August 19, 1997

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

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SUBJECT: ISSUANCE OF AMENDMENT - EDWIN I. HATCH NUCLEAR PLANT, UNIT 1 (TAC NO. M98387)

Dear Mr. Sumner:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 207 to Facility Operating License DPR-57 for the Edwin I. Hatch Nuclear Plant, Unit 1. The amendment consist of changes to the Technical Specifications (TS) in response to your application dated April 29, 1997, as supplemented by letter dated May 28, 1997.

The amendment revises the Hatch Unit 1 reactor vessel pressure and temperature limits to reflect data collected from the material sample recovered during the March 1996 Unit 1 outage.

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

Sincerely,

ORIGINAL SIGNED BY:

Ngoc B. (Tommy) Le, Project Manager
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-321

Enclosures: 1. Amendment No. 207 to DPR-57
2. Safety Evaluation

cc w/encl: See next page

DOCUMENT NAME: G:\HATCH\HAT98387.AMD

OFFICE	PD22/PM	PD22/LA	EMEB	OGC	DRPE/PD22/D
NAME	NBLE:cn	LBERRY	for KWICHMAN	C Marco	HBERKOW
DATE	7/19/97	7/9/97	7/28/97	8/8/97	8/18/97
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

August 19, 1997

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Vice President
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Docket No. 50-321

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2. Safety Evaluation

cc w/encl: See next page

Edwin I. Hatch Nuclear Plant
Units 1 and 2

cc:
Mr. Ernest L. Blake, Jr.
Shaw, Pittman, Potts and Trowbridge
2300 N Street, NW.
Washington, DC 20037

Mr. D. M. Crowe
Manager, Licensing
Southern Nuclear Operating
Company, Inc.
P. O. Box 1295
Birmingham, Alabama 35201-1295

Resident Inspector
U.S. Nuclear Regulatory Commission
11030 Hatch Parkway North
Baxley, Georgia 31513

Regional Administrator, Region II
U.S. Nuclear Regulatory Commission
Atlanta Federal Center
61 Forsyth Street, S.W., Suite 23T85
Atlanta, Georgia 30303

Mr. Charles H. Badger
Office of Planning and Budget
Room 610
270 Washington Street, SW.
Atlanta, Georgia 30334

Harold Reheis, Director
Department of Natural Resources
205 Butler Street, SE., Suite 1252
Atlanta, Georgia 30334

Steven M. Jackson
Senior Engineer - Power Supply
Municipal Electric Authority
of Georgia
1470 Riveredge Parkway, NW
Atlanta, Georgia 30328-4684

Charles A. Patrizia, Esquire
Paul, Hastings, Janofsky & Walker
10th Floor
1299 Pennsylvania Avenue
Washington, DC 20004-9500

Chairman
Appling County Commissioners
County Courthouse
Baxley, Georgia 31513

Mr. W. G. Harriston, III
President and Chief Executive
Officer
Southern Nuclear Operating
Company, Inc.
P. O. Box 1295
Birmingham, Alabama 35201-1295

Mr. J. D. Woodard
Executive Vice President
Southern Nuclear Operating
Company, Inc.
P. O. Box 1295
Birmingham, Alabama 35201-1295

Mr. P. W. Wells
General Manager, Edwin I. Hatch
Nuclear Plant
Southern Nuclear Operating
Company, Inc.
P. O. Box 439
Baxley, Georgia 31513

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. 207
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. (Southern Nuclear), acting for itself, Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, and City of Dalton, Georgia (the licensees), dated April 29, 1997, as supplemented May 28, 1997, 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. 207 , 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 from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Herbert N. Berkow

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: August 19, 1997

ATTACHMENT TO LICENSE AMENDMENT NO. 207

FACILITY OPERATING LICENSE NO. DPR-57

DOCKET NO. 50-321

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.

