December 31, 1990

Docket No. 50-364

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Mr. W. G. Hairston, III
Senior Vice President
Alabama Power Company
40 Inverness Center Parkway
Post Office Box 1295
Birmingham, Alabama 35201

Dear Mr. Hairston:

SUBJECT: ISSUANCE OF AMENDMENT NO. 81 TO FACILITY OPERATING LICENSE NO. NPF-8 REGARDING PRESSURE-TEMPERATURE LIMITS RELATING TO GENERIC LETTER 88-11 - JOSEPH M. FARLEY NUCLEAR PLANT, UNIT 2, (TAC NO. 77542)

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 81 to Facility Operating License NPF-8 for the Joseph M. Farley Nuclear Plant, Unit 2. The amendment consists of changes to the Technical Specifications in response to your submittal dated August 27, 1990.

The amendment changes the Technical Specifications to provide heatup and cool down curves applicable to the first 14 effective full power years for the reactor.

A copy of our related Safety Evaluation is also enclosed. A Notice of Issuance will be included in the Commission's bi-weekly Federal Register notice.

Sincerely,

Original Signed By:

Stephen T. Hoffman, Project Manager
Project Directorate II-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 81 to NPF-8
2. Safety Evaluation

cc w/enclosures:
See next page

OFC	:LA:PD21:DRPR	:PM:PD21:DRPR	:D:PD21:DRPR	:	:	:	:
NAME	: PAnderson	: SHoffman:sw	: EAdensam	:	:	:	:
DATE	: 12/27/90	: 12/28/90	: 12/31/90	:	:	:	:

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Document Name: FARLEY AMENDMENT77542

Mr. W. G. Hairston, III
Alabama Power Company

Joseph M. Farley Nuclear Plant

cc:

Mr. R. P. McDonald
Executive Vice President
Nuclear Operations
Alabama Power Company
P. O. Box 1295
Birmingham, Alabama 35201

Mr. B. L. Moore
Manager, Licensing
Alabama Power Company
P. O. Box 1295
Birmingham, Alabama 35201

Mr. Louis B. Long, General Manager
Southern Company Services, Inc.
Houston County Commission
P. O. Box 2625
Birmingham, Alabama 35202

Mr. D. N. Morey
General Manager - Farley Nuclear Plant
P. O. Box 470
Ashford, Alabama 36312

Mr. J. D. Woodward
Vice-President - Nuclear
Farley Project
Alabama Power Company
P. O. Box 1295
Birmingham, Alabama 35201

Resident Inspector
U.S. Nuclear Regulatory Commission
P. O. Box 24 - Route 2
Columbia, Alabama 36319

Regional Administrator, Region II
U.S. Nuclear Regulatory Commission
101 Marietta Street, Suite 2900
Atlanta, Georgia 30323

Chairman
Houston County Commission
Dothan, Alabama 36301

Claude Earl Fox, M.D.
State Health Officer
State Department of Public Health
State Office Building
Montgomergy, Alabama 36130

James H. Miller, III, Esq.
Balch and Bingham
P. O. Box 306
1710 Sixth Avenue North
Birmingham, Alabama 35201

AMENDMENT NO. 81 TO FACILITY OPERATING LICENSE NO. NPF-8 - FARLEY, UNIT 2

Docket File
NRC PDR
Local PDR
PDII-1 Reading
S. Varga (14E4)
G. Lainas
E. Adensam
P. Anderson
S. Hoffman
OGC
D. Hagan (MNBB 3302)
E. Jordan (MNBB 3302)
G. Hill (4) (P1-137)
Wanda Jones (P-130A)
J. Calvo (11D3)
C. Cheng
J. Tsao
ACRS (10)
GPA/PA
OC/LFMB

cc: Farley Service List

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

ALABAMA POWER COMPANY

DOCKET NO. 50-364

JOSEPH M. FARLEY NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 81
License No. NPF-8

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Alabama Power Company (the licensee), dated August 27, 1990, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations 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;

   D. The issuance of this license 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 amended by changes to the Technical Specifications, as indicated in the attachment to this license amendment; and paragraph 2.C.(2) of Facility Operating License No. NPF-8 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 81, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Original Signed By:

Elinor G. Adensam, Director
Project Directorate II-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: December 31, 1990

