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April 14, 1997

Docket No. 50-321 HL-5345

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

Edwin I. Hatch Nuclear Plant - Unit 1
Technical Specifications Revision Request for:
Unit 1 Pressure/Temperature Limits

Gentlemen:

In accordance with the provisions of 10 CFR 50.90, as required by 10 CFR 50.59(c)(1), Southern Nuclear Operating Company (SNC) hereby proposes changes to the Plant Hatch Unit 1 Technical Specifications, Appendix A to Operating License DPR-57.

Changes are proposed to Unit 1 Specification 3.4.9, reactor coolant system pressure and temperature limits. Specifically, changes are proposed to the figures which contain the pressure/temperature limits for 1) reactor vessel pressure test, 2) non-nuclear heatup and cooldown and 3) criticality. The changes are made to reflect the latest information from the removal and evaluation of the Unit 1 reactor vessel material capsule. The capsule was removed during the Unit 1 spring outage of 1996.

The evaluation and results of the material capsule surveillance are presented in General Electric report GE-NE-B1100691-01R1, dated March of 1997. This report was submitted to you on March 25, 1997.

In accordance with the requirements of 10 CFR 50.91, a copy of this letter and all applicable enclosures will be sent to the designated State official of the Environmental Protection Division of the Georgia Department of Natural Resources.

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Mr. H. L. Sumner, Jr., states he is Vice President of Southern Nuclear Operating Company and 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.

Sincerely,

Lewis Sumner

H. L. Sumner, Jr.

Sworn to and subscribed before me this 14th *day of* April 1997.

Janet B. Layne
Notary Public

Commission Expiration Date: 11-2-98

OCV/eb

Enclosures:
1. Description and Justification for Proposed Changes
2. 10 CFR 50.92 Evaluation
3. Page Change Instructions and Revised Technical Specifications Pages

cc: *Southern Nuclear Operating Company*
Mr. P. H. Wells, Nuclear Plant General Manager
NORMS

U.S. Nuclear Regulatory Commission, Washington, D.C.
Mr. K. Jabbour, Licensing Project Manager - Hatch

U.S. Nuclear Regulatory Commission, Region II
Mr. L. A. Reyes, Regional Administrator
Mr. B. L. Holbrook, Senior Resident Inspector - Hatch

State of Georgia
Mr. J. D. Tanner, Commissioner - Department of Natural Resources

Enclosure 1

Edwin I. Hatch Nuclear Plant - Unit 1
Technical Specifications Revision Request for:
Unit 1 Pressure/Temperature Limits

Description and Justification for Proposed Changes

Description of Change

Southern Nuclear Operating Company (SNC) proposes to revise the Unit 1 reactor vessel pressure and temperature limits to reflect the data collected from the material sample recovered during the March 1996 Unit 1 Outage. Changes are proposed to Figure 3.4.9-1 (pressure/temperature limits for inservice hydrostatic and inservice leakage tests), Figure 3.4.9-2 (pressure/temperature limits for non-nuclear heatup, low power physics tests, and cooldown following a shutdown) and Figure 3.4.9-3 (pressure/temperature limits for criticality). These changes pertain to the Unit 1 Technical Specifications only.

Justification for Change

Pressure/Temperature (P/T) limits are provided to prevent brittle fracture of the reactor vessel, and are required per Appendix G of 10 CFR 50. Appendix H requires that samples of the RPV material be periodically removed from the reactor vessel. The samples are evaluated for changes in the fracture toughness properties of ferritic materials in the RPV beltline region, which result from exposure to neutron irradiation and the thermal environment.

The RPV surveillance capsule removed contained flux wires for neutron flux monitoring and Charpy V impact and tensile test specimens. The irradiated material properties were compared to available unirradiated properties to determine the effect of irradiation on material toughness for the base and weld materials through Charpy testing. Irradiated tensile testing results are compared with unirradiated data to determine the effect of irradiation on the stress-strain relationship of the materials. These evaluations were performed per the methods of 10 CFR 50 Appendix H.

General Electric report GENE-B1100691-01R1, "Plant Hatch Unit 1 RPV Surveillance Materials Testing and Analysis," March, 1997, documents the data and conclusions from the review of the Unit 1 material sample pulled from the reactor vessel. From the results of this report, we are revising the Hatch 1 Technical Specification Pressure Temperature limit curves to reflect this latest information. There are no changes proposed to the Limiting Condition for Operation or to any of the surveillances of specification 3.4.9. Additionally, the Unit 1 pressure test curve (Figure 3.4.9-1) is being revised to contain the latest exposure dependencies, starting from 16 EFPY to 32 EFPY. All the curves were generated based on the approved methodologies of 10 CFR 50 Appendix G.

