

March 6, 1996

Mr. J. T. Beckham, Jr.
Vice President - Hatch
Georgia Power Company
P.O. Box 1295
Birmingham, AL 35201

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SUBJECT: ISSUANCE OF AMENDMENTS - EDWIN I. HATCH NUCLEAR PLANT,
UNITS 1 AND 2 (TAC NOS. M94046 AND M94047)

Dear Mr. Beckham:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 200 to Facility Operating License DPR-57 and Amendment No. 141 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 November 10, 1995.

The amendments revise the TS for containment systems to reflect the adoption of the requirements of 10 CFR Part 50, Appendix J, Option B, and the implementation of a performance-based containment leak-rate testing program at the Edwin I. Hatch Nuclear Plant, Units 1 and 2.

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

Sincerely,

Original signed by:

Kahtan N. Jabbour, Senior Project Manager
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket Nos. 50-321 and 50-366

- Enclosures:
1. Amendment No. 200 to DPR-57
 2. Amendment No. 141 to NPF-5
 3. Safety Evaluation

cc w/encl: See next page

DOCUMENT NAME: G:\HATCH\HAT94046.AMD

OFFICE	DRPE/PD22/LA	DRPE/PD22/PM	SCSB	OGC	DRPE/PD22/D
NAME	L.BERRY	KJABBOUR:cn	CBERLINGER	E.HOLLER	H.BERKON
DATE	2/19/96	2/13/96	JCP 2/14/96	2/23/96	3/6/96
COPY	YES NO	YES NO	YES NO	YES NO	YES NO

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UNITED STATES
NUCLEAR REGULATORY COMMISSION

WASHINGTON, D.C. 20555-0001

March 6, 1996

Mr. J. T. Beckham, Jr.
Vice President - Plant Hatch
Georgia Power Company
P.O. Box 1295
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SUBJECT: ISSUANCE OF AMENDMENTS - EDWIN I. HATCH NUCLEAR PLANT,
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Kahtan N. Jabbour

Kahtan N. Jabbour, Senior Project Manager
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket Nos. 50-321 and 50-366

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cc w/encl: See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

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. 200
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 the Georgia Power Company, acting for itself, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, and City of Dalton, Georgia (the licensees), dated November 10, 1995, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as 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.

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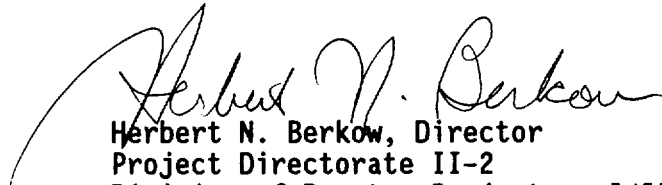
2. Accordingly, the license is hereby amended by page changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-57 is hereby amended to read as follows:

Technical Specifications

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 200 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and shall be implemented within 90 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION


Herbert N. Berkow, Director
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification
Changes

Date of Issuance: March 6, 1996



UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

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. 141
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 the Georgia Power Company, acting for itself, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, and City of Dalton, Georgia (the licensees), dated November 10, 1995, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as 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.

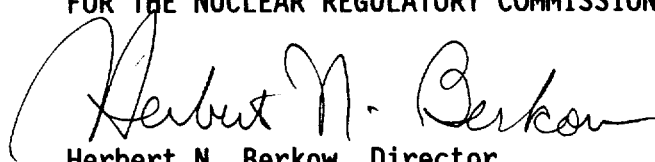
2. Accordingly, the license is hereby amended by page changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-5 is hereby amended to read as follows:

Technical Specifications

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 141 are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and shall be implemented within 90 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION



Herbert N. Berkow, Director
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification
Changes

Date of Issuance: March 6, 1996

ATTACHMENT TO LICENSE AMENDMENT NO. 200

FACILITY OPERATING LICENSE NO. DPR-57

DOCKET NO. 50-321

AND

TO LICENSE AMENDMENT NO. 141

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.