OGC concurrence subject to noted changes on letter, S.E., Tech Specs and notice of issuance

OFC	:LA:PD21:DRPR	:PM:PD21:DRPR	: OGC	:D:PD21:DRPR	:	:
NAME	: PAnderson	: SHoffman:sw	:	: EAdensam	:	:
DATE	: 12/10/90	: 12/12/90 12/28/90	: 12/1?/90	: 12/31/90	:	:

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ATTACHMENT TO LICENSE AMENDMENT NO. 81

FACILITY OPERATING LICENSE NO. NPF-8

DOCKET NO. 50-364

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised areas are indicated by marginal lines.

Remove Pages	Insert Pages
3/4 4-29	3/4 4-29
3/4 4-30	3/4 4-30
B 3/4 4-7	B 3/4 4-7
B 3/4 4-10	B 3/4 4-10
B 3/4 4-14	B 3/4 4-14

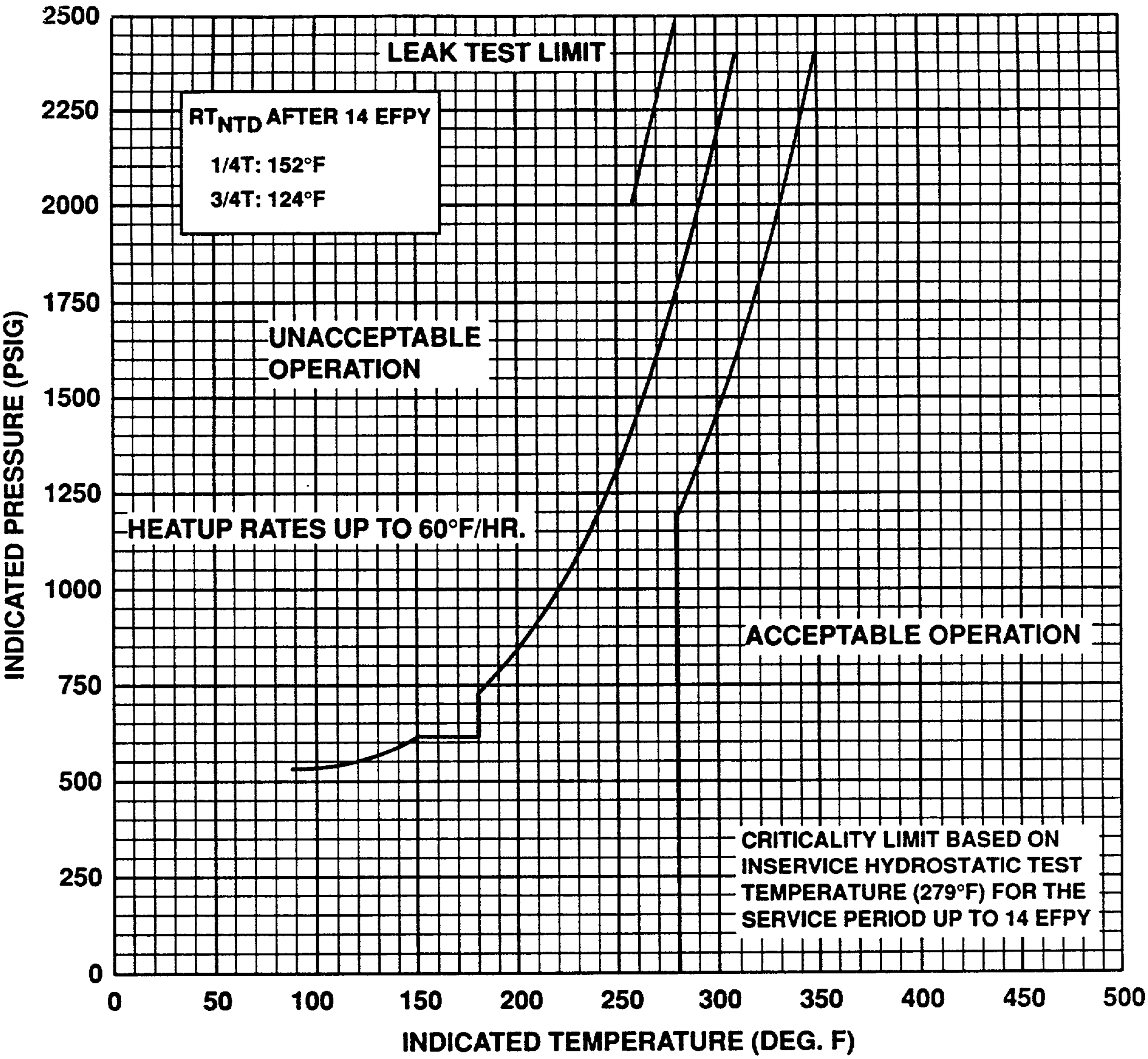

Figure 3.4-2 Farley Unit 2 Reactor Coolant System Heatup Limitations Applicable for the First 14 EFPY.

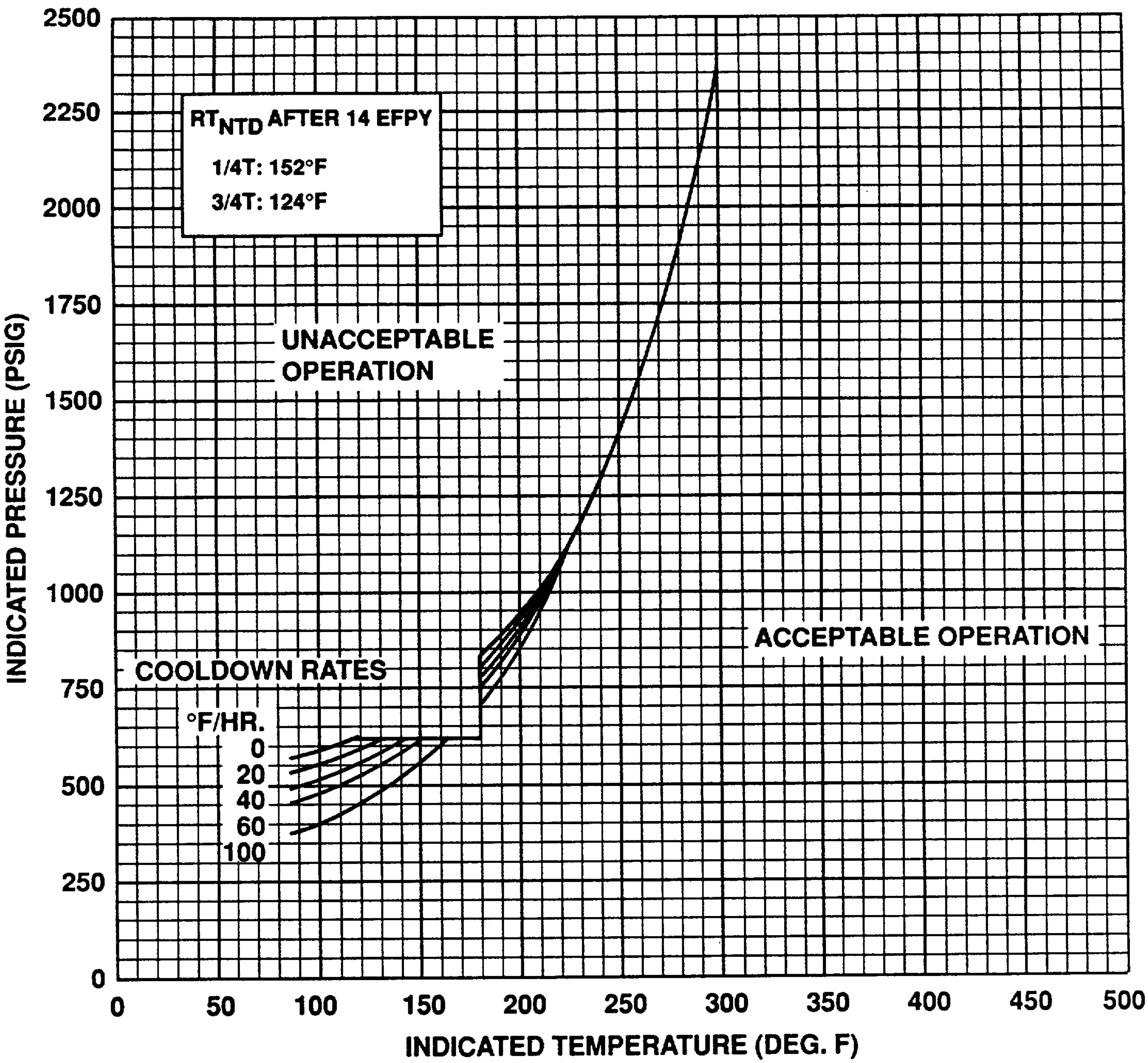

Figure 3.4-3 Farley Unit 2 Reactor Cooling System Cooldown Limitations Applicable for the First 14 EFPY.

