H. L. Sumner, Jr.
Vice President
Hatch Project

Southern Nuclear Operating Company, Inc.
Post Office Box 1295
Birmingham, Alabama 35201

Tel 205.992.7279

SOUTHERN COMPANY
Energy to Serve Your World℠

December 1, 2003

NL-03-2353

Docket Nos.: 50-321
50-366

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D. C. 20555-0001

Edwin I. Hatch Nuclear Plant
Proposed Technical Specifications Revision to
Primary Containment Leakage Rate Testing Program

Ladies and Gentlemen:

In accordance with the provisions of 10 CFR 50.90, Southern Nuclear Operating Company (SNC) hereby proposes a change to the Plant Hatch Units 1 and 2 Technical Specifications (TS), Appendix A to operating licenses DPR-57 and NPF-5, respectively.

This request proposes to change the peak calculated post accident primary containment internal pressure, P_a, in section 5.5.12, "Primary Containment Leakage Rate Testing Program", of the Unit 1 and 2 TS. The change is proposed to support a 10 psi increase in the nominal Unit 1 and 2 reactor operating pressure.

The Unit 1 TS value for P_a is proposed to be changed from 50.5 psig to 50.8 psig. The corresponding Unit 2 value is proposed to be changed from 46.9 psig to 47.3 psig. The TS maximum operating pressure, high pressure scram setpoints, maximum thermal power and SRV setpoints are not changed by this request.

Enclosure 1 provides a description and justification for the change. Enclosure 2 contains the 10 CFR 50.92 evaluation and the justification for the categorical exclusion from performing an environmental assessment. Enclosure 3 provides the marked-up Technical Specifications pages and Enclosure 4 provides the clean typed pages. There are no corresponding Bases pages associated with this Technical Specifications change.

(Affirmation and signature are on the following page).

A017

Mr. H. L. Sumner, Jr. states he is a Vice President of Southern Nuclear Operating Company, is authorized to execute this oath on behalf of Southern Nuclear Operating Company and to the best of his knowledge and belief, the facts set forth in this letter are true.

This letter contains no NRC commitments. If you have any questions, please advise.

Respectfully submitted,

SOUTHERN NUCLEAR OPERATING COMPANY

Lewis Sumner

H. L. Sumner, Jr.

Sworn to and subscribed before me this 1 *day of* December*, 2003.*

Valerie A. O'Brien
Notary Public

My commission expires: 4/28/07

HLS/OCV/sdl

Enclosures:
1. Description and Justification for TS Change
2. No Significant Hazards and the Environmental Assessment
3. Marked-up Pages for TS
4. Clean typed Pages for TS

cc: Southern Nuclear Operating Company
Mr. J. D. Woodard, Executive Vice President
Mr. G. R. Frederick, General Manager – Plant Hatch
Document Services RTYPE: CHA02.004

U. S. Nuclear Regulatory Commission
Mr. L. A. Reyes, Regional Administrator
Mr. S. D. Bloom, NRR Project Manager – Hatch
Mr. D. S. Simpkins, Senior Resident Inspector – Hatch

State of Georgia
Mr. L. C. Barrett, Commissioner – Department of Natural Resources

Enclosure 1

Edwin I. Hatch Nuclear Plant
Technical Specifications Revision
To Primary Containment Leakage Rate Testing Program

Description and Justifications for Change

Description

The Plant Hatch primary containment leakage rate test program ensures that the primary containment meets the leakage-rate test requirements of 10 CFR 50, Appendix J. This program is designed to provide assurance that primary containment leakage does not exceed allowable rates specified in the Technical Specifications (TS), and that the integrity of the containment structure is maintained during its service life. The allowable leakage rate (L_a) with margin is specified in the TS.

Currently, Plant Hatch is optimizing the efficiency of the main turbine by adjusting reactor steam dome pressure within the current reactor coolant system TS requirements, Limiting Condition for Operation (LCO) 3.4.10. An adjustment of a 10 psig increase in the nominal reactor steam dome pressure from 1035 psig to 1045 psig to allow for additional flow control margin for the high pressure turbine is a part of that optimization. For the reasons discussed in the change justification section below, the higher operating pressure results in an increase in the post-accident peak primary containment pressure (P_a). The peak containment pressure is derived from analytical models (Refs. 1 and 2) and is explicitly listed in TS section 5.5.12, "Primary Containment Leakage Rate Testing Program". Consequently, to allow this flow optimization, a TS change is required.

