Docket

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FEB 13 1981

Docket No. 50-348

RECEIVED FEB 23 1981 U.S. NUCLEAR REGULATORY COMMISSION

Mr. F. L. Clayton
Senior Vice President
Alabama Power Company
Post Office Box 2641
Birmingham, Alabama 35291

Dear Mr. Clayton:

The Commission has issued the enclosed Amendment No. 18 to Facility Operating License No. NPR-2 for the Joseph M. Farley Nuclear Plant, Unit No. 1. The amendment consists of changes to the Technical Specifications in response to your application transmitted by letter dated March 28, 1980, supplemented by letter dated January 5, 1981 and your application transmitted by letter dated January 29, 1981, supplemented by letter dated January 23, 1981.

The amendment provides the following Technical Specification changes:

1. A new rod bow penalty curve and bases, and

2. New heatup and cooldown curves and bases. These are based on the Capsule Y analyses after 1.3 effective full power years of operation.

Minor changes were made to some of your proposals. These changes have been discussed with your staff who concur with our changes.

Copies of the Safety Evaluation and the Notice of Issuance are also enclosed.

Sincerely,

Original signed by:
S. A. Varga

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing

Enclosures:
1. Amendment No. 18 to NPF-1
2. Safety Evaluation
3. Notice Of Issuance

cc: w/enclosures
See next page

legal review of letter, amendment and FR notice only — note that the actual letter does not correspond to this copy SET

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SURNAME	CParrish	EReeves:ds	SVarga	TNovak	STurk		
DATE	2/10/81	2/11/81	2/11/81	2/12/81	2/12/81		

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

February 13, 1981

Docket No. 50-348

Mr. F. L. Clayton
Senior Vice President
Alabama Power Company
Post Office Box 2641
Birmingham, Alabama 35291

Dear Mr. Clayton:

The Commission has issued the enclosed Amendment No. 18 to Facility Operating License No. NPF-2 for the Joseph M. Farley Nuclear Plant, Unit No. 1. The amendment consists of changes to the Technical Specifications in response to your application transmitted by letter dated March 28, 1980, supplemented by letter dated January 5, 1981 and to your application transmitted by letter dated January 9, 1981, supplemented by letter dated January 23, 1981.

The amendment provides the following Technical Specification changes:

1. A new rod bow penalty curve and bases, and
2. New heatup and cooldown curves and bases. These are based on the Capsule Y analyses after 1.13 effective full power years of operation.

Minor changes were made to some of your proposals. These changes have been discussed with your staff who concur with our changes.

Copies of the Safety Evaluation and the Notice of Issuance are also enclosed.

Sincerely,

Steven A. Varga

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing

Enclosures:
1. Amendment No. 18 to NFP-1
2. Safety Evaluation
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cc: w/enclosures
See next page

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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. 18
License No. NPF-2

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

   A. The applications for amendment by Alabama Power Company (the licensee) dated March 28, 1980 (supplemented by letter dated January 5, 1981) and January 9, 1981 (supplemented by letter dated January 23, 1981), comply 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 applications, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

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

   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.

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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 N 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. 18, 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 the date of its issuance.

FOR THE NUCLEAR REGUALTORY COMMISSION

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: February 13, 1981

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 18 TO FACILITY OPERATING LICENSE NO. NPF-1

DOCKET NO. 50-348

Revise Appendix A as follows:

Remove Pages	Insert Pages
XI	XI
3/4 2-10a	3/4 2-10a
3/4 4-25	3/4 4-25
3/4 4-26	3/4 4-26
3/4 4-27	3/4 4-27
3/4 4-28	3/4 4-28
B 2-2	B 2-2
B 3/4 2-5	B 3/4 2-5
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 4-11	B 3/4 4-11
———	B 3/4 4-12
———	B 3/4 4-13
———	B 3/4 4-14
———	B 3/4 4-6a