4) The pressurizer heatup and cooldown rates shall not exceed 100°F/hr and 200°F/hr respectively. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 320°F.

5) System preservice hydrotests and in-service leak and hydrotests shall be performed at pressures in accordance with the requirements of ASME Boiler and Pressure Vessel Code, Section XI.

The fracture toughness properties of the ferritic materials in the reactor vessel are determined in accordance with ASTM E185-82, and in accordance with additional reactor vessel requirements. These properties are then evaluated in accordance with Appendix G of the 1976 Summer Addenda to Section III of the ASME Boiler and Pressure Vessel Code and the calculation methods described in WCAP-7924-A, "Basis for Heatup and Cooldown Limit Curves, April 1975."

Heatup and cooldown limit curves are calculated using the most limiting value of the nil-ductility reference temperature, RT_{ndt}, at the end of 14 effective full power years (EFPY) of service life. The 14 EFPY service life period is chosen such that the limiting RT_{ndt} at the 1/4T location in the core region is greater than the RT_{ndt} of the limiting unirradiated material. The selection of such a limiting RT_{ndt} assures that all components in the Reactor Coolant System will be operated conservatively in accordance with applicable Code requirements.

The reactor vessel materials have been tested to determine their initial RT_{ndt}; the results of these tests are shown in Table B 3/4.4-1. Reactor operation and resultant fast neutron (E greater than 1 MEV) irradiation can cause an increase in the RT_{ndt}. Therefore, an adjusted reference temperature, based upon the fluence and the nickel and copper content of the material in question, can be predicted using WCAP-12471 and the recommendations of Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials." The heatup and cooldown limit curves of Figures 3.4-2 and 3.4-3 include predicted adjustments for this shift in RT_{ndt} at the end of 14 EFPY.

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The use of the composite curve is necessary to set conservative heatup limitations because it is possible for conditions to exist such that over the course of the heatup ramp the controlling condition switches from the inside to the outside and the pressure limit must at all times be based on analysis of the most critical criterion.

Finally, the 10CFR Part 50, Appendix G Rule which addresses the metal temperature of the closure head flange and vessel flange must be considered. This Rule states that the minimum metal temperature of the closure flange regions be at least 120°F higher than the limiting RT_{ndt} for these regions when the pressure exceeds 20 percent of the preservice hydrostatic test pressure (621 psig for Farley Unit 2). In addition, the new 10CFR Part 50 Rule states that a plant specific fracture evaluation may be performed to justify less limiting requirements. Based upon such a fracture analysis for Farley Unit 2, the 14 EFPY heatup and cooldown curves are impacted by the new 10CFR Part 50 Rule as shown on Figures 3.4-2 and 3.4-3.

Although the pressurizer operates in temperature ranges above those for which there is reason for concern of non-ductible failure, operating limits are provided to assure compatibility of operation with the fatigue analysis performed in accordance with the ASME Code requirements.

The OPERABILITY of two RHR relief valves or an RCS vent opening of greater than or equal to 2.85 square inches ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10CFR Part 50 when one or more of the RCS cold legs are less than or equal to 310°F. Either RHR relief valve has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either (1) the start of an idle RCP with the secondary water temperature of the steam generator less than or equal to 50°F above the RCS cold leg temperatures or (2) the start of 3 charging pumps and their injection into a water solid RCS.

3/4.4.11 STRUCTURAL INTEGRITY

The inservice inspection and testing programs for ASME Code Class 1, 2 and 3 components ensure that the structural integrity and operational readiness of these components will be maintained at an acceptable level throughout the life of the plant. These programs are in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10CFR Part 50.55a(g) except where specific written relief has been granted by the Commission pursuant to 10CFR Part 50.55a(g)(6)(i).