The TS value for P_a is proposed to be changed from 50.5 psig to 50.8 psig for Unit 1. The Unit 2 value is proposed to be changed from 46.9 psig to 47.3 psig. No other TS values are being changed by this proposal. For example, the Unit 1 and 2 TS maximum operating steam dome pressures are unchanged by this proposal since the new operating pressure remains below the TS LCO value of 1058 psig. This value is an assumed initial condition for the overpressurization event. The high pressure scram setpoint (1085 psig for both units) and the SRV setpoints (1150 psig for all eleven SRVs on both units) are not changing since adequate operating margin exists between the TS allowable value and the nominal setpoint.

Justification

Section 6.2.3 of the Unit 2 Final Safety Analysis Report (FSAR) provides the analysis results of the containment response to various postulated accidents for both Hatch Units 1 and 2. Operation with the adjustment in steam dome pressure changes some of the initial conditions assumed in the current licensing basis containment analysis. The short-term analysis is directed primarily at determining the drywell pressure response during the initial blowdown of the reactor vessel inventory to the containment following a large break loss of coolant accident (LOCA) inside the drywell. The long-term analysis is directed primarily at the suppression pool temperature response, considering the long-term heat addition to the suppression pool. The short-term design basis (DBA) LOCA containment response during the reactor blowdown is governed by the blowdown flow rate. This blowdown flow rate is dependent on the reactor initial thermal-hydraulic conditions such as the mass and energy of the vessel fluid inventory, which change slightly with the nominal operating pressure change. The long-term heat-up of the suppression pool following a LOCA or plant transients is governed by the ability of the Residual Heat

Removal (RHR) system to remove decay heat. Since the decay heat depends on the initial reactor power level (which is not changed with this request), the long-term containment response is unaffected by the nominal operating pressure change.

The short-term analysis covers the blowdown period during which the maximum drywell pressures and differential pressures between the drywell and wetwell occur. The licensing basis short-term containment response analyses (Ref. 1) were reevaluated for the limiting DBA LOCA, which assumes a double-ended guillotine break of a recirculation suction line, to determine the effect of operating at a 10 psig increase in nominal operating pressure (Ref. 2). These analyses were performed at 100.5% of the current license power level using methods previously reviewed and accepted by the NRC during the Mark I Containment Long Term Program (LTP).

The effect of the 10 psi nominal operating pressure increase was evaluated using the General Electric M3CPT built-in blowdown model to provide a consistent analysis basis for comparison to the current licensing basis analysis. Unit 1 and Unit 2 initial vessel parameters (e.g., steam flow, feedwater flow, recirculation flow), were used as inputs to the break flow model at 100.5 % power, 100% core flow and normal feedwater temperature operating conditions. The results are summarized in Table 1 below.

Table 1
Short-Term Containment Performance Results

Parameter	Unit 1		Unit 2	
	Current	Proposed	Current	Proposed
Peak Drywell Pressure (psig)	50.5	50.8[1]	46.9	47.3[1]
Peak Bulk Pool Temperature (°F)	208	208	208	208

[1]Design Pressure = 56 psig, 62 psig code allowable maximum

As shown by the results, the calculated peak containment pressure values with the 10 psig adjustment in the nominal reactor operating pressure change slightly and are bounded by the design pressure.

References

1. GE-NE-A13-00402-04, Extended Power Uprate Evaluation Task Report for Edwin I. Hatch Plant Units 1 and 2, Containment Evaluation, Revision 1, September, 1998.

2. GE-NE-0000-0003-0634-01 Revision 1, Project Report – Southern Nuclear Operating Company Edwin I. Hatch Nuclear Plant, Units 1 and 2, 10-PSI Dome Pressure Increase, July, 2003.

Enclosure 2

Edwin I. Hatch Nuclear Plant
Technical Specifications Revision
To Primary Containment Leakage Rate Testing Program

No Significant Hazards Evaluation and Environmental Assessment

Proposed Change

The peak post accident primary containment pressure (P_a), given in Technical Specifications section 5.5.12, is being increased from 50.5 to 50.8 psig on Unit 1 and from 46.9 to 47.3 psig on Unit 2. This change is needed to support an increase in the Units 1 and 2 reactor operating pressures from 1035 psig to 1045 psig.

10 CFR 50.92 Evaluation

10CFR50.92(c), the NRC provides the following standards to be used in determining the existence of a significant hazards consideration:

> ...a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

Southern Nuclear Operating Company (SNC) has reviewed the proposed licensing amendment and has concluded that the change does not involve a significant hazards based upon the following discussion:

1. *Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?*

The proposed change to TS section 5.5.12, "Primary Containment Leakage Rate Testing Program", involves an increase to the peak post accident primary containment pressure. It does not involve physical changes to the primary containment structure itself, nor to any of its support systems and components, nor does it involve changes to any other systems and components designed for the prevention of previously analyzed events. Consequently, the proposed amendment does not involve a significant increase in the probability of occurrence of a previously evaluated event.

The increase in operating pressure for the Hatch reactors from 1035 psig to 1045 psig results in an increase to the peak post-accident primary containment internal pressure. This pressure increases from 50.5 to 50.8 psig for Unit 1 and from 46.9 to 47.3 psig for Unit 2. This is a very small increase with respect to the Unit 1 and 2 primary containment design pressure of 56 psig and with the maximum code allowable pressure of 62 psig. The primary containment thus remains capable of withstanding the post accident pressure and thus the consequences of a previously evaluated event are not increased.

2. *Does the proposed change create the possibility of a new or different kind of accident from any previously evaluated?*

The primary containment boundary will not be altered by the proposed change to Technical Specifications sections 5.5.12, Primary Containment Leakage Rate Testing Program. Furthermore, the primary containment will function as presently described in the Updated Final Safety Analysis Report and will be subject to the same structural and functional requirements. The containment will be operated, maintained and surveilled as before, with the exception of the increased peak post accident pressure, which changes the post accident test pressure acceptance criteria. As a result, no new modes of operation are introduced by this Technical Specifications change and therefore, the possibility of a new or different kind of accident from any previously evaluated is not created.

3. *Does the proposed change involve a significant decrease in the margin of safety?*

The change in the analyzed peak post accident containment pressure will require that the containment be tested to ensure that it meets leakage acceptance criteria at the new pressures of 50.8 psig and 47.3 psig for Units 1 and 2 respectively. Therefore, the primary containment's ability to sustain the slightly higher pressures will be verified during leak rate testing at the required intervals.

The Unit 1 peak pressure increases from 50.5 to 50.8 psig and the Unit 2 pressure increases from 46.9 to 47.3 psig. The primary containment design pressure is 56 psig for both units and the maximum code allowable pressure is 62 psig. Therefore, the margin to the design and maximum code allowable pressures has not been significantly affected. As a result, this proposed Technical Specifications change does not significantly reduce the margin of safety associated with the primary containment function.

Based upon the preceding information, SNC has concluded that the requested change does not involve a significant hazards consideration.

Environmental Assessment

10 CFR 51.22(c)(9) provides criteria for identification of licensing and regulatory actions eligible for categorical exclusion from performing an environmental assessment. A proposed amendment to an operating license for a facility requires no environmental assessment if operation of the facility in accordance with the proposed license amendment does not:

1. Involve a significant hazards consideration,

2. Result in a significant change in the types, or a significant increase in the amount, of any effluents that may be released offsite, or

3. Result in a significant increase in individual or cumulative occupational radiation exposure.

Southern Nuclear has evaluated the proposed changes and determined the changes do not involve (1) a significant hazards consideration, (2) a significant change in the types or significant increase in the amounts of any effluents that may be released off-site, or (3) a significant increase in the individual or cumulative occupational exposure. Accordingly, the proposed change meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9), and an environmental assessment of the proposed change is not required.

Enclosure 3
Edwin I. Hatch Nuclear Plant
Technical Specifications Revision
To Primary Containment Leakage Rate Testing Program

Marked-up Technical Specifications Pages

5.5 Programs and Manuals (continued)

5.5.12 <u>Primary Containment Leakage Rate Testing Program</u>

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, as modified by the following exception to NEI 94-01, Rev. 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J":

> Section 9.2.3: The first Type A test after the April 1993 Type A test shall be performed no later than April 2008.

The peak calculated primary containment internal pressure for the design basis loss of coolant accident, P_a, is ~~595~~ psig.
50.8

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 (continued)

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 ~~[illegible]~~ 47.3 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)

Enclosure 4
Edwin I. Hatch Nuclear Plant
Technical Specifications Revision
To Primary Containment Leakage Rate Testing Program

Clean Typed Technical Specifications Pages

5.5 Programs and Manuals (continued)

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, as modified by the following exception to NEI 94-01, Rev. 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J":

Section 9.2.3: The first Type A test after the April 1993 Type A test shall be performed no later than April 2008.

The peak calculated primary containment internal pressure for the design basis loss of coolant accident, P_a, is 50.8 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 $\geq$ 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 (continued)

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 47.3 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 ≤ 1.0 L_a. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are ≤ 0.60 L_a for the combined Type B and Type C tests, and ≤ 0.75 L_a for Type A tests;

b. Air lock testing acceptance criteria are:

    1) Overall air lock leakage rate is ≤ 0.05 L_a when tested at ≤ P_a,

    2) For each door, leakage rate is ≤ 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)