Remove	Insert
3.4-25	3.4-25
3.4-26	3.4-26
3.4-27	3.4-27

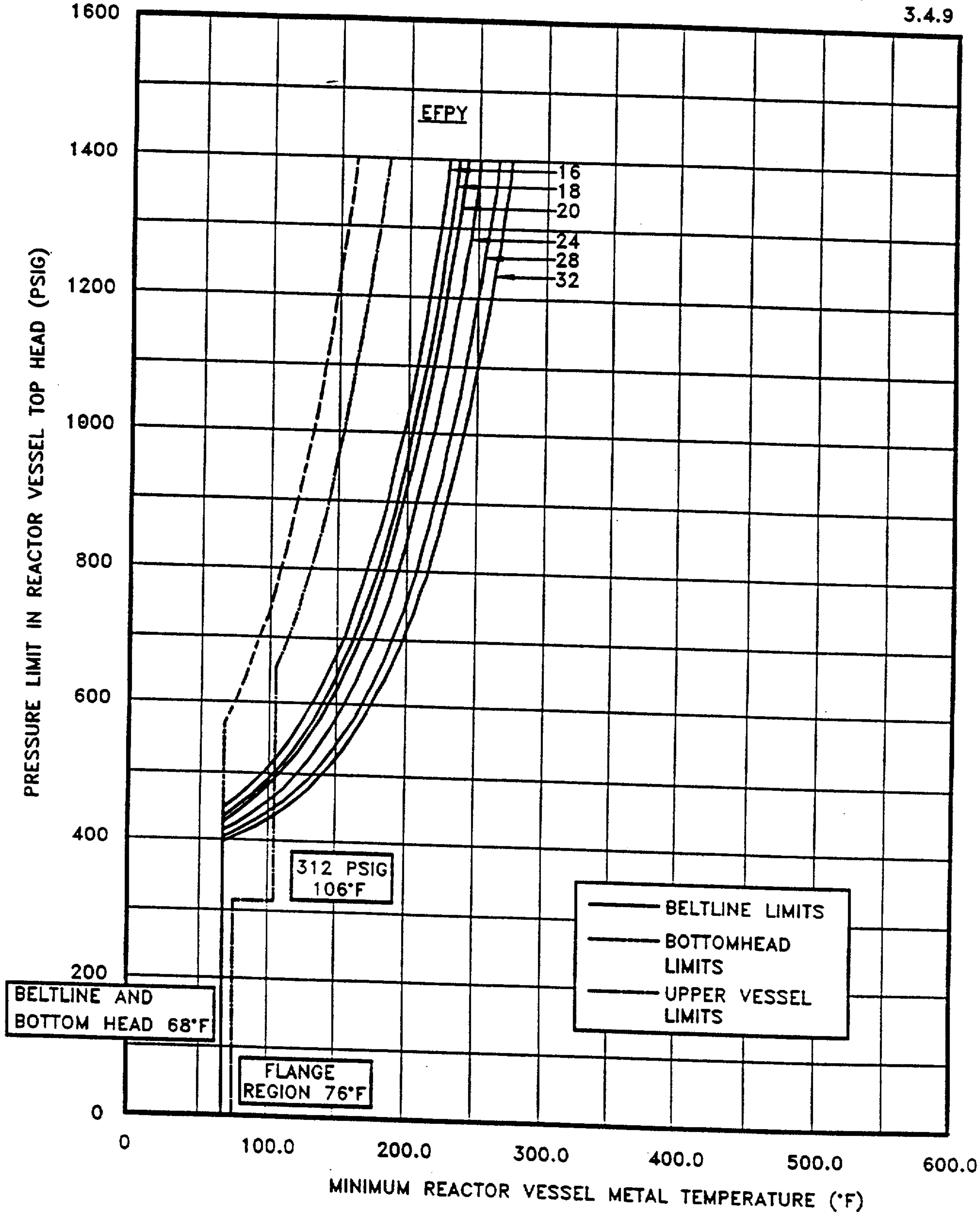

Figure 3.4.9-1 (Page 1 of 1)
Pressure/Temperature Limits for
Inservice Hydrostatic and Inservice Leakage Tests

