April 21, 1986

Docket No. 50-364

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

Dear Mr. McDonald:

The Commission has issued the enclosed Amendment No.55 to Facility Operating License NPF-8 for the Joseph M. Farley Nuclear Plant, Unit No. 2. The amendment consists of changes to the Technical Specifications in response to your application transmitted by letter dated September 30, 1985, as supplemented March 27, 1986.

The amendment modifies Technical Specifications Table 4.4-5 and Figures 3.4-2 and 3.4-3 based on results of analysis of Capsule "U" Reactor Vessel Surveillance Program. Changes are in accordance with 10 CFR 50 Appendix G and H and are acceptable for eight effective full power years of operation.

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,

/s/

Edward A. Reeves, Project Manager
PWR Project Directorate #2
Division of PWR Licensing-A

Enclosures:
1. Amendment No.55 to NPF-8
2. Safety Evaluation

cc: w/enclosures:
See next page

LA:PAD#2	PM:PAD#2	D:PAD#2	OELD
DMiller	EReeves:hc	LRubenstein	
4/10/86	4/10/86	4/2/86	4/16/86

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

ALABAMA POWER COMPANY

DOCKET NO. 50-364

JOSEPH M. FARLEY NUCLEAR PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 55
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 September 30, 1985, as supplemented March 27, 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-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. 55 , 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

L. S. Rubenstein

Lester S. Rubenstein, Director
PWR Project Directorate #2
Division of PWR Licensing-A
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 21, 1986

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 55 FACILITY OPERATING LICENSE NO. NPF-8

DOCKET NO. 50-364

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. The corresponding overleaf pages are also provided to maintain document completeness.

Remove Pages	Insert Pages
3/4 4-29	3/4 4-29
3/4 4-30	3/4 4-30
B 3/4 4-6	B 3/4 4-6
B 3/4 4-7	B 3/4 4-7
B 3/4 4-8	B 3/4 4-8
B 3/4 4-9	B 3/4 4-9
B 3/4 4-10	B 3/4 4-10
B 3/4 4-14	B 3/4 4-14

3/4.4.8 CHEMISTRY

The limitations on Reactor Coolant System chemistry ensure that corrosion of the Reactor Coolant System is minimized and reduces the potential for Reactor Coolant System leakage or failure due to stress corrosion. Maintaining the chemistry within the Steady State Limits provides adequate corrosion protection to ensure the structural integrity of the Reactor Coolant System over the life of the plant. The associated effects of exceeding the oxygen, chloride and fluoride limits are time and temperature dependent. Corrosion studies show that operation may be continued with contaminant concentration levels in excess of the Steady State Limits, up to the Transient Limits, for the specified limited time intervals without having a significant effect on the structural integrity of the Reactor Coolant System. The time interval permitting continued operation within the restrictions of the Transient Limits provides time for taking corrective actions to restore the contaminant concentrations to within the Steady State Limits.

The surveillance requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

3/4.4.9 SPECIFIC ACTIVITY

The limitations on the specific activity of the primary coolant ensure that the resulting 2 hour doses at the site boundary will not exceed an appropriately small fraction of Part 100 limits following a steam generator tube rupture accident in conjunction with an assumed steady state primary-to-secondary steam generator leakage rate of 1.0 GPM. The values for the limits on specific activity represent limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters of the Farley site, such as site boundary location and meteorological conditions, were not considered in this evaluation.

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the primary coolant's specific activity greater than 1.0 microcuries/gram DOSE EQUIVALENT I-131, but within the allowable limit shown on Figure 3.4-1, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER. Operation with specific activity levels exceeding 1.0 microcuries/gram DOSE EQUIVALENT I-131 but within the limits shown on Figure 3.4-1 must be restricted to no more than 800 hours per year (approximately 10 percent of the unit's yearly operating time) since the activity levels allowed by Figure 3.4-1 increase the 2 hour thyroid dose at the site boundary by a factor of up to 20 following a postulated steam generator tube rupture. The reporting of cumulative operating time over 500 hours in any 2 consecutive calendar quarters period with greater than 1.0 microcuries/gram DOSE EQUIVALENT I-131 will allow sufficient time for Commission evaluation of the circumstances prior to reaching the 800 hour limit.

