June 20, 1989

Docket Nos.: 50-321
and 50-366

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

Dear Mr. Hairston:

SUBJECT: ISSUANCE OF AMENDMENT NO. 164 TO FACILITY OPERATING LICENSE DPR-57 AND AMENDMENT NO. 101 TO FACILITY OPERATING LICENSE NPF-5 - EDWIN I. HATCH NUCLEAR PLANT, UNITS 1 AND 2 (TACS 72034/72035)

The Commission has issued the enclosed Amendment No. 164 to Facility Operating License DPR-57 and Amendment No. 101 to Facility Operating License NPF-5 for the Edwin I. Hatch Nuclear Plant, Units 1 and 2. The amendments consist of changes to the Technical Specifications (TS) in response to your application dated February 3, 1989.

The amendments change the TS related to primary containment isolation valves.

A copy of our Safety Evaluation is 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
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 164 to DPR-57
2. Amendment No. 101 to NPF-5
3. Safety Evaluation

cc w/ enclosures:
See next page

PDII-3	PDII-3	PDII-3
MRood	LCrocker:ls	DMatthews
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DF01 1/1

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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. 164
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 licensee) dated February 3, 1989, 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. 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. 164, 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

DBMatthews

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: June 20, 1989

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) <u>Technical Specifications</u>

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 101, 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

Original Signed By:
David B. Matthews

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: June 20, 1989

OFFICIAL RECORD COPY

LA:PDII-3	PM:PDII-3	DEST:SPLB	OGC-WF	D:PDII-3
MRood	LCrocker:	JCraig		DBMatthews
04/4/89	05 ~~04~~/04/89	04/ /89	06/19/89	06/15/89

ATTACHMENT TO LICENSE AMENDMENT NO. 164

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 area of change.

Remove Page	Insert Page
3.7-6	3.7-6
3.7-6a	3.7-6a
3.7-16	3.7-16
3.7-18	3.7-18
3.7-18a	3.7-18a
3.7-18b	3.7-18b
3.7-21	3.7-21
3.7-26a	3.7-26a
3.7-27	3.7-27

4.7.A.2.e. Type B Tests - Leak Tests of Penetrations with Seals and Bellows (Continued) (Tables 3.7-2 and 3.7-3)

(1) Primary containment components which seal or penetrate the pressure containing boundary of the containment shall be tested at a pressure not less than P_a. These components shall be tested at each major refueling shutdown or at intervals not to exceed 2 years.

(2) (a) The primary containment airlock shall be tested at 6-month intervals at P_a by pressurizing the compartment between the two airlock doors. The leakage shall not exceed 0.05 L_a.

(b) If the primary containment airlock is opened during periods when primary containment integrity is not required, the test required by 4.7.A.2.e.(2)(a) shall be performed at the end of such periods.

(c) If the primary containment airlock is opened during periods when primary containment integrity is required, it shall be tested within 3 days of being opened by pressurizing the gap between the door seals to ≥ 10 psig for at least 15 minutes. The leakage for each set of door seals shall not exceed 0.01 L_a.

(d) If primary containment is required and the primary containment airlock is being opened more frequently than once every 3 days, the test required by 4.7.A.2.e.(2)(c) shall be performed at least once per 3 days during the period of frequent openings.

*All Type B and Type C Leakage Tests (i.e., Local Leak Rate Tests) that fail (i.e., test leakage is such that an LER would be required) during an outage shall be reported according to 10 CFR 50.73 by one, 30-day written report that is due within 30 days of the first leakage test failure in the outage. All other leakage test failures discovered during the outage will be reported in a revision to the original report due within 30 days after the end of the outage.

(e) At least once per 6 months it shall be verified that only one door in the air-lock can be opened at a time.

f. Type C Tests-Local Leak Tests of Containment Isolation Valves (Tables 3.7-1 and 3.7-4)

Type C tests shall be performed under the program established in Appendix J of 10 CFR Part 50.