ACAD FIG34-25

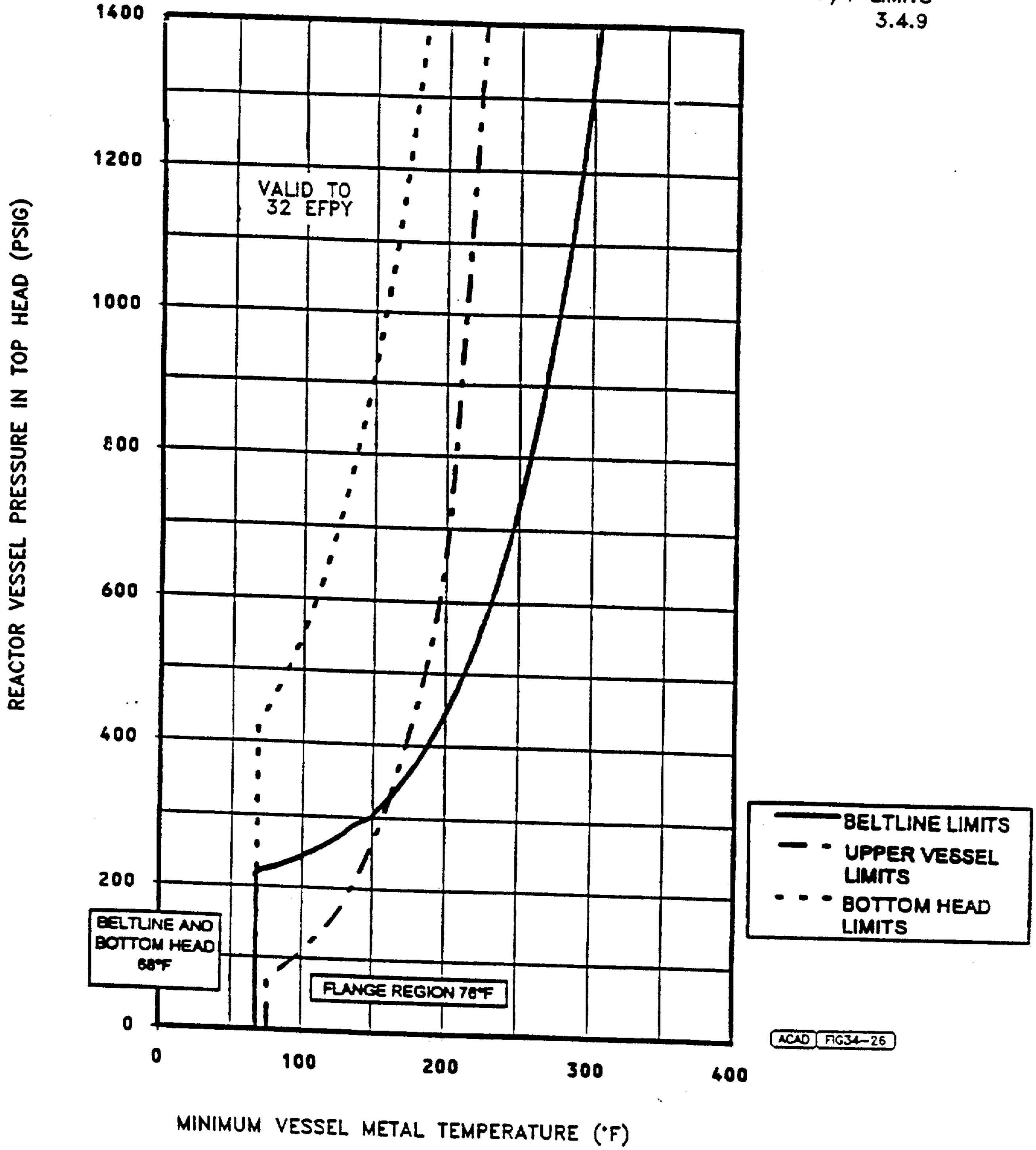

Figure 3.4.9-2 (Page 1 of 1)
Pressure/Temperature Limits for Non-Nuclear Heat-up, Low Power Physics Tests, and Cooldown Following a Shutdown

