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

February 20, 2002

SUBJECT: EDWIN I. HATCH NUCLEAR PLANT, UNIT 1 RE: ISSUANCE OF AMENDMENT (TAC NO. MB2842)

Dear Mr. Sumner:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 226 to Facility Operating License DPR-57 for the Edwin I. Hatch Nuclear Plant, Unit 1. The amendment consists of changes to the Technical Specifications (TS) in response to your application dated August 31, 2001, as supplemented by letter dated January 24, 2002.

The amendment revises TS 5.5.12 to allow a one-time deferral of the Type A Containment Integrated Leak Rate Test based on the risk-informed guidance in Regulatory Guide 1.174.

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/
Leonard N. Olshan, Senior Project Manager, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-321

Enclosures:
1. Amendment No. 226 to DPR-57
2. Safety Evaluation

cc w/encls: See next page

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SOUTHERN NUCLEAR OPERATING COMPANY, INC.

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. 226
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 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 August 31, 2001, as supplemented by letter dated January 24, 2002, 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. 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. 226, 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/

Richard J. Laufer, Acting Chief, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification
Changes

Date of Issuance: February 20, 2002

ATTACHMENT TO LICENSE AMENDMENT NO. 226

FACILITY OPERATING LICENSE NO. DPR-57

DOCKET NO. 50-321

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
5.0-16a	5.0-16a
5.0-16b	5.0-16b

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 226 TO FACILITY OPERATING LICENSE DPR-57

SOUTHERN NUCLEAR OPERATING COMPANY, INC., ET AL.

EDWIN I. HATCH NUCLEAR PLANT, UNIT 1

DOCKET NO. 50-321

1.0 INTRODUCTION

By letter dated August 31, 2001, as supplemented by letter dated January 24, 2002, Southern Nuclear Operating Company, Inc. (Southern Nuclear, the licensee), et al., proposed a license amendment to change the Technical Specifications (TS) for the Edwin I. Hatch Nuclear Plant, Unit 1 (Hatch Unit 1). The proposed change would revise TS 5.5.12 to allow a one-time deferral of the Type A Containment Integrated Leak Rate Test (ILRT) based on the risk-informed guidance in Regulatory Guide (RG) 1.174. The supplemental letter dated January 24, 2002, provided clarifying information that did not change the scope of the August 31, 2001, application nor the initial proposed no significant hazards consideration determination.

2.0 BACKGROUND

Title 10 of the Code of Federal Regulations (10 CFR) Part 50, Appendix J, Option B requires that a Type A test be conducted at a periodic interval based on historical performance of the overall containment system. Hatch Unit 1 TS 5.5.12 requires that a program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR Part 50, Appendix J, Option B, as modified by approved exemptions. It further requires that this program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, “Performance-Based Containment Leak-Test Program,” dated September 1995, as modified by exceptions set forth in the TS. This regulatory guide endorses, with certain exceptions, NEI 94-01, Revision 0, “Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J,” dated July 26, 1995.

A Type A test is an overall (integrated) leakage rate test of the containment structure. NEI 94-01 specifies an initial test interval of 48 months, but allows an extended interval of 10 years, based upon two consecutive successful tests. There is also a provision for extending the test interval an additional 15 months in certain circumstances.

The most recent two Type A tests at Hatch Unit 1 have been successful, so their current interval requirement is 10 years.

The licensee is requesting an addition to TS 5.5.12, "Primary Containment Leakage Rate Testing Program," which would indicate that they are allowed to take an exception from the guidelines of RG 1.163 regarding the Type A test interval. Specifically, the proposed TS says that the first Type A test performed after the April 1993 Type A test shall be performed no later than April 2008. This would make the interval 15 years between tests.

3.0 EVALUATION

The licensee has performed a risk impact assessment of extending the Type A test interval to 15 years. In performing the risk assessment, the licensee considered the guidelines of NEI 94-01, the methodology used in EPRI TR-104285, "Risk Impact Assessment of Revised Containment Leak Rate Testing," and RG 1.174, "An Approach For Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis."

The basis for the current 10-year test interval is provided in Section 11.0 of NEI 94-01, Revision 0, and was established in 1995 during development of the performance-based Option B to Appendix J. Section 11.0 of NEI 94-01 states that NUREG-1493, "Performance-Based Containment Leak-Test Program," September 1995, provided the technical basis to support rulemaking to revise leakage rate testing requirements contained in Option B to Appendix J. The basis consisted of qualitative and quantitative assessments of the risk impact (in terms of increased public dose) associated with a range of extended leakage rate test intervals. To supplement the NRC's rulemaking basis, NEI undertook a similar study. The results of that study are documented in EPRI TR-104285.

The EPRI study used an analytical approach similar to that presented in NUREG-1493 for evaluating the incremental risk associated with increasing the interval for Type A tests. The EPRI study estimated that relaxing the test frequency from 3 in 10 years to 1 in 10 years, will increase the average time that a leak detectable only by a Type A test goes undetected from 18 to 60 months. Since Type A tests only detect about 3 percent of leaks (the rest are identified during local leak rate tests based on industry leakage rate data gathered from 1987 to 1993), this results in a 10 percent increase in the overall probability of leakage. The risk contribution of pre-existing leakage, in percent of person-rem/year, for the pressurized water reactor representative plant was estimated to increase from .032 percent to .035 percent. This confirmed the NUREG-1493 conclusion that a reduction in the frequency of Type A tests from 3 per 10 years to 1 per 10 years leads to an "imperceptible" increase in risk.

Building upon the methodology of the EPRI study, the licensee assessed the change in the predicted person-rem/year frequency. The licensee quantified the leakage from sequences that have the potential to result in large releases if a pre-existing leak were present. Since the Option B rulemaking in 1995, the staff has issued RG 1.174 on the use of probabilistic risk assessment in risk-informed changes to a plant's licensing basis. The licensee has proposed using RG 1.174 to assess the acceptability of extending the Type A test interval beyond that established during the Option B rulemaking. RG 1.174 defines very small changes in the risk-acceptance guidelines as increases in core damage frequency (CDF) less than 10^{-6} per reactor year and increases in large early release frequency (LERF) less than 10^{-7} per reactor year. Since the Type A test does not impact CDF, the relevant criterion is the change in LERF. The licensee has estimated the change in LERF for the proposed change and the cumulative change from the original 3 in 10-year interval. RG 1.174 also discusses defense-in-depth and encourages the use of risk analysis techniques to help ensure and show that key principles, such as the defense-in-depth philosophy, are met. The licensee has provided information for estimating the change in the conditional containment failure probability to demonstrate that the defense-in-depth philosophy is met.

The licensee provided analyses to estimate all of these risk metrics. The methodology used is consistent with previously approved submittals. The following conclusions can be drawn from the licensee's analyses associated with extending the Type A test frequency:

1. A slight increase in risk is predicted when compared to that estimated from current requirements. Given the change from a 10-year test interval to a 15-year test interval, the increase in the total integrated plant risk is estimated to be 0.07 percent. The increase in the total integrated plant risk, given the change from a 3 in 10-year test interval to a 15-year test interval, was found to be 0.19 percent. This is reasonable when compared to the range of risk increase, 0.02 to 0.14 percent, estimated in NUREG-1493 when going from a 3 in 10-year test interval to a 10-year interval. NUREG-1493 concluded that a reduction in the frequency of tests from 3 per 10 years to 1 per 10 years leads to an "imperceptible" increase in risk. Therefore, the increase in the total integrated plant risk for the proposed change is considered small and supportive of the proposed change.

2. RG 1.174 provides guidance for determining the risk impact of plant-specific changes to the licensing basis. RG 1.174 defines very small changes in the risk-acceptance guidelines as increases in CDF less than 10^{-6} per reactor year and increases in LERF less than 10^{-7} per reactor year. Since the Type A test does not impact CDF, the relevant criterion is LERF. The increase in LERF resulting from a change in the Type A test interval from 1 in 10 years to 1 in 15 years is the increase in Class 3B frequency which is estimated to be 1.4 x 10^{-8}/year. The increase in LERF resulting from a change in the Type A test interval from the original 3 in 10 years to 1 in 15 years is estimated to be 4.0 x 10^{-8}/year. Increasing the Type A interval to 15 years is considered to be a very small change in LERF.

3. RG 1.174 also encourages the use of risk analysis techniques to help ensure and show that the proposed change is consistent with the defense-in-depth philosophy. Consistency with the defense-in-depth philosophy is maintained if a reasonable balance is preserved among prevention of core damage, prevention of containment failure, and consequence mitigation. For the purpose of this evaluation, the conditional containment failure probability is one minus the fraction of sequences where the containment is intact or the leakage from containment is small divided by the core damage frequency. The change in the conditional containment failure probability is estimated to increase by 0.0011 for the proposed change and 0.0032 for the cumulative change of going from a test interval of 3 in 10 years to 1 in 15 years. The staff finds that the defense-in-depth philosophy is maintained based on the change in the conditional containment failure probability for the proposed change.

Hatch Unit 1 is a General Electric boiling water reactor with a Mark I primary steel containment which consists of a drywell, a wetwell, vents connecting the drywell and wetwell, access penetrations, and process piping and electrical penetrations. The integrity of the penetrations and isolation valves is verified through Type B and Type C local leak rate tests (LLRT) as required by 10 CFR Part 50, Appendix J, and the overall integrity of the containment structure is verified through an ILRT. These tests are performed to verify the essentially leaktight characteristics of the containment structure at the design basis accident pressure. As stated in the request, Hatch Unit 1 has undergone six (6) operational ILRTs (Type A tests) in addition to the pre-operational Type A test during the period of its operating license. The completion dates of these six Type A tests are: May 26, 1974; June 30, 1978; February 2, 1982; April 19, 1986; November 30, 1988; and April 28, 1993. The test results show that the containment leakage rate of all these tests is less than the allowable primary containment leakage rate, L_a, of 1.2 percent of primary containment air weight per day. Based on the six successful Type A tests at Hatch Unit 1 and the requirements of 10 CFR Part 50, Appendix J, Option B, the TS was revised in 1996 to require Type A, B, and C testing in accordance with RG 1.163, "Performance-Based Containment Leak Test Program," which specifies a method acceptable to the NRC by approving the use of NEI 94-01, subject to several regulatory positions in the guide. On this basis, the current interval requirement is 10 years.

With the requested extension of the ILRT time interval, the next overall verification of the containment leaktight integrity will be performed within 15 years from the last ILRT, but no later than April 2008. Because the leak rate testing requirements (ILRT and LLRTs) of Option B of Appendix J, and the containment inservice inspection (ISI) requirements mandated by 10 CFR 50.55a complement each other in ensuring the leaktightness and structural integrity of the containment, the licensee provided additional information related to the ISI of the containment and potential areas of weaknesses in the containment that may not be apparent in the risk assessment. In addition, in the letter dated January 24, 2002, the licensee provided information to explicitly address the five issues raised by the staff during its ILRT review of other plants. The staff's evaluation of the licensee's response to these issues is discussed in the following paragraphs.

Regarding the containment ISI program, the licensee stated that the ISI program at Hatch Unit 1was developed in accordance with the requirements of Subsection IWE of ASME Section XI, 1992 Edition (with the 1992 Addenda), including the NRC-approved requests for relief from certain code requirements. The licensee also stated that the combination of examinations under the Hatch Unit 1 ISI program and visual examination of the accessible and immersed surfaces of the containment in accordance with the requirements of 10 CFR 50.65 and licensing commitments for General Letter 98-04 will provide assurance that the leaktight integrity and the containment structural integrity will be maintained during the extended ILRT period.

For the issue related to the application of any augmented examination and findings at Hatch Unit 1, the licensee stated that periodic examinations in accordance with Table IWE-2500-1, Examination Category E-A, were performed for the interior submerged surfaces of suppression pool (torus), interior torus surfaces exposed to periodic wetting and drying, bottom interior of torus adjacent to safety/relief valve discharge lines, torus seismic restraints, exterior drywell shell below the 114 feet floor elevation (sand cushion area), and drywell equipment hatches and personnel air lock. The results of these examinations performed for these items indicate that the performance of augmented examinations per Table IWE-2500-1, Examination Category E-C, is not presently warranted.

With regard to the issue related to the ISI of seals, gaskets and pressure retaining bolted connections, the licensee explained that with the approved relief requests in these areas, under Option B of Appendix J, the containment leak tight integrity will be, by following the NEI 94-01 guidelines, pressure tested periodically during Type B test. In addition, the licensee stated that the examination of seals and gaskets will continue to be performed each time a specific joint is disassembled. For the bolted connections, testing associated with the primary containment pressure boundary is accomplished by VT-1 examination in accordance with the requirements of Subsection IWE, when disassembled, or examined in place if not disassembled during the interval. Bolting that has not been disassembled and reassembled during the inspection will also be VT-1 examined in the event that the bolting is disassembled. The staff finds that the licensee's ISI program provides reasonable assurance that the integrity of the containment pressure boundary will be maintained during the period of the ILRT extension.

In addressing the staff's question related to the integrity of stainless steel bellows, the licensee stated that because stainless steel expansion bellows are typically covered by a guard plate which encloses the bellows and is welded to the penetration assembly, the guard plate must be removed in order to perform any meaningful examinations. Removing the guard plate will pose the risk of damaging the bellows assembly. Therefore, all bellows are Type B tested in accordance with 10 CFR Part 50, Appendix J, Option B to ensure the leaktight integrity of these bellows. Under Option B of Appendix J testing, these bellows will be pressure-tested every 6 years.

The staff raised a question related to the effect of degradations in areas of the steel containment vessel that cannot be visually examined. Because ILRTs help to identify areas of through-wall degradations when the containment vessel is pressurized, the staff questioned how the potential leakages due to age-related degradation were considered in risk assessment of the extended ILRT. In addressing this staff's concern, the licensee stated that the potential for containment leakage was included in the risk assessment. By definition, the intact containment case, EPRI Containment Failure Class 1, includes a leakage term that is independent of the ILRT interval. Similarly, the Containment Failure Classes 3a and 3b model the potential leakage impact of the ILRT interval extension. These cases include the potential that the leakage is due to a containment shell failure. The assessment shows that even with the increased potential to have an undetected containment flaw or leak path, the increase in risk is insignificant.

In providing further assurance of the containment leaktight integrity, the licensee noted that as a result of nitrogen insertion, the primary containment is maintained at a slightly positive pressure during power operation. The primary containment pressure is continuously recorded and verified by TS surveillance on a frequency of every 12 hours from the main control room. Because the containment is continuously pressurized and is periodically monitored, this design feature provides assurance that any gross containment leakage that may be developed during power operation will be detected.

The staff's review of the basis provided in the TS change request and the response to the five general concerns, the staff finds that (1) the structural integrity of the containment vessel is verified through the periodic inservice inspections conducted as required by Subsections IWE and IWL of the ASME Code, Section XI; (2) the integrity of the penetrations, containment isolation valves and mechanical bellows is periodically verified through Type B and Type C tests as required by 10 CFR Part 50, Appendix J and Hatch Unit 1 TS; and (3) the potential for large leakage from the areas that cannot be examined by the ISI has been explicitly modeled in performing the risk assessment. In addition, the system pressure tests for containment pressure boundary (i.e., Appendix J tests, as applicable) are required to be performed following repair and replacement activities in accordance with Article IWE-5000 of the ASME Code, Section XI. Significant degradation of the primary containment pressure boundary is required to be reported under 10 CFR 50.72 and 10 CFR 50.73.

Based on these conclusions, the staff finds that the increase in predicted risk due to the proposed change is within the acceptance criteria while maintaining the defense-in-depth philosophy of RG 1.174 and, therefore, is acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Georgia State official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes 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 amendment involves 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 amendment involves no significant hazards consideration, and there has been no public comment on such finding (66 FR 52802). Accordingly, the amendment meets 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 amendment.

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.

Principal Contributor: T. Cheng
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Date: February 20, 2002

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