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 10CFR 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 for the first full-power service period.

   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 8 effective full power years of service life. The 8 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 content of the material in question, can be predicted using Figure B 3/4.4-1 and the recommendations of Regulatory Guide 1.99, Revision 1, "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 8 EFPY.

Values of ΔRT_{ndt} determined in this manner may be used until the next results from the material surveillance program, evaluated according to ASTM E185, are available. Capsules will be removed in accordance with the requirements of ASTM E185-82 and 10 CFR 50, Appendix H. The surveillance specimen withdrawal schedule is shown in Table 4.4-5. The heatup and cooldown curves must be recalculated when the ΔRT_{ndt} determined from the next surveillance capsule exceeds the calculated ΔRT_{ndt} for the equivalent capsule radiation exposure.

Allowable pressure-temperature relationships for various heatup and cooldown rates are calculated using methods derived from Appendix G in Section III of the ASME Boiler and Pressure Vessel Code as required by Appendix G to 10 CFR Part 50 and these methods are discussed in detail in WCAP-7924-A.

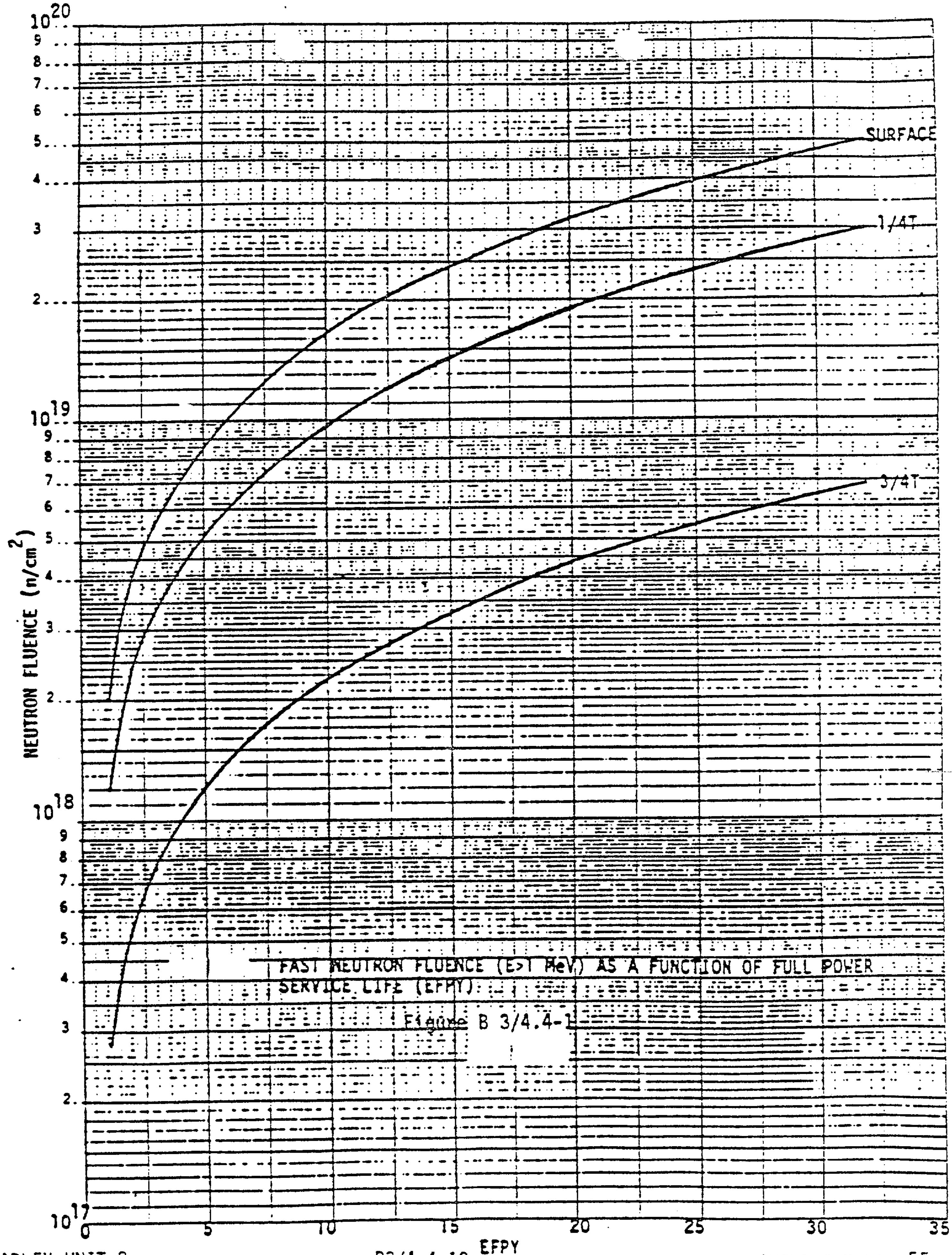
The general method for calculating heatup and cooldown limit curves is based upon the principles of the linear elastic fracture mechanics (LEFM) technology. In the calculation procedures a semi-elliptical surface defect with a depth of one-quarter of the wall thickness, T, and a length of 3/2T is assumed to exist at the inside of the vessel wall as well as at the outside of the vessel wall. The dimensions of this postulated crack, referred to in Appendix G of ASME Section III as the reference flaw, amply exceed the current capabilities of inservice inspection techniques. Therefore, the reactor operation limit curves developed for this reference crack are conservative and provide sufficient safety margins for protection against non-ductile failure. To assure that the radiation embrittlement effects are accounted for in the calculation of the limit curves, the most limiting value of the nil ductility reference temperature, RT_{ndt}, is used and this includes the radiation induced shift, ΔRT_{ndt}, corresponding to the end of the period for which heatup and cooldown curves are generated.

