December 24, 1992

Docket No. 50-483

Mr. Donald F. Schnell
Senior Vice President - Nuclear
Union Electric Company
Post Office Box 149
St. Louis, Missouri 63166

Dear Mr. Schnell:

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Wanda Jones

SUBJECT: AMENDMENT NO. 76 TO FACILITY OPERATING LICENSE NO. NPF-30 (TAC NO. M82656)

The Commission has issued the enclosed Amendment No.76 to Facility Operating License No. NPF-30 for the Callaway Plant, Unit 1. This amendment revises the Technical Specifications (TS) in response to your application dated December 18, 1991.

The amendment revises Technical Specification Section 4.4.9.1.2, Figures 3.4-2, 3.4-3, and 3.4-4, Tables 4.4-5 and B 3/4.4-1 and associated Bases for Reactor Coolant System Pressure/Temperature Limits by modifying plant heatup and cooldown curves and the maximum allowable PORV setpoint curve for cold overpressure protection.

A copy of the Safety Evaluation is also enclosed. The notice of issuance will be included in the Commission's next biweekly Federal Register notice.

Sincerely,

original signed by

L. Raynard Wharton, Project Manager
Project Directorate III-3
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No.76 to License No. NPF-30
2. Safety Evaluation

cc: See next page

LA/PD33	PM/PD33	PD/PD33	OGC
PKreutzer	RWharton	JHannon	
12/1/92	12/8/92	12/11/92	12/18/92

OFFICIAL RECORD
DOCUMENT NAME: g:\callaway\CAL82656.AMD

DF01

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

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Sincerely,

L. Raynard Wharton

L. Raynard Wharton, Project Manager
Project Directorate III-3
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No.76 to
   License No. NPF-30
2. Safety Evaluation

cc: See next page

Mr. D. F. Schnell
Union Electric Company

Callaway Plant
Unit No. 1

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

UNION ELECTRIC COMPANY

CALLAWAY PLANT, UNIT 1

DOCKET NO. 50-483

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 76
License No. NPF-30

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

   A. The application for amendment filed by Union Electric Company (UE, the licensee) dated December 18, 1991, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

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

   D. The issuance of this 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-30 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 76 , and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into the license. UE shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance. The Technical Specifications are to be implemented within 30 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Anthony T. Gody, Jr. for

John N. Hannon, Director
Project Directorate III-3
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of issuance: December 24, 1992

ATTACHMENT TO LICENSE AMENDMENT NO. 76

OPERATING LICENSE NO. NPF-30

DOCKET NO. 50-483

Revise Appendix A Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages are identified by the captioned amendment number and contain marginal lines indicating the area of change. The corresponding overleaf pages are provided to maintain document completeness.

REMOVE	INSERT
3/4 4-29	3/4 4-29
3/4 4-30	3/4 4-30
3/4 4-31	3/4 4-31
3/4 4-32	
3/4 4-36	3/4 4-36
B 3/4 4-7	B 3/4 4-7
B 3/4 4-8	B 3/4 4-8
B 3/4 4-11	B 3/4 4-11
B 3/4 4-16	B 3/4 4-16

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 1-hour period,

b. A maximum cooldown of 100°F in any 1-hour period, and

c. A maximum temperature change of less than or equal to 10°F in any 1-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 to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System; determine that the Reactor Coolant System 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