3/4.4.12 REACTOR VESSEL HEAD VENTS

The OPERABILITY of the Reactor Head Vent System ensures that adequate core cooling can be maintained in the event of the accumulation of non-condensable gases in the reactor vessel. This system is in accordance with 10CFR50.44(c)(3)(iii).

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 81 TO FACILITY OPERATING LICENSE NO. NPF-8

ALABAMA POWER COMPANY

JOSEPH M. FARLEY NUCLEAR PLANT, UNIT 2

DOCKET NO. 50-364

1.0 INTRODUCTION

By letter dated November 23, 1988, Alabama Power Company (the licensee) responded to Generic Letter 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Effect on Plant Operations." In its response, the licensee stated that for Joseph M. Farley Nuclear Plant (Farley), Unit 2, the pressure/temperature (P/T) limits contained in the Technical Specifications required revision. By letter dated August 27, 1990, the licensee requested a license amendment to revise the P/T limits. The reqested amendment revises the P/T limits from 8 to 14 effective full power years (EFPY). The proposed P/T limits were developed based on the data from actual surveillance capsules. The proposed revision provides up-to-date P/T limits for the operations of the reactor coolant system during heatup, cooldown, criticality, and hydrotest.

To evaluate the P/T limits, the staff uses the following NRC regulations and guidance: Appendices G and H of 10 CFR Part 50; the American Society of Testing Materials (ASTM) Standards and the American Society of Mechnical Engineers Boiler and Pressure Vessel Code (ASME Code), which are referenced in Appendices G and H; 10 CFR 50.36(c)(2); Regulatory Guide (RG) 1.99, Revision 2; Standard Review Plan (SRP) Section 5.3.2; and Generic Letter 88-11.

Each licensee authorized to operate a nuclear power reactor is required by 10 CFR 50.36 to provide Technical Specifications for the operation of the plant. In particular, 10 CFR 50.36(c)(2) requires that limiting conditions of operation be included in the Technical Specifications. Appendices G and H to 10 CFR Part 50 describe specific requirements for fracture toughness and reactor vessel material surveillance that must be considered in setting P/T limits. An acceptable method for constructing the P/T limits is described in SRP Section 5.3.2.

Appendix G to 10 CFR Part 50 specifies fracture toughness and testing requirements for reactor vessel materials in accordance with the ASME Code and, in particular, that the beltline materials in the surveillance capsules be tested in accordance with the Appendix H of 10 CFR Part 50. Appendix H, in turn, refers to ASTM Standards for surveillance testing requirements. These surveillance tests define the extent of vessel embrittlement at the time of capsule withdrawal in terms of the increase in reference temperature. Appendix G also requires the licensee to predict the effects of neutron irradiation on vessel embrittlement by calculating the adjusted reference temperature (ART) and Charpy upper shelf energy (USE). Generic Letter 88-11 requested that licensees and permittees use the methods in RG 1.99, Revision 2, to predict the effect of neutron irradiation on reactor vessel materials. This guide defines the ART as the sum of unirradiated reference temperature, the increase in reference temperature resulting from neutron irradiation, and a margin to account for uncertainties in the prediction method.

Appendix H to 10 CFR Part 50 requires the licensee to establish a surveillance program to periodically withdraw surveillance capsules from the reactor vessel. Appendix H refers to the ASTM Standards which, in turn, require that the capsules be installed in the vessel before startup and that they contain test specimens made from plate, weld, and heat-affected-zone (HAZ) materials of the reactor beltline.

2.0 EVALUATION

The staff evaluated the effect of neutron irradiation embrittlement on each beltline material in the Farley, Unit 2, reactor vessel. The amount of irradiation embrittlement was calculated in accordance with RG 1.99, Revision 2. The staff has determined that the material with the highest ART at 1/4T (T = reactor vessel beltline thickness) at 14 EFPY was intermediate shell plate B7212-1 with 0.20% copper (Cu), 0.60% nickel (Ni), and an initial RT_{ndt} of -10°F. The material with the highest ART at 3/4T was intermediate shell plate B7203-1 with 0.14% copper (Cu), 0.60% nickel (Ni), and an initial RT_{ndt} of 15°F.