TABLE B3/4.4-1

REACTOR VESSEL TOUGHNESS DATA

Component	Code No.	Grade	Cu (%)	P (%)	Ni (%)	T_{NDT} (°F)	RT_{NDT} (°F)	Average Upper Shelf Energy: Normal to Principal Working Direction (ft-lb)	Average Upper Shelf Energy: Principal Working Direction (ft-lb)
CL. HD. Dome	B7215-1	A533,B,CL.1	0.17	0.010	0.49	-30	16[a]	83[a]	128
CL. HD. Flange	B7207-1	A508,CL.2	0.14	0.011	0.65	60[a]	60[a]	>56[a]	>86[c]
VES. Flange	B7206-1	A508,CL.2	0.10	0.012	0.67	60[a]	60[a]	>71[a]	>109
Inlet Noz.	B7218-2	A508,CL.2	-	0.010	0.68	50[a]	50[a]	103[a]	158
Inlet Noz.	B7218-1	A508,CL.2	-	0.010	0.71	32[a]	32[a]	112[a]	172
Inlet Noz.	B7218-3	A508,CL.2	-	0.010	0.72	60[a]	60[a]	98[a]	150
Outlet Noz.	B7217-1	A508,CL.2	-	0.010	0.73	60[a]	60[a]	100[a]	154
Outlet Noz.	B7217-2	A508,CL.2	-	0.010	0.72	6[a]	6[a]	108[a]	167
Outlet Noz.	B7217-3	A508,CL.2	-	0.010	0.72	48[a]	48[a]	103[a]	158
Upper Shell	B7216-1	A508,CL.2	-	0.010	0.73	30	30[a]	97[a]	149
Inter Shell	B7203-1	A533,B,CL.1	0.14	0.010	0.60	-40	15	99	140
Inter Shell	B7212-1	A533,B,CL.1	0.20	0.018	0.60	-30	-10	99	134
Lower Shell	B7210-1	A533,B,CL.1	0.13	0.010	0.56	-40	18	103	128
Lower Shell	B7210-2	A533,B,CL.1	0.14	0.015	0.57	-30	0	99	145
Trans. Ring	B7208-1	A508,CL.2	-	0.010	0.73	40	40[a]	89[a]	137
Bot. HD. Dome	B7214-1	A533,B,CL.1	0.11	0.007	0.48	-30	-2[a]	87[a]	134
Inter. Shell	A1.46	SMAW	0.02	0.009	0.96	0[a]	0[a]	>131	-
Long Seams	A1.40	SMAW	0.02	0.010	0.93	-60	-60	>106	-
Inter Shell to Lower Shell	G1.50	SAW	0.13	0.016	<.20[b]	-40	-40	>102	-
Lower Shell Long Seams	G1.39	SAW	0.05	0.006	<.20[b]	-70	-70	>126	-

(a) Estimate per NUREG 0800 "USNRC Standard Review Plan" Branch Technical Position MTEB 5-2.
(b) Estimated.
(c) Upper shelf not available, value represents minimum energy at the highest test temperature.



FAST NEUTRON FLUENCE (E>1 MeV) AS A FUNCTION OF FULL POWER SERVICE LIFE (EFPY)

Figure B 3/4.4-1

HEATUP

Three separate calculations are required to determine the limit curves for finite heatup rates. As is done in the cooldown analysis, allowable pressure-temperature relationships are developed for steady-state conditions as well as finite heatup rate conditions assuming the presence of a 1/4T defect at the inside of the vessel wall. The thermal gradients during heatup produce compressive stresses at the inside of the wall that alleviate the tensile stresses produced by internal pressure. The metal temperature at the crack tip lags the coolant temperature; therefore, the K_{IR} for the 1/4T crack during heatup is lower than the K_{IR} for the 1/4T crack during steady-state conditions at the same coolant temperature. During heatup, especially at the end of the transient, conditions may exist such that the effects of compressive thermal stresses and different K_{IR}'s for steady-state and finite heatup rates do not offset each other and the pressure-temperature curve based on steady-state conditions no longer represents a lower bound of all similar curves for finite heatup rates when the 1/4T flaw is considered. Therefore, both cases have to be analyzed in order to assure that at any coolant temperature the lower value of the allowable pressure calculated for steady-state and finite heatup rates is obtained.

The second portion of the heatup analysis concerns the calculation of pressure-temperature limitations for the case in which a 1/4T deep outside surface flaw is assumed. Unlike the situation at the vessel inside surface, the thermal gradients established at the outside surface during heatup produce stresses which are tensile in nature and thus tend to reinforce any pressure stresses present. These thermal stresses, of course, are dependent on both the rate of heatup and the time (or coolant temperature) along the heatup ramp. Furthermore, since the thermal stresses, at the outside are tensile and increase with increasing heatup rate, a lower bound curve cannot be defined. Rather, each heatup rate of interest must be analyzed on an individual basis.

Following the generation of pressure-temperature curves for both the steady-state and finite heatup rate situations, the final limit curves are produced as follows. A composite curve is constructed based on a point-by-point comparison of the steady-state and finite heatup rate data. At any given temperature, the allowable pressure is taken to be the lesser of the three values taken from the curves under consideration.

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 2). In addition, the new 10 CFR 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 8 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 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).

MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL	:	R. V. INTERMEDIATE SHELL
COPPER CONTENT	:	0.20 WT%
PHOSPHORUS CONTENT	:	0.018 WT%
INITIAL RT_{NDT}	:	$-10^{\circ}F$
RT_{NDT} AFTER 8 EFPY	:	1/4T, $146^{\circ}F$
	:	3/4T, $83^{\circ}F$

CURVES APPLICABLE FOR HEATUP RATES UP TO $60^{\circ}F/HR$ FOR THE SERVICE PERIOD UP TO 8 EFPY

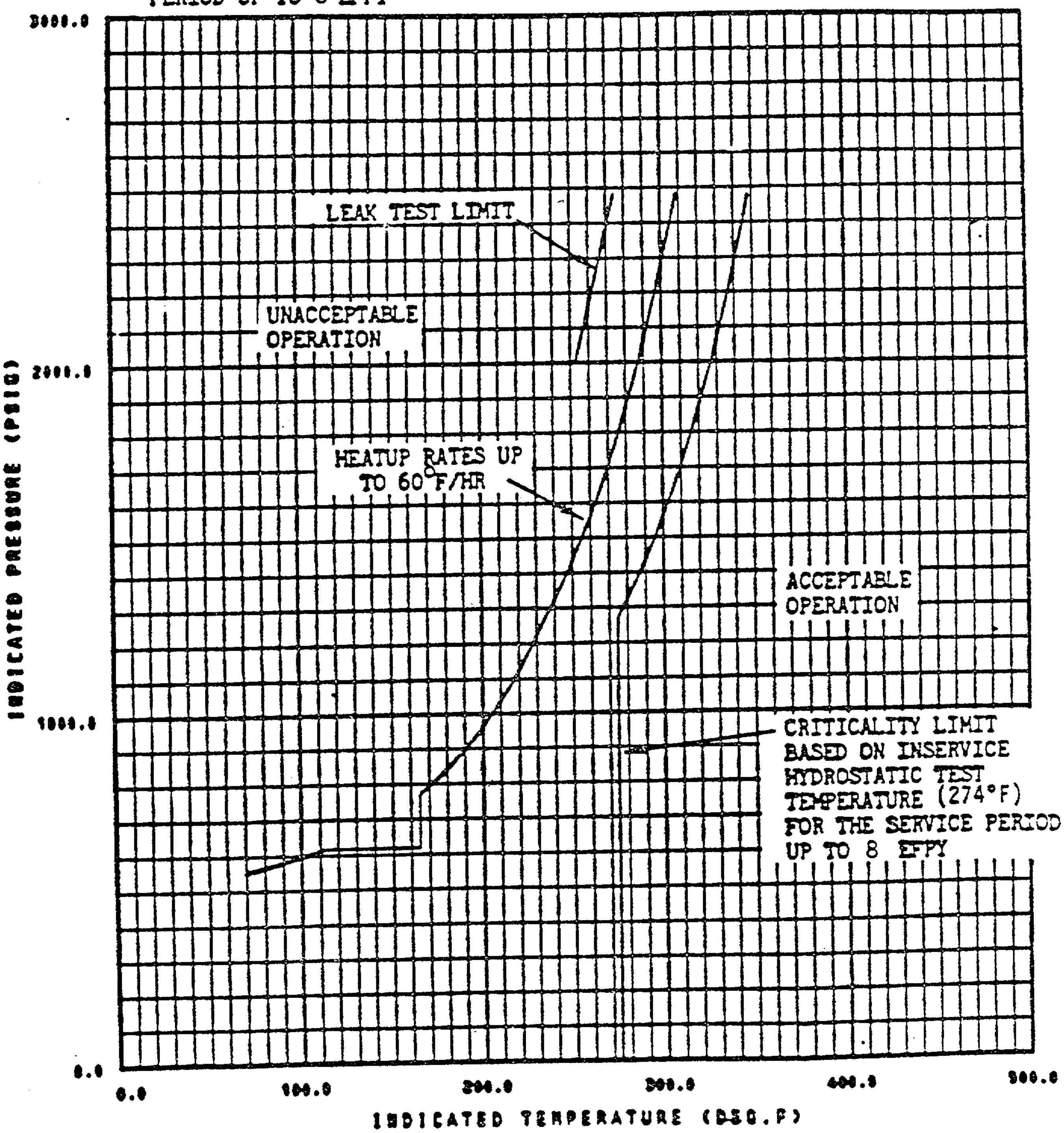


FIGURE 3.4-2 FARLEY UNIT 2 REACTOR COOLANT SYSTEM HEATUP LIMITATIONS APPLICABLE FOR THE FIRST 8 EFPY

MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL	:	R. V. INTERMEDIATE SHELL
COPPER CONTENT	:	0.20 WT%
PHOSPHORUS CONTENT	:	0.018 WT%
INITIAL RT_{NDT}	:	$-10^{\circ}F$
RT_{NDT} AFTER 8 EFPY	:	1/4T, $146^{\circ}F$
	:	3/4T, $83^{\circ}F$