	<u>Remove Pages</u>	<u>Insert Pages</u>
Unit 1	1.1-3	1.1-3
	3.6-2	3.6-2
	3.6-7	3.6-7
	3.6-14	3.6-14
	5.0-16	5.0-16
	---	5.0-16a
	---	5.0-16b
Unit 2	1.1-3	1.1-3
	3.6-2	3.6-2
	3.6-7	3.6-7
	3.6-14	3.6-14
	3.6-15	3.6-15
	5.0-16	5.0-16
	---	5.0-16a
	---	5.0-16b

1.1 Definitions

DOSE EQUIVALENT I-131
(continued)

ICRP 30, Supplement to Part 1, page 192-212, Table titled, "Committed Dose Equivalent in Target Organs or Tissues per Intake of Unit Activity."

END OF CYCLE
RECIRCULATION PUMP TRIP
(EOC-RPT) SYSTEM RESPONSE
TIME

The EOC-RPT SYSTEM RESPONSE TIME shall be that time interval from initial signal generation by the associated turbine stop valve limit switch or from when the turbine control valve hydraulic control oil pressure drops below the pressure switch setpoint to complete suppression of the electric arc between the fully open contacts of the recirculation pump circuit breaker. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.

LEAKAGE

LEAKAGE shall be:

a. Identified LEAKAGE

1. LEAKAGE into the drywell, such as that from pump seals or valve packing, that is captured and conducted to a sump or collecting tank; or
2. LEAKAGE into the drywell atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE;

b. Unidentified LEAKAGE

All LEAKAGE into the drywell that is not identified LEAKAGE;

c. Total LEAKAGE

Sum of the identified and unidentified LEAKAGE;

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.6.1.1.1 Perform required visual examinations and leakage rate testing except for primary containment air lock testing, in accordance with the Primary Containment Leakage Rate Testing Program.</p>	<p>In accordance with the Primary Containment Leakage Rate Testing Program</p>
<p>SR 3.6.1.1.2 Verify drywell to suppression chamber differential pressure does not decrease at a rate > 0.25 inch water gauge per minute tested over a 10 minute period at an initial differential pressure of 1 psid.</p>	<p>18 months <u>AND</u> -----NOTE----- Only required after two consecutive tests fail and continues until two consecutive tests pass ----- 9 months</p>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.6.1.2.1 -----NOTES-----</p> <ol style="list-style-type: none"> 1. An inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test. 2. Results shall be evaluated against acceptance criteria applicable to SR 3.6.1.1.1. <p>-----</p> <p>Perform required primary containment air lock leakage rate testing in accordance with the Primary Containment Leakage Rate Testing Program.</p>	<p>In accordance with the Primary Containment Leakage Rate Testing Program</p>
<p>SR 3.6.1.2.2 -----NOTE-----</p> <p>Only required to be performed upon entry or exit through the primary containment air lock when the primary containment is de-inerted.</p> <p>-----</p> <p>Verify only one door in the primary containment air lock can be opened at a time.</p>	<p>184 days</p>

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
SR 3.6.1.3.6 Verify the isolation time of each MSIV is ≥ 3 seconds and ≤ 5 seconds.	In accordance with the Inservice Testing Program
SR 3.6.1.3.7 Verify each automatic PCIV, excluding EFCVs, actuates to the isolation position on an actual or simulated isolation signal.	18 months
SR 3.6.1.3.8 Verify each reactor instrumentation line EFCV actuates to restrict flow to within limits.	18 months
SR 3.6.1.3.9 Remove and test the explosive squib from each shear isolation valve of the TIP system.	18 months on a STAGGERED TEST BASIS
SR 3.6.1.3.10 Verify leakage rate through each MSIV is ≤ 11.5 scfh when tested at ≥ 28.0 psig.	In accordance with the Primary Containment Leakage Rate Testing Program

(continued)

5.5 Programs and Manuals

5.5.10 Safety Function Determination Program (SFDP) (continued)

A loss of safety function exists when, assuming no concurrent single failure, a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

- a. A required system redundant to system(s) supported by the inoperable support system is also inoperable; or
- b. A required system redundant to system(s) in turn supported by the inoperable supported system is also inoperable; or
- c. A required system redundant to support system(s) for the supported systems (a) and (b) above is also inoperable.

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

5.5.11 Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

- a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.
- b. Licensees may make changes to Bases without prior NRC approval provided the changes do not involve either of the following:
 1. A change in the TS incorporated in the license; or
 2. A change to the FSAR or Bases that involves an unreviewed safety question as defined in 10 CFR 50.59.
- c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the FSAR.

