentation, primary containment isolation instrumentation, and the emergency core cooling system actuation instrumentation will be changed.

There is no change to system or component maintenance or testing. The safety-related systems and components whose operation may be initiated by a Level 3 low water level signal will still operate in the same manner as they do currently.

2.0 EVALUATION

Low water level in the reactor vessel indicates that the reactor is in danger of being inadequately cooled. Should the water level decrease too far, fuel damage could occur. The purpose of the low water level (Level 3) scram is to prevent fuel damage following events involving a loss of inventory or loss-of-coolant accidents (LOCA) that result in a decreasing reactor vessel water level. The setting of the scram signal is chosen far enough below normal operational levels to avoid spurious scrams but high enough above the top of active fuel to assure that enough water is available to account for evaporation losses and displacements of coolant following the most severe abnormal operational transient involving a level decrease.

A decrease in the reactor vessel water level to Level 3 will generate: (1) a reactor scram; (2) a permissive signal for the Automatic Depressurization System (ADS) to verify that the reactor vessel water level is, in fact, low; (3) a signal which shifts the recirculation pump to slow speed; (4) a signal which causes the Residual Heat Removal (RHR) System shutdown cooling isolation valves to close (during shutdown cooling operation only); (5) a signal which causes RHR low pressure coolant injection (LPCI) line valves and RHR vessel head spray line valves to close; and (6) a signal which isolates certain containment isolation valves. Except for the initiation of a reactor scram, the other Level 3 water level instrumentation isolation functions are anticipatory, and credit for the actual setpoint is not explicitly assumed in safety analyses.

The licensee proposed to lower the current scram setpoint for the low reactor water level scram and isolation function from its current value of greater than 10 inches above instrument zero to greater than 0 inches. Reducing the scram water level setpoint will provide several additional seconds for operator actions in the event of a feedwater transient and may avert an unnecessary reactor scram.

The proposed reduction of the reactor vessel water level (Level 3) scram setpoint potentially affects the system response to transient and accident conditions. Specifically, the reduction of the setpoint will impact the analyses for Loss of Offsite Power (LOOPS), Appendix R Event, High Energy Line Break (HELB), and the design basis accident event for the LOCA analyses. The licensee stated that analyses have been completed which support a 12-inch reduction in the Level 3 setpoint (the proposed 0-inches above instrument zero setpoint was selected for convenience and is bounded by the supporting analyses). The licensee stated that the Reactor Core Isolation Cooling (RCIC) is capable of preventing reactor water level from reaching Level 1 following system initiation at Level 2. Also, LOCA analyses were performed with the water level setpoints Level 3 reduced by 12.5 inches. The results demonstrated that there is no effect on the calculated peak clad temperature for the design basis accident event.

The modified Level 3 low water level signal will still generate signals to initiate systems/components in the same manner as they currently are. The following discusses the influence of the reduced level setpoint on the signals that are generated by the level 3 signal.

The Reactor Core Isolation Cooling (RCIC) system is designed to provide adequate makeup inventory to prevent the reactor water level from reaching low-low-low (Level 1) following system initiation at low-low (Level 2) regardless of the scram function generated by the Level 3 scram setpoint. Furthermore, the licensee has performed analyses that demonstrate that RCIC has the capability to prevent the reactor water level from reaching Level 1 for events in which there is a loss of normal feedwater.

The RHR Shutdown Cooling valves also receive automatic closure signals upon receipt of a low reactor water level (Level 3) signal. Lowering the Level 3 signal 10 inches will have an insignificant impact on this isolation function. The modified Level 3 setpoint will still be approximately 14 feet above the top of active fuel, and sufficient water inventory will be available to assure the fuel is covered.

The ADS logic requires a Level 3 signal, in addition to Level 1 signal and high drywell signal, to start the 2-minute ADS timer. A Level 3 signal and a sustained Level 1 signal can initiate a timer which will bypass the high drywell signal in approximately 13 minutes and start a 2-minute timer. If the 2-minute timer expires and a signal is present indicating a low-pressure ECCS pump is running, ADS will actuate. Since the Level 3 input would occur prior to the Level 1 input, lowering the setpoint to 0 inches will have no impact on ADS operation.

The LPCI injection valves are designed to close on a low water level signal (Level 3) to prevent inadvertent flow from the reactor vessel to the suppression pool. A continued lowering of vessel water level would result in an LPCI initiation signal on Level 1 (low-low-low water level). The LPCI isolation signal is only meaningful when the injection valves are open and thus, would only have an impact during the shutdown mode. No change in system operation will result from the proposed change in the Level 3 setpoint. Lowering the low water level setpoint from 10 inches to 0 inches will not significantly decrease the safety margin.

Certain containment isolation valves (CIVs) receive closure signals generated by the Level 3 signal. There are CIVs in the purge and inerting system, sampling system, process radiation monitoring system, drywell equipment and floor drain sump discharge lines, traversing in-core probe (TIP) guide tubes and nitrogen purge lines, drywell pneumatic compressor's suction line, torus drainage and purification line, RHR-to-radwaste line, and the RHR heat exchanger sample line. These isolations are anticipatory and credit for the actual setpoint is not explicitly assumed in the safety analysis. Lowering the trip setpoint as proposed will not have a significant impact on the safety analysis.

Furthermore, the modified Level 3 signal may avert an unnecessary scram since the reduced setpoint will provide a few additional seconds for operator action in the event of a feedwater transient.

Based on the discussion above, the NRC staff concludes that the proposed change in reduction of the Level 3 scram setpoint to greater than or equal to 0 above the instrument zero is acceptable.

The staff has reviewed the Edwin I. Hatch Nuclear Plant, Units 1 & 2 submittal and has found that the proposed changes to the TS to reduce the Level 3 scram setpoint approximately 10 inches below the current Level 3 setting 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. 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 27045). 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 30, 1991