July 14, 1987

Dockets Nos.: 50-321
and 50-366

Mr. James P. O'Reilly
Senior Vice President - Nuclear Operations
Georgia Power Company
P. O. Box 4545
Atlanta, Georgia 30302

Dear Mr. O'Reilly:

Subject: Issuance of Amendment Nos. 143 and 78 to Facility Operating Licenses DPR-57 and NPF-5 - Edwin I. Hatch Nuclear Plant, Units 1 and 2 (TACS 59542/56049)

The Commission has issued the enclosed Amendments Nos. 143 and 78 to Facility Operating Licenses DPR-57 and NPF-5, for the Edwin I. Hatch Nuclear Plant, Units 1 and 2. The amendments consist of changes to the Technical Specifications in response to your applications dated September 5, 1984, August 20, 1985, and January 7, 1986, and supplemented June 26, 1986.

The amendments modify the Technical Specifications by adding limiting conditions for operation, trip setpoints, and surveillance requirements for the monitors which provide the high radiation closure signals to the containment purge and vent valves.

A copy of our Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's Bi-Weekly Federal Register Notice.

Sincerely,

/S/

Lawrence P. Crocker, Project Manager
Project Directorate II-3
Division of Reactor Projects-I/II

Enclosures:
1. Amendment No. 143 to DPR-57
2. Amendment No. 78 to NPF-5
3. Safety Evaluation

cc w/enclosures:
See next page

PD#II-3/DRP-I/II	PD#II-3/DRP-I/II	PD#II-3/DRP-I/II
MDuncan/mac	LCrocker	BJYoungblood
06/10/87	06/10/87	~~06/ /87~~ 7/14/87

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PDR ADOCK 05000321
P PDR

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:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 143, 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

/S/

B. J. Youngblood, Director
Project Directorate II-3
Division of Reactor Projects-I/II

Attachment:
Changes to the Technical Specifications

Date of Issuance: July 14, 1987

	ML		ML
PD#II-3/DRP-I/II	PD#II-3/DRP-I/II	OGC-Bethesda	PD#II-3/DRP-I/II
MDuncan/mac	LCrocker	M. Karman for	BJYoungblood
~~04~~/10/87	~~04~~/10/87	~~04~~/ /87 7/9	~~04~~/ /87 7/14/87

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:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 78, 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

/S/

B. J. Youngblood, Director
Project Directorate II-3
Division of Reactor Projects-I/II

Attachment:
Changes to the Technical
Specifications

Date of Issuance: July 14, 1987

PD#II-3/DRP-I/II	PD#II-3/DRP-I/II	OGC-Bethesda	PD#II-3/DRP-I/II
MDuncan/mac	LCrocker	M. Karman	BJYoungblood
06/10/87	06/10/87	07/9/87	~~06/ /87~~ 7/14/87

DATED July 14, 1978

AMENDMENT NO. 143 TO FACILITY OPERATING LICENSE DPR-57, EDWIN I. HATCH, UNITS 1 & 2
AMENDMENT NO. 78 TO FACILITY OPERATING LICENSE NPF-05, EDWIN I. HATCH, UNITS 1 & 2

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

July 14, 1987

Dockets Nos.: 50-321
and 50-366

Mr. James P. O'Reilly
Senior Vice President - Nuclear Operations
Georgia Power Company
P. O. Box 4545
Atlanta, Georgia 30302

Dear Mr. O'Reilly:

Subject: Issuance of Amendment Nos. 143 and 78 to Facility Operating Licenses DPR-57 and NPF-5 - Edwin I. Hatch Nuclear Plant, Units 1 and 2 (TACS 59542/56049)

The Commission has issued the enclosed Amendments Nos. 143 and 78 to Facility Operating Licenses DPR-57 and NPF-5, for the Edwin I. Hatch Nuclear Plant, Units 1 and 2. The amendments consist of changes to the Technical Specifications in response to your applications dated September 5, 1984, August 20, 1985, and January 7, 1986, and supplemented June 26, 1986.

The amendments modify the Technical Specifications by adding limiting conditions for operation, trip setpoints, and surveillance requirements for the monitors which provide the high radiation closure signals to the containment purge and vent valves.

A copy of our Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's Bi-Weekly Federal Register Notice.

Sincerely,

Lawrence P. Crocker

Lawrence P. Crocker, Project Manager
Project Directorate II-3
Division of Reactor Projects-I/II

Enclosures:
1. Amendment No. 143 to DPR-57
2. Amendment No. 78 to NPF-5
3. Safety Evaluation

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~~2pp~~

cc w/enclosures:
See next page

Mr. James P. O'Reilly
Georgia Power Company

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Units Nos. 1 and 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 NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 143
License No. DPR-57

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

   A. The applications 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 licensee) dated August 20, 1985, and January 7, 1986, and supplemented June 26, 1986, 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 application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

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

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

2. Accordingly, the license is 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:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 143, 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

Lawrence P. Crocker

for B. J. Youngblood, Director
Project Directorate II-3
Division of Reactor Projects-I/II

Attachment:
Changes to the Technical
Specifications

Date of Issuance: July 14, 1987

ATTACHMENT TO LICENSE AMENDMENT NO. 143

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 Page	Insert Page
3.7-16	3.7-16
3.7-20	3.7-20
3.2-3	3.2-3
3.2-25	3.2-25

Table 3.2-1 (Cont.)

Ref. No. (a)	Instrument	Trip Condition Nomenclature	Required Operable Channels per Trip System (b)	Trip Setting	Action to be taken if number of channels is not met for both trip systems (c)	Remarks (d)
5	Main Steam Line Pressure	Low	2	$\geq$825 psig	Initiate an orderly load reduction and close MSIVs within 8 hours.	Initiates Group 1 isolation. Only required in RUN mode, therefore activated when Mode Switch is in RUN position.
6	Main Steam Line Flow	High	2	$\leq$140% rated flow ($\leq$120 psid)	Initiate an orderly load reduction and close MSIVs within 8 hours.	Initiates Group 1 isolation.
7	Main Steam Line Tunnel Temperature	High	2	$\leq$200°F	Initiate an orderly load reduction and close MSIVs within 8 hours.	Initiates Group 1 isolation.
8	Reactor Water Cleanup System Differential Flow	High	1	20-80 gpm	Isolate reactor water cleanup system.	Final trip setting will be determined during startup test program.
9	Reactor Water Cleanup Equipment Room Temperature	High	2	100-150°F	Isolate reactor water cleanup system.	Final trip setting will be determined during startup test program.
10	Reactor Water Cleanup Equipment Room Differential Temperature	High	2	0-100°F	Isolate reactor water cleanup system	Final trip setting will be determined during startup test program.
11	Condenser Vacuum	Low	2	$\geq$7" Hg. vacuum	Initiate an orderly load reduction and close MSIVs within 8 hours.	Initiate Group 1 isolation.
12	Drywell Radiation	High	1	$\leq$138 R/HR.	Close the affected isolation valves within 24 hours or be in Hot Shutdown within the next 6 hours and in Cold Shutdown within the next 30 hours	Isolates containment purge and vent valves

Table 4.2-1 (Cont'd)

Ref. No. (a)	Instrument	Instrument Check Minimum Frequency	Instrument Functional Test Minimum Frequency (b)	Instrument Calibration Minimum Frequency (c)
10	Reactor Water Cleanup Equipment Room Differential Temperature	None	(d)	Every 3 months
11	Condenser Vacuum	None	(d)	Every 3 months
12	Drywell Radiation	Once/Day	Once/Month	Once/Operating Cycle

Notes for Table 4.2-1

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

b. Instrument functional tests are not required when the instruments are not required to be operable or are tripped. However, if functional tests are missed, they shall be performed prior to returning the instrument to an operable status.

c. Calibrations are not required when the instruments are not required to be operable. However if calibrations are missed, they shall be performed prior to returning the instrument to an operable status.

d. Initially once per month or according to Figure 4.1-1 with an interval of not less than one month nor more than three months. The compilation of instrument failure rate date may include data obtained

TABLE 3.7-1

PRIMARY CONTAINMENT ISOLATION VALVES

Isolation Group (b)	Valve Identification (d)	Number of Power Operated Valves Inside	Outside	Maximum Operating Time (sec)	Normal Position (a)	Action on Initiation Signal (a)
1	Main steam line (B21-F022 A,B,C,D; B21-F028 A,B,C,D)	4	4	$3 < T < 5$	O	GC
1	Main steam line drain (B21-F016, B21-F019)	1	1	15	C	SC
1	Reactor water sample line (B31-F019, B31-F020)	1	1	5	C	SC
2	H_2 -O_2 Analyzer system (P33-P001)		2 each line	5	O	GC
2(f)	Drywell purge inlet (T48-F307, T48-F308)		2	5	C	SC
2(f)	Drywell main exhaust (T48-F319, T48-F320)		2	5	O	GC
2	Drywell exhaust valve bypass to standby gas treatment (T48-F341, T48-F340)		2	5	C	SC
2	Drywell nitrogen make-up line (normal operation) (T48-F118A)		1	5	O	GC
2(f)	Suppression chamber purge inlet (T48-F309, T48-F324)		2	5	C	SC
2(f)	Suppression chamber main exhaust (T48-F318, T48-F326)		2	5	O	GC

Notes to Table 3.7-1 (Concluded)

(f) Requires a Group 2 signal or a Primary Containment high radiation isolation signal.

For all entries in Table 3.7-1 where the number of isolation valves is equal to one outside containment and none inside containment the valve is in a series path with at least one other containment isolated valve in the Table. For example, T48-F118 is in a series path with T48-F104, thus providing two in series power operated containment isolation valves.

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. 78
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 licensee) dated September 5, 1984, as supplemented June 26, 1986, 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 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:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 78 , 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

Lawrence P. Crocker

for B. J. Youngblood, Director
Project Directorate II-3
Division of Reactor Projects-I/II

Attachment:
Changes to the Technical Specifications

Date of Issuance: July 14, 1987

ATTACHMENT TO LICENSE AMENDMENT NO. 78

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 Page	Insert Page
3/4 3-11	3/4 3-11
3/4 3-15	3/4 3-15
3/4 3-16	3/4 3-16
3/4 3-21	3/4 3-21

TABLE 3.3.2-1

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	VALVE GROUPS OPERATED BY SIGNAL (a)	MINIMUM NUMBER OPERABLE CHANNELS PER TRIP SYSTEM(b) (c)	APPLICABLE OPERATIONAL CONDITION	ACTION
1. PRIMARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level				
1. Low (Level 3) (2B21-N680 A,B,C,D)	2, 6, 10, 11, 12,	2	1, 2, 3	20
2. Low-Low (Level 2) (2B21-N682 A,B,C,D)	5, #,*	2	1, 2, 3	20
3. Low-Low-Low (Level 1) (2B21-N681 A,B,C,D)	1	2	1, 2, 3	20
b. Drywell Pressure - High (2C71-N650 A, B, C, D)	2, 6, 7, 10 12, #,*	2	1, 2, 3	20
c. Main Steam Line				
1. Radiation - High (2D11-K603 A,B,C,D)	1, 12, #, (d)	2	1, 2, 3	21
2. Pressure - Low (2B21-N015 A,B,C,D)	1	2	1	22
3. Flow - High (2B21-N686 A,B,C,D) (2B21-N687 A,B,C,D) (2B21-N688 A,B,C,D) (2B21-N689 A,B,C,D)	1, #	2/line	1, 2, 3	21
d. Main Steam Line Tunnel High Temperature - High (2B21-N623 A,B,C,D) (2B21-N624 A,B,C,D) (2B21-N625 A,B,C,D) (2B21-N626 A,B,C,D)	1	2/line(e)	1, 2, 3	21
e. Condenser Vacuum - Low (2B21-N056 A, B, C, D)	1	2	1,2(f),3(f)	23
f. Turbine Building Area Temperature - High (2U61-R001, 2U61-R002, 2U61-R003, 2U61-R004)	1	2(e)	1, 2, 3	21
g. Drywell Radiation-High (2D11-K621 A,B)	(j)	1	1, 2, 3	29

ACTION

ACTION 20 - Be in at least HOT SHUTDOWN within 6 hours and in COLD SHUTDOWN within the next 30 hours.

ACTION 21 - Be in at least STARTUP with the main steam line isolation valves closed within 2 hours or be in at least HOT SHUTDOWN within 6 hours and in COLD SHUTDOWN within the next 30 hours.

ACTION 22 - Be in at least STARTUP within 2 hours.

ACTION 23 - Be in at least STARTUP with the Group 1 isolation valves closed within 2 hours or in at least HOT SHUTDOWN within 6 hours.

ACTION 24 - Establish SECONDARY CONTAINMENT INTEGRITY with the standby gas treatment system operating within one hour.

ACTION 25 - Isolate the reactor water cleanup system.

ACTION 26 - Close the affected system isolation valves and declare the affected system inoperable.

ACTION 27 - Verify power availability to the bus at least once per 12 hours or close the affected system isolation valves and declare the affected system inoperable.

ACTION 28 - Close the shutdown cooling supply and reactor vessel head spray isolation valves unless reactor steam dome pressure $\leq$ 145 psig.

ACTION 29 - Either close the affected isolation valves within 24 hours or be in hot shutdown within the next 6 hours and in cold shutdown within the next 30 hours.

NOTES

Actuates operation of the main control room environmental control system in the pressurization mode of operation.

* Actuates the standby gas treatment system.

** When handling irradiated fuel in the secondary containment.

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

b. A channel may be placed in an inoperable status for up to 2 hours for required surveillance without placing the trip system in the tripped condition provided at least one other OPERABLE channel in the same trip system is monitoring that parameter.

c. With a design providing only one channel per trip system, an inoperable channel need not be placed in the tripped condition where this would cause the Trip Function to occur. In these cases, the inoperable channel shall be restored to OPERABLE status within 2 hours or the ACTION required by Table 3.3.2-1 for that Trip Function shall be taken.

d. Trips the mechanical vacuum pumps.

e. A channel is OPERABLE if 2 of 4 instruments in that channel are OPERABLE.

f. May be bypassed with all turbine stop valves closed.

g. Closes only RWCU outlet isolation valve 2G31-F004.

h. Alarm only.

i. Adjustable up to 60 minutes.

j. Isolates containment purge and vent Valves.

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)	$\geq$ 8.5 inches*	$\geq$ 8.5 inches*
2. Low Low (Level 2)	$\geq$ -55 inches*	$\geq$ -55 inches*
3. Low Low Low (Level 1)	$\geq$ -121.5 inches*	$\geq$ -121.5 inches*
b. Drywell Pressure - High	$\leq$ 1.85 psig	$\leq$ 1.85 psig
c. Main Steam Line		
1. Radiation - High	$\leq$ 3 x full power background	$\leq$ 3 x full power background
2. Pressure - Low	$\geq$ 825 psig	$\geq$ 825 psig
3. Flow - High	$\leq$ 138% of rated flow	$\leq$ 138% of rated flow
d. Main Steam Line Tunnel Temperature - High	$\leq$ 194°F	$\leq$ 194°F
e. Condenser Vacuum - Low	$\geq$ 7" Hg vacuum	$\geq$ 7" Hg vacuum
f. Turbine Building Area Temp. - High	$\leq$ 200°F	$\leq$ 200°F
g. Drywell Radiation-High	$\leq$ 138 R/hr	$\leq$ 138 R/hr
2. SECONDARY CONTAINMENT ISOLATION		
a. Reactor Building Exhaust Radiation - High	$\leq$ 60 mr/hr	$\leq$ 60 mr/hr
b. Drywell Pressure - High	$\leq$ 1.85 psig	$\leq$ 1.85 psig
c. Reactor Vessel Water Level - Low Low (Level 2)	$\geq$ -55 inches*	$\geq$ -55 inches*
d. Refueling Floor Exhaust Radiation - High	$\leq$ 20 mr/hr	$\leq$ 20 mr/hr

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

TABLE 4.3.2-1

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
1. PRIMARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level				
1. Low (Level 3)	S	M	R	1, 2, 3
2. Low Low (Level 2)	S	M	R	1, 2, 3
3. Low Low Low (Level 1)	S	M	R	1, 2, 3
b. Drywell Pressure - High	S	M	R	1, 2, 3
c. Main Steam Line				
1. Radiation - High	D	W(a)	R	1, 2, 3
2. Pressure - Low	NA	M	Q	1
3. Flow - High	S	M	R	1, 2, 3
d. Main Steam Line Tunnel Temperature - High	S	M	R	1, 2, 3
e. Condenser Vacuum - Low	NA	M	Q	1, 2#, 3#
f. Turbine Building Area Temp. - High	NA	M	R	1, 2, 3
g. Drywell Radiation-High	D	M	R	1, 2, 3
2. SECONDARY CONTAINMENT ISOLATION				
a. Reactor Building Exhaust Radiation - High	D	M(a)	R	1, 2, 3, 5 and *
b. Drywell Pressure - High	S	M	R	1, 2, 3
c. Reactor Vessel Water Level - Low Low (Level 2)	S	M	R	1, 2, 3
d. Refueling Floor Exhaust Radiation - High	D	M(a)	Q	1, 2, 3, 5 and *

*When handling irradiated fuel in the secondary containment.
#May be bypassed with all turbine stop valves closed.
(a) Instrument alignment using a standard current source.

Amendment No. 78

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENTS NOS. 143 AND 78 TO

FACILITY OPERATING LICENSES DPR-57 AND NPF-5

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

EDWIN I. HATCH NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-321 AND 50-366

INTRODUCTION

By letters dated September 5, 1984 (Ref. 1), August 20, 1985 (Ref. 2) and January 7, 1986 (Ref. 3), Georgia Power Company (GPC) proposed to change the Technical Specifications (TS) for the Edwin I. Hatch Nuclear Plant, Units 1 and 2, to provide for operation of both units in accordance with TMI Action Plan II.E.4.2.(7), which requires that containment purge and vent valves close automatically on a high containment radiation signal. On April 28, 1986, the staff forwarded to the licensee a Request for Additional Information (RAI). In response to this RAI, by letter dated June 26, 1986 (Ref. 4), the licensee provided details regarding the design and operation of the various containment vent and purge lines and systems, sufficient to enable the staff to perform independent audit calculations. The information provided in the June 26, 1986, letter did not alter the TS amendment requests as described in References 1, 2, and 3. On February 3, 1987, during a conference call with representatives of the licensee, the staff obtained additional clarification of certain details regarding system design and operation. This clarification also had no effect on the original TS amendment requests.

The proposed changes for the Hatch 1 and 2 Technical Specifications (TS) add limiting conditions for operation, trip setpoints and surveillance requirements for the monitors which provide the high radiation closure signals to the containment purge and vent valves.

The staff review of the submittals and supplemental information consisted of an evaluation of the licensee's supplemental safety analysis and radiological dose calculation analysis, and comparison of the analyses with an audit calculation performed by the staff. The staff then determined that the radiological consequences of the previously evaluated design basis LOCA remained within the 10 CFR Part 100 dose exposure guidelines after adding the incremental contribution of the containment purge/vent system prior to isolation.

EVALUATION

The staff performed audit calculations based on a conservative estimate of a steam release of 2000 lbs prior to the post-LOCA closure of the vent/purge valves, and a coolant activity iodine spiking factor using the maximum dose equivalent I-131 reactor coolant concentration of 4.0 microcuries given in the Hatch Unit 1 TS 3.6.F.1 and Unit 2 TS 3.4.5.a. It was assumed that containment isolation would be achieved before the onset of any fuel failures resulting from the accident. In addition, the worst 0.5% probability directionally dependent case X/Q values consistent with ground level releases for the condition of fumigation were used in the dose calculations, based on the methodology in Regulatory Guide 1.145. A list of the applicable parameters is presented in the attached Table I.

The staff estimates that the steam released through the purge/vent lines (2000 lbs) would result in an incremental dose of 0.26 Rem to the thyroid at the Exclusion Area Boundary (EAB) and 0.131 Rem to the thyroid at the outer boundary of the Low Population Zone (LPZ). These doses at the EAB and LPZ outer boundary do not exceed a small fraction (10%) of the dose guideline values of 10 CFR 100; i.e., 2.5 Rem and 30 Rem respectively, to the whole body and thyroid. Furthermore, when these doses are added to the LOCA doses at the EAB and outer boundary of the LPZ for each unit as calculated in the staff Safety Evaluation Reports, the combined doses are still within the applicable guidelines of 10 CFR 100 (See Table II attached).

The staff previously had performed a generic evaluation of the radiological consequences of accidental release through BWR 2-inch vent and purge lines, and an evaluation of BWR vent and purge radiation monitor setpoints. The results of these evaluations were provided to the BWR Owners Group by letter dated May 6, 1986 (Ref. 5). A copy of this letter is provided as Enclosure 1 to this Safety Evaluation.

The radiation monitor setpoint was calculated by the licensee such that no accident would result in radiation dose at the site boundary exceeding the EPA Protection Action Guide limits of 1 Rem whole body and 5 Rem thyroid, based on the assumption of a delay of 30 minutes from the time the setpoint is exceeded to the time of valve closure. The nominal setpoint of 100 Rem/Hr is approximately an order of magnitude lower than the actual containment radiation level at which isolation would need to occur to satisfy the 10 CFR 100 offsite dose criterion (Ref. 5). In reality, the isolation of containment purge/vent valves should not exceed 7 seconds as proposed to be specified in the Hatch Unit 1 TS Table 3.6.3-1 and Unit 2 TS Table 3.7-1 (for more than 2" diameter vent/purge lines); thus, the impact on the radiological consequences of an accident is less than calculated by the licensee. The licensee proposed to use the EPA dose guidelines to establish setpoint values. The staff does not agree that compliance with the EPA guidelines or with comparable NRC regulations (10 CFR Part 20) demonstrates conformance to 10 CFR 100, which applies to accidents. However, based on the staff calculations, we conclude that the proposed setpoint values are acceptable.

We have reviewed the submittal proposing changes to the containment purge and vent valve closure Technical Specifications. The audit calculations performed by the staff are within the 10 CFR 100 dose exposure guidelines, and the setpoint value corresponds to a small fraction of the dose guidelines of 10 CFR 100. Therefore, the proposed TS changes to permit operation of both units in accordance with TMI Action Plan II.E.4.2(7) are acceptable and have been implemented in the surveillance testing procedures of each unit.

ENVIRONMENTAL CONSIDERATIONS

The amendments involve a change in use of facility components located within the restricted area as defined in 10 CFR Part 20 and changes in surveillance requirements. The 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 should be 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. 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.

CONCLUSION

The Commission made proposed determinations that the amendments involve no significant hazards consideration which were published in the Federal Register (49 FR 45952) on November 21, 1984, and (50 FR 38915) on September 25, 1985, and consulted with the state of Georgia. No public comments were received, and the state of Georgia did not have any comments.

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

REFERENCES

1. Letter from J. T. Beckham, Jr., GPC to D. Muller, NRC, "Edwin I. Hatch Nuclear Plant Unit 2 Request to Amend Technical Specifications" dated September 5, 1984.

2. Letter from J. T. Beckham, Jr., GPC to J. F. Stolz, NRC, "Edwin I Hatch Nuclear Plant Unit 1 Request to Revise Technical Specifications Purge/Vent Isolation", dated August 20, 1985.

3. Letter from J. T. Beckham, Jr., GPC to D. Muller, NRC, "Revision to Proposed Technical Specification Amendment Purge/Vent Isolation", dated January 7, 1986.

4. Letter from L. T. Gucwa, GPC to D. Muller, NRC, "Edwin I. Hatch Nuclear Plant Units 1 and 2 Isolation of Containemnt Purge and Vent Valves", dated June 26, 1986.

5. Letter from R. Bernero, NRC, to J. Fulton, Chairman BWR Owners Group, dated May 7, 1986.

Principal Contributors: U. Cheh, RSB
L. Crocker, PDII-3

Dated: July 14, 1978

TABLE I

ASSUMPTIONS USED TO EVALUATE THE CONTAINMENT PURGING/VENTING CONTRIBUTION TO THE LOCA CASE

X/Q value (0-2 hours, EAB, ground level release), sec/m^3	1.4×10^{-4} 3/
Purge valve closure time, sec	less than or equal to 7
Amount of steam released through the purging/venting valves prior to post-LOCA closure, lbm mass	2000
Maximum technical specification primary coolant limit, dose-equivalent I-131, µCi/gm	4

TABLE II

RADIOLOGICAL CONSEQUENCES

Thyroid Doses 2/

	EAB 0-2 Hours		LPZ 0-8 Hours	
	Unit 1	Unit 2	Unit 1	Unit 2
Containment Purge/Vent Contributions	0.261 Rem	0.131 Rem	0.261 Rem	0.131 Rem
SER LOCA Dose Estimate	8 Rem 1/	65 Rem 1/	75 Rem 1/	182 Rem 1/
Effective LOCA Dose	8.261 Rem	65.131 Rem	75.015 Rem	182.131 Rem

Notes:

1. The LOCA dose were taken from the Safety Evaluation Report related to the Operations of the Edwin I. Hatch Unit 1 and 2, Docket No. 50-321/366 GPC, dated May 11, 1973 and June 1978, respectively.

2. The whole body dose are not listed because they would be negligible when compared to the guideline values.

3. The X/Q values are provided in the updated NRC evaluation file dated May 1983, and are based upon the criteria in Regulatory Guide 1.145.

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

MAY 07 1986

Mr. Jack M. Fulton, Chairman
BWR Owners Group
Boston Edison Company
800 Boylston Street
Boston, Massachusetts 02199

Dear Mr. Fulton:

Analyses provided by the Boiling Water Reactor Owners Group (BWROG) contained in submittals of June 14, 1982 (T. Dente to D. Eisenhut); June 27, 1981 (T. Dente to D. Eisenhut) and NSEO-78-0882 have been reviewed with respect to the need to automatically isolate small diameter lines on a high radiation signal as required by NUREG-0737 Item II.E.4.2(7). We have also considered material presented in a technical appeal meeting on June 20, 1985, including a request for staff comments on setpoints for radiation detection instrumentation to be used for isolation of larger lines.

Our review of the BWROG evaluations and request for comment has resulted in two conclusions. First, BWR lines of 2 inches in diameter or smaller need not be provided with a radiation isolation signal, provided that a licensee demonstrates on his docket that the BWROG generic evaluation is applicable to his plant. This demonstration should include an assessment of the ability of the operators to assess and isolate leakages that would not cause other isolation signals (e.g., verification of the BWROG generic assertion of a maximum of 30 minutes). The generic staff safety evaluation of this relief is provided as Enclosure 1 and may be referenced by individual licensees. Secondly, Enclosure 2 summarizes staff comments on radiation signals still required for the larger lines.

I have asked Jerry Hulman of my staff to respond to any technical questions on this matter. He may be reached on (301) 492-7941. Individual NRC licensing Project Managers should be consulted with respect to docketing requests for relief for specific plants.

Sincerely,

Robert M. Bernero

Robert M. Bernero, Director
Division of BWR Licensing

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Enclosures:
1. Safety Evaluation
2. BWR Vent & Purge Radiation Monitor Set Points

ENCLOSURE 1

EVALUATION OF THE RADIOLOGICAL CONSEQUENCES FOR ACCIDENTAL RELEASES THROUGH BWR 2-INCH VENT AND PURGE LINES

INTRODUCTION

NUREG-0737, Item II.E.4.2(7) required that the containment purge and vent isolation valves must close on a high radiation signal. This position was added to the original NUREG-0578 requirements of Recommendation 2.1.4 as a result of further staff evaluation of features needed to improve containment isolation dependability.

One basis for the implementation of II.E.4.2(7) was the additional protection it would provide against low rates of reactor coolant leakage and releases to the environment which would not initiate the other automatic isolation signals of reactor low water level and high drywell pressure. The BWR Owners Group (BWROG) previously transmitted an evaluation of offsite radiological consequences for accidental releases through BWR vent and purge lines which do not meet the requirement of NUREG-0737, Item II.E.4.2(7) in a letter from T. J. Dente of the BWROG to D. G. Eisenhut of the NRC, dated June 14, 1982. In a June 20, 1985 meeting, the BWROG requested that the staff review its evaluation for small (2-inch diameter) vent and purge lines.

DISCUSSION

The staff has reviewed the BWROG evaluation which provides calculations of the radiological consequences of the limiting reactor coolant system break which would not initiate automatic isolation with the current design. The limiting event was conservatively modeled as a reactor coolant system break such that the drywell atmosphere would contain saturated steam at a pressure just below the containment isolation setpoint. Steam release through one vent or purge line was assumed to pass directly to the environment with no credit given for holdup or dilution, or for filtering by the standby gas treatment system. The fraction of the iodine postulated to become airborne and available for release to the atmosphere, without credit for plateout, was assumed to equal the fraction of the coolant flashing to steam. The BWROG evaluation provided calculations for a typical plant as well as a generic analytical procedure.

Independent calculations of the radiological consequences of the limiting reactor coolant system break were performed by the staff. The staff conservatively estimated a mass release value of 492 cubic feet per minute of saturated steam at 2 psig over a 30 minute duration until the one purge and vent line would be isolated by other actions.

The assumptions used in this staff analysis were as follows:

1. Drywell atmosphere is saturated steam and at a pressure equal to the containment isolation setpoint (psig).

2. Operator action time to close the purge or vent valve is assumed to be 30 minutes.

3. Vent pipe length is conservatively assumed to be 10 ft. for purposes of flow calculations.

4. Elevation changes have been neglected.

The BWR Owners Group analysis used formulas described in NEDM-10363-13, "Hydraulic Analyses Procedure for BWR Piping Systems." The staff used similar formulas, which are described in the Crane Flow of Fluid Manual and the above assumptions, and obtained similar results to those provided by the BWR Owners Group.

The staff, using the above release rate, performed plant specific calculations of the radiological consequences for Pilgrim Unit 1, Hatch Units 1 and 2, Peach Bottom Units 2 and 3, and Limerick Unit 1. The staff's calculation of offsite doses differed from the procedure outlined in the BWROG's evaluation in two respects. First, the staff used short term diffusion estimates typical of other conservative regulatory evaluations of accidents; the BWROG used annual average relative concentrations typical of a realistic evaluation of doses from routine releases. Second, the staff used conservative reactor coolant iodine concentrations assuming a pre-accident iodine spike for those plants with a technical specification iodine spiking limit. For Pilgrim Unit 1, which has no technical specification iodine spiking limit, the staff used the maximum technical specification equilibrium concentration with an accident-initiated spike, modeled by increasing the equilibrium fission product activity release rate from the fuel by a factor of 500. The staff's iodine spiking model is typical of regulatory analyses involving accidental releases of primary coolant, as outlined in Section 15.6.2 of the Standard Review Plan (NUREG-0800). The BWROG's evaluation assumed equilibrium iodine concentrations with an accident-initiated spike using a 95% cumulative probability iodine spiking model.

RESULTS

The staff estimates of the thyroid and whole body doses at the exclusion area and low population zone outer boundaries for the 6 units are presented in Table I (attached). Although specific acceptance criteria do not exist for this postulated accident, the radiological consequences and frequency of occurrence for this accident would tend to be similar to that of the failure of small lines carrying primary coolant outside containment. The staff concluded that the use of the acceptance criteria for the failure of small lines, which appear in Section 15.6.2 of the Standard Review Plan, would be appropriate for use in this evaluation. Thus, the radiological consequences of this postulated accident would be acceptable if the calculated whole-body and thyroid doses at the exclusion area and low population zone outer boundaries do not exceed a small fraction (10%) of the dose guideline values of 10 CFR Part 100, viz., 2.5 rem and 30 rem respectively, for whole body and thyroid doses. As summarized in Table I, the estimated doses are a small fraction of these dose guideline values of 10 CFR Part 100.

TABLE 1

RADIOLOGICAL CONSEQUENCES FOR ACCIDENTAL RELEASES THROUGH BWR 2-INCH VENT AND PURGE LINES

	Exclusion Area Boundary (0-2 hr), rems		Low Population Zone Boundary (0-8 hr), rems	
	Thyroid	Whole Body	Thyroid	Whole Body
Limerick Unit 1	0.4	0.007	0.08	0.002
Peach Bottom Units 2 & 3	0.4	0.004	0.007	0.00006
Hatch Unit 1*	0.08	0.0008	0.04	0.0004
Hatch Unit 2*	0.08	0.002	0.04	0.0008
Pilgrim Unit 1	3.3	0.03	0.2	0.002

*The difference in whole body doses between Hatch Unit 1 and Unit 2 was a result of different Technical Specification primary coolant activity limits.

The magnitudes of these doses calculated by the staff are higher than would realistically be expected because of the many conservative assumptions in the staff's methodology, particularly with respect to iodine spiking behavior and to meteorology. For example, coolant iodine concentration levels generally are small fractions of equilibrium technical specification levels, iodine spiking does not always occur coincident with the transients, the iodine spiking concentrations assumed to occur are well in excess of any level recorded at an operating boiling water reactor, and the probability of better meteorological conditions is quite high. A more realistic analysis would yield dose estimates about 1/100th or less of the values noted above.

Since this evaluation assumes that operator action to close the purge or vent valve is taken within 30 minutes, for the BWROG evaluation to be acceptable the licensee must verify that the 30 minute operating time is valid based upon location and accessibility of the valve operators, and insturmentation necessary to determine the need for manual closure, and that plant procedure and operator training are sufficient to support the approach.

ENCLOSURE 2

BWR VENT & PURGE RADIATION MONITOR SET POINTS

BACKGROUND - In a meeting on June 20, 1985 the BWR Owners Group requested that the staff establish set point criteria for isolation signals for vent and purge line radiation monitors required under TMI Action Item II.E.4.2(7) of NUREG-0737. The monitors are not considered safety related, but are to be provided solely to assure diverse isolation signals in the event of an accident.

EVALUATION - Radiation monitors with vent and purge line isolation capability are required as a post TMI item to ensure containment isolation. Other diverse isolation signals, such as drywell pressure and reactor water level, are also provided. A review of the regulations indicates there are no explicit dose guidelines that apply to such monitors in the event of accidents, other than the siting values in 10 CFR 100. The Standard Review Plan contains design basis accident dose acceptance criteria which have previously been evaluated with respect to purge and vent valve closure time criteria. As discussed in Enclosure 1, the staff concluded that the use of acceptance criteria of calculated whole-body and thyroid doses at the exclusion area and low population zone outer boundaries which do not exceed a small fraction (10%) of the guideline values of 10 CFR Part 100 would be appropriate for use in the evaluation of the radiological consequences of accidental releases through open vent and purge lines. As a minimum requirement, vent and purge radiation monitor set points should be established such that this acceptance criteria is met.

The staff notes, however, that a guiding principle in establishing set point values for radiation monitors used to limit doses is to establish them as low as possible to avoid unnecessary exposures. If set too low, however, spurious signals resulting from minor changes in instrument detectability or background activity levels not representative of accident conditions can occur. As a practical matter, for radiation detectors which are located on the vent or purge line set points which do not exceed the highest radiation level expected in normal operation should provide suitable warning of accidents and avoid most spurious signals.

POSITION - Radiation monitors provided for assuring diverse isolation signals for BWR vent and purge valves should be set low enough to effectively limit accidental releases of radioactivity from being released offsite when such valves are open during operation. While such set points should be established as low as possible to limit offsite accident releases, the set points should not cause unnecessary isolation signals resulting from instrument uncertainties or non-accident variations in radiation levels. As a minimum requirement, vent and purge radiation monitor set points should be established such that the radiological consequences of accidental releases through open vent and purge lines do not exceed a small fraction (10%) of the dose guideline values of 10 CFR Part 100. As a practical matter, for well shielded monitors which directly measure activity levels in the flow past such valves, set points at a level which does not exceed the highest radiation level expected in normal operation should provide adequate assurance of accident isolation.