May 2, 1985

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

Dear Mr. McDonald:

The Commission has issued the enclosed Amendment No. 58 to Facility Operating License 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 April 20, 1984.

The amendment modifies Technical Specifications (Table 4.4-5 and Figures 3.4-2 and 3.4-3) relating to the reactor vessel material surveillance schedule and heatup and cooldown curves extending the curves to seven effective full power years of operation. These changes are based on WCAP-10474 provided by licensee letter dated March 1, 1984 and are in accordance with 10 CFR 50, Appendix H, published May 31, 1983.

Additionally, page 6-25 is deleted as an administrative correction to our error in instructions for License Amendment No. 57 dated February 19, 1985. Page 6-25 is identical to page 6-27, which we deleted by License Amendment No. 33 dated October 14, 1983. We regret any inconvenience the error may have caused.

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

Sincerely,

/s/EAReeves

Edward A. Reeves, Project Manager
Operating Reactors Branch #1
Division of Licensing

Enclosures:
1. Amendment No. 58 to NPF-2
3. Safety Evaluation

cc: w/enclosures:
See next page

ORB#1:DL	ORB#1:DL	BC-ORB#1:DL	OELD	AD/OR:DL
CParrish	EReeves/ts	SVarga	W.D. Paton	GLainas
04/11/85	04/17/85	04/22/85	04/24/85	04/29/85

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

May 2, 1985

Docket No. 50-348

Mr. R. P. McDonald
Senior Vice President
Alabama Power Company
Post Office Box 2641
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Edward A. Reeves

Edward A. Reeves, Project Manager
Operating Reactors Branch #1
Division of Licensing

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1. Amendment No. 58 to NPF-2
3. Safety Evaluation

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. 58
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 April 20, 1984, 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:

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PDR ADOCK 05000348
P PDR

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 58 , 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

Steven A. Varga

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: May 2, 1985

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 58 FACILITY OPERATING LICENSE NO. NPF-2

DOCKET NO. 50-348

Revised Appendix A as follows:

Remove Pages	Insert Pages
3/4 4-28	3/4 4-28
3/4 4-29	3/4 4-29
3/4 4-30	3/4 4-30
B 3/4 4-7	B 3/4 4-7
B 3/4 4-8	B 3/4 4-8
B 3/4 4-9	B 3/4 4-9
6-25	None*

* Page was page 6-27 but was removed by Amendment No. 33 dated October 14, 1983.

TABLE 4.4-5

REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM-WITHDRAWAL SCHEDULE

CAPSULE	VESSEL LOCATION	LEAD FACTOR	WITHDRAWAL TIME
Y	343°	3.12	Removed 1.13 EFPY
U	107°	3.12	Removed 3.02 EFPY
X	287°	3.12	6 EFPY
W	110°	2.70	12 EFPY
V	290°	2.70	21 EFPY
Z	340°	2.70	Standby

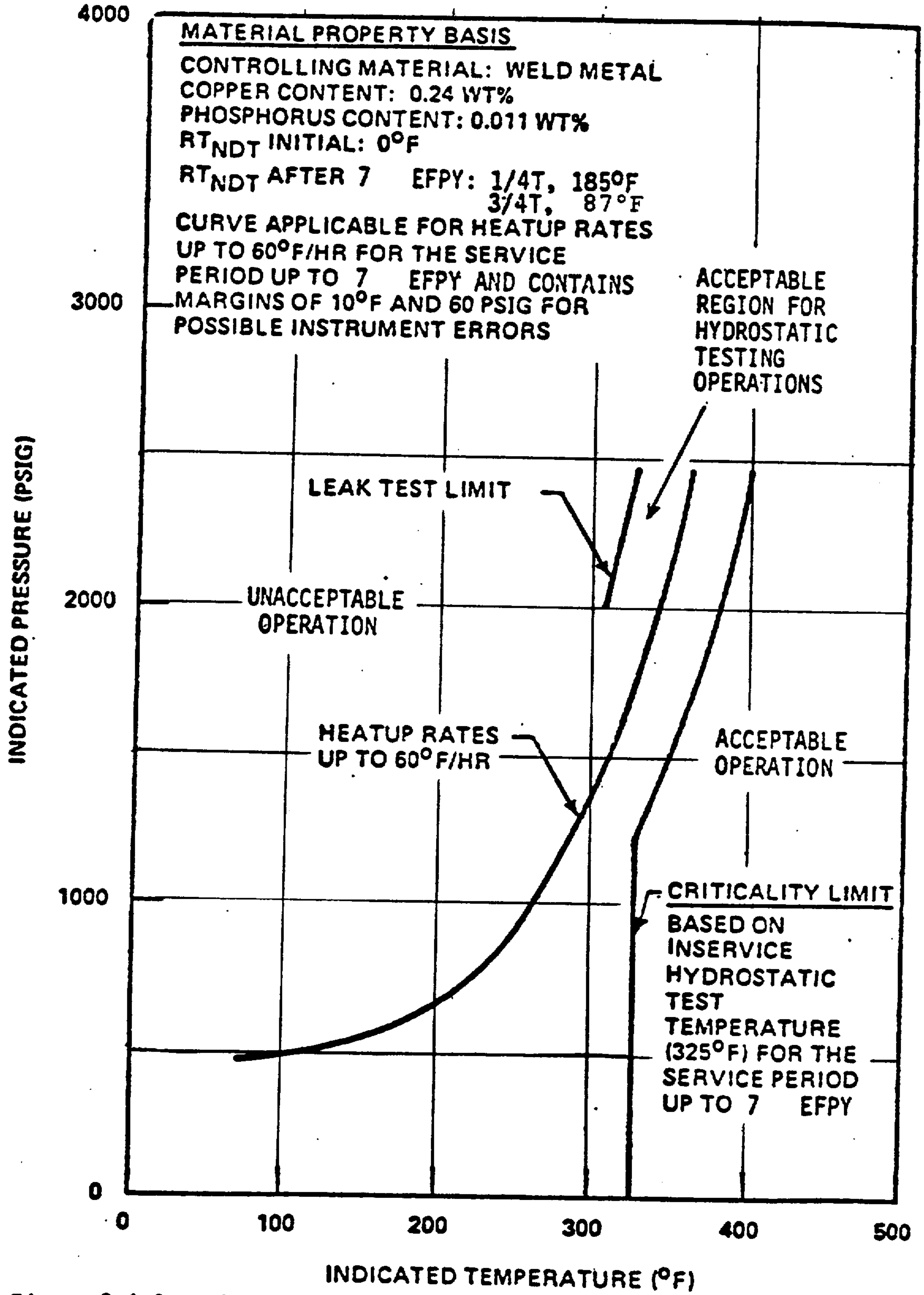

Figure 3.4-2 Farley Unit 1 Reactor Coolant System Heatup Limitations Applicable For The First 7 EFPY