The licensee has removed three surveillance capsules from Farley, Unit 2. The results from capsules U, W, and X were published in Westinghouse Reports WCAP-10425, WCAP-11438, and WCAP-12471, respectively. All surveillance capsules contained Charpy impact specimens and tensile specimens made from base metal, weld metal, and HAZ metal.

For the limiting beltline materials, plates B7212-1 and B7203-1, the staff calculated the ART to be 149.5°F at 1/4T and 121.2°F for 3/4T at 14 EFPY. The staff used a neutron fluence of 9.43E18 n/cm^2 at 1/4T and 3.66E18 n/cm^2 at 3/4T. The ART for plate B7212-1 was determined by the least squares extrapolation method using the Farley, Unit 2, surveillance data. The least squares method is described in Section 2.1 of RG 1.99, Revision 2. The ART for plate B7203-1 was determined using Section 1 of RG 1.99, Revision 2, because plate B7203-1 was not in the surveillance capsules.

The licensee calculated an ART of 152°F and 124°F at 1/4T and 3/4T, respectively, for the limiting material. The licensee's ARTs are more conservative than the staff's ARTs; therefore, they are acceptable. Substituting the ART of 149.5°F into equations in SRP 5.3.2, the staff verified that the proposed P/T limits for heatup, cooldown, and hydrotest meet the beltline material requirements of Appendix G to 10 CFR Part 50.

In addition to beltline materials, Appendix G to 10 CFR Part 50 also imposes P/T limits based on the reference temperature for the reactor vessel closure flange materials. 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 reference temperature of 60°F, the staff has determined that the proposed P/T limits satisfy Section IV.2 of Appendix G.

Section IV.A.1 of Appendix G requires that the predicted Charpy USE at end-of-life (EOL) be above 50 ft-lb. The material with the lowest predicted Charpy EOL USE was intermediate shell plate B7212-1 with an unirradiated USE of 99 ft-lb. Using Figure 2 of RG 1.99, Revision 2, the staff calculated that the EOL USE would be 60.4 ft-lb. This is greater than 50 ft-lb and, therefore, is acceptable.

3.0 SUMMARY

The staff concludes that the proposed P/T limits for the reactor coolant system for heatup, cooldown, leak test, and criticality are valid through 14 EFPY as the limits conform to the requirements of Appendices G and H to 10 CFR Part 50. The licensee's submittal also satisfies Generic Letter 88-11 as the licensee used the method in RG 1.99, Revision 2, to calculate the ART. Hence, the proposed P/T limits may be incorporated into the Farley, Unit 2, Technical Specifications.

4.0 ENVIRONMENTAL CONSIDERATION

This amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The 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 off site, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration, and there has been no public comment on such finding. Accordingly, this 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 this amendment.

5.0 CONCLUSION

The Commission made a proposed determination that this amendment involves no significant hazards consideration which was published in the Federal Register (55 FR 40456) on October 3, 1990, and consulted with the State of Alabama. No public comments or requests for hearing were received, and the State of Alabama did not have any comments.

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

6.0 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. November 23, 1988, Letter from W. G. Hairston, III (APCo) to USNRC Document Control Desk, Subject: Generic Letter 88-11.

4. August 27, 1990, Letter from W. G. Hairston, III (APCo) to USNRC Document Control Desk, Subject: Farley 2 RCS Heatup and Cooldown Limit Curves.

5. M. K. Kunka, et al., "Analysis of Capsule U from the Alabama Power Company Joseph M. Farley Unit 2 Reactor Vessel Radiation Surveillance Program," WCAP-10425, Westinghouse Electric Company, October 1983.

6. R. P. Shogan, et al., "Analysis of Capsule W from the Alabama Power Company Joseph M. Farley Unit 2 Reactor Vessel Radiation Surveillance Program," WCAP-11438, Westinghouse Electric Company, April 1987.

7. E. Terek, et al., "Analysis of Capsule X from the Alabama Power Company Joseph M. Farley Unit 2 Reactor Vessel Radiation Surveillance Program," WCAP-12471, Westinghouse Electric Company, December 1989.

Dated: December 31, 1990

Principal Contributor: J. Tsao