In 10 CFR 50.92c, 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 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 an accident of a new and different kind from any previously evaluated; or (3) involve a significant reduction in the margin of safety.

Southern Nuclear Operating Company has reviewed the proposed license amendment request and determined its adoption does not involve a significant hazards consideration based on the following discussion.

Basis for no significant hazards consideration determination:

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

   Pressure and Temperature (P/T) limits for the reactor pressure vessel are established to the requirements of 10 CFR 50, Appendix G to ensure brittle fracture of the vessel does not occur.

   This revision changes the P/T curves in the Unit 1 Technical Specifications to reflect the material capsule surveillance results from the sample removed during the Spring outage of 1996.

   The RPV surveillance capsule contained flux wires for neutron flux monitoring and Charpy V notch impact and tensile test specimens. The irradiated material properties were compared to available unirradiated properties to determine the effect of irradiation on material toughness for the base and weld materials through Charpy testing. Irradiated tensile testing results are compared with unirradiated data to determine the effect of irradiation on the stress-strain relationship of the materials.

   The P/T curves are modified to reflect the results of the above examination. These curves and their operating limits were evaluated using the approved methodologies of 10 CFR 50 Appendix G and ASME Code Appendix G. The new curves therefore represent the latest information available on the state of the reactor vessel materials.

The P/T curves are generated for reactor vessel protection against brittle fracture, they do not affect the recirculation piping. Accordingly, the probability of

occurrence of a design basis Loss of Coolant Accident (LOCA) is not increased. Likewise, no other previously evaluated accident and transients, as defined in Chapter 14 of the Final Safety Analysis Report (FSAR) are affected by this proposed change to the Unit 1 P/T curves. Additionally, this proposed revision does not affect the design, operation, or maintenance of any safety related system designed for the mitigation or prevention of previously analyzed events.

Since no previously evaluated accidents or transients are being affected by this change, their probability of occurrence is not increased and their consequences are not made worse.

2. *Do the proposed changes create the possibility of a new and different type of accident from any previously evaluated?*

Implementing the proposed P/T curves into the Unit 1 Technical Specifications does not alter the design or operation of any system or piece of equipment designed for the prevention or mitigation of accidents and transients. As a result, no new operating modes are introduced from which a new type accident becomes possible. Existing systems will continue to be operated per present design basis assumptions.

The proposed P/T limits were generated from the evaluation of the material capsule removed during the Spring Unit 1 outage of 1996. As a result, these limits include the latest available information on the reactor vessel materials. Furthermore, they will continue to be monitored per the requirements of the Technical Specifications and 10 CFR 50 Appendices G and H. For the above reasons, the changes do not create the possibility of a new type of accident.

3. *Do the proposed changes involve a significant reduction in the margin of safety?*

The purpose of the P/T limits is to avoid a brittle fracture of the reactor vessel. As such, material capsules are removed periodically to determine the effects of neutron irradiation on reactor vessel materials. This change to the Unit 1 P/T curves is proposed to incorporate the evaluation results of the latest capsule removed during the Spring Unit 1 outage of 1996. Accordingly, these curves represent the latest information available on the reactor vessel materials. Also, the curves were generated using the approved methodologies of 10 CFR 50 Appendix G.

The pressure test curve (Figure 3.4.9-1) is also being revised to reflect exposure dependencies. These curves were generated for exposures of 16, 18, 20, 24, 28, and 32 EFPY. As previously described, each of these curves were generated using approved methodologies and all reflect the results of this latest material capsule report.

The proposed change does not affect the evaluation of any FSAR Unit 1 Chapter 14 transient and accident. Furthermore, the proposed change does not affect the operation of systems or equipment important to safety.

The Limiting Condition for Operation of Specification 3.4.9 will not change. Also, no Technical Specification surveillances or surveillance frequencies are revised as a result of this Technical Specification submittal, besides the fact that the P/T surveillances will now refer to the revised curves. Procedures regarding the monitoring of the P/T limits during reactor startup, cooldown, and leakage testing will not change as a result of this proposed Technical Specification change with respect to frequency of the surveillance or the methods used to perform the surveillances. Thus, the P/T limits will continue to be surveilled as before per the same procedures and the same frequencies.

No other Technical Specifications are affected by the proposed revision. The margin of safety to any Technical Specifications safety limit therefore is not reduced.

For the above reasons the new curves do not represent a significant reduction in the margin of safety.

Enclosure 3

Edwin I. Hatch Nuclear Plant - Unit 1
Technical Specifications Revision Request for:
Unit 1 Pressure/Temperature Limits

Page Change Instructions

Unit 1

Page	Instruction
3.4-25	Replace
3.4-26	Replace
3.4-27	Replace

Attachment to Enclosure 3

Technical Specifications

Unit 1

Marked-Up Pages