(continued) |

5.5 Programs and Manuals

5.5.11 Technical Specifications (TS) Bases Control Program (continued)

- d. Proposed changes that meet the criteria of b. above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

5.5.12 Primary Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the primary containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995.

The peak calculated primary containment internal pressure for the design basis loss of coolant accident, P_a , is 49.6 psig.

The maximum allowable primary containment leakage rate, L_a , at P_a is 1.2% of primary containment air weight per day.

Leakage rate acceptance criteria are:

- a. Primary containment overall leakage rate acceptance criterion is $\leq 1.0 L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60 L_a$ for the combined Type B and Type C tests, and $\leq 0.75 L_a$ for Type A tests;
- b. Air lock testing acceptance criteria are:
- 1) Overall air lock leakage rate is $\leq 0.05 L_a$ when tested at $\geq P_a$,
 - 2) For each door, leakage rate is $\leq 0.01 L_a$ when the gap between the door seals is pressurized to ≥ 10 psig for at least 15 minutes.

The provisions of SR 3.0.2 do not apply to the test frequencies specified in the Primary Containment Leakage Rate Testing Program.

(continued)

5.5 Programs and Manuals

5.5.12 Primary Containment Leakage Rate Testing Program (continued)

The provisions of SR 3.0.3 are applicable to the Primary Containment Leakage Rate Testing Program.

1.1 Definitions

DOSE EQUIVALENT I-131 (continued)	ICRP 30, Supplement to Part 1, page 192-212, Table titled, "Committed Dose Equivalent in Target Organs or Tissues per Intake of Unit Activity."
EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME	The ECCS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ECCS initiation setpoint at the channel sensor until the ECCS equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays, where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.
END OF CYCLE RECIRCULATION PUMP TRIP (EOC-RPT) SYSTEM RESPONSE TIME	The EOC-RPT SYSTEM RESPONSE TIME shall be that time interval from initial signal generation by the associated turbine stop valve limit switch or from when the turbine control valve hydraulic control oil pressure drops below the pressure switch setpoint to complete suppression of the electric arc between the fully open contacts of the recirculation pump circuit breaker. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.
ISOLATION SYSTEM RESPONSE TIME	The ISOLATION SYSTEM RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its isolation initiation setpoint at the channel sensor until the isolation valves travel to their required positions. Times shall include diesel generator starting and sequence loading delays, where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.6.1.1.1 Perform required visual examinations and leakage rate testing except for primary containment air lock testing, in accordance with the Primary Containment Leakage Rate Testing Program.</p>	<p>In accordance with the Primary Containment Leakage Rate Testing Program</p>
<p>SR 3.6.1.1.2 Verify drywell to suppression chamber differential pressure does not decrease at a rate > 0.25 inch water gauge per minute tested over a 10 minute period at an initial differential pressure of 1 psid.</p>	<p>18 months</p> <p><u>AND</u></p> <p>-----NOTE----- Only required after two consecutive tests fail and continues until two consecutive tests pass -----</p> <p>9 months</p>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.6.1.2.1 -----NOTES-----</p> <ol style="list-style-type: none"> 1. An inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test. 2. Results shall be evaluated against acceptance criteria applicable to SR 3.6.1.1.1. <p>-----</p> <p>Perform required primary containment air lock leakage rate testing in accordance with the Primary Containment Leakage Rate Testing Program.</p>	<p>In accordance with the Primary Containment Leakage Rate Testing Program</p>
<p>SR 3.6.1.2.2 -----NOTE-----</p> <p>Only required to be performed upon entry or exit through the primary containment air lock when the primary containment is de-inerted.</p> <p>-----</p> <p>Verify only one door in the primary containment air lock can be opened at a time.</p>	<p>184 days</p>

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
<p>SR 3.6.1.3.6 Verify the isolation time of each MSIV is ≥ 3 seconds and ≤ 5 seconds.</p>	<p>In accordance with the Inservice Testing Program</p>
<p>SR 3.6.1.3.7 Verify each automatic PCIV, excluding EFCVs, actuates to the isolation position on an actual or simulated isolation signal.</p>	<p>18 months</p>
<p>SR 3.6.1.3.8 Verify each reactor instrumentation line EFCV actuates to restrict flow to within limits.</p>	<p>18 months</p>
<p>SR 3.6.1.3.9 Remove and test the explosive squib from each shear isolation valve of the TIP System.</p>	<p>18 months on a STAGGERED TEST BASIS</p>
<p>SR 3.6.1.3.10 Verify the combined leakage rate for all secondary containment bypass leakage paths is $\leq 0.009 L_a$ when pressurized to $\geq P_a$.</p>	<p>In accordance with the Primary Containment Leakage Rate Testing Program</p>

