June 23, 1987

Docket No. 50-348

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Mr. R. P. McDonald
Senior Vice President
Alabama Power Company
Post Office Box 2641
Birmingham, Alabama 35291-0400

Dear Mr. McDonald:

SUBJECT: TECHNICAL SPECIFICATION AMENDMENT REGARDING HEATUP AND COOLDOWN LIMITATIONS, JOSEPH M. FARLEY NUCLEAR PLANT, UNIT 1 (TAC NO. 60075)

The Commission has issued the enclosed Amendment No. 71 to Facility Operating License NPF-2 for the Joseph M. Farley Nuclear Plant, Unit 1 The amendment consists of changes to the Technical Specifications, in response to your application transmitted by letter dated October 25, 1985, as supplemented September 29, 1986.

The amendment revises Technical Specification Figures 3.4-2 and 3.4-3, heatup and cooldown limitations based on results of analysis of Capsule "U" Reactor Vessel Material Radiation Surveillance Program. Analysis results are included in the Westinghouse Report, WCAP-10934, Revision 2.

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

Sincerely,

Edward A. Reeves, Project Manager
Project Directorate II-1
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 71 to NPF-2
3. Safety Evaluation

cc: w/enclosures:
See next page

*See Previous Concurrence

LA:PDII-1	PM:PDII-1	PD:PDII-1	OGC
PAnderson	EReeves:bg	EAdensam	BVogler*
6/23/87	6/23/87	6/22/87	6/ /87

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

June 23, 1987

Docket No. 50-348

Mr. R. P. McDonald
Senior Vice President
Alabama Power Company
Post Office Box 2641
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SUBJECT: TECHNICAL SPECIFICATION AMENDMENT REGARDING HEATUP AND COOLDOWN LIMITATIONS, JOSEPH M. FARLEY NUCLEAR PLANT, UNIT 1 (TAC NO. 60075)

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Edward A. Reeves

Edward A. Reeves, Project Manager
Project Directorate II-1
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 71 to NPF-2
3. Safety Evaluation

cc: w/enclosures:
See next page

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Alabama Power Company

Joseph M. Farley Nuclear Plant

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

ALABAMA POWER COMPANY

DOCKET NO. 50-348

JOSEPH M. FARLEY NUCLEAR PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 71
License No. NPF-2

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

   A. The application for amendment by Alabama Power Company (the licensee) dated October 25, 1985, as supplemented September 29, 1986, 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, as amended, the provisions of the Act, and the 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-2 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 71 , 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 and shall be implemented within 30 days of receipt of this amendment.

FOR THE NUCLEAR REGULATORY COMMISSION

Elinor G. Adensam

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: June 23, 1987

ATTACHMENT TO LICENSE AMENDMENT NO. 71

FACILITY OPERATING LICENSE NO. NPF-2

DOCKET NO. 50-348

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages as indicated. The revised pages are identified by amendment number and contain vertical lines indicating the area of change.

Remove Pages	Insert Pages
3/4 4-29	3/4 4-29
3/4 4-30	3/4 4-30
B3/4 4-6	B3/4 4-6
B3/4 4-7	B3/4 4-7
B3/4 4-9	B3/4 4-9
B3/4 4-10	B3/4 4-10
--	B3/4 4-10A
B3/4 4-14	B3/4 4-14

MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL	:	LOWER SHELL (PLATE NO. B6919-2)
COPPER CONTENT	:	0.14 WT%
NICKEL CONTENT	:	0.56 WT%
INITIAL RT_{NDT}	:	5^{o} F
RT_{NDT} AFTER 16 EFPY	:	1/4T, 146.4^{o} F
	:	3/4T, 121.5^{o} F

CURVES APPLICABLE FOR HEATUP RATES UP TO 60^{o} F/HR FOR THE SERVICE PERIOD UP TO 16 EFPY

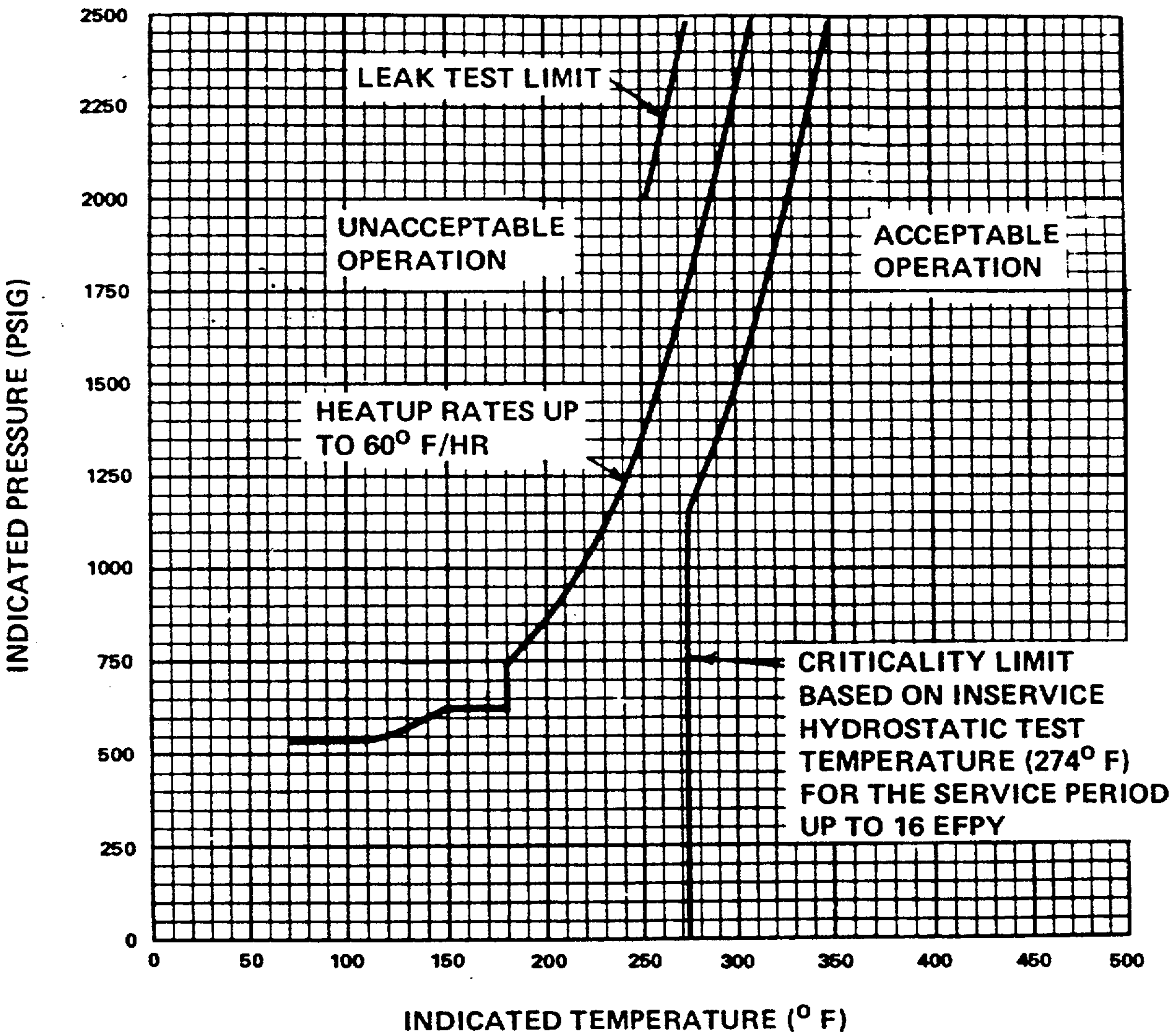

FIGURE 3.4-2 FARLEY UNIT 1 REACTOR COOLANT SYSTEM HEATUP LIMITATIONS APPLICABLE FOR THE FIRST 16 EFPY

MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL	:	LOWER SHELL (PLATE NO. B6919-2)
COPPER CONTENT	:	0.14 WT%
NICKEL CONTENT	:	0.56 WT%
INITIAL RT_{NDT}	:	5° F
RT_{NDT} AFTER 16 EFPY	:	1/4T, 146.4° F
	:	3/4T, 121.5° F