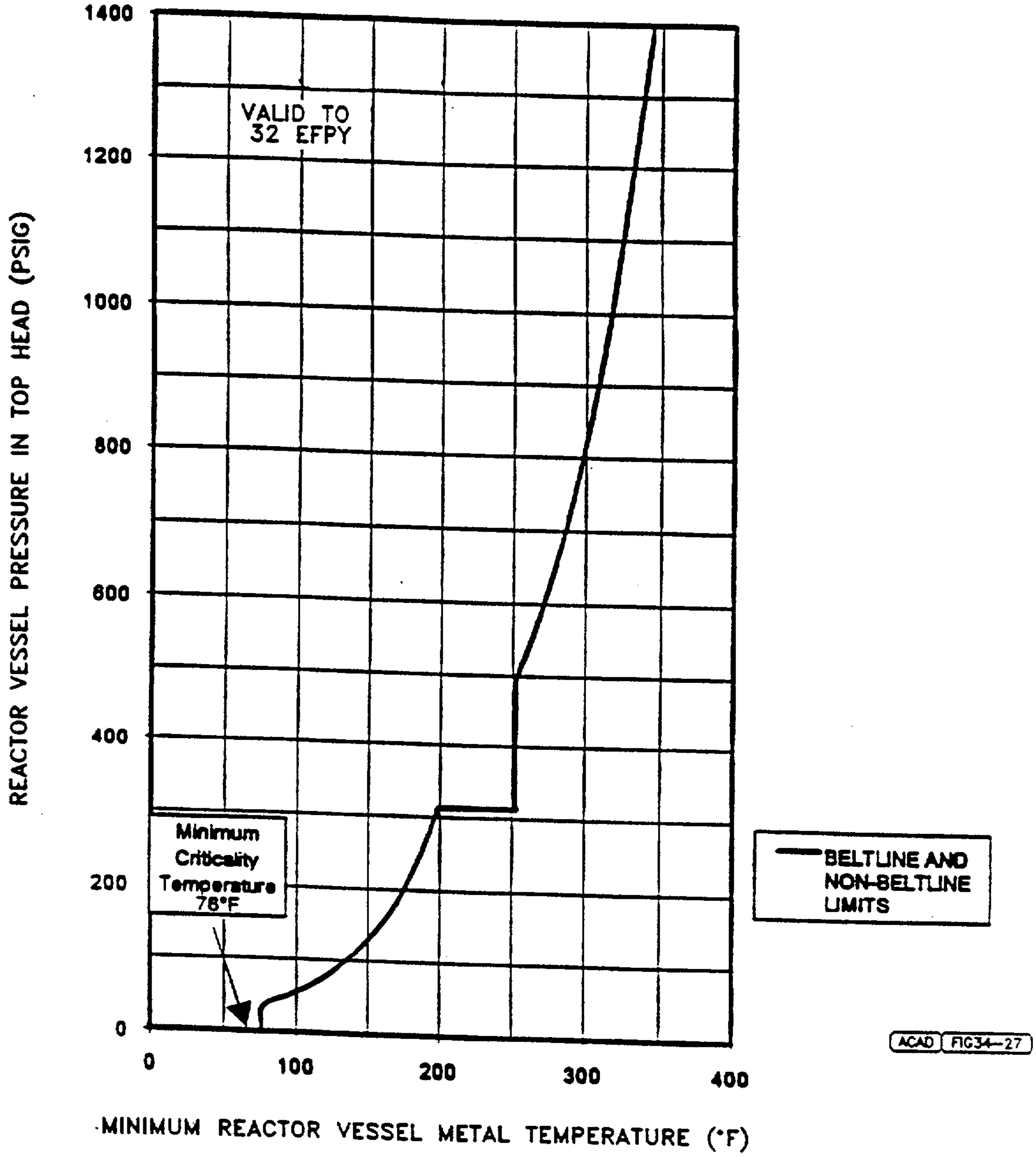

Figure 3.4.9-3 (Page 1 of 1)
Pressure/Temperature Limits for Criticality

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 207 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 April 29, 1997, as supplemented May 28, 1997, Southern Nuclear Operating Company, Inc. (Southern Nuclear), et al. (the licensee) proposed a license amendment to change the Technical Specifications (TS) for the Edwin I. Hatch Nuclear Plant, Unit 1. The proposed changes would revise the Unit 1 reactor vessel pressure and temperature limits to reflect data collected from the material sample recovered during the March 1996 Unit 1 outage.

2.0 EVALUATION

By letter dated April 14, 1997, the licensee submitted proposed changes related to the pressure-temperature (P-T) limits in the Hatch Unit 1 TS. The submittal was incomplete and was superseded by a new submittal dated April 29, 1997. Additional information regarding initial reference temperatures was also supplied in a letter dated May 28, 1997. The changes are the result of removal and evaluation of the surveillance capsule at the 120° azimuthal location in the Hatch Unit 1 reactor vessel. The capsule was removed at 14.3 effective full power years (EFPY). The licensee revised the P-T limits to provide new limits that are valid to 32 EFPY.

Previously, a safety evaluation (SE) for the Hatch Units 1 and 2 P-T limits was completed and issued by letter dated April 4, 1997. The April 4, SE did not reflect the results from the evaluation of the Hatch Unit 1 surveillance capsule since the information had not yet been provided.

In addition, separate limits were approved for the upper vessel and bottom head regions. The separate curves were developed from the generic pressure (P) vs. temperature minus RT_{NDT} ($T-RT_{NDT}$) values from a General Electric (GE) analysis for a large boiling water reactor/6 (BWR/6) reactor pressure vessel (see Section 2.2 of the April 4 SE for more detail).

PRESSURE-TEMPERATURE LIMITS