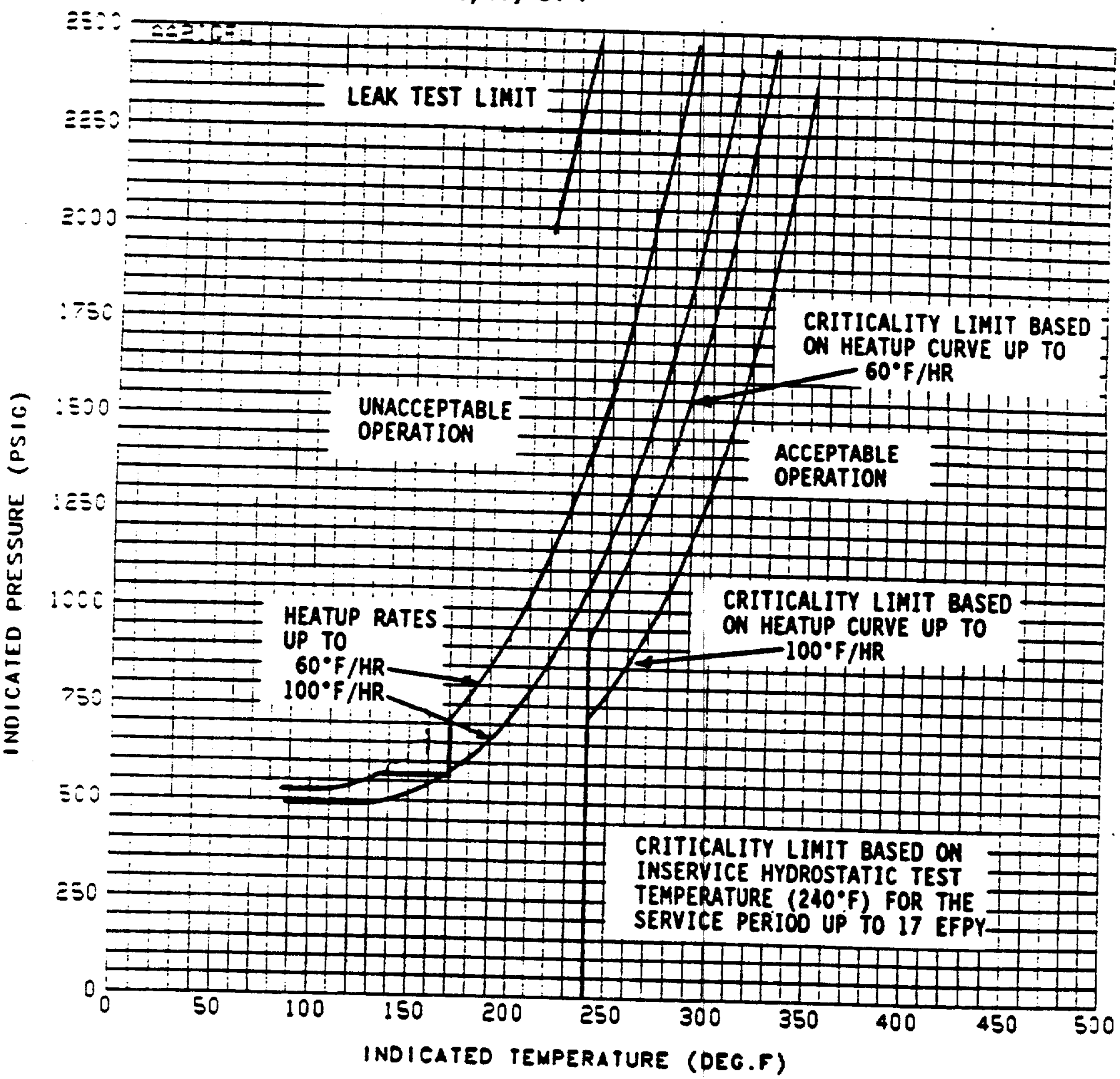
4.4.9.1.1 The Reactor Coolant System temperature and pressure shall be determined to be within the limits at least once per 30 minutes during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

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 Part 50, Appendix H. The results of these examinations shall be used to update Figures 3.4-2, 3.4-3, and 3.4-4.

Material Property Basis
1/4T Limiting Material: Plate, R2708-3
Copper Content: 0.07 wt. %
Nickel Content: 0.59 wt. %
Initial RT_{NDT}: 20°F

3/4T Limiting Material: Plate, R2708-1
Copper Content: 0.07 wt. %
Nickel Content: 0.59 wt. %
Initial RT_{NDT}: 50°F

Limiting ART after 17 EFPY: 1/4T, 95°F
3/4T, 84°F



Callaway Unit 1 Reactor Coolant System Heatup Limitations (Heat up rates up to 60°F/hr and 100°F/hr) Applicable for the First 17 EFPY (With Margins 10°F and 60 psig For Instrumentation Errors)

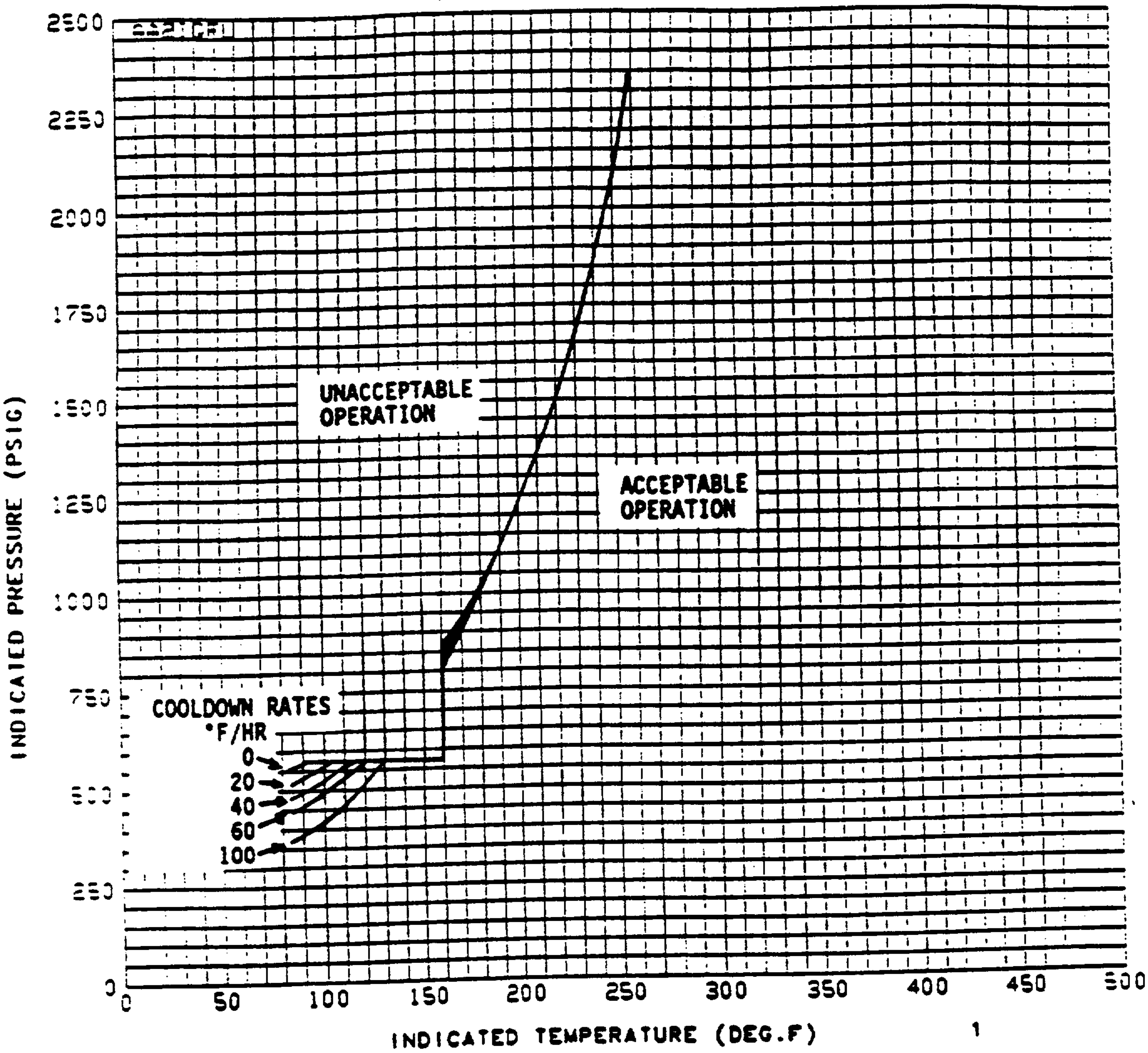
FIGURE 3.4-2

Material Property Basis

1/4T Limiting Material: Plate, R2708-3
Copper Content: 0.07 wt. %
Nickel Content: 0.59 wt. %
Initial RT_{NDT}: 20°F

3/4T Limiting Material: Plate, R2708-1
Copper Content: 0.07 wt. %
Nickel Content: 0.59 wt. %
Initial RT_{NDT}: 50°F

Limiting ART after 17 EFPY: 1/4T, 95°F
3/4T, 84°F



SCP COOLDOWN CURVES REG. GUIDE 1.99,REV.2 WITH MARGIN

Callaway Unit 1 Reactor Coolant System Cooldown (Cooldown rates up to 100°F/hr) Limitations Applicable for the First 17 EFPY (With Margins 10°F and 60 psig For Instrumentation Errors)

FIGURE 3.4-3

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SURVEILLANCE REQUIREMENTS

4.4.9.3.1 Each PORV shall be demonstrated OPERABLE by:

a. Performance of an ANALOG CHANNEL OPERATIONAL TEST on the PORV actuation channel, but excluding valve operation, within 31 days prior to entering a condition in which the PORV is required OPERABLE and at least once per 31 days thereafter when the PORV is required OPERABLE;

b. Performance of a CHANNEL CALIBRATION on the PORV actuation channel at least once per 18 months; and

c. Verifying the PORV isolation valve is open at least once per 72 hours when the PORV is being used for overpressure protection.

4.4.9.3.2 Each RHR suction relief valve shall be demonstrated OPERABLE when the RHR suction relief valves are being used for cold overpressure protection as follows:

a. For RHR suction relief valve 8708B:

By verifying at least once per 72 hours that RHR RCS suction isolation valves (RRSIV) EJ-HV-8701B and BB-PV-8702B are open.

b. For RHR suction relief valve 8708A:

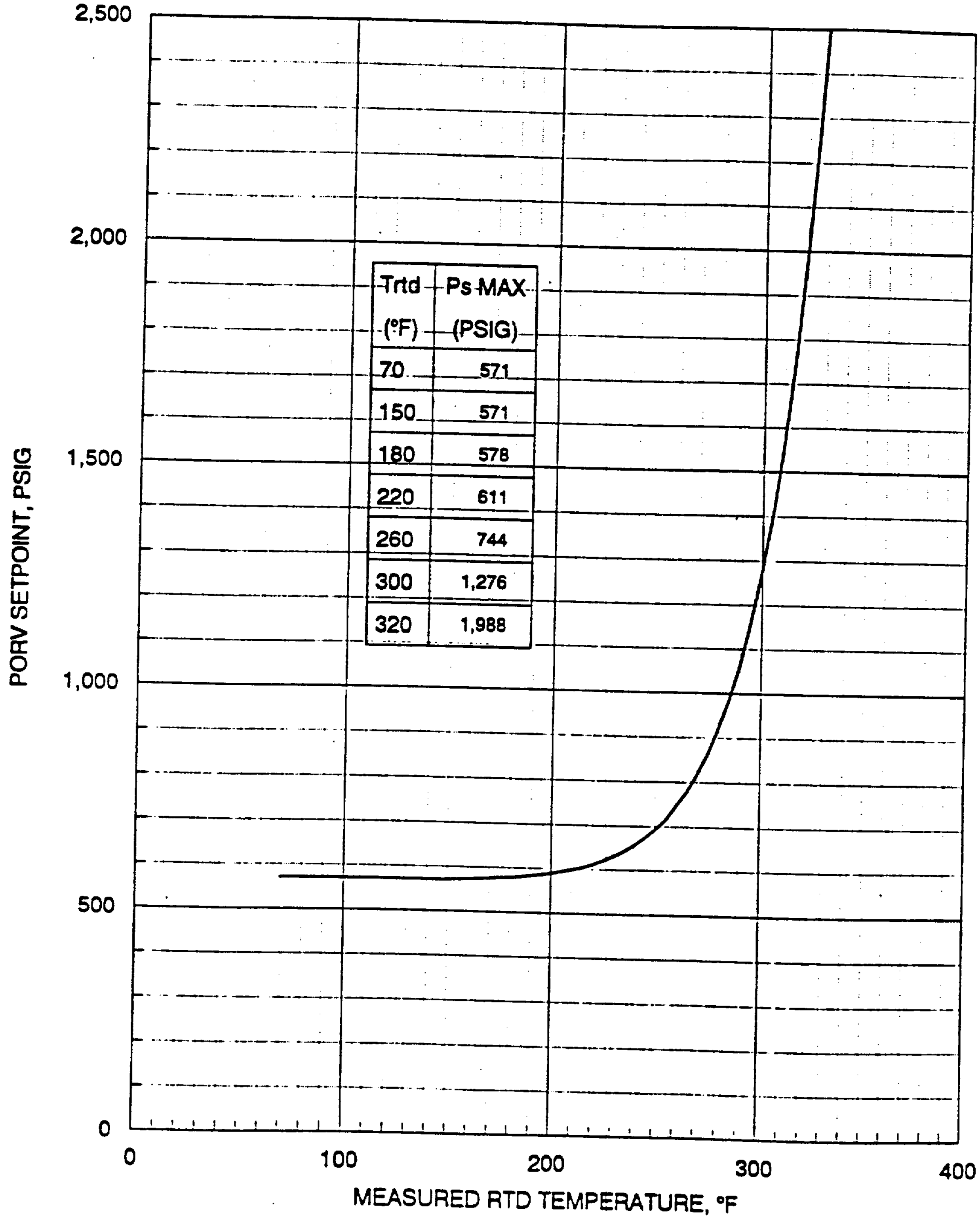
By verifying at least once per 72 hours that RRSIV EJ-HV-8701A and BB-PV-8702A are open.

c. Testing pursuant to Specification 4.0.5.

4.4.9.3.3 The RCS vent(s) shall be verified to be open at least once per 12 hours* when the vent(s) is being used for overpressure protection.

*Except when the vent pathway is provided with a valve which is locked, sealed, or otherwise secured in the open position, then verify these valves open at least once per 31 days.

FIGURE 3.4-4
MAXIMUM ALLOWED PORV SETPOINT
FOR THE COLD OVERPRESSURE MITIGATION SYSTEM



Trtd (°F)	Ps MAX (PSIG)
70	571
150	571
180	578
220	611
260	744
300	1,276
320	1,988

PRESSURE/TEMPERATURE LIMITS (Continued)

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/h and 200°F/h, respectively. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 583°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 the 1972 Winter 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 17 effective full power years (EFPY) of service life. The 17 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 and phosphorus content of the material in question, can be predicted using Figure B 3/4.4-1 and the largest value of ΔRT_{NDT} computed by either Regulatory Guide 1.99, Revision 2, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials," or the Westinghouse Copper Trend Curves shown in Figure B 3/4.4-2. 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 17 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 Part 50, Appendix H. The lead factor represents the

PRESSURE/TEMPERATURE LIMITS (Continued)

relationship between the fast neutron flux density at the location of the capsule and the inner wall of the reactor vessel. Therefore, the results obtained from the surveillance specimens can be used to predict the future radiation damage to the reactor vessel material by using the lead factor and the withdrawal time of the capsule. 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 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 nonductile 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.

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,

TABLE B 3/4.4-1 (Continued)

REACTOR VESSEL TOUGHNESS

COMPONENT	COMP CODE	ASME MATERIAL TYPE	CU (%)	P (%)	T_{NDT} (°F)	50 FT-LB 35 Mil TEMP (°F)	RT_{NDT} (°F)	AVG. UPPER SHELF NMWD* (FT-LB)	AVG. UPPER SHELF MWD** (FT-LB)
Bottom Head Torus	R2714-1	A533B, CL.1	0.15	0.010	-20	40	-20	139	----
Bottom Head Dome	R2715-1	A533B, CL.1	0.17	0.011	-40	20	-40	152	----
Inter. & Lower Shell Long. Weld Seams	G2.03	SAW	0.04	0.008	-60	<0	-60	143	----
Inter. to Lower Shell Girth Weld Seams	E3.14	SAW	0.04	0.006	-60	<0	-60	112	----