CURVES APPLICABLE FOR COOLDOWN UP TO 100° F/HR FOR THE SERVICE PERIOD UP TO 16 EFPY

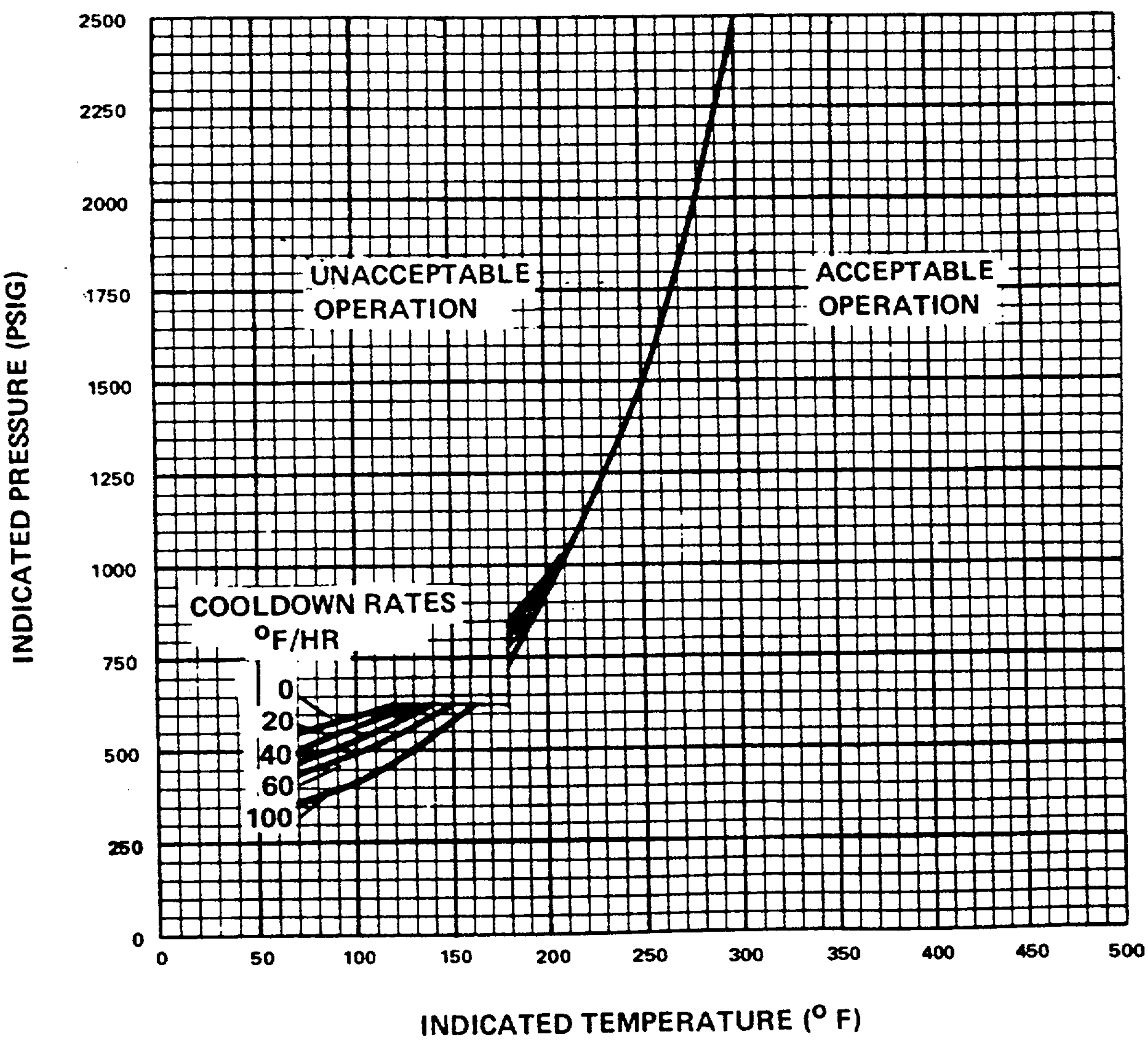

FIGURE 3.4-3 FARLEY UNIT 1 REACTOR COOLANT SYSTEM COOLDOWN LIMITATIONS APPLICABLE FOR THE FIRST 16 EFPY

Reducing T_{avg} to less than 500°F prevents the release of activity should a steam generator tube rupture since the saturation pressure of the primary coolant is below the lift pressure of the atmospheric steam relief valves. The surveillance requirements provide adequate assurance that excessive specific activity levels in the primary coolant will be detected in sufficient time to take corrective action. Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

3/4.4.10 PRESSURE/TEMPERATURE LIMITS

The temperature and pressure changes during heatup and cooldown are limited to be consistent with the requirements given in the ASME Boiler and Pressure Vessel Code, Section III, Appendix G as required per 10 CFR Part 50 Appendix G.

1) The reactor coolant temperature and pressure and system heatup and cooldown rates (with the exception of the pressurizer) shall be limited in accordance with Figures 3.4-2 and 3.4-3.

    a) Allowable combinations of pressure and temperature for specific temperature change rates are below and to the right of the limit lines shown. Limit lines for cooldown rates between those presented may be obtained by interpolation.

    b) Figures 3.4-2 and 3.4-3 define limits to assure prevention of nonductile failure only. For normal operation, other inherent plant characteristics, e.g., pump heat addition and pressurizer heater capacity, may limit the heatup and cooldown rates that can be achieved over certain pressure-temperature ranges.

2) These limit lines shall be calculated periodically using methods provided below.

3) The secondary side of the steam generator must not be pressurized above 200 psig if the temperature of the steam generator is below 70°F.

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 16 effective full power years (EFPY) of service life. The 16 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 copper and nickel content of the material in question, can be predicted using Figure B 3/4.4-1, Figure B 3/4.4-2 and the recommendations of Regulatory Guide 1.99, Revision 2, "Effects of Residual Elements on Predicted Radiation Damage to 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 16 EFPY.

TABLE B 3/4.4-1

FARLEY UNIT 1 REACTOR VESSEL TOUGHNESS PROPERTIES

Component	Code No.	Material Type	Cu (%)	P (%)	Ni (%)	T_{ndt} (°F)	RT_{ndt} (°F)	Upper Shell Energy MWD[c]	Upper Shell Energy NMWD[d]
Closure head dome	B6901	A533,B,Cl.1	0.16	0.009	0.50	-30	-20[a]	140	-
Closure head segment	B6902-1	A533,B,Cl.1	0.17	0.007	0.52	-20	-20[a]	138	-
Closure head flange	B6915-1	A508, Cl.2	0.10	0.012	0.64	60[a]	60[a]	75[a]	-
Vessel flange	B6913-1	A508, Cl.2	0.17	0.011	0.69	60[a]	60[a]	106[a]	-
Inlet nozzle	B6917-1	A508, Cl.2	-	0.010	0.83	60[a]	60[a]	-	110
Inlet nozzle	B6917-2	A508, Cl.2	-	0.008	0.80	60[a]	60[a]	-	80
Inlet nozzle	B6917-3	A508, Cl.2	-	0.008	0.87	60[a]	60[a]	-	98
Outlet nozzle	B6916-1	A508, Cl.2	-	0.007	0.77	60[a]	60[a]	-	96.5
Outlet nozzle	B6916-2	A508, Cl.2	-	0.011	0.78	60[a]	60[a]	-	97.5
Outlet nozzle	B6916-3	A508, Cl.2	-	0.009	0.78	60[a]	60[a]	-	100
Nozzle shell	B6914-1	A508, Cl.2	-	0.010	0.68	30	30[a]	148	-
Inter. shell	B6903-2	A533,B,Cl.1	0.13	0.011	0.60	0	0	151.5	97
Inter. shell	B6903-3	A533,B,Cl.1	0.12	0.014	0.56	10	10	134.5	100
Lower shell	B6919-1	A533,B,Cl.1	0.14	0.015	0.55	-20	15	133	90.5
Lower shell	B6919-2	A533,B,Cl.1	0.14	0.015	0.56	-10	5	134	97
Bottom head ring	B6912-1	A508, Cl.2	-	0.010	0.72	10	10[a]	163.5	-
Bottom head segment	B6906-1	A533,B,Cl.1	0.15	0.011	0.52	-30	-30[a]	147	-
Bottom head dome	B6907-1	A533,B,Cl.1	0.17	0.014	0.60	-30	-30[a]	143.5	-
Inter. shell long. weld seam	M1.33	Sub Arc Weld	0.25	0.017	0.21	0[a]	0[a]	-	-
Inter. to lower shell weld seams	G1.18	Sub Arc Weld	0.22	0.011	<0.20[b]	0[a]	0[a]	-	-
Lower shell long. weld seams	G1.08	Sub Arc Weld	0.17	0.022	<0.20[b]	0[a]	0[a]	-	-

[a] Estimate per NUREG-0800 "USNRC Standard Review Plan" Branch Technical Position MTEB 5-2.
[b] Estimated (low nickel weld wire used in fabricating vessel weld seams).
[c] Major working direction.
[d] Normal to major working direction.