The staff evaluates the P-T limits based on the following NRC regulations and guidance: Appendix G to 10 CFR Part 50; Generic Letters (GLs) 88-11 and 92-01; Regulatory Guide (RG) 1.99, Revision 2; and Standard Review Plan (SRP) Section 5.3.2. Appendix G to 10 CFR Part 50 requires that P-T limits for the reactor vessel must be at least as conservative as those obtained by Appendix G to Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (ASME Code). GL 88-11 requires that licensees use the methods in RG 1.99, Revision 2, to predict the effect of neutron irradiation on the adjusted reference temperature (ART) of reactor vessel materials. The ART is defined as the sum of initial nil-ductility transition reference temperature (RT_{NDT}) of the material, the increase in RT_{NDT} caused by neutron irradiation (ΔRT_{NDT}), and a margin to account for uncertainties in the prediction method.

The increase in RT_{NDT} is calculated from the product of a chemistry factor (CF) and a fluency factor. The chemistry factor may be calculated using credible surveillance data, obtained by the licensee's surveillance program, as directed by Position 2 of RG 1.99, Revision 2. If credible surveillance data are not available, the chemistry factor is calculated dependent upon the amount of copper and nickel in the vessel material as specified in Table 1 of RG 1.99, Revision 2. GL 92-01 requires licensees to submit reactor vessel materials data, which the staff uses in the review of the P-T limit submittals.

Standard Review Plan 5.3.2 provides guidance on calculation of the P-T limits using linear elastic fracture mechanics methodology specified in Appendix G to Section III of the ASME Code. The linear elastic fracture mechanics methodology postulates sharp surface defects that are normal to the direction of maximum stress and have a depth of one-fourth of the reactor vessel beltline thickness (1/4T) and a length of 1-1/2 times the beltline thickness. The critical locations in the vessel for this methodology are the 1/4T and 3/4T locations, which correspond to the maximum depth of the postulated inside surface and outside surface defects, respectively.