BASES

SECTION	PAGE

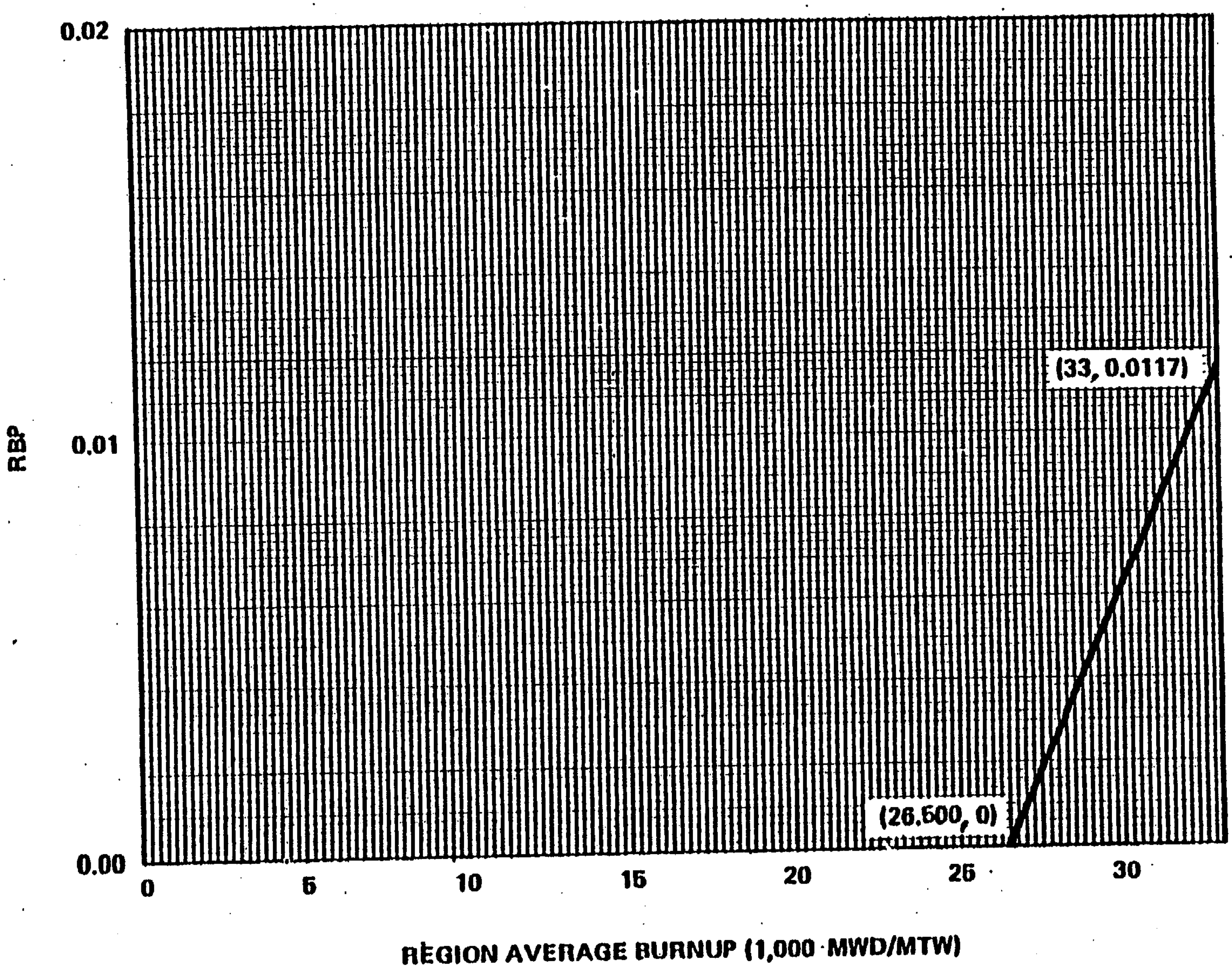

Figure 3.2-3 Rod Bow Penalty (RBP) as a Function of Burnup.

REACTOR COOLANT SYSTEM

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.9.1 The Reactor Coolant System (except the pressurizer) temperature and pressure shall be limited in accordance with the limit lines shown on Figures 3.4-2 and 3.4-3 during heatup, cooldown, criticality, and inservice leak and hydrostatic testing with:

a. A maximum heatup of 100°F in any one hour period.

b. A maximum cooldown of 100°F in any one hour period.

c. A maximum temperature change of less than or equal to 10°F in any one hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves.

APPLICABILITY: At all times.

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limit within 30 minutes; perform an engineering evaluation or inspection to determine the effects of the out-of-limit condition on the fracture toughness of the Reactor Pressure Vessel; determine that the Reactor Pressure Vessel remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the RCS T_{avg} and pressure to less than 200°F and 500 psig, respectively, within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.9.1.1 The Reactor Coolant System temperature and pressure shall be determined to be within the limits at least once per hour during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

4.4.9.1.2 The reactor vessel material irradiation surveillance specimens shall be removed and examined, to determine changes in material properties, as required by 10 CFR 50, Appendix H in accordance with the schedule in Table 4.4-5. The results of these examinations shall be used to update Figures 3.4-2 and 3.4-3.