CURVES APPLICABLE FOR COOLDOWN RATES UP TO $100^{\circ}F$/HR FOR THE SERVICE PERIOD UP TO 8 EFPY

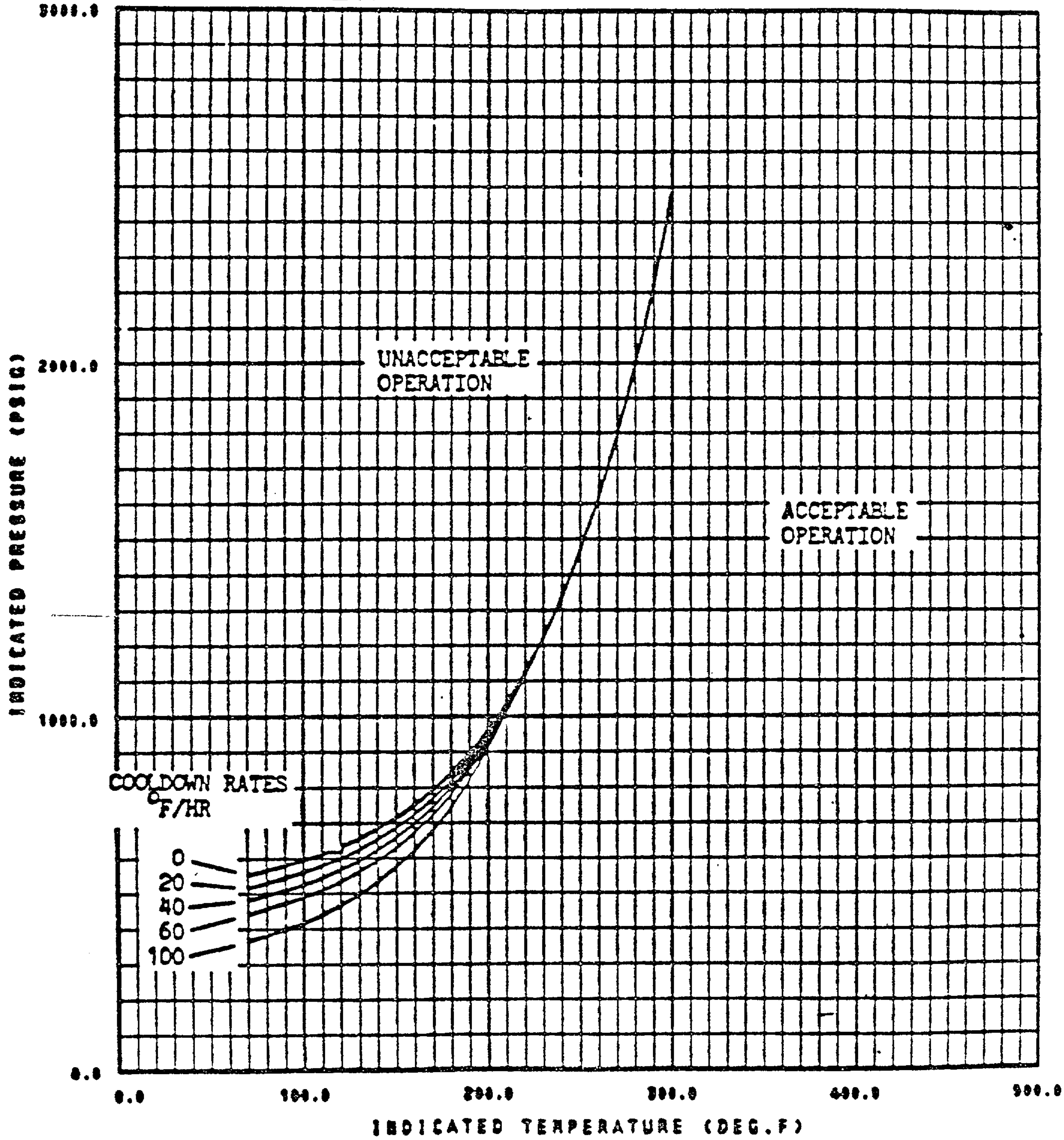


FIGURE 3.4-3 FARLEY UNIT 2 REACTOR COOLANT SYSTEM COOLDOWN LIMITATIONS APPLICABLE FOR THE FIRST 8 EFPY

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 55 TO FACILITY OPERATING LICENSE NO. NPF-8

ALABAMA POWER COMPANY

JOSEPH M. FARLEY NUCLEAR PLANT, UNIT NO. 2

DOCKET NO. 50-364

Introduction

In a letter from R. P. McDonald to S. A. Varga dated September 30, 1985, Alabama Power Company (the licensee, APCo) requested changes to the Joseph M. Farley Nuclear Plant, Unit 2 heatup/cooldown curves and supporting bases. The curves and bases are contained in Figures 3.4-2 and 3.4-3 and Bases Section 3/4.4.10 of the Farley 2 Technical Specifications. The effect of neutron irradiation on the Farley 2 reactor vessel beltline materials is documented in Westinghouse Report WCAP-10910, which is enclosed in the licensee's letter of September 30, 1985. The effect of boltup, pressure and thermal stresses on the reactor vessel closure flange region is documented in Attachment 2 to the licensee's letter of September 30, 1985 and in a previous letter from R. P. McDonald to S. A. Varga dated June 18, 1984. By letter dated March 27, 1986, APCo provided supplementary information following discussions with the NRC staff. Our discussion and evaluation follows.

Discussion and Evaluation