EQUIVALENT MARGIN ANALYSIS

Appendix G also requires that the predicted Charpy upper-shelf energy (USE) at end-of-license (EOL) for vessel beltline materials be above 50 ft-lb or that licensees demonstrate that lower values of Charpy USE will provide margins of safety equivalent to those required by Appendix G of Section XI of the ASME Code. ASME Code Case N-512 and Appendix K contain analytical procedures and acceptance criteria for demonstrating that reactor vessel beltline materials with low Charpy USE will have margins of safety against fracture equivalent to Appendix G of the ASME Code.

In a December 9, 1993, letter to L.A. England from J.T. Wiggins (USNRC), the staff issued the Safety Evaluation Report of the GE topical report NEDO-32205, Revision 1, "10 CFR 50, Appendix G, Equivalent Margin Analysis for Low Upper Shelf Energy in BWR/2 through BWR/6 Vessels." The staff concluded that the reactor pressure vessels of the participating utilities should have margins of safety against ductile fracture in low USE plates and welds until their expiration of licenses (EOL) for level A, B, C, and D conditions, and meet the criteria of ASME Code Case N-512 and Appendix K. Individual licensees that reference the topical report as their basis for addressing the USE requirements of 10 CFR Part 50, Appendix G, were requested to confirm the plant-specific applicability of the topical report by comparing the predicted percentage drop in the USE to the allowable decrease in the USE from the topical report.

The April 4, P-T limits SE that was issued by the staff also requested that the licensee address the plant-specific applicability of the Hatch reactor pressure vessel materials to the GE topical report NEDO-32205, Revision 1, "10 CFR 50, Appendix G, Equivalent Margin Analysis for Low Upper Shelf Energy in BWR/2 through BWR/6 Vessels." The plant-specific equivalent margins analysis (EMA) for Hatch Unit 1 was included in the current submittal. The results from the EMA were compared to the allowable decrease in USE from topical report NEDO-32205.

PRESSURE-TEMPERATURE LIMITS

As part of the review, the basis for the initial reference temperature values for all beltline materials were revisited since many of the values differed from the values that were reported in response to GL 92-01. In response to GL 92-01, the limiting plate initial RT_{NDT} of 10°F was conservatively applied to all beltline plates even though data were available for the other plates. Similarly, the limiting weld initial RT_{NDT} of -10°F was conservatively applied to all beltline welds. By letter dated May 28, 1997, the licensee provided detailed justification for all initial RT_{NDT} values including the certified material test reports. The staff reviewed all calculations and raw data that were used to determine the initial RT_{NDT} and found all values acceptable. Table 1 shows a comparison of the previous values and the values reported in this submittal. The changes will be included in the next update of the reactor vessel integrity database.

RG 1.99, Revision 2, Position 2.1, requires that the chemistry factor (CF) for welds be adjusted based on credible surveillance capsule test results. In this procedure, the adjustment is based on the ratio of the chemistry factor from the surveillance material and the chemistry (copper and nickel) of the weld. The licensee used a similar procedure for its plate surveillance material. The CF that results from the least squares fit of the surveillance data is 221.7°F. From Table 2 of RG 1.99, Revision 2, the CF for the surveillance plate with copper = 0.12% and nickel = 0.70% is 84.5°F. Therefore, the plate adjustment is 2.62 (221.7/84.5). This adjustment was conservatively applied to all of the beltline plates. No unirradiated data were available for the weld material, so the CFs for the welds were calculated using Position 1 of RG 1.99, Revision 2.

For the Hatch Unit 1 reactor vessel, the licensee determined that the most limiting material at the 1/4T and 3/4T locations is the lower-intermediate plate G-4803-7. This plate was fabricated using plate heat C4337-1. The licensee calculated an ART of 153°F at the 1/4T location and 112°F at the 3/4T location at 32 EFPY. The neutron fluency used in the ART calculation was 1.3×10^{18} n/cm^2 at the 1/4T location and 0.6×10^{18} n/cm^2 at the 3/4T location. The initial RT_{NDT} for the limiting plate was -20°F. The margin term used in calculating the ART for the limiting plate was 17°F as permitted by Position 2.1 of RG 1.99, Revision 2.