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
<p>SR 3.6.1.3.11 Verify leakage rate through each MSIV is ≤ 100 scfh, and a combined maximum pathway leakage ≤ 250 scfh for all four main steam lines, when tested at ≥ 28.8 psig.</p> <p>However, the leakage rate acceptance criteria for the first test following discovery of leakage through an MSIV not meeting the 100 scfh limit, shall be ≤ 11.5 scfh for that MSIV.</p>	<p>In accordance with the Primary Containment Leakage Rate Testing Program</p>
<p>SR 3.6.1.3.12 Replace the valve seat of each 18 inch purge valve having a resilient material seat.</p>	<p>18 months</p>
<p>SR 3.6.1.3.13 Cycle each 18 inch excess flow isolation damper to the fully closed and fully open position.</p>	<p>18 months</p>

5.5 Programs and Manuals

5.5.10 Safety Function Determination Program (SFDP) (continued)

A loss of safety function exists when, assuming no concurrent single failure, a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

- a. A required system redundant to system(s) supported by the inoperable support system is also inoperable; or
- b. A required system redundant to system(s) in turn supported by the inoperable supported system is also inoperable; or
- c. A required system redundant to support system(s) for the supported systems (a) and (b) above is also inoperable.

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

5.5.11 Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

- a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.
- b. Licensees may make changes to Bases without prior NRC approval provided the changes do not involve either of the following:
 1. A change in the TS incorporated in the license; or
 2. A change to the FSAR or Bases that involves an unreviewed safety question as defined in 10 CFR 50.59.
- c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the FSAR.

(continued) |

5.5 Programs and Manuals

5.5.11 Technical Specifications (TS) Bases Control Program (continued)

- d. Proposed changes that meet the criteria of b. above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

5.5.12 Primary Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the primary containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995.

The peak calculated primary containment internal pressure for the design basis loss of coolant accident, P_a , is 45.5 psig.

The maximum allowable primary containment leakage rate, L_a , at P_a is 1.2% of primary containment air weight per day.

Leakage rate acceptance criteria are:

- a. Primary containment overall leakage rate acceptance criterion is $\leq 1.0 L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60 L_a$ for the combined Type B and Type C tests, and $\leq 0.75 L_a$ for Type A tests;
- b. Air lock testing acceptance criteria are:
- 1) Overall air lock leakage rate is $\leq 0.05 L_a$ when tested at $\leq P_a$,
 - 2) For each door, leakage rate is $\leq 0.01 L_a$ when the gap between the door seals is pressurized to ≥ 10 psig for at least 15 minutes.

The provisions of SR 3.0.2 do not apply to the test frequencies specified in the Primary Containment Leakage Rate Testing Program.

(continued)

5.5 Programs and Manuals

5.5.12 Primary Containment Leakage Rate Testing Program (continued)

The provisions of SR 3.0.3 are applicable to the Primary Containment Leakage Rate Testing Program.



UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
RELATED TO AMENDMENT NO.200 TO FACILITY OPERATING LICENSE DPR-57
AND AMENDMENT NO. 141 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 November 10, 1995, Georgia Power Company, et al. (the licensee), proposed license amendments to change the Technical Specifications (TS) for the Edwin I. Hatch Nuclear Plant, Units 1 and 2. The proposed changes would revise the TS for containment systems to reflect the adoption of the requirements of 10 CFR Part 50, Appendix J, Option B, and the implementation of a performance-based containment leak-rate testing program. The program "Primary Containment Leakage Rate Testing Program" references Regulatory Guide (RG) 1.163, September 1995, "Performance-Based Containment Leak Test Program" which specifies a method acceptable to the NRC for complying with Option B.

2.0 BACKGROUND

On September 12, 1995, the NRC approved issuance of a revision to 10 CFR Part 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors" which was subsequently published in the Federal Register on September 26, 1995, and became effective on October 26, 1995. The NRC added Option B, "Performance-Based Requirements," to allow licensees to voluntarily replace the prescriptive testing requirements of 10 CFR Part 50, Appendix J, with testing requirements based on both overall leakage rate performance and the performance of individual components.