Containment isolation valves (except for main steam line isolation valves) shall be tested at a pressure not less than P_a. Type C tests shall be performed at each major refueling shutdown or at intervals not to exceed 2 years.*

g. Acceptance Criteria for Type B and Type C Tests

The combined leakage rate of components subject to Type B and C tests shall be determined under the program established in Appendix J of 10 CFR Part 50 and shall not exceed 0.6 L_a.*

h. Main Steam Line Isolation Valves

The main steam line isolation valves shall be tested at a pressure of 28 psig for leakage at least once per operating cycle. If a total leak rate of 11.5 scf per hour for any one main steam line isolation valve is exceeded, repairs and retest shall be performed to correct this condition.

*All Type B and Type C Leakage Tests (i.e., Local Leak Rate Tests) that fail (i.e., test leakage is such that an LER would be required) during an outage shall be reported according to 10 CFR 50.73 by one, 30-day written report that is due within 30 days of the first leakage test failure in the outage. All other leakage test failures discovered during the outage will be reported in a revision to the original report due within 30 days after the end of the outage.

TABLE 3.7-1

PRIMARY CONTAINMENT ISOLATION VALVES WHICH
RECEIVE A PRIMARY CONTAINMENT ISOLATION SIGNAL

Isolation Group (b)	Valve Identification (d)	Number of Power Operated Valves Inside	Outside	Maximum Operating Time (sec)	Normal Position (a)	Action on Initiating 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	20	C	SC
1	Reactor water sample line (B31-F019, B31-F020)	1	1	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	C	SC
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	C	SC
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	C	SC

TABLE 3.7-1 (Cont'd)

PRIMARY CONTAINMENT ISOLATION VALVES WHICH
RECEIVE A PRIMARY CONTAINMENT ISOLATION SIGNAL

Isolation Group (b)	Valve Identification (d)	Number of Power Operated Valves Inside	Number of Power Operated Valves Outside	Maximum Operating Time (sec)	Normal Position (a)	Action on Initiating Signal (a)
6	RHR reactor shutdown cooling suction (supply) (E11-F008, E11-F009)	1	1	24	C	SC
6	RHR reactor head spray (E11-F022, E11-F023)	1	1	20/12	C	SC
3	HPCI - turbine steam (E41-F002, E41-F003)	1	1	50	0	GC
4	RCIC - turbine steam (E51-F007, E51-F008)	1	1	20	0	GC
5	Reactor water cleanup from recirculation loop (G31-F001, G31-F004)	1	1	30	0	GC
2	Post-accident sampling system supply (B21-F111, B21-F112)		2	5	C	SC
2	Post-accident sampling system return (E41-F122, E41-F121)		2	5	C	SC
2	Core spray test line to suppression pool (E21-F015A,B)		1 each line	57	C	SC

TABLE 3.7-1 (Cont'd)

PRIMARY CONTAINMENT ISOLATION VALVES WHICH
RECEIVE A PRIMARY CONTAINMENT ISOLATION SIGNAL

Isolation Group (b)	Valve Identification (d)	Number of Power Operated Valves Inside	Number of Power Operated Valves Outside	Maximum Operating Time (sec)	Normal Position (a)	Action on Initiating Signal (a)
2	HPCI turbine exhaust vacuum breaker (E41-F111, E41-F104)		2	20	O	GC
2	RCIC turbine exhaust vacuum breaker (E51-F105, E51-F104)		2	20	O	GC
2	Torus drainage and purification suction (G51-F011, G51-F012)		2	12	C	SC
2	RHR drywell spray (E11-F016A,B)		1 each line	11	C	SC
2	RHR test line to the suppression pool (E11-F024A,B; E11-F028A,B)		2 each line	110/26	C	SC
2	RHR to torus spray header (E11-F027A,B; E11-F028A,B)		2 each line	13/26	C	SC
2	RHR heat exchanger to the suppression pool (E11-F011A,B; E11-F026A,B)		2 each line	24	C	SC

TABLE 3.7-1 (Cont'd)