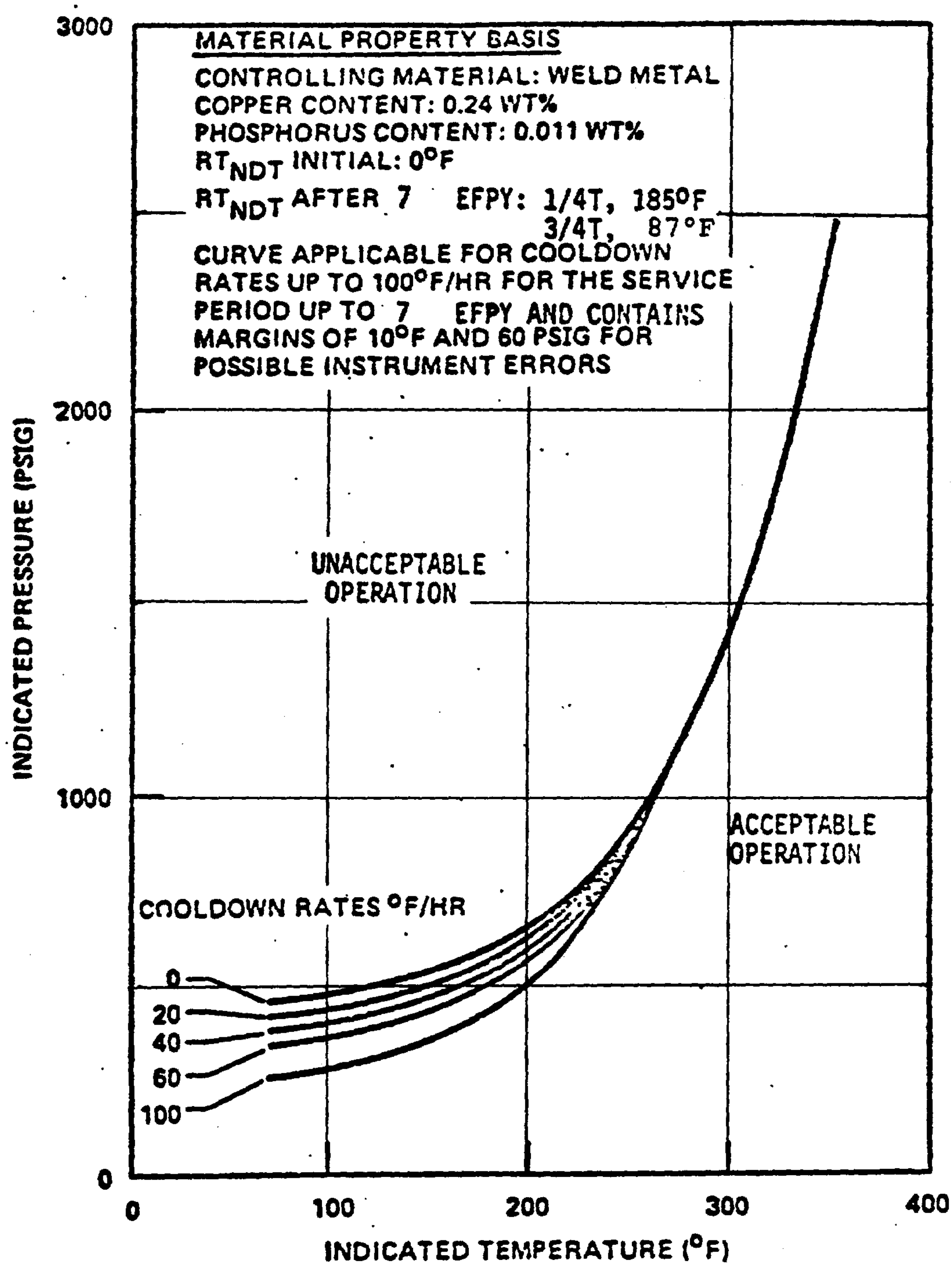

Figure 3.4-3 Farley Unit 1 Reactor Coolant System Cooldown Limitations Applicable For The First 7 EFPY

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 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 E185-82, are available. Capsules will be removed in accordance with the requirements of ASTM E185-82 and 10CFR50, 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 10CFR 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 B 3/4.4-1

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.1	-	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. lo lower shell weld seam			0.24	0.011	0(a)	-	<60	0	-	-
Lower shell long. weld seam			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

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 58 TO FACILITY OPERATING LICENSE NO. NPF-2

ALABAMA POWER COMPANY

JOSEPH M. FARLEY NUCLEAR PLANT, UNIT NO. 1

DOCKET NO. 50-348

INTRODUCTION

In letters from F. L. Clayton, Jr. to S. A. Varga dated March 1, 1984 and April 20, 1984 the Alabama Power Company submitted reactor vessel material surveillance test data and changes to the Farley Unit 1 Technical Specifications, respectively. The reactor vessel material surveillance test data were detailed in WCAP-10474, "Analysis of Capsule U from the Alabama Power Company Joseph M. Farley Unit 1 Reactor Vessel Radiation Surveillance Program." The proposed technical specifications were changes to the pressure-temperature limits and the reactor vessel material surveillance program withdrawal schedule.

