May 28, 2004

Mr. H. L. Sumner, Jr.
Vice President - Nuclear
Hatch Project
Southern Nuclear Operating
Company, Inc.
Post Office Box 1295
Birmingham, Alabama 35201-1295

SUBJECT: EDWIN I. HATCH NUCLEAR PLANT, UNITS 1 AND 2 RE: ISSUANCE OF AMENDMENTS REVISING THE TECHNICAL SPECIFICATIONS FOR THE PRIMARY CONTAINMENT LEAKAGE RATE TESTING PROGRAM (TAC NOS. MC1432 AND MC1433)

Dear Mr. Sumner:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 241 to Renewed Facility Operating License DPR-57 and Amendment No. 184 to Renewed 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 in response to your application dated December 1, 2003, as supplemented on March 10 and March 30, 2004.

The amendments revise the technical specification regarding the primary leakage rate testing program.

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,

/RA/

Christopher Gratton, Senior Manager, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-321 and 50-366

Enclosures:
1. Amendment No. 241 to DPR-57
2. Amendment No. 184 to NPF-5
3. Safety Evaluation

cc w/encls: See next page

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Dated: May 28, 2004

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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 RENEWED FACILITY OPERATING LICENSE

Amendment No. 241
Renewed 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) Renewed Facility Operating License No. DPR-57 filed by Southern Nuclear Operating Company, Inc. (the licensee), acting for itself, Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, and City of Dalton, Georgia (the owners), dated December 1, 2003, as supplemented on March 10 and 30, 2004, 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 Renewed Facility Operating License No. DPR-57 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 241, are hereby incorporated in the license. Southern Nuclear 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 30 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Stephanie M. Coffin, Acting Chief, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification
Changes

Date of Issuance: May 28, 2004

ATTACHMENT TO LICENSE AMENDMENT NO. 241

RENEWED FACILITY OPERATING LICENSE NO. DPR-57

DOCKET NO. 50-321

Replace the following page of the Appendix A Technical Specifications with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

Remove	Insert
5.0-16	5.0-16

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

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 RENEWED FACILITY OPERATING LICENSE

Amendment No. 184
Renewed 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) Renewed Facility Operating License No. NPF-5 filed by Southern Nuclear Operating Company, Inc. (the licensee), acting for itself, Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, and City of Dalton, Georgia (the owners), dated December 1, 2003, as supplemented on March 10 and 30, 2004, 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 Renewed Facility Operating License No. NPF-5 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 184, are hereby incorporated in the license. Southern Nuclear 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 30 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Stephanie M. Coffin, Acting Chief, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification
Changes

Date of Issuance: May 28, 2004

<u>ATTACHMENT TO LICENSE AMENDMENT NO. 184</u>

<u>RENEWED FACILITY OPERATING LICENSE NO. NPF-5</u>

<u>DOCKET NO. 50-366</u>

Replace the following page of the Appendix A Technical Specifications with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

Remove	Insert
5.0-16	5.0-16

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO

AMENDMENT NO. 241 TO RENEWED FACILITY OPERATING LICENSE DPR-57

AND AMENDMENT NO. 184 TO RENEWED FACILITY OPERATING LICENSE NPF-5

SOUTHERN NUCLEAR OPERATING COMPANY, INC., ET AL.

EDWIN I. HATCH NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-321 AND 50-366

1.0 INTRODUCTION

By letter dated December 1, 2003 (Reference 1), Southern Nuclear Operating Company (SNC, the licensee), proposed a change to the Edwin I. Hatch (Hatch) Nuclear Plant, Units 1 and 2, Technical Specifications (TSs), Appendix A to operating licenses DPR-57 and NPF-5, respectively. The licensee proposes to change the peak calculated post-accident primary containment internal pressure values, P_a, in TS Section 5.5.12, “Primary Containment Leakage Rate Testing Program,” for Units 1 and Unit 2. The proposed change supports a 10 psi increase in the nominal reactor steam dome operating pressure at each unit. In response to the Nuclear Regulatory Commission (NRC) staff’s request for additional information (RAI), the licensee provided supplemental information to support the proposed change by letters dated March 10, 2004 (Reference 2) and March 30, 2004 (Reference 3). The supplemental letters provided clarifying information that did not change the scope of the December 1, 2003, application nor the initial proposed no significant hazards consideration determination.