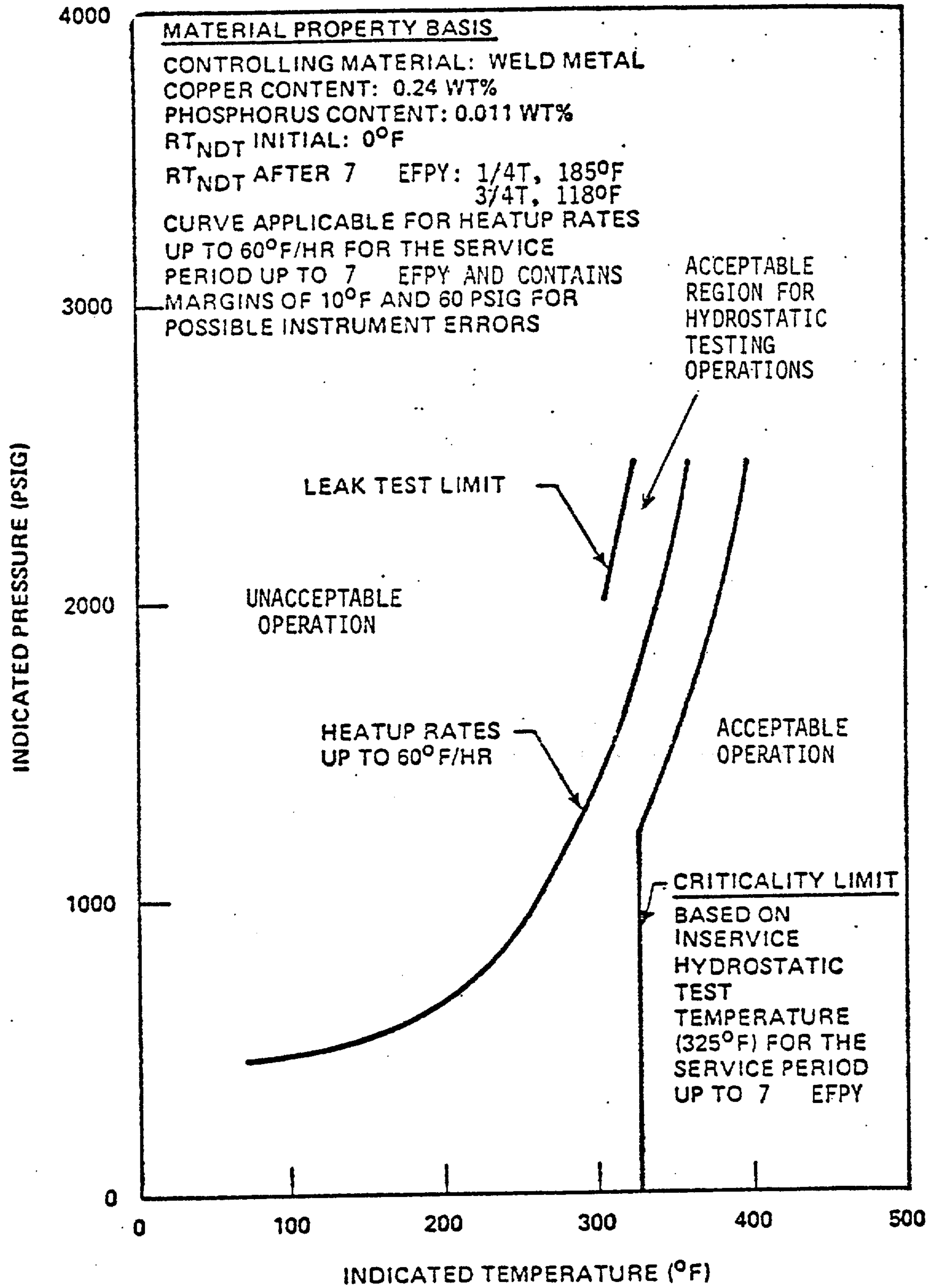

Figure 3.4-2 Farley Unit 1 Reactor Coolant System Heatup Limitations Applicable For The First 7 EFPY