The staff performed an independent calculation of the ART values for the limiting material using the methodology in RG 1.99, Revision 2. Based on these calculations, the staff verified that the licensee's limiting material for the Hatch Unit 1 reactor vessel is the lower-intermediate plate G-4803-7 that was fabricated using plate heat C4337-1. The staff's calculated ART value for the limiting material agreed with the licensee's calculated ART value.

In support of the initial reference temperature evaluation, the staff reviewed weld wire heat data in the reactor vessel integrity database (RVID). Other initial reference temperature values from plants with data from welds fabricated using the same heats of weld wire as in the Hatch Unit 1 welds were reported for heats 13253 and 33A377.

If generic initial reference temperature values are used for weld wire heats 13253 and 33A377, the resulting ART values increase slightly. However, the welds do not become limiting.

Substituting the ART values for the Hatch Unit 1 limiting plate into equations in SRP 5.3.2, the staff verified that the proposed P-T limits satisfy the requirements in Paragraph IV.A.2 of Appendix G of 10 CFR Part 50.

In addition to beltline materials, Appendix G of 10 CFR Part 50 also imposes a minimum temperature at the closure head flange based on the reference temperature for the flange material. Section IV.A.2 of Appendix G states that when the pressure exceeds 20% of the preservice system hydrostatic test pressure, the temperature of the closure flange regions highly stressed by the bolt preload must exceed the reference temperature of the material in those regions by at least 120°F for normal operation and by 90°F for hydrostatic pressure tests and leak tests. Based on the flange RT_{NDT} of 16°F for Hatch Unit 1 provided by the licensee, the staff has determined that the proposed P-T limits have satisfied the requirement for the closure flange region during normal operation and hydrostatic pressure test and leak test.

EQUIVALENT MARGIN ANALYSIS

Methods acceptable to the staff for determining the percentage decrease in USE are documented in RG 1.99, Revision 2. Figure 2 in the RG indicates that the percentage decrease in USE increases with increasing amounts of copper and neutron fluency. However, the percentage decrease in USE could be affected by surveillance test results. If surveillance data indicate that the percentage decrease in USE is greater than the amount predicted by Figure 2 in the RG, the percentage decrease in USE for the material must be increased. If surveillance data indicate that the percentage decrease in USE is less than the amount predicted by Figure 2, the percentage decrease in USE for the material may be decreased from the amount predicted by Figure 2.

In the current submittal, the licensee compared the USE decrease from the surveillance plate to the amount of decrease predicted by RG 1.99, Revision 2, and the allowable limits in NEDO-32205. The licensee reported that the percent decrease in USE values for the surveillance plate are less than that from using Figure 2 in the RG. In addition, the predicted USE decrease of 18% for the limiting plate is less than the allowable limit of 21% for plates from NEDO-32205. The predicted USE decrease of 28% for the limiting weld is less than the allowable limit of 34% for welds from NEDO-32205. Since no unirradiated data were available for the weld material, the licensee used Figure 2 in the RG to obtain the percent decrease in USE value. The percent decrease in USE from the first to the second capsule was reported for information only, and was not used to obtain the USE decrease of 28% for the limiting weld. Therefore, both plates and welds meet the allowable limits of NEDO-32205.

3.0 STAFF CONCLUSION

PRESSURE-TEMPERATURE LIMITS

The staff has performed an independent analysis to verify the licensee's proposed P-T limits. The staff concludes that the proposed P-T limits are valid to 32 EFPY since the limits conform to the requirements of Appendix G of 10 CFR Part 50 and GL 88-11. Hence, the proposed P-T limits may be incorporated in the Hatch Unit 1 Technical Specifications.