TABLE NOTATIONS

*NMWD - Normal to Major Working Directions
**MWD - Major Working Directions

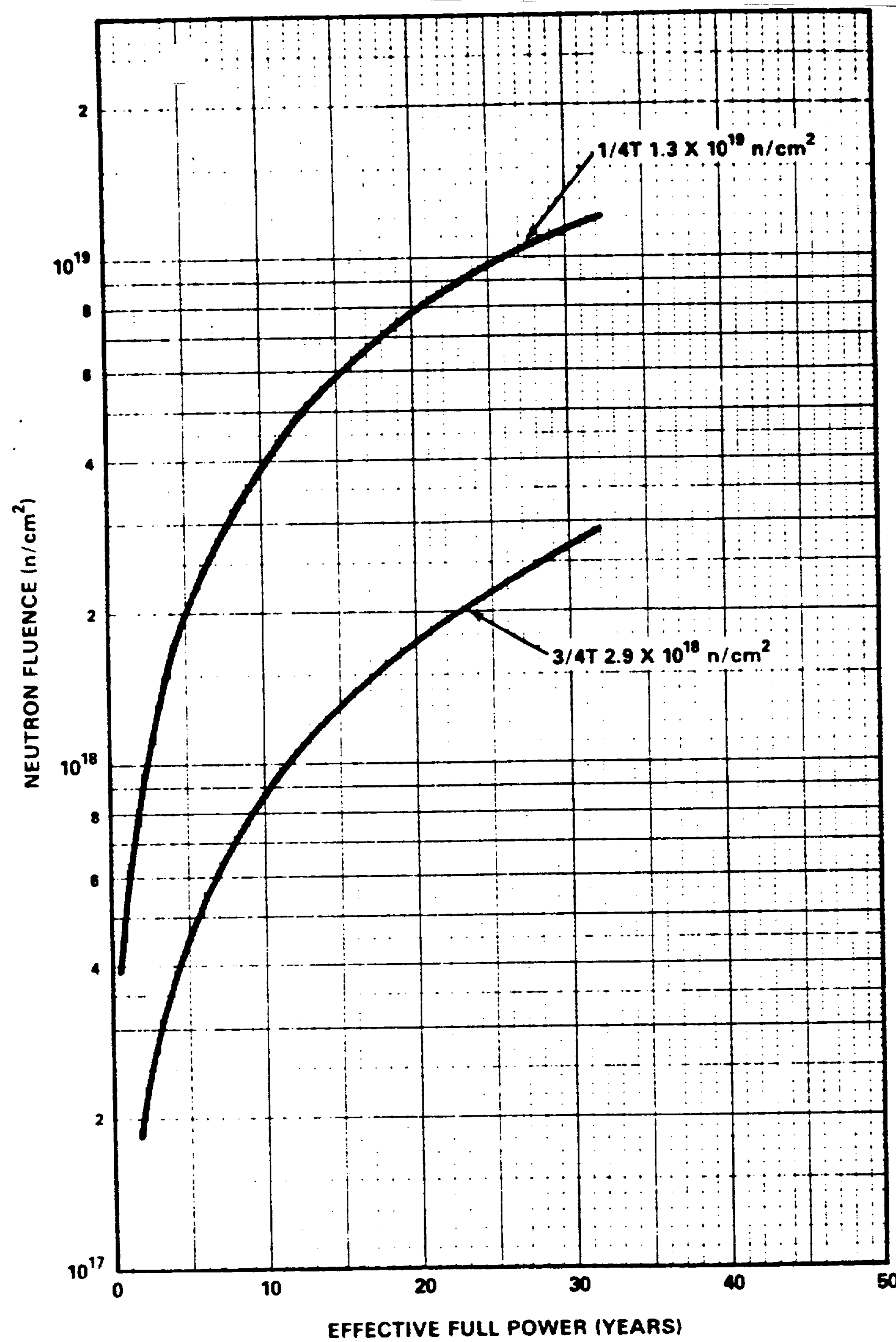


FIGURE B3/4.4-1
FAST NEUTRON FLUENCE (E > 1 MeV) AS A FUNCTION OF
FULL POWER SERVICE LIFE

COLD OVERPRESSURE (Continued)

possible by the geometrical relationship of the RHR suction line and the RCS wide range temperature indicator used for COMS; 3) instrument uncertainties; and 4) single failure. To ensure mass and heat input transients more severe than those assumed cannot occur, technical specifications require lockout of both safety injection pumps and all but one centrifugal charging pump while in MODES 4, 5 and 6 with the reactor vessel head installed and disallow start of an RCP if secondary temperature is more than 50°F above primary temperature. Exceptions to these mode requirements are acceptable as described below.

Operation above 350°F but less than 375°F with only one centifugal charging pump OPERABLE and no safety injection pumps OPERABLE is allowed for up to 4 hours. As shown by analysis LOCA's occurring at low temperature, low pressure conditions can be successfully mitigated by the operation of a single centrifugal charging pump and a single RHR pump with no credit for accumulator injection. Given the short time duration that the condition of having only one centrifugal charging pump OPERABLE is allowed and the probability of a LOCA occurring during this time, the failure of the single centrifugal charging pump is not assumed.

Operation below 350°F but greater than 325°F with all centrifugal charging and safety injection pumps OPERABLE is allowed for up to 4 hours. During low pressure, low temperature operation all automatic safety injection actuation signals except Containment Pressure - High are blocked. In normal conditions a single failure of the ESF actuation circuitry will result in the starting of at most one train of safety injection (one centrifugal charging pump, and one safety injection pump). For temperatures above 325°F, an overpressure event occurring as a result of starting two pumps can be successfully mitigated by operation of both PORV's without exceeding Appendix G limit. Given the short time duration that this condition is allowed and the low probability of a single failure causing an overpressure event during this time, the single failure of a PORV is not assumed. Initiation of both trains of safety injection during this 4-hour time frame due to operator error or a single failure occurring during testing of a redundant channel are not considered to be credible accidents.

Although COMS is required to be OPERABLE when RCS temperature is less than 368°F, operation with all centrifugal charging pumps and both safety injection pumps OPERABLE is acceptable when RCS temperature is greater than 350°F. Should an inadvertent safety injection occur above 350°F, a single PORV has sufficient capacity to relieve the combined flow rate of all pumps. Above 350°F, two RCP and all pressurizer safety valves are required to be OPERABLE. Operation of an RCP eliminates the possibility of a 50°F difference existing between indicated and actual RCS temperature as a result of heat transport effects. Considering instrument uncertainties only, an indicated RCS temperature of 350°F is sufficiently high to allow full RCS pressurization in accordance with Appendix G limitations. Should an overpressure event occur in these conditions, the pressurizer safety valves provide acceptable and redundant overpressure protection.

The Maximum Allowed PORV setpoint for the Cold Overpressure Mitigation System will be updated based on the results of examinations of reactor vessel material irradiation surveillance specimens performed as required by 10 CFR Part 50, Appendix H.

HEATUP (Continued)

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 nonductile 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 PORVs, or two RHR suction relief valves, or an RCS vent opening of at least 2 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 368°F. Either PORV or either RHR suction 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 a centrifugal charging pump and its injection into a water-solid RCS.

In addition to opening RCS vents to meet the requirement of Specification 3.4.9.3c., it is acceptable to remove a pressurizer Code safety valve, open a PORV block valve and remove power from the valve operator in conjunction with disassembly of a PORV and removal of its internals, or otherwise open the RCS.

COLD OVERPRESSURE

The Maximum Allowed PORV Setpoint for the Cold Overpressure Mitigation System (COMS) is derived by analysis which models the performance of the COMS assuming various mass input and heat input transients. Operation with a PORV setpoint less than or equal to the maximum setpoint ensures that Appendix G criteria will not be violated with consideration for 1) a maximum pressure overshoot beyond the PORV setpoint which can occur as a result of time delays in signal processing and valve opening; 2) a 50°F heat transport effect made

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 76 TO FACILITY OPERATING LICENSE NO. NPF-30

UNION ELECTRIC COMPANY

CALLAWAY PLANT, UNIT 1

DOCKET NO. 50-483

1.0 INTRODUCTION

By letter dated December 18, 1991, Union Electric Company (the licensee) requested changes to the pressure/temperature (P/T) limits in the Callaway Plant, Unit 1 Technical Specifications, Section 3.4. The proposed P/T limits are valid for 17 effective full power years (EFPY) and were developed using Regulatory Guide (RG) 1.99, "Radiation Embrittlement of Reactor Vessel Materials," Revision 2. Generic Letter 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Effect on Plant Operations," recommends RG 1.99, Rev. 2, be used in calculating P/T limits, unless the use of different methods can be justified. The P/T limits provide for the operation of the reactor coolant system during heatup, cooldown, criticality, and hydrotest.

To evaluate the P/T limits, the staff used the following NRC regulations and guidance: Appendices G and H of 10 CFR Part 50; the ASTM Standards and the ASME Code, which are referenced in Appendices G and H; 10 CFR 50.36(c)(2); RG 1.99, Rev. 2; Standard Review Plan (SRP) Section 5.3.2; and Generic Letter 88-11.

Background

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

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

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

2.0 EVALUATION

The staff evaluated the effect of neutron irradiation embrittlement on each beltline material in the Callaway 1 reactor vessel. The amount of irradiation embrittlement was calculated in accordance with RG 1.99, Rev. 2. The staff has determined that the material with the highest ART at 17 EFPY at 1/4T (T = reactor vessel beltline thickness) was the lower shell plate R2708-3 with 0.07% Cu, 0.59% Ni, and an initial RT_{ndt} of 20°F. The material with the highest ART at 17 EFPY at 3/4T was the lower shell plate R2708-1 with 0.07% copper (Cu), 0.59% nickel (Ni), and an initial RT_{ndt} of 50°F.

The licensee has removed two surveillance capsules from Callaway 1. The results from capsules U and Y in Unit 1 were published in Westinghouse reports WCAP-11374 and WCAP-12946, respectively. All surveillance capsules contained Charpy impact specimens and tensile specimens made from base metal, weld metal, and HAZ metal.

For the limiting beltline material at 1/4T, lower shell plate R2708-3, the staff calculated the ART to be 94.6°F at 1/4T at 17 EFPY. The ART for 1/4T was determined by Section 1 of RG 1.99, Rev. 2, because plate R2708-3 is not in the surveillance capsules.

For the limiting beltline material at 3/4T, lower shell plate R2708-1, the staff calculated the ART to be 83.8°F at 17 EFPY. The staff used a neutron fluence of 7.62E18 n/cm^2 at 1/4T and 2.74E18 n/cm^2 at 3/4T. The ART for 3/4T was determined by the least squares extrapolation method using the Callaway 1 surveillance data. The least squares method is described in Section 2.1 of RG 1.99, Rev. 2.

The licensee used the method in RG 1.99, Rev. 2, to calculate an ART of 95°F at 17 EFPY at 1/4T for the same limiting plate material. The staff determines that the licensee's ART of 95°F is more conservative than the staff's ART of 94.6°F, and it is acceptable. Substituting the ART of 95°F into equations in SRP 5.3.2, the staff verified that the proposed P/T limits for heatup, cooldown, and hydrotest meet the beltline material requirements in Appendix G of 10 CFR Part 50.

In addition to beltline materials, Appendix G of 10 CFR Part 50 also imposes P/T limits based on the reference temperature for the reactor vessel closure flange materials. Section IV.2 of Appendix G states that when the pressure exceeds 20% of the preservice system hydrostatic test pressure, the temperature of the closure flange regions highly stressed by the bolt preload must exceed the reference temperature of the material in those regions by at least 120°F for normal operation and by 90°F for hydrostatic pressure tests and leak tests. Based on the flange reference temperature of 40°F, the staff has determined that the proposed P/T limits satisfy Section IV.2 of Appendix G.

Section IV.B of Appendix G requires that the predicted Charpy USE at end of life be above 50 ft-lb. The material with the lowest predicted EOL USE is intermediate shell plate R2707-1 with a unirradiated USE of 78 ft-lb. Using Figure 2 of RG 1.99, Rev. 2, the staff calculated that the EOL USE at 1/4T for this material is 61.9 ft-lb. This is greater than 50 ft-lb and, therefore, is acceptable.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Missouri State official was notified of the proposed issuance of the amendment. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

This amendment involves a change to a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 or a change to a surveillance requirement. 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 (57 FR 28206). Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

5.0 CONCLUSION

The staff has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

6.0 REFERENCES

1. Regulatory Guide 1.99, Radiation Embrittlement of Reactor Vessel Materials, Revision 2, May 1988

2. NUREG-0800, Standard Review Plan, Section 5.3.2: Pressure-Temperature Limits

3. December 18, 1991, letter from D. F. Schnell (UECo) to USNRC Document Control Desk, subject: Callaway Plant Revision to Technical Specification 3/4.4.9 Pressure/Temperature Limits

4. R. G. Lott, et al., Analysis of Capsule U from the Union Electric Company Callaway Unit 1 Reactor Vessel Radiation Surveillance Program, WCAP-11374, Westinghouse Electric Corporation, June 1987

5. E. Terek, et al., Analysis of Capsule Y from the Union Electric Company Callaway Unit 1 Reactor Vessel Radiation Surveillance Program, WCAP-12946, Westinghouse Electric Corporation, June 1991

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