The purpose of the pressure increase in the nominal reactor steam dome pressure is to allow for additional flow control margin for the high pressure turbine. This flow margin is needed to operate the plants at 100 percent of the recently increased (Reference 4) rated thermal power level of 2804 MW(t).

2.0 REGULATORY EVALUATION

Hatch’s primary containment leakage rate testing program (PCLRTP), TS Section 5.5.12, “Primary Containment Leakage Rate Testing Program,” satisfies the requirement that the primary containment meets the leakage rate test requirements of Appendix J to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50 (Reference 5). Hatch uses Option B which identifies the performance-based requirements and criteria for preoperational and subsequent periodic leakage rate testing. This program was designed to ensure that (a) leakage through the primary containment or systems and components penetrating the primary containment does not exceed the allowable leakage rates specified in the TSs and (b) integrity of the containment structure is maintained during its service life.

The PCLRTP satisfies the requirements in General Design Criterion (GDC) 52, “Capability for Containment Leakage Rate Testing,” GDC 53, “Provisions for Containment Testing and Inspection,” and GDC 54, “Systems Penetrating Containment,” and has been previously accepted by the NRC staff. The PCLRTP is consistent with ANSI/ANS-56.8-199 (Reference 6), and the guidance in Regulatory Guide (RG) 1.163 (Reference 7).

This safety evaluation addresses the safety analyses for the increased reactor steam dome nominal operating pressure to establish the post-accident primary containment internal pressure values, P_a, in TS Section 5.5.12 for Units 1 and 2.

3.0 TECHNICAL EVALUATION

The licensee evaluated the 10 psi increase in the nominal reactor steam dome operating pressure under the provisions of 10 CFR 50.59, “Changes, tests and experiments.” The licensee reviewed the balance of plant and the nuclear steam supply system systems, structures, analyses, transients and special events. The impact of the pressure increase on applicable plant programs, such as motor operator valve testing, equipment qualification and Appendix J testing, was also examined by the licensee. As a result of these analyses and evaluations, the licensee determined that, with the exception of the Appendix J testing, the 10 psi increase can be accomplished under the provisions of 10 CFR 50.59 since:

- The increased nominal pressure, from 1035 psig to 1045 psig, is within the reactor steam dome pressure TS 3.4.10 limiting condition for operation requirement of 1058 psig.

- No change is required to any TS allowable value associated with instrument settings that initiate protective functions, including (a) the reactor pressure vessel (RPV) steam dome pressure trip analytical limit or TS allowable value, and (b) the RPV water level trip analytical limits or TS allowable values.

- No change is required to the currently approved rated thermal power level of 2804 MW(t).

- No change is required to the safety relief valve setpoints.

- No change is required to the maximum reactor core flow or power-to-flow map.

Plant implementation of the 10 psi increase will be accomplished through the SNC design change process. Following completion, the affected final safety analysis report (FSAR) sections will be submitted under the provisions of 10 CFR 50.71(e), “Maintenance of records, making of reports,” to reflect the change.

The licensee analyzed the containment response to the design-basis loss-of-coolant accident (LOCA) for the 10 psi increase in the nominal reactor steam dome operating pressure. This pressure increase resulted in a 1 °F increase in the saturation temperature in the steam dome. All other plant parameters remained at their previously accepted values. The previously accepted analytical method, NEDO-10320 (Reference 8), was used for the analyses. The Unit 1 peak containment pressure increased from 50.5 psig to 50.8 psig. The Unit 2 peak containment pressure increased from 46.9 psig to 47.3 psig. The design pressure is 56 psig for both units.

The NRC staff finds the licensee's analyses of the peak containment pressure acceptable since the safety analysis method used has been previously accepted by the NRC staff and, with the exception of the increased pressure and temperature in the reactor steam dome, the other plant parameters remained at their previously acceptable values.

The licensee addressed the long-term suppression pool heatup in response to the NRC staff's RAI. The impact of the pressure and temperature increase on the peak suppression pool temperature was evaluated by the licensee. The previously accepted single bounding analysis from the extended power uprate (Reference 9) long-term containment (peak temperature) analysis was used for the assessment. This analysis, described in the Unit 2 FSAR Section 6.2.3.1.2.2, "Containment Long-Term Response," used conservative inputs and assumptions to maximize the calculated suppression pool temperature response. The calculated peak suppression pool temperature was 208 °F. The licensee's conservative evaluation concluded that the effect of the increased pressure and temperature in the reactor steam dome (sensible energy) on the long-term containment design-basis LOCA response is insignificant (less than 0.3 °F increase in the peak suppression pool temperature). The suppression pool design temperature is 281 °F for Unit 1 and 340 °F for Unit 2.

The existing Hatch long-term analysis takes no credit for passive structural heat sinks in the drywell, and suppression chamber (airspace and pool). After the RPV is depressurized, the suppression pool is the only heat sink considered in the analysis. Since the decay heat, the heat source, is dependent on the initial power level, which is unchanged with the pressure increase, the long-term containment response will not be significantly impacted by the 10 psi reactor steam dome nominal operating pressure increase. The main effect of the pressure increase is the increase in the initial stored sensible energy in the fluid and the solid components within the RPV. The additional sensible energy in the RPV is an insignificant contributor to the overall peak suppression pool temperature response when compared to the other conservative input assumptions used for the long-term containment analysis. The NRC staff agrees with the licensee's assessment that the peak suppression pool temperature during the design-basis LOCA remains bounded by the currently accepted value of 208 °F.

3.1 SUMMARY

The NRC staff finds the licensee's revised safety analyses for the design-basis LOCA, used to determine the values for the proposed change to the peak calculated post-accident primary containment internal pressure value, P_a, in TS Section 5.5.12, "Primary Containment Leakage Rate Testing Program," acceptable. The safety analysis method used has been previously accepted by the NRC staff and, with the exception of the increased pressure and temperature in the reactor steam dome, the other plant parameters remained at their previously accepted values.

The NRC staff finds the licensee's proposed change to TS 5.5.12, increasing P_a from 50.5 psig to 50.8 psig for Unit 1 and from 46.9 psig to 47.3 psig for Unit 2, acceptable. The revised values for P_a, the calculated peak containment internal pressure for the design-basis LOCA, were conservatively calculated using an acceptable safety analysis method.

4.0 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.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendments involve no significant increase in the amounts and no significant change in the types of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (69 FR 2747). 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.

6.0 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.

7.0 REFERENCES

1. H. L. Sumner, Jr., Southern Nuclear Operating Company, letter NL-03-2353, "Edwin I. Hatch Nuclear Plant Proposed Technical Specifications Revision to Primary Containment Leakage Rate Testing Program," to U.S. Nuclear Regulatory Commission, December 1, 2003 (ML033381137).

2. H. L. Sumner, Jr., Southern Nuclear Operating Company, letter NL-04-0371," Edwin I. Hatch Nuclear Plant Clarification of the Proposed Technical Specifications Revision to Primary Containment Leakage Rate Testing Program," to U.S. Nuclear Regulatory Commission, March 10, 2004 (ML040720423).

3. H. L. Sumner, Jr., Southern Nuclear Operating Company, letter NL-04-0392, "Edwin I. Hatch Nuclear Plant Request for Additional Information on the Proposed Technical Specifications Revision to Primary Containment Leakage Rate Testing Program," to U.S. Nuclear Regulatory Commission, March 30, 2004 (ML040930186).

4. S. Bloom, U.S. Nuclear Regulatory Commission, letter, "Edwin I. Hatch Nuclear Plant, Unit 1 and 2 - Issuance of Amendments Regarding Appendix K Measurement Uncertainty Recovery," to H.L. Sumner, Jr., Southern Nuclear Operating Company, September 23, 2003 (ML032590944).

5. Appendix J to Part 50 -- Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors, Option B -- Performance-Based Requirements.

6. American National Standard ANSI/ANS-56.8-1994, “Containment System Leakage Testing Requirements,” dated August 4, 1994.

7. RG 1.163, “Performance-Based Containment Leak-Test Program,” dated September 1995.

8. NEDO-10320, “The General Electric Pressure Suppression Containment Analytical Model,” General Electric Company, April 1972; Supplement 1 May 1971; Supplement 2 January 1973.

9. NEDC-32749P, “Extended Power Uprate Safety Analysis Report for Edwin I. Hatch Nuclear Plants Units 1 and 2,” July 1997.

Principal Contributor: E. Throm

Date: May 28, 2004

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