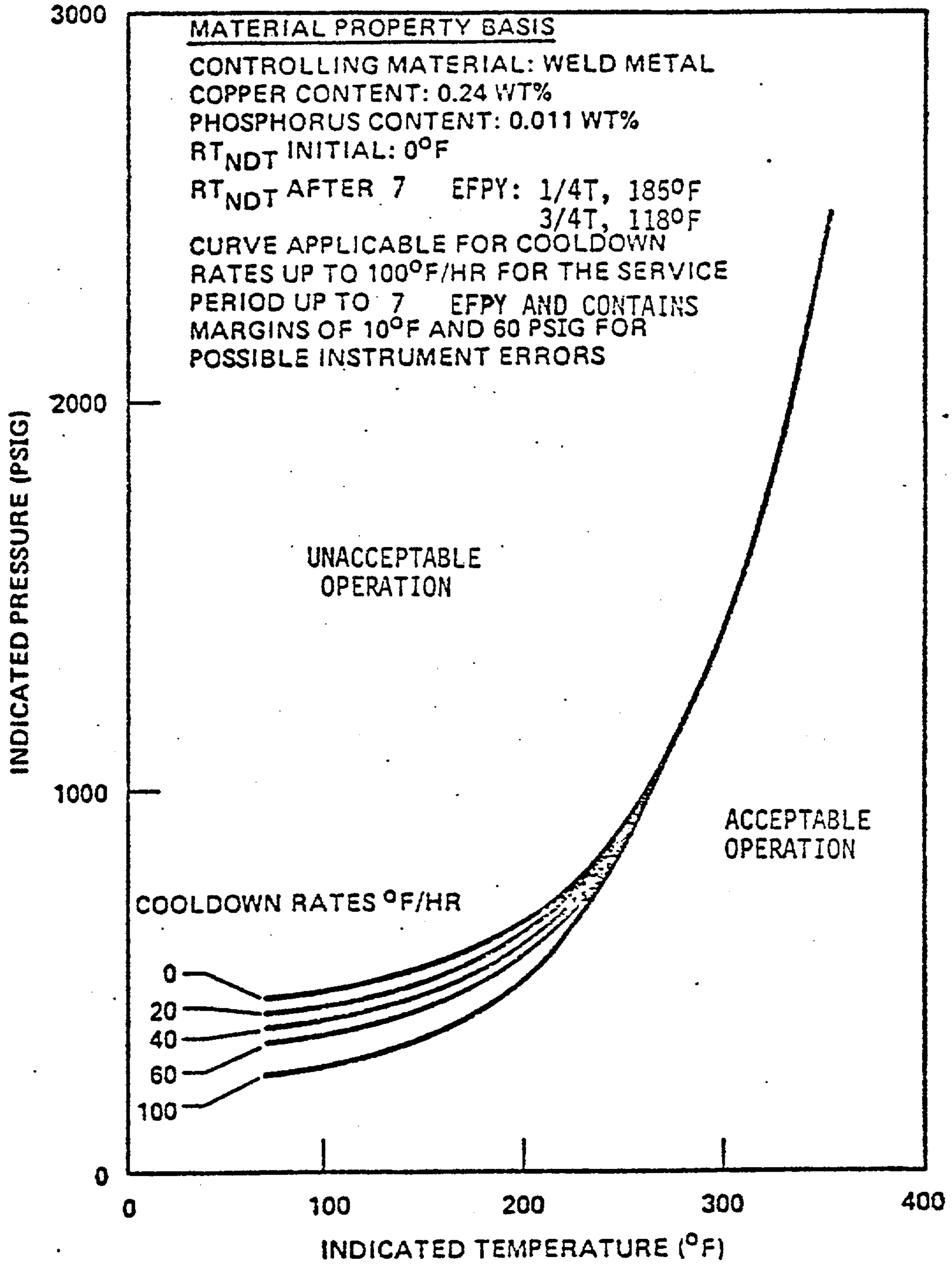

Figure 3.4-3 Farley Unit 1 Reactor Coolant System Cooldown Limitations Applicable For The First 7 EFPY

TABLE 4.4-5

REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM-WITHDRAWAL SCHEDULE

CAPSULE	VESSEL LOCATION	LEAD FACTOR	WITHDRAWAL TIME
Y	343°	3.5	1st Refueling Outage
U	107°	3.5	3 EFPY
X	287°	3.5	6 EFPY
W	110°	2.9	11 EFPY
V	290°	2.9	20 EFPY
Z	340°	2.9	STBY

SAFETY LIMITS

BASES

The curves are based on an enthalpy hot channel factor, $F^N_{\Delta H}$, of 1.55 and a reference cosine with a peak of 1.55 for axial power shape. An allowance is included for an increase in $F^N_{\Delta H}$ at reduced power based on the expression:

$$F^N_{\Delta H} \leq 1.55 \, [1 + 0.2 \, (1-P)]$$

Where P is the fraction of RATED THERMAL POWER

These limiting heat flux conditions are higher than those calculated for the range of all control rods fully withdrawn to the maximum allowable control rod insertion assuming the axial power imbalance is within the limits of the f_1 (ΔI) function of the Overtemperature trip. When the axial power imbalance is not within the tolerance, the axial power imbalance effect on the Overtemperature ΔT trips will reduce the setpoints to provide protection consistent with core safety limits.

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

The restriction of this Safety Limit protects the integrity of the Reactor Coolant System from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.

The reactor pressure vessel pressurizer and the reactor coolant system piping and fittings are designed to Section III of the ASME Code for Nuclear Power Plant which permits a maximum transient pressure of 110% (2735 psig) of design pressure.

The Safety Limit of 2735 psig is therefore consistent with the design criteria and associated code requirements.

The entire Reactor Coolant System is hydrotested at 3107 psig, 125% of design pressure, to demonstrate integrity prior to initial operation.

a. Abnormal perturbations in the radial power shape, such as from rod misalignment, effect $F^N_{\Delta H}$ more directly than F_Q.

b. Although rod movement has a direct influence upon limiting F to within its limit, such control is not readily available to limit $F^N_{\Delta H}$, and

c. Errors in prediction for control power shape detected during startup physics tests can be compensated for in F_Q by restricting axial flux distributions. This compensation for $F^N_{\Delta H}$ is less readily available.