DISCUSSION AND EVALUATION

Pressure-temperature limits must be calculated in accordance with the requirements of Appendix G, 10 CFR 50, which became effective on July 26, 1983. Pressure-temperature limits that are calculated in accordance with the requirements of Appendix G, 10 CFR 50 are dependent upon the initial RT_{NDT} for the limiting materials in the beltline and closure flange regions of the reactor vessel and the increase in RT_{NDT} resulting from neutron irradiation damage to the limiting beltline material.

The Farley Unit 1 reactor vessel was procured to ASME Code requirements, which did not specify fracture toughness testing to determine RT_{NDT} for each reactor vessel material. Hence, the initial RT_{NDT} for some materials in the closure flange and beltline regions of the reactor vessel could not be determined in accordance with the test requirements of the ASME Code. The RT_{NDT} for these materials were estimated using the method recommended by the staff in Section 5.3.2 of the NRC Standard Review Plan. The initial RT_{NDT} values for the limiting materials in the beltline and closure flange regions are 0°F and 60°F, respectively.

The increase in RT_{NDT} resulting from neutron irradiation damage was estimated by the licensee using the method recommended in Regulatory Guide 1.99, Rev. 1, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials." In Table I we have compared the increase in RT_{NDT} predicted by the regulatory guide with that measured from the surveillance material, which was reported in WCAP-10474. The method of predicting neutron irradiation damage in Reg. Guide 1.99, Rev. 1 provides conservative estimates, because the increase in RT_{NDT} predicted by the regulatory guide exceeds that from the surveillance material.

The amount of time that pressure-temperature limits are effective depends upon the amount of neutron irradiation damage. Utilizing the method recommended in Regulatory Guide 1.99, Rev. 1 to predict the neutron irradiation damage, the neutron fluence estimates in Technical Specification Figure B 3/4.4-1 and initial RT_{NDT} values for the limiting materials in the beltline and closure flange region of 0°F and 60°F, respectively, we have determined that the proposed pressure-temperature limits are acceptable for 7 effective full power years.

Appendix H, 10 CFR 50 contains the regulatory requirements for a reactor vessel materials surveillance program. Appendix H requires that the proposed withdrawal schedule be approved prior to implementation and references ASTM E 185-82, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessel." For the Farley Unit 1 reactor vessel, ASTM E 185-82 recommends that there be minimum of 5 capsules and that these capsules be withdrawn when the neutron fluence received by the capsules corresponds to the amount identified in Table I of ASTM E 185-82. This table recommends that the capsules be withdrawn at various neutron fluences throughout the plant's life and that the fifth (last) capsule be withdrawn at a neutron fluence not less than once or greater than twice the peak end-of-life (EOL) vessel fluence.

The peak neutron fluence to be received by the Farley Unit 1 reactor vessel is estimated at 6.4 X 10^{19} n/cm^2 (E>1MeV). We have compared the expected neutron fluence to be received by each capsule to that required by ASTM E 185-82 and conclude that the withdrawal schedule for the Farley Unit 1 capsules meets the intent of ASTM E 185-82. Hence, we consider acceptable the proposed revision to the Farley Unit 1 Technical Specification.

TABLE I

COMPARISON OF REG. GUIDE 1.99 PREDICTION MODEL
AND FARLEY UNIT 1 SURVEILLANCE TEST DATA
FROM WCAP 10474

Material	Increase in Reference Temperature, ΔRT_{NDT} (F°) From Surveillance Test Data	Predicted by Reg. Guide 1.99, Rev. 1
Plate 6919-1 (transverse)	90	173
Plate 6919-1 (longitudinal)	105	173
Weld Metal	80	180

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 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.

Dated: May 2, 1985

Principal Contributor:

B. J. Elliot