Compliance with Appendix J provides assurance that the primary containment, including those systems and components, which penetrate the primary containment, do not exceed the allowable leakage rate specified in the TS and Bases. The allowable leakage rate is determined so that the leakage assumed in the safety analyses is not exceeded.

On February 4, 1992, the NRC published a notice in the Federal Register (57 FR 4166) discussing a planned initiative to begin eliminating requirements marginal to safety which impose a significant regulatory burden. Part 50 of 10 CFR, Appendix J, "Primary Containment Leakage Testing for Water-Cooled Power Reactors," was considered for this initiative and the staff undertook a study of possible changes to this regulation. The study examined the

previous performance history of domestic containments and examined the effect on risk of a revision to the requirements of Appendix J. The results of this study are reported in NUREG-1493, "Performance-Based Leak-Test Program."

Based on the results of this study, the staff developed a performance-based approach to containment leakage rate testing. On September 12, 1995, the NRC approved issuance of this revision to 10 CFR Part 50, Appendix J, which was subsequently published in the Federal Register on September 26, 1995, and became effective on October 26, 1995. The revision added Option B, "Performance-Based Requirements," to Appendix J to allow licensees to voluntarily replace the prescriptive testing requirements of Appendix J with testing requirements based on both overall and individual component leakage rate performance.

RG 1.163, September 1995, "Performance-Based Containment Leak Test Program," was developed as a method acceptable to the NRC staff for implementing Option B. This RG states that the Nuclear Energy Institute (NEI) guidance document NEI 94-01, Rev. 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," provides methods acceptable to the NRC staff for complying with Option B with four exceptions which are described therein.

Option B requires that the regulatory guide or other implementation document used by a licensee to develop a performance-based leakage testing program must be included, by general reference, in the plant TS. The licensee has referenced RG 1.163 in the Hatch TS.

RG 1.163 specifies an extension in Type A test frequency to at least one test in 10 years based upon two consecutive successful tests. Type B tests may be extended up to a maximum interval of 10 years based upon completion of two consecutive successful tests and Type C tests may be extended up to 5 years based on two consecutive successful tests.

By letter dated October 20, 1995, NEI proposed TS to implement Option B. After some discussion, the staff and NEI agreed on final TS which were transmitted to NEI in a letter dated November 2, 1995. These TS are to serve as a model for licensees to develop plant-specific TS in preparing amendment requests to implement Option B.

In order for a licensee to determine the performance of each component, factors that are indicative of or affect performance, such as an administrative leakage limit, must be established. The administrative limit is selected to be indicative of the potential onset of component degradation. Although these limits are subject to NRC inspection to assure that they are selected in a reasonable manner, they are not TS requirements. Failure to meet an administrative limit requires the licensee to return to the minimum value of the test interval.

Option B requires that the licensee maintain records to show that the criteria for Type A, B and C tests have been met. In addition, the licensee must maintain comparisons of the performance of the overall containment system

and the individual components to show that the test intervals are adequate. These records are subject to NRC inspection.

3.0 EVALUATION

The licensee's November 10, 1995, letter to the NRC proposes to establish a "Primary Containment Leakage Rate Testing Program" and proposes to add this program to the Hatch TS. The program references RG 1.163, September 1995, "Performance-Based Containment Leak Test Program," which specifies methods acceptable to the NRC for complying with Option B. This requires a change to existing TS 1.1, 3.6.1.1.1, 3.6.1.2.1, 3.6.1.3.10, and the addition of the "Primary Containment Leakage Rate Testing Program" as TS 5.5.12; these changes apply to both Hatch units. For Hatch Unit 2, TS 3.6.1.3.11 is also changed. Corresponding Bases were also modified.

Option B permits a licensee to choose Type A; Type B and C; or Type A, B and C testing to be done on a performance basis. The licensee has elected to perform Type A, B, and C testing on a performance basis.

The TS changes proposed by the licensee are in compliance with the requirements of Option B and consistent with the guidance of RG 1.163, September 1995, and (with minor editorial differences) the model TS of the November 2, 1995, letter and are therefore 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 surveillance requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (60 FR 65679 dated December 20, 1995). 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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