Fuel rod bowing reduces the value of DNB ratio. Sufficent credit is available to offset this reduction. This credit comes from generic design margins totaling 9.1% and 3% margin in the difference between the 1.3 DNBR safety limit and the minimum DNBR calculated for the Complete Loss of Flow event. The penalties applied to $F^N_{\Delta H}$ to account for Rod Bow (Figure 3.2-3) as a function of burnup are consistent with those described in Mr. John F. Stolz's (NRC) letter to T. M. Anderson (Westinghouse) dated April 5, 1979, and WCAP-8691, Rev. 1 (partial rod bow test data).

The radial peaking factor, Fxy (z), is measured periodically to provide additional assurance that the hot channel factor, F_Q (z), remains within its limit. The Fxy (z) limits were determined from expected power control maneuvers over the full range of burnup conditions in the core.

3/4.2.4 QUADRANT POWER TILT RATIO

The quardrant power tilt ratio limit assures that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during startup testing and periodically during power operation.

The limit of 1.02 at which corrective action is required provides DNB and linear heat generation rate protection with x-y plane power tilts. A limiting tilt of 1.025 can be tolerated before the margin for uncertainty in F_Q is depleted. The limit of 1.02 was selected to provide an allowance for the uncertainty associated with the indicated power tilt.

The two hour time allowance for operation with a tilt condition greater than 1.02 but less than 1.09 is provided to allow identification and correction of a dropped or misaligned rod. In the event such action does not correct the tilt, the margin for uncertainty on F_Q is reinstated by reducing the power by 3 percent for each percent of tilt in excess of 1.0.

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the primary coolant's specific activity > 1.0 µCi/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 µCi/gram DOSE EQUIVALENT I-131 but within the limits shown on Figure 3.4-1 must be restricted to no more than 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.

Reducing T_{avg} to <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.

3/4.4.9 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.

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 non-ductile 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°/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 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 E185-73, 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.

Heatup and cooldown limit curves are calculated using the most limiting value of the nil-ductility reference temperature, RT_{NDT}, at the end of 7 effective full power years of service life. The 7 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 7 EFPY (as well as adjustments for possible errors in the pressure and temperature sensing instruments).

Values of ΔRT_{NDT} determined in this manner may be used until the results from the material surveillance program, evaluated according to ASTM E185, are available. Capsules will be removed in accordance with the requirements of ASTM E185-73 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 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 the following paragraphs.

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 4.4

REACTOR VESSEL TOUGHNESS DATA

Component	Code No.	Material Type	Cu (%)	P (%)	T_{NDT} (°F)	MWD (°F)	NMWD (°F)	RT_{NDT} (°F)	Upper Shelf Energy MWD	Upper Shelf Energy NMWD
Closure head dome	B6901	A533B, Cl.1	0.16	0.009	-30	20	40[a]	-20	140	-
Closure head segment	B6902-1	A533B, Cl.1	0.17	0.007	-20	-10	10[a]	-20	138	-
Closure head flange	B6915-1	A508, Cl. 2	0.10	0.012	60[a]	-20	0[a]	60	75[a]	-
Vessel flange	B6913-1	A508, Cl. 2	0.17	0.011	60[a]	-30	-10[a]	60	106[a]	-
Inlet nozzle	B6917-1	A508, Cl. 2	-	0.010	60[a]	-	45	60	-	110
Inlet nozzle	B6917-2	A508, Cl. 2	-	0.008	60[a]	-	115	60	-	80
Inlet nozzle	B6917-3	A508, Cl. 2	-	0.008	60[a]	-	35	60	-	98
Outlet nozzle	B6916-1	A508, Cl. 2	-	0.007	60[a]	-	60	60	-	96.5
Outlet nozzle	B6916-2	A508, Cl. 2	-	0.011	60[a]	-	30	60	-	97.5
Outlet nozzle	B6916-3	A508, Cl. 2	-	0.009	60[a]	-	50	60	-	100
Nozzle shell	B6914-1	A508, Cl. 2	-	0.010	30	70	90[a]	30	148	-
Inter. shell	B6903-2	A533B, Cl. 1	0.13	0.011	0	-25	40	0	151.5	97
Inter. shell	B6903-3	A533B, Cl. 1	0.12	0.014	10	5	52	10	134.5	100
Lower shell	B6919-1	A533B, Cl. 1	0.14	0.015	-20	-5	75	15	133	90.5
Lower shell	B6919-2	A533B, Cl. 1	0.14	0.015	-10	0	65	5	134	97
Bottom head ring	B6912-1	A508, Cl. 2	-	0.010	10	-25	-5[a]	10	163.5	-
Bottom head segment	B6906-1	A533B, Cl. 1	0.15	0.011	-30	-50	-30[a]	-30	147	-
Bottom head dome	B6907-1	A533B, Cl. 1	0.17	0.014	-30	-10	10[a]	-30	143.5	-
Inter. shell long. weld seams			0.27	0.015	0[a]	-	<60	0	-	-
Inter. to lower shell weld seam			0.24	0.011	0[a]	-	<60	0	-	-
Lower shell long. weld seams			0.17	0.022	0[a]	-	<60	0	-	-

(a) Estimated per NRC Regulatory Standard Review Plan, section 5.3.2.