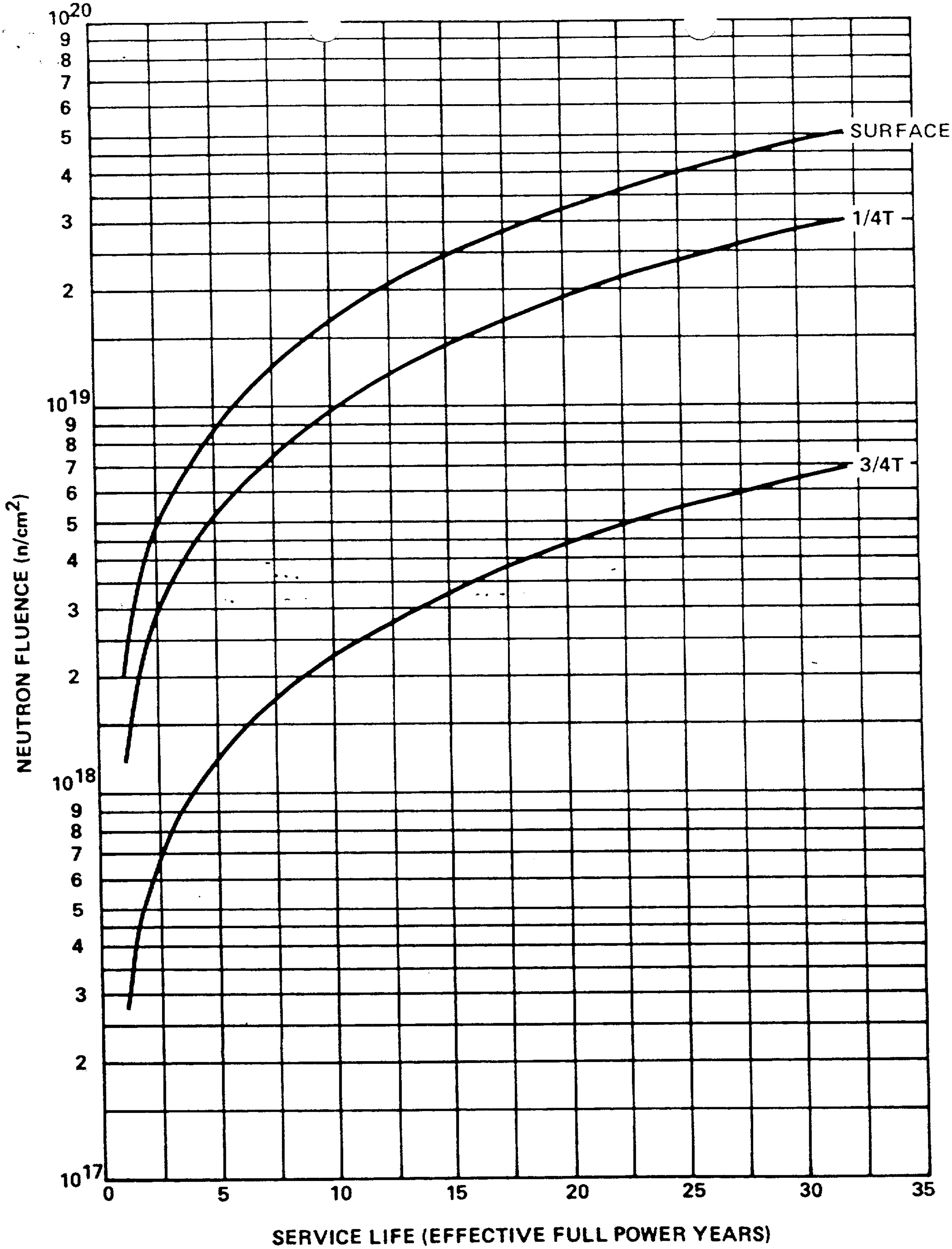

FIGURE B 3/4.4-1 FAST NEUTRON FLUENCE (E>1 MeV) AS A FUNCTION OF FULL POWER SERVICE LIFE (EFPY)

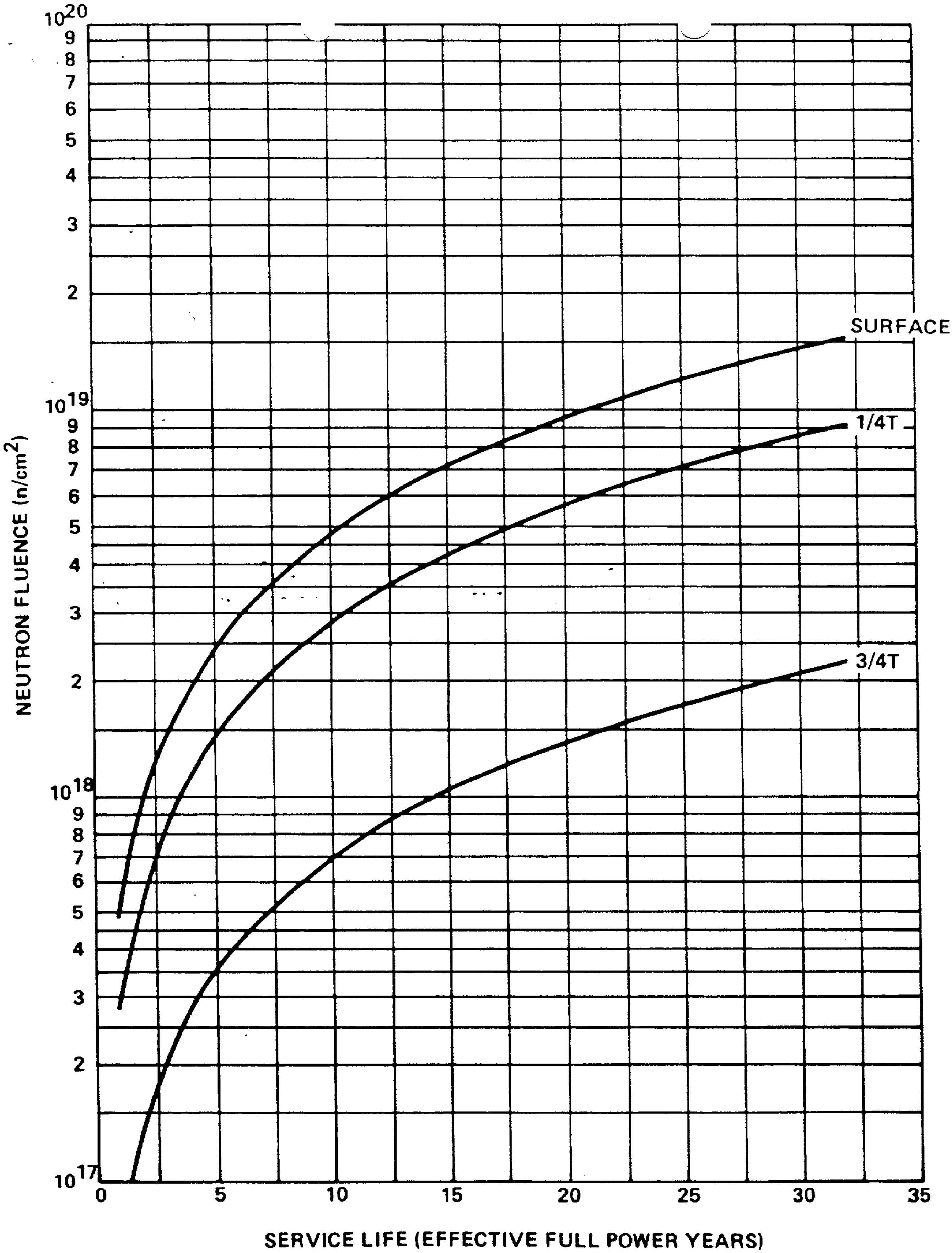

FIGURE B 3/4.4-2 FAST NEUTRON FLUENCE (E>1 MeV) AT 45° AS A FUNCTION OF FULL POWER SERVICE (EFPY)

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 10 CFR 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 1). In addition, the new 10 CFR Part 50 Rule states that a plant specific fracture evaluation may be performed to justify less limiting requirements. As a result, such a fracture analysis was performed for Farley Unit 2. These Farley Unit 2 fracture analysis results are applicable to Farley Unit 1 since the pertinent parameters are identical for both plants. Based upon this fracture analysis, the 16 EFPY heatup and cooldown curves are impacted by the new 10 CFR 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-ductile 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 10 CFR 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

RELATED TO AMENDMENT NO. 71 TO FACILITY OPERATING LICENSE NO. NPF-2

ALABAMA POWER COMPANY

JOSEPH M. FARLEY NUCLEAR PLANT, UNIT NO. 1

DOCKET NO. 50-348

INTRODUCTION

In a letter from R. P. McDonald to S. A. Varga dated October 25, 1985, the Alabama Power Company (the licensee) requested changes to the Joseph M. Farley Nuclear Plant, Unit 1 (Farley-1) Heatup/Cooldown Curves and supporting bases. The staff reviewed the licensee's submittal and determined that the effect of neutron irradiation on all beltline materials had not been assessed by the licensee. The staff's concerns are documented in a letter from E. A. Reeves to R. P. McDonald dated June 16, 1986. In response to this letter, the licensee provided a revised set of heatup and cooldown curves, which were to be applicable for 16 effective full power years (EFPY). The revised curves and their bases were submitted in a letter from R. P. McDonald to L. S. Rubenstein dated September 29, 1986. The curves and bases are to be contained in Figures 3.4-2 and 3.4-3 and Bases Section 3/4.4.10 of the Farley-1 Technical Specifications.