EQUIVALENT MARGIN ANALYSIS

Based on its evaluation, the staff concludes that the projected decreases in USE for the beltline materials are less than the allowable decreases in USE from topical report NEDO-32205. Consequently, the applicability requirements of NEDO-32205 have been satisfied and the conclusions of the topical report are applicable to the Hatch Unit 1 reactor vessel. As a result, the Hatch Unit 1 reactor vessel satisfies the criteria in ASME Code Case N-512 and Appendix K, and is projected to have margins of safety against fracture that are equivalent to those required by Appendix G of the ASME Code at expiration of license. Therefore, the Hatch Unit 1 reactor vessel also meets the requirements of Appendix G to 10 CFR Part 50.

TABLE 1

COMPARISON OF PREVIOUS AND CURRENT INITIAL REFERENCE TEMPERATURE VALUES

BELTLINE DESCRIPTION/WELD TYPE	HEAT IDENTIFICATION	PREVIOUS INITIAL REFERENCE TEMPERATURE VALUE (°F)	CURRENT REFERENCE TEMPERATURE VALUE (°F)
PLATES:			
Lower			
G-4805-1	C4112-1	10	8
G-4805-2	C4112-2	10	10
G-4805-3	C4149-1	10	-10
Lower-Intermediate			
G-4803-7	C4337-1	10	-20
G-4804-1	C3985-2	10	-20
G-4804-2	C4114-2	10	-20
WELDS:			
Lower Longitudinal 1-307	13253	-10	-50
Lower-Intermediate Longitudinal 1-308	IP2809	-10	-50
	IP2815	-10	-50
Lower to Lower-Int. Girth 1-313	90099	-10	-10
	33A277	-10	-50

3.0 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.

4.0 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 (62 FR 38138 dated July 16, 1997). 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.

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 amendment will not be inimical to the common defense and security or to the health and safety of the public.

Attachment: References

Principal Contributor: A. D. Lee

Date: August 19, 1997

REFERENCES

1. Regulatory Guide 1.99, Radiation Embrittlement of Reactor Vessel Materials, Revision 2, May 1988.

2. NUREG-0800, Standard Review Plan, Section 5.3.2: Pressure-Temperature Limits.

3. Code of Federal Regulations, Title 10, Part 50, Appendix G, Fracture Toughness Requirements.

4. Generic Letter 88-11, NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Impact on Plant Operations, July 12, 1988.

5. ASME Boiler and Pressure Vessel Code, Section III, Appendix G for Nuclear Power Plant Components, Division 1, "Protection Against Nonductile Failure."

6. October 23, 1996, Letter from D. Pickett to P. Telthorst, Subject: Issuance of Amendment No. 109 to Facility Operating License No. NPF-62-Clinton Power Station, Unit 1 (TAC # M94887).

7. February 28, 1997, Letter from J.F. Stang to I. Johnson, Subject: Issuance of Amendments (TAC NOS. M96898, M96899, M96900 AND M96901).

8. April 4, 1997, Letter from K.N. Jabbour (USNRC) to H.L. Sumner, Jr., Subject: Issuance of Amendments - Edwin I. Hatch Nuclear Plant, Units 1 and 2 (TAC NOS. M96609 and M96610).

9. April 14, 1997, Letter from H.L. Sumner, Jr., to USNRC Document Control Desk, Subject: Edwin I. Hatch Nuclear Plant - Unit 1 Technical Specifications Revision Request for: Unit 1 Pressure/Temperature Limits.

10. April 29, 1997, Letter from H.L. Sumner, Jr., to USNRC Document Control Desk, Subject: Edwin I. Hatch Nuclear Plant - Unit 1 Technical Specifications Revision Request for: Unit 1 Pressure/Temperature Limits.

11. May 28, 1997, Letter from H.L. Sumner, Jr., to USNRC Document Control Desk, Subject: Edwin I. Hatch Nuclear Plant - Unit 1 Response to Verbal Request for Information on P-T limits Technical Specification Revision Request.