MWD - Major Working Direction
NMWD - Normal to Major Working Direction

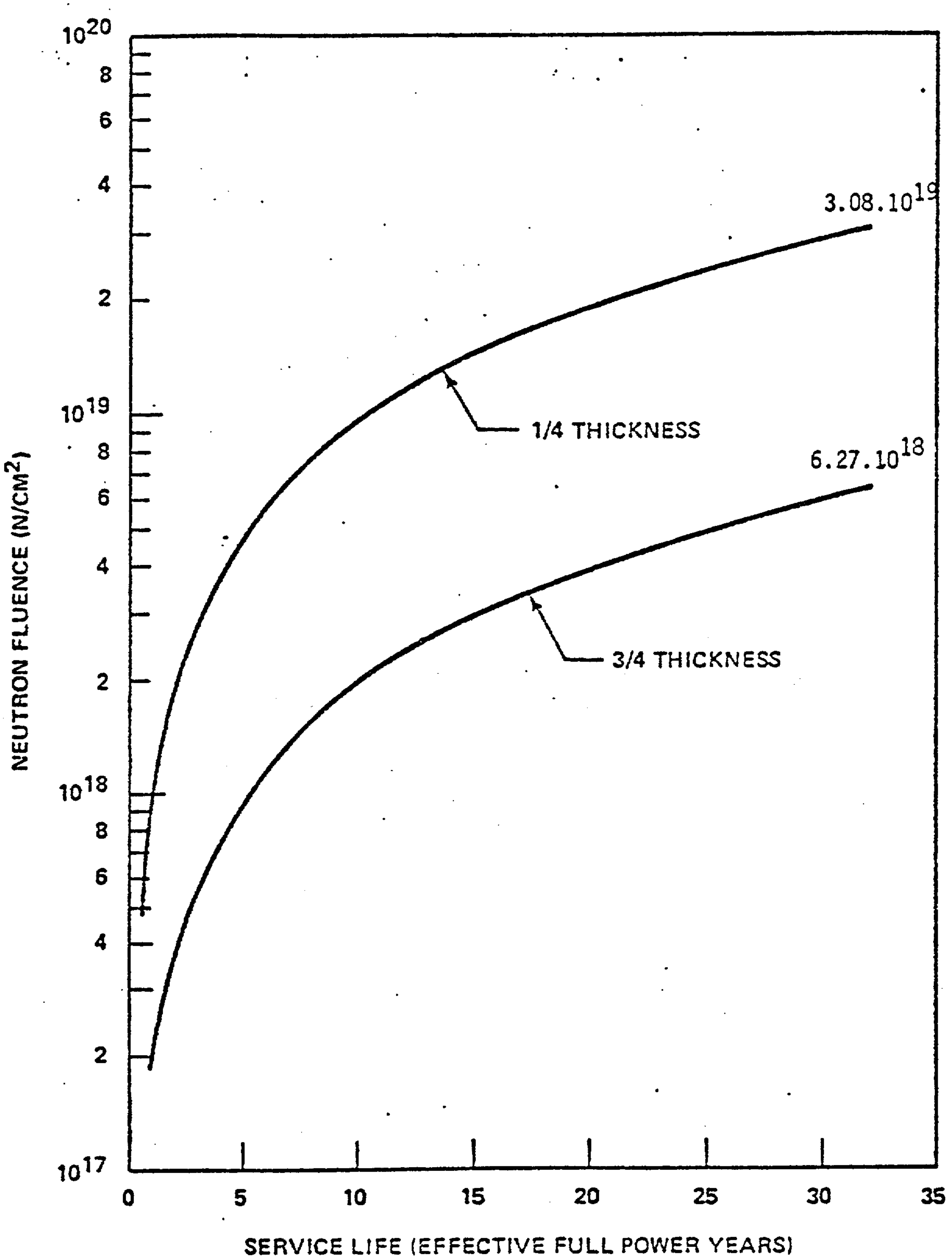

Figure B 3/4.4-1 Fast-Neutron Fluence (E > 1 Mev) as a Function of Full-Power Service Life

The ASME approach for calculating the allowable limit curves for various heatup and cooldown rates specifies that the total stress intensity factor, K_I, for the combined thermal and pressure stresses at any time during heatup or cooldown cannot be greater than the reference stress intensity factor, K_{IR}, for the metal temperature at that time. K_{IR} is obtained from the reference fracture toughness curve, defined in Appendix G to the ASME Code. The K_{IR} curve is given by the equation:

$$K_{IR} = 26.78 + 1.223 \exp [0.0145(T-RT_{NDT} + 160)] \qquad (1)$$

where K_{IR} is the reference stress intensity factor as a function of the metal temperature T and the metal nil ductility reference temperature RT_{NDT}. Thus, the governing equation for the heatup-cooldown analysis is defined in Appendix G of the ASME Code as follows:

$$C\ K_{IM} + K_{It} \leq K_{IR} \qquad (2)$$

Where, K_{IM} is the stress intensity factor caused by membrane (pressure) stress.