DISCUSSION

Heatup/Cooldown curves must be calculated in accordance with the requirements of Appendix G of 10 CFR 50, which became effective on July 26, 1983. Appendix G of 10 CFR 50 requires that the reactor vessel beltline and closure flange region materials meet the safety margins of Appendix G of the ASME Code Section III. To calculate pressure-temperature limits in accordance with these requirements, the effect of neutron irradiation, boltup, pressure and thermal stresses on the limiting reactor vessel beltline and closure flange region materials must be estimated. The effect of neutron irradiation on the Farley-1 beltline materials is documented in Westinghouse Report WCAP-10934, Rev. 2, dated June 1986, which is in Attachment 2 to the licensee's submittal dated September 29, 1986. The effect of boltup, pressure and thermal stresses on the reactor vessel closure flange region are documented in letters from R. P. McDonald to S. A. Varga dated June 18, 1984, and October 25, 1985.

EVALUATION

The methods recommended by the NRC staff for calculating the effect of neutron irradiation damage are documented in Regulatory Guide 1.99, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials." The relationships documented in Regulatory Guide 1.99 were empirically derived from materials that were irradiated in commercial nuclear reactor surveillance capsules. The most pertinent empirical relationships are contained in Revision 2 to the guide. Revision 2 has been reviewed by the NRC staff and published for comment. The licensee has used the methods recommended in Regulatory Guide 1.99, Revision 2, to calculate the effect of neutron irradiation on the Farley-1 beltline materials.

Neutron irradiation damage is measured by an increase in a material's reference temperature. The value of the reference temperature that results from neutron irradiation damage is called the material's adjusted reference temperature, ART. The limiting ART was used to calculate pressure-temperature limits for the Farley-1 beltline materials. These limits were calculated in accordance with the requirements in Appendix G of the ASME Code Section III. The NRC staff has evaluated these limits using the calculation methods recommended in Standard Review Plan (SRP) 5.3.2, "Pressure-Temperature Limits."

The stresses in the closure flange region resulting from pressure, thermal effects, and boltup were calculated by the licensee using finite element analysis. The closure head and vessel flange geometry used in the finite element analysis was modelled for a typical 4-loop reactor vessel. The Farley-1 plant is a 3-loop reactor vessel. The geometry of the closure flange region in the Farley-1 reactor vessel is slightly different than that of the typical 4-loop reactor vessel. To account for these differences, the licensee used the computation method of Reference 1 to perform a stress analysis of the Farley-1 vessel based on the finite element analysis and an analytical comparison of critical dimensions of the two types of vessels. Their analysis indicates that the typical 4-loop reactor vessel and the Farley-1 reactor vessel have essentially equivalent pressure and boltup stresses at the critical closure flange region. Hence, the stresses from boltup and pressure used for the typical 4-loop plant were used in the fracture mechanics evaluation for Farley-1. The stresses at the critical closure flange region resulting from thermal conditions during heatup or cooldown of the Farley-1 vessel were determined by the computation method in Reference 1 to be significantly less than those calculated for the typical 4-loop plant.

Fracture mechanics evaluations at three discontinuity locations in the closure flange region were performed in accordance with the methodology in Appendix A of ASME Code Section XI. In this analysis the licensee used all the safety factors required by Appendix G of the ASME Code, except for the Code recommended flaw size, to determine the closure flange location that would be considered the critical location. The location with the highest stress intensity factor after applying safety margins was considered the critical closure flange location. The critical location was determined to be the outside surface at the discontinuity between the flange and upper shell of the reactor vessel.

The postulated flaw size recommended by Appendix G of the ASME Code was used for evaluating the beltline region, but was not used in evaluating the closure flange region. The postulated flaw size recommended by Appendix G has a depth of ¼ the section thickness (¼ T) and a length of 1½ times the section thickness. The section thickness at the critical flange location for Farley-1 is 9.125 inches. Appendix G of the ASME Code indicates that smaller defect sizes may be used on an individual case basis, if a smaller size of maximum postulated defect can be assured. The postulated defect used in the licensee's analysis was a 0.625 inch deep by 3.75 inches long surface flaw. The licensee's justification for using a smaller flaw size in evaluating the closure flange region than that used in evaluating the beltline region is that the volumetric examination of the closure flange location will assure detection of the critical size flaw.

Volumetric examination of the reactor vessel flange-to-upper shell weld and specified adjacent base material is accomplished by two ultrasonic scan routines. Coverage from the flange side of the weld involves use of angled longitudinal waves from the flange seal surface. Beam angles are selected based on their ability to provide coverage of the weld and specified adjacent base material to the extent practical and provide near normal incidence to the plane of the weld. Refracted beam angles in the range 0° to 16° are typically used for these examinations. Examinations from the shell side of the weld involve 0°, 45°, and 60° refracted angle beam coverage from the vessel inside diameter surface. Angle beam scanning is performed in two directions, parallel to the weld and perpendicular to the weld from the shell side. Access for the shell side examination is limited to the Ten Year Inservice Inspection outage when the core barrel is removed from the reactor vessel.

The licensee indicates that the fact that postulated flaws are surface related is significant from a detection probability point of view. Incipient cracks starting at right angles to a given surface (OD or ID) provide favorable conditions for detection via ASME Code specified 45° shear wave ultrasonic examinations from the opposite surface. Circumferential flaws are oriented favorably for detection during axial scanning. Axial flaws are oriented favorably for detection during circumferential scans. Circumferentially oriented flaws in the vessel flange weld region also provide favorable conditions for detection during ultrasonic examinations from the flange seal surface.

Additional justifications for permitting smaller postulated flaws in the closure flange region than the size postulated for the beltline region are described in the staff's regulatory analysis of public comments which is in Enclosure 4 to the staff's report SECY-83-80, "10 CFR Part 50-General Revision of Appendices G and H, Fracture Toughness and Reactor Vessel Material Surveillance Requirements," February 25, 1983.

As previously reported, the licensee's fracture mechanics evaluation was performed in accordance with the methodology in Appendix A of ASME Code Section XI. In this method, the stress intensity factors at the crack tip are calculated by linearizing the stress around the postulated flaw. The linearized stress is divided into membrane and bending stresses. The Appendix A method of linearizing stress resulted in negative membrane stresses when considering boltup, pressure and thermal condition during heat-up. The licensee considered the negative membrane stresses equal to zero when determining the stress intensity factor resulting from thermal conditions during heat-up. The NRC staff considers this acceptable, since it conservatively represents the stress condition resulting from heat-up.

The licensee used the negative value of membrane stress when determining the stress intensity factor resulting from boltup and pressure conditions. The negative membrane stress will result in a reduction in the calculated stress intensity factor, since the stress intensity factor is the sum of a positive bending stress and a negative membrane stress. A negative value of membrane stress does not represent the real membrane stress resulting from boltup and pressure conditions. However, the non-conservatism resulting from a negative valued membrane stress will be offset by a high value for the bending stress that results from the linearizing method. Several methods of calculating stress intensity factors for a stress distribution similar to that in the closure flange region were evaluated in Reference 2. The Appendix A method of linearizing the stress around the postulated flaw produced conservative stress intensity factors when compared to those calculated using a finite element analysis method, an ASME Code Section III Appendix G method recommended for nonlinear stress distributions, and a poly-nomial method (Reference 3). This comparison indicates that the Appendix A method of linearizing stress will result in an acceptable fracture mechanics analysis for evaluating flaws in the closure flange region of the reactor vessel.

Using the stress intensity factors calculated in accordance with Appendix A of the ASME Code Section XI and the safety margins of Appendix G of the ASME Code with a postulated flaw of 0.625 inch deep by 3.75 inches long, the licensee proposed pressure-temperature limits for the closure flange region materials. The pressure-temperature limits for the closure flange region materials were combined with the limits for the beltline region to develop the Farley-1 Heatup/Cooldown Curves.

SAFETY SUMMARY

Based on the method documented in Regulatory Guide 1.99, Revision 2, for evaluating the effect of neutron irradiation on reactor vessel beltline materials, and the method of calculating pressure-temperature limits in SRP 3.6.2, the licensee's proposed Heatup/Cooldown Curves for 16 EFPY meet the safety margins of Appendix G of the ASME Code.

Based on the licensee's finite element analysis, the fracture mechanics analysis performed in accordance with Appendix A of Section XI of the ASME Code, and the licensee's and NRC staff's justification for considering smaller postulated flaw sizes based on SECY-83-80, the licensee's proposed pressure-temperature limits for the closure flange region meet the safety margins of Appendix G of the ASME Code.

Based on the above two conclusions, the proposed Heatup/Cooldown Curves that are contained in the licensee's letter dated September 29, 1986, meet the safety margins of Appendix G, 10 CFR 50 for 16 EFPY and are acceptable Farley-1 Technical Specification.

ENVIRONMENTAL CONSIDERATION

This amendment involves a change in the 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 offsite, 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.

CONCLUSION

We have concluded, based on the consideration 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, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

References

1. "Tentative Structural Design Basis for Reactor Pressure Vessels and Directly Associated Components (Pressurized, Water Cooled Systems), U.S. Department of Commerce, December 1, 1958 and February 27, 1959, pp. 58, 59, 60, Addendum No. 1.

2. Bloom, J.M., and Van Der Sluys, W.A., "Determination of Stress Intensity Factors for Gradient Stress Fields," Journal of Pressure Vessel Technology, Vol. 99, August 1977.

3. Buchalet, C.B., and Bamford, W.H., "Stress Intensity Factor Solutions for Continuous Surface Flaws in Reactor Pressure Vessels," Mechanics of Crack Growth, ASTM STP 590, American Society for Testing and Materials, 1976.

Principal Contributor: B. Elliot

Dated: June 23, 1987

June 23, 1987

Docket No. 50-348

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Mr. R. P. McDonald
Senior Vice President
Alabama Power Company
Post Office Box 2641
Birmingham, Alabama 35291-0400

Dear Mr. McDonald:

SUBJECT: TECHNICAL SPECIFICATION AMENDMENT REGARDING HEATUP AND COOLDOWN LIMITATIONS, JOSEPH M. FARLEY NUCLEAR PLANT, UNIT 1 (TAC NO. 60075)

The Commission has issued the enclosed Amendment No. to Facility Operating License NPF-2 for the Joseph M. Farley Nuclear Plant, Unit 1 The amendment consists of changes to the Technical Specifications, in response to your application transmitted by letter dated October 25, 1985, as supplemented September 29, 1986.

The amendment revises Technical Specification Figures 3.4-2 and 3.4-3, heatup and cooldown limitations based on results of analysis of Capsule "U" Reactor Vessel Material Radiation Surveillance Program. Analysis results are included in the Westinghouse Report, WCAP-10934, Revision 2.

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

Sincerely,

Edward A. Reeves, Project Manager
Project Directorate II-1
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 71 to NPF-2
3. Safety Evaluation

cc: w/enclosures:
See next page

*See Previous Concurrence

LA:PDII-1 for	PM:PDII-1	PD:PDII-1	OGC
PAnderson	EReeves:bg	EAdensam	BVogler*
6/23/87	6/23/87	6/22/87	6/ /87

Mr. R. P. McDonald
Alabama Power Company

Joseph M. Farley Nuclear Plant

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

ALABAMA POWER COMPANY

DOCKET NO. 50-348

JOSEPH M. FARLEY NUCLEAR PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 71
License No. NPF-2

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

   A. The application for amendment by Alabama Power Company (the licensee) dated October 25, 1985, as supplemented September 29, 1986, 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, as amended, the provisions of the Act, and the 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-2 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 71 , 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 and shall be implemented within 30 days of receipt of this amendment.

FOR THE NUCLEAR REGULATORY COMMISSION

Elinor G. Adensam

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: June 23, 1987

ATTACHMENT TO LICENSE AMENDMENT NO. 71

FACILITY OPERATING LICENSE NO. NPF-2

DOCKET NO. 50-348

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages as indicated. The revised pages are identified by amendment number and contain vertical lines indicating the area of change.

Remove Pages	Insert Pages
3/4 4-29	3/4 4-29
3/4 4-30	3/4 4-30
B3/4 4-6	B3/4 4-6
B3/4 4-7	B3/4 4-7
B3/4 4-9	B3/4 4-9
B3/4 4-10	B3/4 4-10
--	B3/4 4-10A
B3/4 4-14	B3/4 4-14

MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL	:	LOWER SHELL (PLATE NO. B6919-2)
COPPER CONTENT	:	0.14 WT%
NICKEL CONTENT	:	0.56 WT%
INITIAL RT_{NDT}	:	5° F
RT_{NDT} AFTER 16 EFPY	:	1/4T, 146.4° F
	:	3/4T, 121.5° F

CURVES APPLICABLE FOR HEATUP RATES UP TO 60° F/HR FOR THE SERVICE PERIOD UP TO 16 EFPY

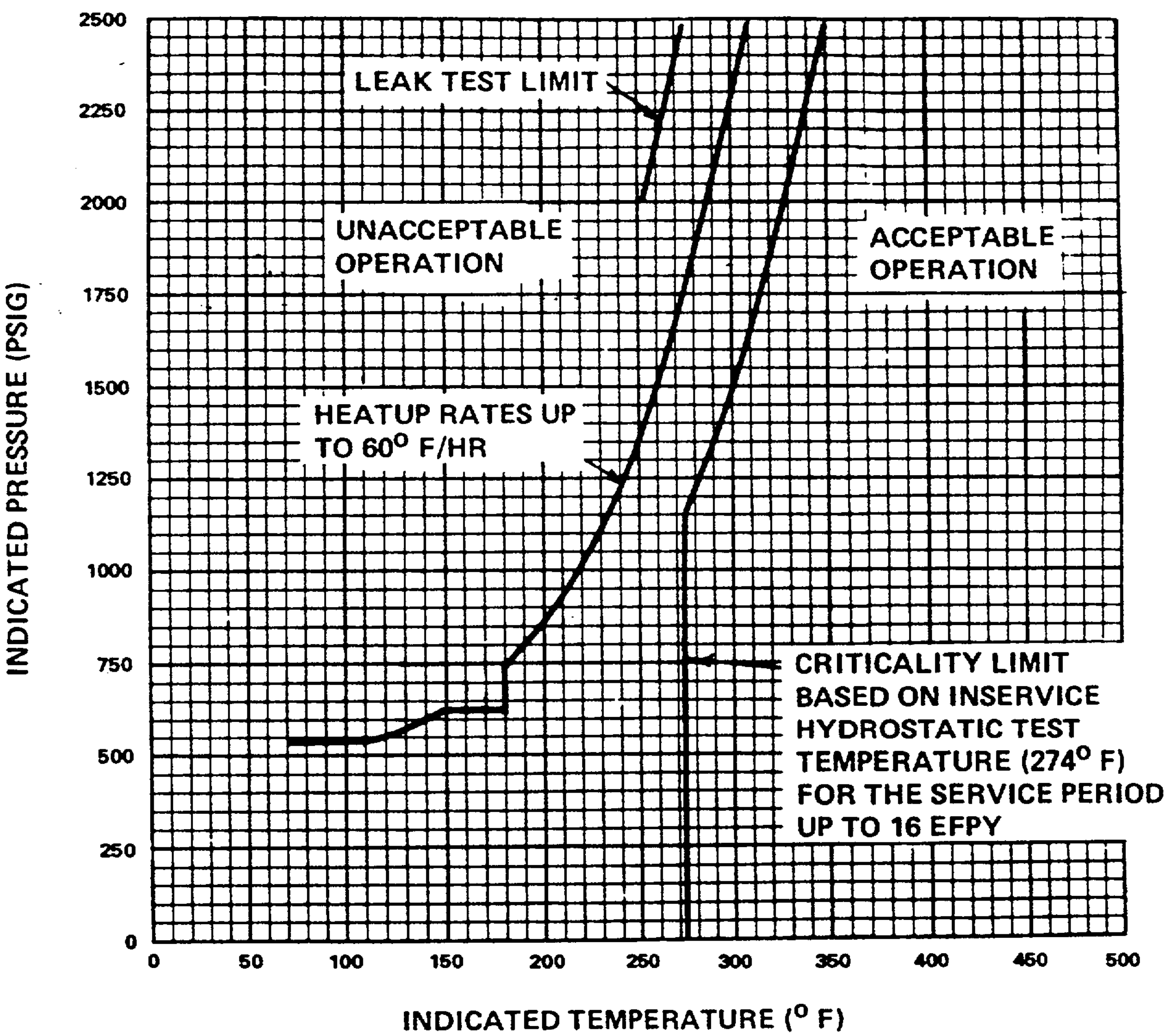

FIGURE 3.4-2 FARLEY UNIT 1 REACTOR COOLANT SYSTEM HEATUP LIMITATIONS APPLICABLE FOR THE FIRST 16 EFPY

MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL	:	LOWER SHELL (PLATE NO. B6919-2)
COPPER CONTENT	:	0.14 WT%
NICKEL CONTENT	:	0.56 WT%
INITIAL RT_{NDT}	:	5° F
RT_{NDT} AFTER 16 EFPY	:	1/4T, 146.4° F
	:	3/4T, 121.5° F

CURVES APPLICABLE FOR COOLDOWN UP TO 100° F/HR FOR THE SERVICE PERIOD UP TO 16 EFPY

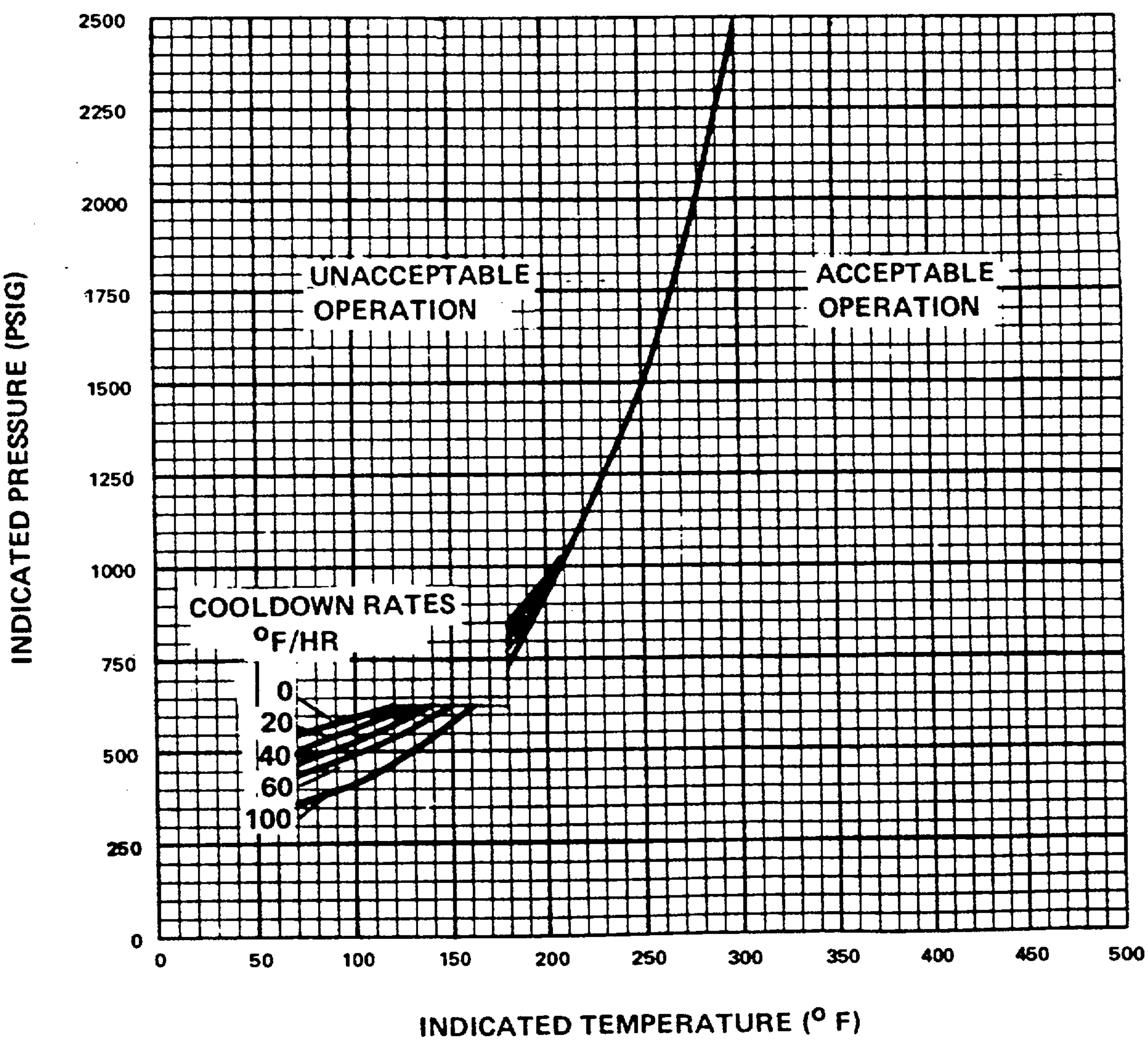

FIGURE 3.4-3 FARLEY UNIT 1 REACTOR COOLANT SYSTEM COOLDOWN LIMITATIONS APPLICABLE FOR THE FIRST 16 EFPY

Reducing T_{avg} to less than 500°F prevents the release of activity should a steam generator tube rupture since the saturation pressure of the primary coolant is below the lift pressure of the atmospheric steam relief valves. The surveillance requirements provide adequate assurance that excessive specific activity levels in the primary coolant will be detected in sufficient time to take corrective action. Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

3/4.4.10 PRESSURE/TEMPERATURE LIMITS

The temperature and pressure changes during heatup and cooldown are limited to be consistent with the requirements given in the ASME Boiler and Pressure Vessel Code, Section III, Appendix G as required per 10 CFR Part 50 Appendix G.

1) The reactor coolant temperature and pressure and system heatup and cooldown rates (with the exception of the pressurizer) shall be limited in accordance with Figures 3.4-2 and 3.4-3.

   a) Allowable combinations of pressure and temperature for specific temperature change rates are below and to the right of the limit lines shown. Limit lines for cooldown rates between those presented may be obtained by interpolation.

   b) Figures 3.4-2 and 3.4-3 define limits to assure prevention of nonductile failure only. For normal operation, other inherent plant characteristics, e.g., pump heat addition and pressurizer heater capacity, may limit the heatup and cooldown rates that can be achieved over certain pressure-temperature ranges.

2) These limit lines shall be calculated periodically using methods provided below.

3) The secondary side of the steam generator must not be pressurized above 200 psig if the temperature of the steam generator is below 70°F.

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 16 effective full power years (EFPY) of service life. The 16 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 copper and nickel content of the material in question, can be predicted using Figure B 3/4.4-1, Figure B 3/4.4-2 and the recommendations of Regulatory Guide 1.99, Revision 2, "Effects of Residual Elements on Predicted Radiation Damage to 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 16 EFPY.

TABLE B 3/4.4-1

FARLEY UNIT 1 REACTOR VESSEL TOUGHNESS PROPERTIES

Component	Code No.	Material Type	Cu (%)	P (%)	Ni (%)	T_{ndt} (°F)	RT_{ndt} (°F)	Upper Shell Energy MWD[c]	Upper Shell Energy NMWD[d]
Closure head dome	B6901	A533,B,Cl.1	0.16	0.009	0.50	-30	-20[a]	140	-
Closure head segment	B6902-1	A533,B,Cl.1	0.17	0.007	0.52	-20	-20[a]	138	-
Closure head flange	B6915-1	A508, Cl.2	0.10	0.012	0.64	60[a]	60[a]	75[a]	-
Vessel flange	B6913-1	A508, Cl.2	0.17	0.011	0.69	60[a]	60[a]	106[a]	-
Inlet nozzle	B6917-1	A508, Cl.2	-	0.010	0.83	60[a]	60[a]	-	110
Inlet nozzle	B6917-2	A508, Cl.2	-	0.008	0.80	60[a]	60[a]	-	80
Inlet nozzle	B6917-3	A508, Cl.2	-	0.008	0.87	60[a]	60[a]	-	98
Outlet nozzle	B6916-1	A508, Cl.2	-	0.007	0.77	60[a]	60[a]	-	96.5
Outlet nozzle	B6916-2	A508, Cl.2	-	0.011	0.78	60[a]	60[a]	-	97.5
Outlet nozzle	B6916-3	A508, Cl.2	-	0.009	0.78	60[a]	60[a]	-	100
Nozzle shell	B6914-1	A508, Cl.2	-	0.010	0.68	30	30[a]	148	-
Inter. shell	B6903-2	A533,B,Cl.1	0.13	0.011	0.60	0	0	151.5	97
Inter. shell	B6903-3	A533,B,Cl.1	0.12	0.014	0.56	10	10	134.5	100
Lower shell	B6919-1	A533,B,Cl.1	0.14	0.015	0.55	-20	15	133	90.5
Lower shell	B6919-2	A533,B,Cl.1	0.14	0.015	0.56	-10	5	134	97
Bottom head ring	B6912-1	A508, Cl.2	-	0.010	0.72	10	10[a]	163.5	-
Bottom head segment	B6906-1	A533,B,Cl.1	0.15	0.011	0.52	-30	-30[a]	147	-
Bottom head dome	B6907-1	A533,B,Cl.1	0.17	0.014	0.60	-30	-30[a]	143.5	-
Inter. shell long. weld seam	M1.33	Sub Arc Weld	0.25	0.017	0.21	0[a]	0[a]	-	-
Inter. to lower shell weld seams	G1.18	Sub Arc Weld	0.22	0.011	<0.20[b]	0[a]	0[a]	-	-
Lower shell long. weld seams	G1.08	Sub Arc Weld	0.17	0.022	<0.20[b]	0[a]	0[a]	-	-

[a] Estimate per NUREG-0800 "USNRC Standard Review Plan" Branch Technical Position MTEB 5-2.
[b] Estimated (low nickel weld wire used in fabricating vessel weld seams).
[c] Major working direction.
[d] Normal to major working direction.

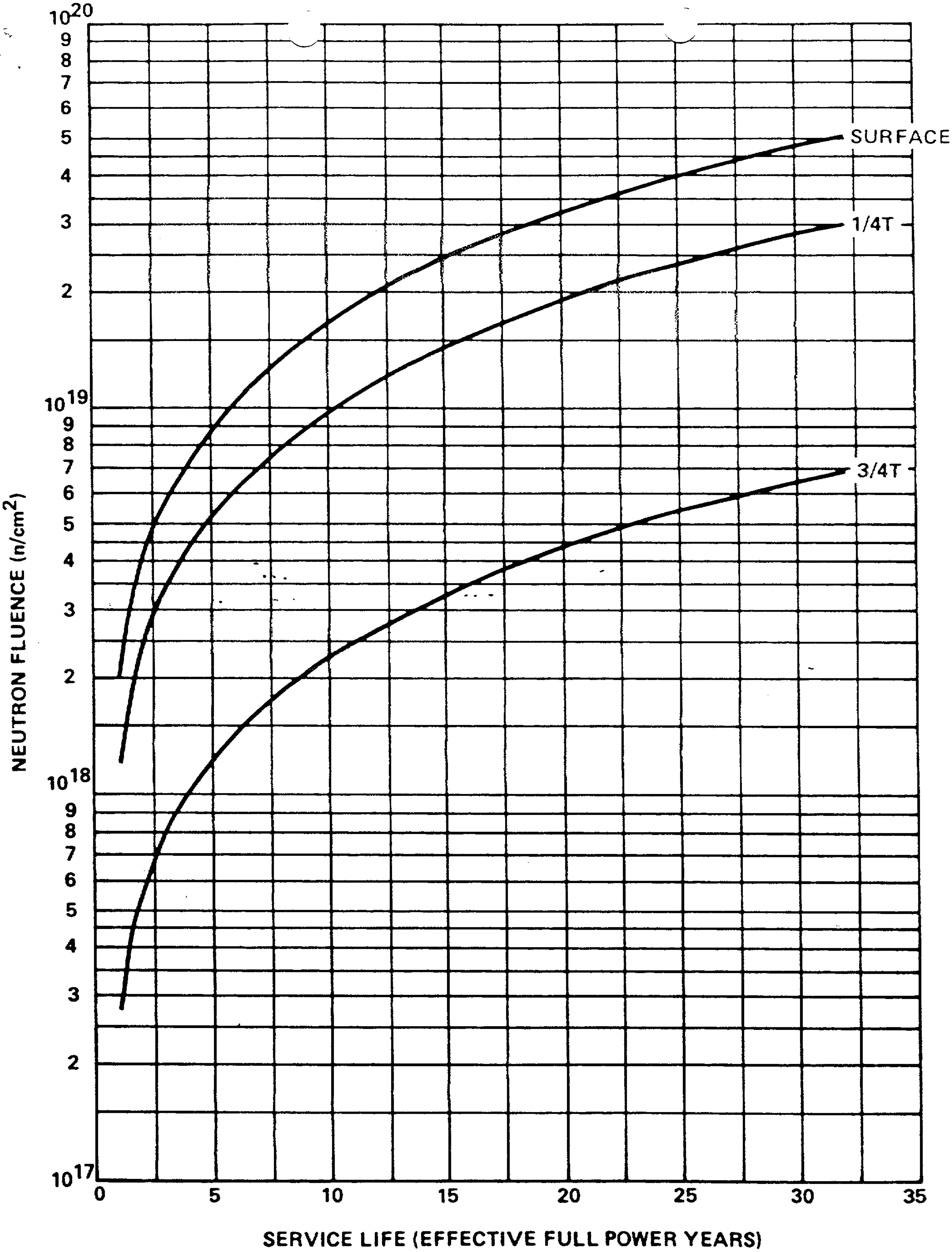

FIGURE B 3/4.4-1 FAST NEUTRON FLUENCE (E>1 MeV) AS A FUNCTION OF FULL POWER SERVICE LIFE (EFPY)

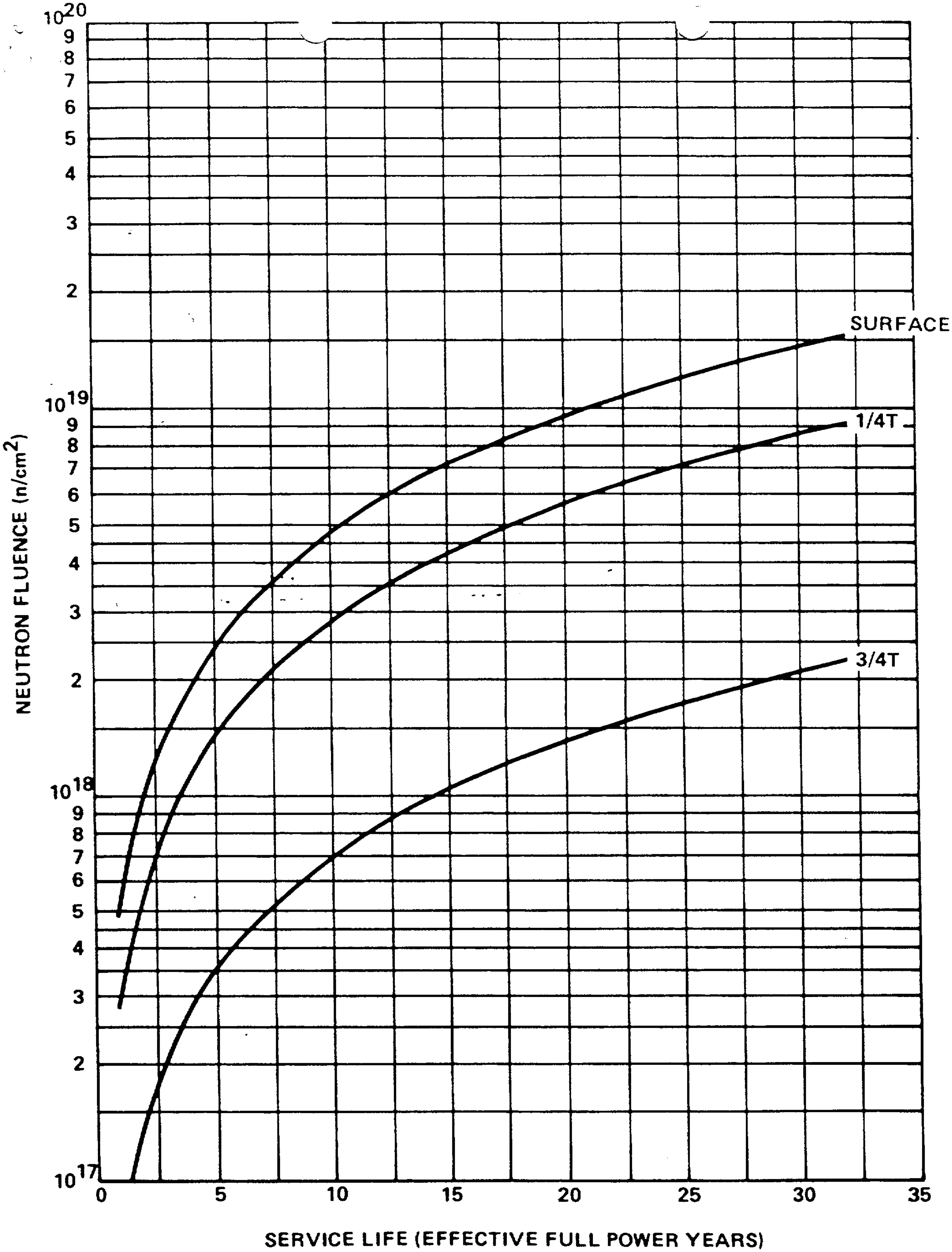

FIGURE B 3/4.4-2 FAST NEUTRON FLUENCE (E>1 MeV) AT 45° AS A FUNCTION OF FULL POWER SERVICE (EFPY)

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 10 CFR 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 1). In addition, the new 10 CFR Part 50 Rule states that a plant specific fracture evaluation may be performed to justify less limiting requirements. As a result, such a fracture analysis was performed for Farley Unit 2. These Farley Unit 2 fracture analysis results are applicable to Farley Unit 1 since the pertinent parameters are identical for both plants. Based upon this fracture analysis, the 16 EFPY heatup and cooldown curves are impacted by the new 10 CFR 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-ductile 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 10 CFR 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

RELATED TO AMENDMENT NO. 71 TO FACILITY OPERATING LICENSE NO. NPF-2

ALABAMA POWER COMPANY

JOSEPH M. FARLEY NUCLEAR PLANT, UNIT NO. 1

DOCKET NO. 50-348

INTRODUCTION

In a letter from R. P. McDonald to S. A. Varga dated October 25, 1985, the Alabama Power Company (the licensee) requested changes to the Joseph M. Farley Nuclear Plant, Unit 1 (Farley-1) Heatup/Cooldown Curves and supporting bases. The staff reviewed the licensee's submittal and determined that the effect of neutron irradiation on all beltline materials had not been assessed by the licensee. The staff's concerns are documented in a letter from E. A. Reeves to R. P. McDonald dated June 16, 1986. In response to this letter, the licensee provided a revised set of heatup and cooldown curves, which were to be applicable for 16 effective full power years (EFPY). The revised curves and their bases were submitted in a letter from R. P. McDonald to L. S. Rubenstein dated September 29, 1986. The curves and bases are to be contained in Figures 3.4-2 and 3.4-3 and Bases Section 3/4.4.10 of the Farley-1 Technical Specifications.

DISCUSSION

Heatup/Cooldown curves must be calculated in accordance with the requirements of Appendix G of 10 CFR 50, which became effective on July 26, 1983. Appendix G of 10 CFR 50 requires that the reactor vessel beltline and closure flange region materials meet the safety margins of Appendix G of the ASME Code Section III. To calculate pressure-temperature limits in accordance with these requirements, the effect of neutron irradiation, boltup, pressure and thermal stresses on the limiting reactor vessel beltline and closure flange region materials must be estimated. The effect of neutron irradiation on the Farley-1 beltline materials is documented in Westinghouse Report WCAP-10934, Rev. 2, dated June 1986, which is in Attachment 2 to the licensee's submittal dated September 29, 1986. The effect of boltup, pressure and thermal stresses on the reactor vessel closure flange region are documented in letters from R. P. McDonald to S. A. Varga dated June 18, 1984, and October 25, 1985.

EVALUATION

The methods recommended by the NRC staff for calculating the effect of neutron irradiation damage are documented in Regulatory Guide 1.99, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials." The relationships documented in Regulatory Guide 1.99 were empirically derived from materials that were irradiated in commercial nuclear reactor surveillance capsules. The most pertinent empirical relationships are contained in Revision 2 to the guide. Revision 2 has been reviewed by the NRC staff and published for comment. The licensee has used the methods recommended in Regulatory Guide 1.99, Revision 2, to calculate the effect of neutron irradiation on the Farley-1 beltline materials.

Neutron irradiation damage is measured by an increase in a material's reference temperature. The value of the reference temperature that results from neutron irradiation damage is called the material's adjusted reference temperature, ART. The limiting ART was used to calculate pressure-temperature limits for the Farley-1 beltline materials. These limits were calculated in accordance with the requirements in Appendix G of the ASME Code Section III. The NRC staff has evaluated these limits using the calculation methods recommended in Standard Review Plan (SRP) 5.3.2, "Pressure-Temperature Limits."

The stresses in the closure flange region resulting from pressure, thermal effects, and boltup were calculated by the licensee using finite element analysis. The closure head and vessel flange geometry used in the finite element analysis was modelled for a typical 4-loop reactor vessel. The Farley-1 plant is a 3-loop reactor vessel. The geometry of the closure flange region in the Farley-1 reactor vessel is slightly different than that of the typical 4-loop reactor vessel. To account for these differences, the licensee used the computation method of Reference 1 to perform a stress analysis of the Farley-1 vessel based on the finite element analysis and an analytical comparison of critical dimensions of the two types of vessels. Their analysis indicates that the typical 4-loop reactor vessel and the Farley-1 reactor vessel have essentially equivalent pressure and boltup stresses at the critical closure flange region. Hence, the stresses from boltup and pressure used for the typical 4-loop plant were used in the fracture mechanics evaluation for Farley-1. The stresses at the critical closure flange region resulting from thermal conditions during heatup or cooldown of the Farley-1 vessel were determined by the computation method in Reference 1 to be significantly less than those calculated for the typical 4-loop plant.

Fracture mechanics evaluations at three discontinuity locations in the closure flange region were performed in accordance with the methodology in Appendix A of ASME Code Section XI. In this analysis the licensee used all the safety factors required by Appendix G of the ASME Code, except for the Code recommended flaw size, to determine the closure flange location that would be considered the critical location. The location with the highest stress intensity factor after applying safety margins was considered the critical closure flange location. The critical location was determined to be the outside surface at the discontinuity between the flange and upper shell of the reactor vessel.

The postulated flaw size recommended by Appendix G of the ASME Code was used for evaluating the beltline region, but was not used in evaluating the closure flange region. The postulated flaw size recommended by Appendix G has a depth of ¼ the section thickness (¼ T) and a length of 1½ times the section thickness. The section thickness at the critical flange location for Farley-1 is 9.125 inches. Appendix G of the ASME Code indicates that smaller defect sizes may be used on an individual case basis, if a smaller size of maximum postulated defect can be assured. The postulated defect used in the licensee's analysis was a 0.625 inch deep by 3.75 inches long surface flaw. The licensee's justification for using a smaller flaw size in evaluating the closure flange region than that used in evaluating the beltline region is that the volumetric examination of the closure flange location will assure detection of the critical size flaw.

Volumetric examination of the reactor vessel flange-to-upper shell weld and specified adjacent base material is accomplished by two ultrasonic scan routines. Coverage from the flange side of the weld involves use of angled longitudinal waves from the flange seal surface. Beam angles are selected based on their ability to provide coverage of the weld and specified adjacent base material to the extent practical and provide near normal incidence to the plane of the weld. Refracted beam angles in the range 0° to 16° are typically used for these examinations. Examinations from the shell side of the weld involve 0°, 45°, and 60° refracted angle beam coverage from the vessel inside diameter surface. Angle beam scanning is performed in two directions, parallel to the weld and perpendicular to the weld from the shell side. Access for the shell side examination is limited to the Ten Year Inservice Inspection outage when the core barrel is removed from the reactor vessel.

The licensee indicates that the fact that postulated flaws are surface related is significant from a detection probability point of view. Incipient cracks starting at right angles to a given surface (OD or ID) provide favorable conditions for detection via ASME Code specified 45° shear wave ultrasonic examinations from the opposite surface. Circumferential flaws are oriented favorably for detection during axial scanning. Axial flaws are oriented favorably for detection during circumferential scans. Circumferentially oriented flaws in the vessel flange weld region also provide favorable conditions for detection during ultrasonic examinations from the flange seal surface.

Additional justifications for permitting smaller postulated flaws in the closure flange region than the size postulated for the beltline region are described in the staff's regulatory analysis of public comments which is in Enclosure 4 to the staff's report SECY-83-80, "10 CFR Part 50-General Revision of Appendices G and H, Fracture Toughness and Reactor Vessel Material Surveillance Requirements," February 25, 1983.

As previously reported, the licensee's fracture mechanics evaluation was performed in accordance with the methodology in Appendix A of ASME Code Section XI. In this method, the stress intensity factors at the crack tip are calculated by linearizing the stress around the postulated flaw. The linearized stress is divided into membrane and bending stresses. The Appendix A method of linearizing stress resulted in negative membrane stresses when considering boltup, pressure and thermal condition during heat-up. The licensee considered the negative membrane stresses equal to zero when determining the stress intensity factor resulting from thermal conditions during heat-up. The NRC staff considers this acceptable, since it conservatively represents the stress condition resulting from heat-up.

The licensee used the negative value of membrane stress when determining the stress intensity factor resulting from boltup and pressure conditions. The negative membrane stress will result in a reduction in the calculated stress intensity factor, since the stress intensity factor is the sum of a positive bending stress and a negative membrane stress. A negative value of membrane stress does not represent the real membrane stress resulting from boltup and pressure conditions. However, the non-conservatism resulting from a negative valued membrane stress will be offset by a high value for the bending stress that results from the linearizing method. Several methods of calculating stress intensity factors for a stress distribution similar to that in the closure flange region were evaluated in Reference 2. The Appendix A method of linearizing the stress around the postulated flaw produced conservative stress intensity factors when compared to those calculated using a finite element analysis method, an ASME Code Section III Appendix G method recommended for nonlinear stress distributions, and a poly-nomial method (Reference 3). This comparison indicates that the Appendix A method of linearizing stress will result in an acceptable fracture mechanics analysis for evaluating flaws in the closure flange region of the reactor vessel.

Using the stress intensity factors calculated in accordance with Appendix A of the ASME Code Section XI and the safety margins of Appendix G of the ASME Code with a postulated flaw of 0.625 inch deep by 3.75 inches long, the licensee proposed pressure-temperature limits for the closure flange region materials. The pressure-temperature limits for the closure flange region materials were combined with the limits for the beltline region to develop the Farley-1 Heatup/Cooldown Curves.

SAFETY SUMMARY

Based on the method documented in Regulatory Guide 1.99, Revision 2, for evaluating the effect of neutron irradiation on reactor vessel beltline materials, and the method of calculating pressure-temperature limits in SRP 3.6.2, the licensee's proposed Heatup/Cooldown Curves for 16 EFPY meet the safety margins of Appendix G of the ASME Code.

Based on the licensee's finite element analysis, the fracture mechanics analysis performed in accordance with Appendix A of Section XI of the ASME Code, and the licensee's and NRC staff's justification for considering smaller postulated flaw sizes based on SECY-83-80, the licensee's proposed pressure-temperature limits for the closure flange region meet the safety margins of Appendix G of the ASME Code.

Based on the above two conclusions, the proposed Heatup/Cooldown Curves that are contained in the licensee's letter dated September 29, 1986, meet the safety margins of Appendix G, 10 CFR 50 for 16 EFPY and are acceptable Farley-1 Technical Specification.

ENVIRONMENTAL CONSIDERATION

This amendment involves a change in the 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 offsite, 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.

CONCLUSION

We have concluded, based on the consideration 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, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

References

1. "Tentative Structural Design Basis for Reactor Pressure Vessels and Directly Associated Components (Pressurized, Water Cooled Systems), U.S. Department of Commerce, December 1, 1958 and February 27, 1959, pp. 58, 59, 60, Addendum No. 1.

2. Bloom, J.M., and Van Der Sluys, W.A., "Determination of Stress Intensity Factors for Gradient Stress Fields," Journal of Pressure Vessel Technology, Vol. 99, August 1977.

3. Buchalet, C.B., and Bamford, W.H., "Stress Intensity Factor Solutions for Continuous Surface Flaws in Reactor Pressure Vessels," Mechanics of Crack Growth, ASTM STP 590, American Society for Testing and Materials, 1976.

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