PRIMARY CONTAINMENT ISOLATION VALVES WHICH
RECEIVE A PRIMARY CONTAINMENT ISOLATION SIGNAL

Isolation Group (b)	Valve Identification (d)	Number of Power Operated Valves Inside	Outside	Maximum Operating Time (sec)	Normal Position (a)	Action on Initiating Signal (a)
2	RHR discharge to radwaste (E11-F049, E11-F040)		2	20/32	C	SC
2	Torus ventilation exhaust (T48-F332A,B; T48-F333A,B)	2	2	5	C	SC
2	Drywell ventilation exhaust (T48-F334A,B; T48-F335A,B)	2	2	5	C	SC
3	HPCI pump minimum flow (E41-F012)		1	11	C	SC
3	HPCI pump suction (E41-F042)		1	84	C	SC
4	RCIC pump minimum flow (E51-F019)		1	15	C	SC
4	RCIC pump suction (E51-F031)		1	33	C	SC

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 X-35-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 B	Construction Drain	(1) (2) (4) (6)

Table 3.7-4
(Continued)

Primary Containment Testable Isolation Valves

Penetration Number	Valve Designation	Notes
X-211B	E11-F027B & E11-F028B	(1) (2) (4) (5) (9)
X-217	P33-F007 & F015	(1) (2) (4) (5) (9)

(1) Test duration for all valves and penetrations listed will generally exceed 1 hour.

(2) Test pressures are at least 59 psig for all valves and penetrations except MSIVs which are tested at 28 psig.

(3) MSIV acceptable leakage limit is 11.5 scfh/valve of air.

(4) The total acceptable leakage for all valves and penetrations other than the MSIVs is 0.6 L_a.

(5) Local leak tests on all testable isolation valves shall be performed each major refueling shutdown but in no case at intervals greater than 2 years.

(6) Local leak tests on all testable penetrations shall be performed each major refueling shutdown but in no case at intervals greater than 2 years.

(7) The personnel airlock shall be tested at intervals not to exceed 6 months.

(8) The personnel airlock door seals are tested at 10 psig per 4.7.A.2.e.(2)(c) and 4.7.A.2.e.(2)(d).

(9) Identifies isolation valves that are tested by applying pressure between the inboard and outboard isolation valves. Inboard valve is not tested in the direction required for isolation but will have equivalent or more conservative leakage results.

(10) Identifies isolation valves that are tested by applying pressure between the isolation valve and a manually operated valve such that the isolation valve is tested in the direction required for isolation.

(11) Identifies isolation valves that are tested by applying pressure between the isolation valves and other system valves. Isolation valves not tested in the direction required for isolation will have equivalent or more conservative results.

(12) The RHR system remains water filled post-LOCA. Isolation valves are tested with water.

(13) Identified blind flange that is tested by applying pressure between the flange and a manually operated valve such that the flange is tested in the direction required for isolation.

(14) Identifies isolation valves that are tested by applying pressure between the inboard and outboard 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. 101
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 February 3, 1989, 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. 101, 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

DBMatthews

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

Attachment:
Changes to the Technical Specifications

Date of Issuance: June 20, 1989

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. 164, 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

Original Signed By:
David B. Matthews

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: June 20, 1989

OFFICIAL RECORD COPY

LA:PDII-3	PM:PDII-3	DEST:SPLB	OGC-WF	D:PDII-3
MRood	LCrocker:	JCraig		DBMatthews
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ATTACHMENT TO LICENSE AMENDMENT NO. 101

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 area of change.

Remove Page	Insert Page
3/4 6-3	3/4 6-3
3/4 6-6	3/4 6-6
3/4 6-17	3/4 6-17
3/4 6-18	3/4 6-18
3/4 6-20	3/4 6-20
3/4 6-21	3/4 6-21
3/4 6-22	3/4 6-22

CONTAINMENT SYSTEMS

PRIMARY CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.2 Primary containment leakage rates shall be limited to:

a. An overall integrated leakage rate of:

1. $\leq L_a$, 1.2 percent by weight of the containment air per 24 hours at P_a, 57.5 psig, or

2. $\leq L_t$, 0.849 percent by weight of the containment air per 24 hours at a reduced pressure of P_t, 28.8 psig.

b. A combined leakage rate of:

1. $\leq 0.60\ L_a$ for all penetrations and valves, except for main steam isolation valves, subject to Type B and C tests when pressurized to P_a, and

2. $\leq 0.009\ L_a$ for the following penetrations*:

(a) Main steam condensate drain, penetration 8;

(b) Deleted

(c) Reactor water cleanup, penetration 14;

(d) Equipment drain sump discharge, penetration 18;

(e) Floor drain sump discharge, penetration 19; and

(f) Chemical drain sump discharge, penetration 55;

(g) Deleted

c. 11.5 scf per hour for any one main steam isolation valve when tested at 28.8 psig.**

APPLICABILITY: When PRIMARY CONTAINMENT INTEGRITY is required per Specification 3.6.1.1.

* Potential bypass leakage paths.
** Exemption to Appendix J of 10 CFR 50.

PRIMARY CONTAINMENT AIR LOCK

LIMITING CONDITION FOR OPERATION

3.6.1.3 The primary containment airlock shall be OPERABLE with:

a. Both doors closed except when the airlock is being used for normal transit entry and exit through the containment, then at least one airlock door shall be closed, and

b. An overall airlock leakage rate of $\leq 0.05\ L_a$ at P_a, 57.5 psig.

APPLICABILITY: CONDITIONS 1, 2* and 3.

ACTION:

a. With one primary containment airlock door inoperable, maintain at least the OPERABLE airlock door closed; restore the inoperable airlock door to OPERABLE status within 24 hours or lock the OPERABLE airlock door closed; operation may then continue until performance of the next required overall airlock leakage test provided that the OPERABLE airlock door is verified to be locked closed at least once per 30 days. The provisions of Specification 3.0.4 are not applicable.

b. With the primary containment airlock inoperable, except as a result of an inoperable airlock door, maintain at least one airlock door closed; restore the inoperable airlock to OPERABLE status within 24 hours.

c. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.3 The primary containment airlock shall be demonstrated OPERABLE:

a. The primary containment airlock shall be tested at 6-month intervals at P_a by pressurizing the compartment between the two airlock doors. The leakage shall not exceed $0.05\ L_a$.

b. If the primary containment airlock is opened during periods when primary containment integrity is not required, the test required by 4.6.1.3.a. shall be performed at the end of such periods.

c. If the primary containment airlock is opened during periods when primary containment integrity is required, it shall be tested within 3 days of being opened by pressurizing the gap between the doors seals to ≥ 10 psig for at least 15 minutes. The leakage for each set of doors seals shall not exceed $0.01\ L_a$.

d. If primary containment is required and the primary containment airlock is being opened more frequently than once every 3 days, the test required by 4.6.1.3.c. shall be performed at least once per 3 days during the period of frequent openings.

e. At least once per 6 months by verifying that only one door in the airlock can be opened at a time.

*See Special Test Exception 3.10.1.

TABLE 3.6.3-1

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER	VALVE GROUP[a]	ISOLATION TIME (Seconds)
A. Automatic Isolation Valves		
1. Main Steam Isolation Valves		
2B21-F022 A, B, C and D	1	$3 \leq t \leq 5$
2B21-F028 A, B, C and D	1	$3 \leq t \leq 5$
2. Main Steam Drain Isolation Valves		
2B21-F016	1	20
2B21-F019	1	20
3. Reactor Water Sample Line Isolation Valves		
2B31-F019	1	5
2B31-F020	1	5
4. Drywell Equipment Drain Sump Discharge Isolation Valves		
2G11-F019	2	20
2G11-F020	2	20
5. Drywell Floor Drain Sump Discharge Isolation Valves		
2G11-F003	2	20
2G11-F004	2	20

[a] See Specification 3.3.2, Table 3.3.2-1, for isolation signals that operate each valve group.

TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER	VALVE GROUP[a]	ISOLATION TIME (Seconds)
A. Automatic Isolation Valves (Continued)		
6. Containment Spray Isolation Valves		
2E11-F016 A[b] and B[b]	*	10
2E11-F028 A[b] and B[b]	*	26
7. RHR Heat Exchanger Drain Isolation Valves		
2E11-F011 A and B	*	24
2E11-F026 A and B	*	24
8. Drywell-to-Torus Differential Pressure System Isolation Valves		
2T48-F209	12	5
2T48-F210	12	5
2T48-F211	12	5
2T48-F212	12	5
9. HPCI Steam Line Isolation Valves		
2E41-F002	3	50
2E41-F003	3	50

[a]See Specification 3.3.2, Table 3.3.2-1, for isolation signals that operate each valve group
[b]May be opened on an intermittent basis under administrative control
*Closes upon actuation of the LPCI mode of RHR via a high drywell pressure signal (see item 2.a of Table 3.3.3-1) or a Low Low Low (Level 1) signal from 2B21-N691A,B,C,D (see item 2.b of Table 3.3.3-1).

TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER	VALVE GROUP[a]	ISOLATION TIME (Seconds)
A. Automatic Isolation Valves (Continued)		
14. Drywell Vent and Purge System Isolation Valves		
2T48-F307	6	5
2T48-F308	6	5
2T48-F103	6	5
2T48-F104	6	5
2T48-F118A	6	5
2T48-F118B	6	5
2T48-F324	6	5
2T48-F319	6	5
2T48-F320	6	5
2T48-F340	6	10
2T48-F341	6	10
2T48-F334 A	6	4
2T48-F334 B	6	4
2T48-F335 A	6	4
2T48-F335 B	6	4
15. Drywell Pneumatic System Isolation Valves		
2P70-F002	6	5
2P70-F003	6	5
16. Fission Products Monitoring System Isolation Valves		
2D11-F050	6	5
2D11-F051	6	5
2D11-F052	6	5
2D11-F053	6	5

[a]See Specification 3.3.2, Table 3.3.2.1, for isolation signals that operate each valve group.

TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER	VALVE GROUP[a]	ISOLATION TIME (Seconds)
A. Automatic Isolation Valves (Continued)		
17. Torus Cleanup Vacuum Drag Isolation Valves		
2G51-F011	7	15
2G51-F012	7	15
18. HPCI Turbine Exhaust Vacuum Breaker Isolation Valves		
2E41-F111	8	20
2E41-F104	8	20
19. RCIC Turbine Exhaust Vacuum Breaker Isolation Valves		
2E51-F104	9	20
2E51-F105	9	20
20. H_2O_2 Sampling System Isolation Valves		
2P33-F004	10	5
2P33-F012	10	5
2P33-F002	10	5
2P33-F010	10	5
2P33-F006	10	5
2P33-F007	10	5
2P33-F014	10	5
2P33-F015	10	5
2P33-F003	10	5
2P33-F011	10	5
2P33-F005	10	5
2P33-F013	10	5

[a] See Specification 3.3.2, Table 3.3.2.1, for isolation signals that operate each valve group.

TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER	VALVE GROUP[a]	ISOLATION TIME (Seconds)
A. Automatic Isolation Valves (Continued)		
21. Core Spray System Flow Test Line Isolation Valves		
2E21-F015 A	*	57
2E21-F015 B	*	57
22. Suppression Pool Vent and Purge System Isolation Valves		
2T48-F338	10	5
2T48-F339	10	5
2T48-F318	10	5
2T48-F326	10	5
2T48-F332 A	10	4
2T48-F332 B	10	4
2T48-F333 A	10	4
2T48-F333 B	10	4
23. RHR Shutdown Cooling Suction Isolation Valves		
2E11-F008	11	24

[a] See Specification 3.3.2, Table 3.3.2-1, for isolation signals that operate each valve group
*Closes upon actuation of Core Spray via a high drywell pressure signal (see item 1.b of Table 3.3.3-1) or a Low Low Low (Level 1) signal from 2B21-N691A,B,C,D (see item 1.a of Table 3.3.3-1).

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NOS. 164 AND 101 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

1.0 INTRODUCTION

By letter dated February 3, 1989, Georgia Power Company (the licensee) requested amendments to the Technical Specifications (TS) for the Edwin I. Hatch Nuclear Plant, Units 1 and 2. Specifically, the proposed amendments would modify the TS for Units 1 and 2 to: (1) Change the maximum operating times for certain primary containment isolation valves (PCIVs) to account for a different method of measuring; (2) exclude several unit 1 containment penetrations and PCIVs from the local leak rate test (LLRT) program; (3) revise Unit 1 TS section 4.7.A.2 and Unit 2 TS section 4.6.1.3 to achieve similarity between the two documents, to comply with current 10 CFR 50 Appendix J testing requirements, and to specify an allowable leakage; (4) delete penetration 218A from Unit 1 TS Table 3.7-2; and (5) remove the isolation valves associated with the primary feedwater and the torus drainage and purification systems from Unit 2 TS section 3.6.1.2.

2.0 EVALUATION

2.1 Proposed Change 1 - Change the maximum operating times for certain primary containment isolation valves (PCIVs) to account for a different method of measuring.

The TS for Hatch Units 1 and 2 now contain table listings (Table 3.7-1 for Unit 1 and Table 3.6.3-1 for Unit 2) of power operated, automatically initiated PCIVs showing maximum operating times for isolating upon receipt of an appropriate signal. The operating times shown in the tables are based upon a "light-to-light" measuring method, which was used during the plant functional testing prior to reactor startup. However, the ASME Code, Section XI, requires that valve stroke times be measured from initiation of the actuating signal to the end of the actuating cycle, more commonly referred to as a "switch-to-light" measuring method. The changes to the maximum stroke times proposed by the licensee are merely to account for the change in stroke time measurement from the "light-to-light" method to the "swtich-to-light" method. Actual valve operation does not change and the new "switch-to-light" operating times maintain the primary containment design bases as defined in the Unit 1 and Unit 2 Final Safety Analysis Reports. Since the actual valve operation remains the same, we find this proposed change to be acceptable.

2.2 Proposed Change 2 - Exclude Unit 1 valves 1E41-F021, F022, F040, and F049; valves 1E51-F001, F002, F028 and F040; and containment penetrations X-212, X-213, X-214 and X-215 from the Appendix J LLRT (Type C) program.

The named valves and the associated containment penetrations are those serving the reactor core isolation cooling (RCIC) turbine exhaust and turbine drain lines, and the high pressure coolant injection (HPCI) turbine exhaust and drain lines. These exhaust and drain lines terminate in the torus below the water level of the suppression pool. Since the torus water level is controlled within narrow limits at all times, continuous coverage of piping terminations is assured, thus providing a water seal between the atmospheres inside and outside the torus. Under this condition, Type C testing is not required by Appendix J. The licensee states, however, that leak rate testing of these valves will still be performed in accordance with Section XI of the ASME code as part of the inservice inspection (ISI) program.

The staff previously has approved the exemption of certain valves and associated penetrations from the Type C testing requirements based upon a similar argument that the piping involved terminated below the water level in the torus (Amendment #131, Hatch Unit 1, October 30, 1986; and Amendment #140, Hatch Unit 1, June 5, 1987).

Accordingly, we find acceptable the licensee's proposal to exclude the listed valves and associated penetrations from the Type C testing program, and the deletion of these valves and penetrations from Unit 1 TS Table 3.7-4.

2.3 Proposed Change 3 - Revise Unit 1 TS 4.7.A.2 and Unit 2 TS 4.6.1.3 to achieve similarity between the two documents, to comply with current 10 CFR Part 50, Appendix J testing requirements, and to specify an allowable leakage.

Appendix J, Section III.D.2.b requires that containment air locks be tested at 6-month intervals after initial fuel loading at an internal pressure of not less than Pa. Air locks that are opened when containment integrity is required shall be tested within 3 days after being opened, at a test pressure as specified in the TS. For Hatch Units 1 and 2, Pa is 57.5 psig and the test pressure specified for the 3-day test requirement is at least 10 psig. The existing Unit 1 TS provide no acceptance criteria for leakage resulting from the 3-day test, while Unit 2 states "no detectable seal leakage." Neither of these is suitable to meet the requirement of 10 CFR Part 50, Appendix J, Section III.D.2.b(iv), which requires that, "The acceptance criteria for air lock testing shall be stated in the Technical Specifications."

The change proposed by the licensee would amend Unit 1 TS 4.7.A.2 and Unit 2 TS 4.6.1.3 to achieve identical wording as regards the test requirements for the containment air locks, both in conformance with the requirements of

Appendix J, Sections III.D.2.b.(i), (ii), and (iii) and would state acceptance criteria for allowable leakage in accordance with Appendix J, Section III.D.2.b(iv) for the leak tests.

For the full pressure (Pa) leak tests, the allowable leakage is 0.05 La. This acceptance criterion is now stated in the TS for each unit and would remain unchanged. For the 3-day tests conducted at pressures $\leq$ 10 psig, an acceptable leakage for each set of door seals would be specified as 0.01 La. This reduced leakage rate is comparable to reduced pressure leak rates previously reviewed and approved by the staff for other plants, and is acceptable.

Accordingly, we find the licensee's proposed TS modifications regarding the containment air locks acceptable.

2.4 Proposed Change 4 - Delete penetration 218A from Unit 1 TS Table 3.7-2.

Amendment No. 140, issued on June 5, 1987, deleted penetration 218A from the listing of containment isolation valves subject to Appendix J leak rate testing (TS Table 3.7-4). Penetration 218A should have been deleted from TS Table 3.7-2 (Testable Penetrations with Double O-Ring Seals) at this same time. This proposed change would correct that oversight. The change is administrative in nature and is acceptable.

2.5 Proposed Change 5 - Delete the isolation valves associated with the primary feedwater and the torus drainage and purification systems from Unit 2 TS 3.6.1.2.

10 CFR Part 50, Appendix J, Section III.C states the conditions under which certain containment isolation valves sealed with fluid may be excluded when determining the total combined leakage rate.

Primary feedwater valves 2B21-F010A & B and 2B21-F077A & B (in penetrations 9A and 9B) are expected to remain covered by water following a design basis Loss-of-Coolant Accident (LOCA). Note 30 to Unit 2 FSAR Table 3.8-5 states that these valves are expected to remain covered by water following a design basis LOCA. Further, Note 8 to the same table states that the system remains filled with water post-LOCA, that the valves are tested with water, and that valve leakage is not included in the 0.60 La type B and C tests to determine total local leakage. These valves, therefore, should not have been listed in TS 3.6.1.2, and their removal from the listing merely amounts to an administrative correction.

Unit 2 FSAR subsection 6.2.1.2.2 states that the torus drainage and purification system valves (2G57-F011 and 2G51-F012), which may be open at the time of an accident, receive a signal to close, and that following closure a water seal is established by the suppression pool water. These valves thus will have no gaseous leakage and, in accordance with 10 CFR Part 50, Appendix J, Section III.C, they need not be considered in determining the combined leakage rate. The removal of these valves from the listing in TS 3.6.1.2 is, therefore, acceptable.

3.0 ENVIRONMENTAL CONSIDERATION

These amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and change surveillance requirements. The 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. 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.

4.0 CONCLUSION

The Commission made a proposed determination that the amendments involve no significant hazards consideration which was published in the Federal Register (41 FR 13765) on April 5, 1989, 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.

Principal Contributor: Lawrence P. Crocker, PDII-3/DRP-I/II

Dated: June 20, 1989

DATED June 20, 1989

AMENDMENT NO. 101 TO FACILITY OPERATING LICENSE NPF-5, EDWIN I. HATCH, UNIT 2
AMENDMENT NO. 164 TO FACILITY OPERATING LICENSE DPR-57, EDWIN I. HATCH, UNIT 1

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