K_{It} is the stress intensity factor caused by the thermal gradients.

K_{IR} is provided by the code as a function of temperature relative to the RT_{NDT} of the material.

C = 2.0 for level A and B service limits, and

C = 1.5 for inservice hydrostatic and leak test operations.

At any time during the heautp or cooldown transient, K_{IR} is determined by the metal temperature at the tip of the postulated flaw, the appropriate value for RT_{NDT}, and the reference fracture toughness curve. The thermal stresses resulting from temperature gradients through the vessel wall are calculated and then the corresponding thermal stress intensity factor, K_{IT}, for the reference flaw is computed. From Equation (2) the pressure stress intensity factors are obtained and from these, the allowable pressures are calculated.

COOLDOWN

For the calculation of the allowable pressure versus coolant temperature during cooldown, the Code reference flaw is assumed to exist at the inside of the vessel wall. During cooldown, the controlling location of the flaw is always at the inside of the wall because the thermal gradients produce tensile stresses at the inside, which increase with increasing cooldown rates. Allowable pressure-temperature relations are generated for both steady-state and finite cooldown rate situations. From these relations composite limit curves are constructed for each cooldown rate of interest.

The use of the composite curve in the cooldown analysis is necessary because control of the cooldown procedure is based on measurement of reactor coolant temperature, whereas the limiting pressure is actually dependent on the material temperature at the tip of the assumed flaw. During cooldown, the 1/4T vessel location is at a higher temperature than the fluid adjacent to the vessel ID. This condition, of course, is not true for the steady-state situation. It follows that at any given reactor coolant temperature, the delta T developed during cooldown results in a higher value of K_{IR} at the 1/4T location for finite cooldown rates than for steady-state operation. Furthermore, if conditions exist such that the increase in K_{IR} exceeds K_{It}, the calculated allowable pressure during cooldown will be greater than the steady-state value.

The above procedures are needed because there is no direct control on temperature at the 1/4T location; therefore, allowable pressures may unknowingly be violated if the rate of cooling is decreased at various intervals along a cooldown ramp. The use of the composite curve eliminates this problem and assures conservative operation of the system for the entire cooldown period.

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 composite curves for the heatup rate data and the cooldown rate data are adjusted for possible errors in the pressure and temperature sensing instruments by the values indicated on the respective curves.

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.

3/4.4.10 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 10 CFR Part 50.55a(g) except where specific written relief has been granted by the Commission pursuant to 10 CFR Part 50.55a (g) (6) (i).

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 18 TO FACILITY OPERATING LICENSE NO. NPF-2

ALABAMA POWER COMPANY

JOSEPH M. FARLEY NUCLEAR PLANT, UNIT NO. 1

DOCKET NO. 50-348

Introduction

By letter dated March 28, 1980 (supplemented by letter dated January 5, 1981) in response to our letter dated December 12, 1980, Alabama Power Company (APCO) proposed Technical Specification changes. These changes related to a proposed new Rod Bow Penalty (RPB) curve and the associated Bases. Additionally, by letter dated January 9, 1981 (supplemented by letter dated January 23, 1981), APCO proposed new heatup and cooldown curves and bases. These new curves resulted from evaluation of a reactor vessel material specimen (Capsule Y) after 1.13 effective full power years of operation.

We have made minor changes to the APCO proposals. These changes have been discussed with the APCO staff who concur with our changes. Our discussion and evaluation are included herein.

Discussion and Evaluation

1. Rod Bow Penalty Curve
(Figure 3.2-3 and Bases)

An $F^N_{\Delta H}$ penalty due to fuel rod bowing was approved for Farley Unit No. 1 Technical Specifications in License Amendment No. 8. The penalty commences at a region-averaged burnup of 400 MWd/MtU and parabolically increases with additional exposure. By letters of March 28, 1980 and January 5, 1981, APCO proposed changes to Technical Specifications Figure 3.2-3 and associated bases. APCO proposed the elimination of the penalty because:

(a) the proposed use of a recently approved thermal-hydraulic model, and

(b) the application of a plant-specific margin available to offset DNBR reductions due to fuel rod bowing.