Heatup/cooldown curves must be calculated in accordance with the requirements of Appendix G, 10 CFR 50, which became effective on July 26, 1983. Appendix G, 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 method recommended by the NRC staff for calculating the effect of neutron irradiation damage is documented in Regulatory Guide 1.99, Rev. 1, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials." This guide indicates that when credible surveillance data becomes available, increases in reference temperatures resulting from neutron irradiation damage may be predicted by extrapolating the surveillance data to higher or lower fluences following the slope of the family of curves in Figure 1 of the guide. The limiting material in the Farley 2 reactor vessel beltline is Plate B7212-1. Samples from this plate were placed in the Farley 2 reactor vessel surveillance capsules for irradiation and testing. Test results on this irradiated plate material were reported in Westinghouse Report WCAP-10425, "Analysis of Capsule U from the Alabama Power Company, Joseph M. Farley Unit 2 Reactor Vessel Radiation Surveillance Program" in APCo letter dated November 10, 1983.

Since material from the limiting plate has been placed in the Farley 2 surveillance capsules, test results after irradiation of the capsules will produce credible surveillance data. Hence, we have evaluated the effect of neutron irradiation on the Farley 2 reactor vessel by extrapolating the surveillance test data from samples of Plate B7212-1. This extrapolation was done in accordance with Regulatory Guide 1.99, Rev. 1. Using the extrapolated values for RT_{NDT}, the licensee's proposed heatup and cooldown curves meet the safety margins of Appendix G of the ASME Code Section III at a neutron fluence of 7.7×10^{18} η/cm^2 (E>1MeV). This corresponds to eight effective full power years (EFPY) of operation.

The licensee's submittal of September 30, 1985, includes a bases section and figures showing the proposed pressure/temperature limits. The pressure/temperature limits are proposed for nine EFPY. But, our review shows that the licensee's proposed calculation method does not include sufficient margin to account for neutron irradiation damage. Hence, the proposed bases section required revision. Also, the proposed heatup/cooldown curves would meet the safety margins of Appendix G of the ASME Code Section III for only 8.0 EFPY instead of the proposed 9.0 EFPY. For these reasons, we advised the licensee and Westinghouse of our evaluation during various telecons. As a result, by letter dated March 27, 1986, the licensee modified their earlier bases section and provided heatup/cooldown curves for 8.0 EFPY which are acceptable.

The stresses resulting from pressure, thermal and boltup on the closure flange region 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 four loop reactor vessel. However, the Farley 2 plant contains a three loop reactor vessel. The geometry of the closure flange region in Farley 2 reactor vessel is slightly different than that of the typical four loop reactor vessel. To account for these differences, the licensee performed a dimensional stress analysis of the two types of vessels. Their analysis indicates that the typical four loop reactor vessel and the Farley 2 reactor vessel have essentially equivalent stresses resulting from pressure and boltup in the critical closure flange region. Hence, the stresses from boltup and pressure used for the typical four loop plant were used in the fracture mechanics evaluation for Farley 2. The stresses resulting from thermal conditions during heatup or cooldown of the Farley 2 vessel were determined by the computation method 1/ to be significantly less than those calculated for the typical four loop plant. The thermal stresses at the critical closure flange region in the Farley 2 reactor vessel were calculated 1/ by reducing the finite element thermal stresses for the typical four loop reactor vessel by the ratio of the thermal stresses in the three loop to those in the four loop.

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.

Fracture mechanics evaluation 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 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 has a depth of 1/4 the section thickness (1/4 T) and a length of 1 1/2 times the section thickness. The section thickness at the critical flange location for Farley 2 is 9.125 inches. Appendix G of the ASME Code indicates that small 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 volumetric examination of the closure flange location will detect this 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 examinations is limited to the ten year ISI outage when the core barrel is removed from the reactor vessel.

The licensee indicates that the fact that the 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 that postulated for the beltline region are described in Enclosure 4 to the NRC 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, ASME Code Section XI 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 staff considers this acceptable, since it conservatively represents the stress condition resulting from heatup. 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 may be offset by a high value for the bending stress that results from the linearizing method. The NRC staff has not completed its evaluation of this issue. The NRC staff is discussing this issue with individuals who are members of the ASME Code Subcommittee on Flaw Evaluation. If we determine that the use of negative-valued membrane stresses and high bending stresses calculated in accordance with the Appendix A, ASME Code Section XI method of linearizing stresses results in non-conservative stress intensity factors, we will supplement this evaluation and inform the licensee that the approved pressure-temperature limits may require further adjustment.

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 material were incorporated into the proposed Farley 2 heatup/cooldown curves.

Safety Summary

1) Based on the method documented in Regulatory Guide 1.99, Rev. 1 used for evaluating the Farley 2 surveillance data and reactor vessel beltline materials, the licensee's proposed heatup/cooldown curves will meet the safety margins of Appendix G of the ASME Code for 8 EFPY.

2) 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 our further justification for considering smaller postulated flaw sizes (based on SECY-83-80), the licensee's proposed pressure-temperature limits for the closure flange region will meet the safety margins of Appendix G of the ASME Code.

Based on our review and on the above two conclusions, we conclude that the modified heatup/cooldown curves provided in licensee letter dated March 27, 1986, meet the safety margins of Appendix G, 10 CFR 50 and are acceptable for eight EFPY of operation.

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 Sec 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 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, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: April 21, 1986

Principal Contributor: B. Elliot