Subsequent to NRC approval of the present $F^N_{\Delta H}$ penalty, Westinghouse submitted 1/ test results on the effects of a bowed rod on critical heat flux. These results showed a significant reduction in the presupposed DNBR penalty associated

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1/ Letter from C. Eicheldinger, Westinghouse, to D. F. Ross, NRC, NS-CE-1580, October 24, 1977.

with a small gap (specifically, the gap was equivalent to 85% closure). Consequently, the NRC approved 2/ the use of a less conservative model for the reduction in DNBR versus gap closure for Westinghouse applications. By letter of March 28, 1980, APCO requested the use of this revised DNBR versus closure model for the Farley Unit No. 1 rod bowing penalty calculations. Since we had approved this model generically, the APCO proposal to use this revised DNBR versus closure model is acceptable.

Using the revised model and available generic thermal margins totaling 9.1% DNBR, APCO first requested a reduction in the DNBR penalty. This proposal was subsequently revised by letter of January 5, 1981 to account for a plant-specific thermal margin of 3% DNBR. The combination of the generic margin plus plant-specific margins are now sufficiently large to completely eliminate the reduction in $F^N_{\Delta H}$ due to fuel rod bowing.

We previously approved 3/ the 9.1% DNBR margin for fuel designs such as is used in Farley Unit No. 1 and find its use in this application to be acceptable. The 3% DNBR margin arises from the difference between the 1.3 DNBR safety limit and the minimum DNBR calculation for the loss-of-flow accident. This 3% margin was also previously approved for the Farley Unit No. 2 Technical Specifications.

Conclusion

Since the thermal-hydraulic analysis of both Farley Units are identical, the application of the margin to Unit No. 1 is acceptable as well. We, therefore, agree with APCO in concluding that there is no longer a need to perform rod bowing penalty calculations for Farley Unit No. 1. This conclusion, however, is premised upon the ability of APCO to maintain at least 2.1% of the present plant-specific 3% DNBR margin. In future cycles where the required 2.1% DNBR margin might not be available APCO should determine the magnitude of any residual $F^N_{\Delta H}$ penalty by use of the new Technical Specifications Figure 3.2-3. Figure 3.2-3 is not presently needed for Unit No. 1 operation although it is retained for the stated reason.

2. Heatup and Cooldown Curves
(Specification 3/4.4.9 and Bases 3/4.4.9)

Discussion and Evaluation

We have reviewed the Updated Reactor Vessel Heatup and Cooldown Curves for the Joseph M. Farley Nuclear Plant, Unit No. 1, submitted in APCO letter dated January 9, 1981.

2/ Letter from J. F. Stolz, NRC, to T. M. Anderson, Westinghouse, Subject: Staff Review of WCAP-8691, dated April 5, 1979.

3/ Memorandum from D. F. Ross and D. G. Eisenhut, NRC to D. B. Vassallo and K. R. Goller, "Interim Safety Evaluation Report on the Effects of Fuel Rod Bowing on Thermal Margin Calculations for Light Water Reactors," dated December 8, 1976.

As part of our review, we have calculated the shift in the nil-ductibility transition temperature of the controlling material in the beltline region of the reactor vessel (the intermediate to lower shell seam weld) to be 192°F after 7.7 EFPY operation. This is compared to an indicated shift of 185°F in the proposed updated curves for the Technical Specifications. At the 1/4 t location, this difference in shift temperature corresponds to the fluence received by the vessel in 0.6 EFPY operation.

Therefore, the Updated Reactor Vessel Heatup and Cooldown Curves should reflect the calculated decrease in fluence. Thus, we consider that the limitations should be applicable for 7 EFPY instead of 7.7 EFPY proposed. Specification 4.4.9 and bases have been modified accordingly.

Conclusion

With the changes which we have made, we conclude that the Pressure-Temperature curves are acceptable and will be in conformance with Appendix G requirements.

Environmental Consideration

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) 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.

Date: February 13, 1981

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UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-348

ALABAMA POWER COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 18 to Facility Operating License No. NPR-2 issued to Alabama Power Company (the licensee), which revised Technical Specifications for operation of the Joseph M. Farley Nuclear Plant, Unit No. 1 (the facility) located in Houston County, Alabama. The amendment is effective as of the date of issuance.

The amendment provides the following:

1. A new rod bow penalty curve and bases, and
2. New heatup and cooldown curves and bases. These are based on the Capsule Y analyses after 1.13 effective full power years of operation.

The applications for the amendment comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment was not required since this amendment does not involve a significant hazards consideration.

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The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the applications for amendment dated March 28, 1980 (supplemented January 5, 1981) and January 9, 1981 (supplemented January 23, 1981), (2) Amendment No. 18 to License No. NPF-2 and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C. and at the George S. Houston Memorial Library 212 W. Berdeshaw Street, Dothan, Alabama 36303. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland, this 13 day of February, 1981.

FOR THE NUCLEAR REGUALTORY COMMISSION

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing