May 24, 1988

Docket No. 50-483

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

Dear Mr. Schnell:

The Commission has issued the enclosed Amendment No. 36 to Facility Operating License No. NPF-30 for the Callaway Plant, Unit 1. This amendment revises the Technical Specifications in response to your application dated July 31, 1987, as supplemented by letter dated February 19, 1988.

The amendment revises the plant heatup and cooldown curves, revises the maximum allowable power operated relief valve (PORV) setpoint curve, and revises the reactor vessel surveillance capsule removal schedule.

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,

/S/

Thomas W. Alexion, Project Manager
Project Directorate III-3
Division of Reactor Projects - III,
IV, V and Special Projects

Enclosures:
1. Amendment No. 36 to License No. NPF-30
2. Safety Evaluation

cc w/enclosures:
See next page

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w/change to FRN

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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. STN 50-483

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 36
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 July 31, 1987, supplemented February 19, 1988, 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.36 , 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 the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Albert W. DeAgazio for

Kenneth E. Perkins, Director
Project Directorate III-3
Division of Reactor Projects - III,
IV, V and Special Projects

Attachment:
Changes to the Technical
Specifications

Date of Issuance: May 24, 1988

ATTACHMENT TO LICENSE AMENDMENT NO. 36

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. Corresponding overleaf pages are provided to maintain document completeness.

REMOVE	INSERT
3/4 4-30	3/4 4-30
3/4 4-31	3/4 4-31
3/4 4-32	3/4 4-32
3/4 4-36	3/4 4-36
B 3/4 4-7	B 3/4 4-7

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

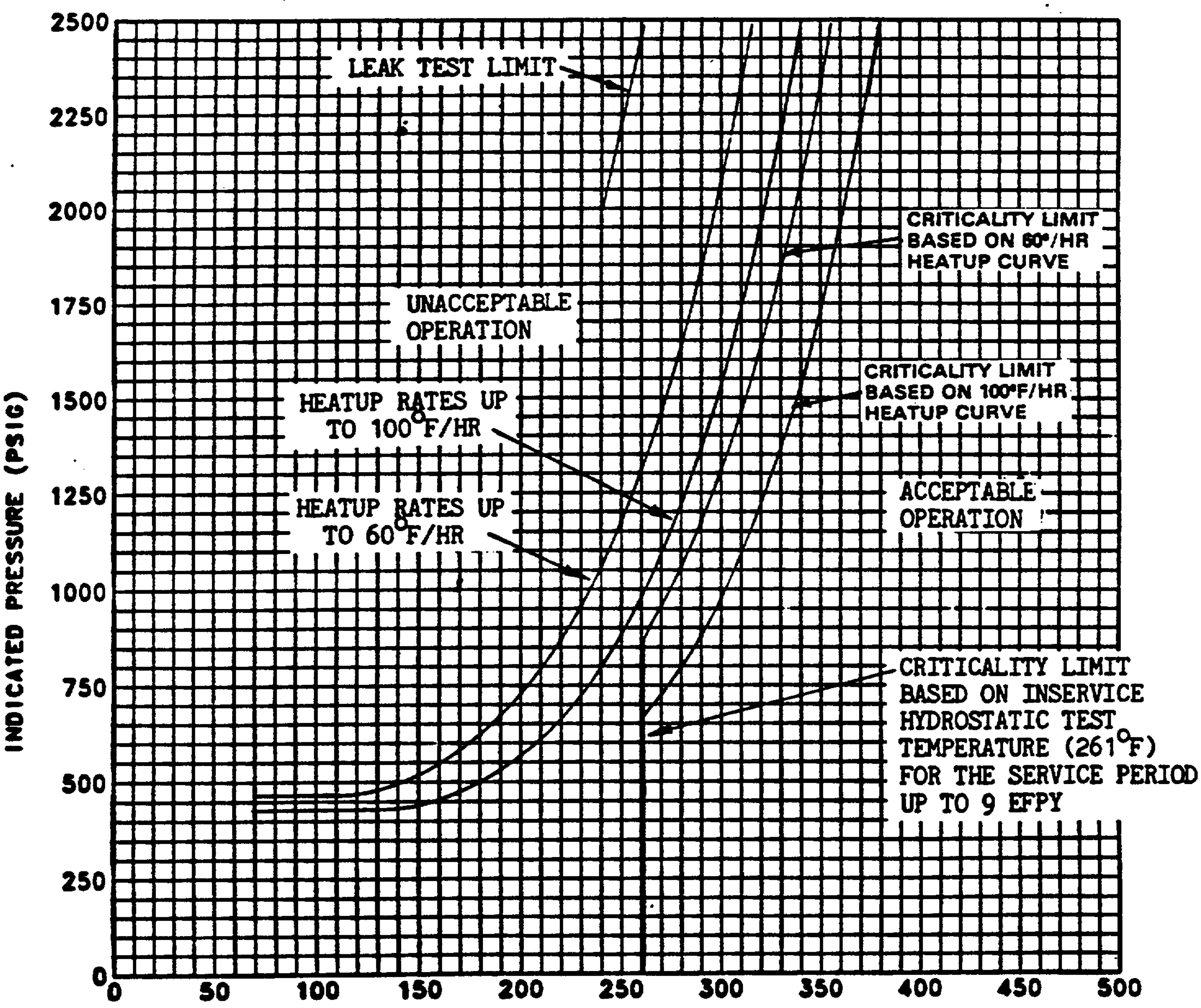
4.4.9.1.1 The Reactor Coolant System temperature and pressure shall be determined to be within the limits at least once per 30 minutes during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

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 in accordance with the schedule in Table 4.4-5. The results of these examinations shall be used to update Figures 3.4-2, 3.4-3, and 3.4-4.

MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL:	R.V. LOWER SHELL
COPPER CONTENT:	0.07 WT%
NICKEL CONTENT:	0.59 WT%
INITIAL RT_{NDT}:	50°F
RT_{NDT} AFTER 9 EFPY:	1/4T, 116.4°F 3/4T, 108.3°F

CURVES APPLICABLE FOR HEATUP RATES UP TO 100°F/HR FOR THE SERVICE PERIOD UP TO 9 EFPY AND CONTAINS MARGINS OF 10°F AND 60 PSIG FOR POSSIBLE INSTRUMENT ERRORS



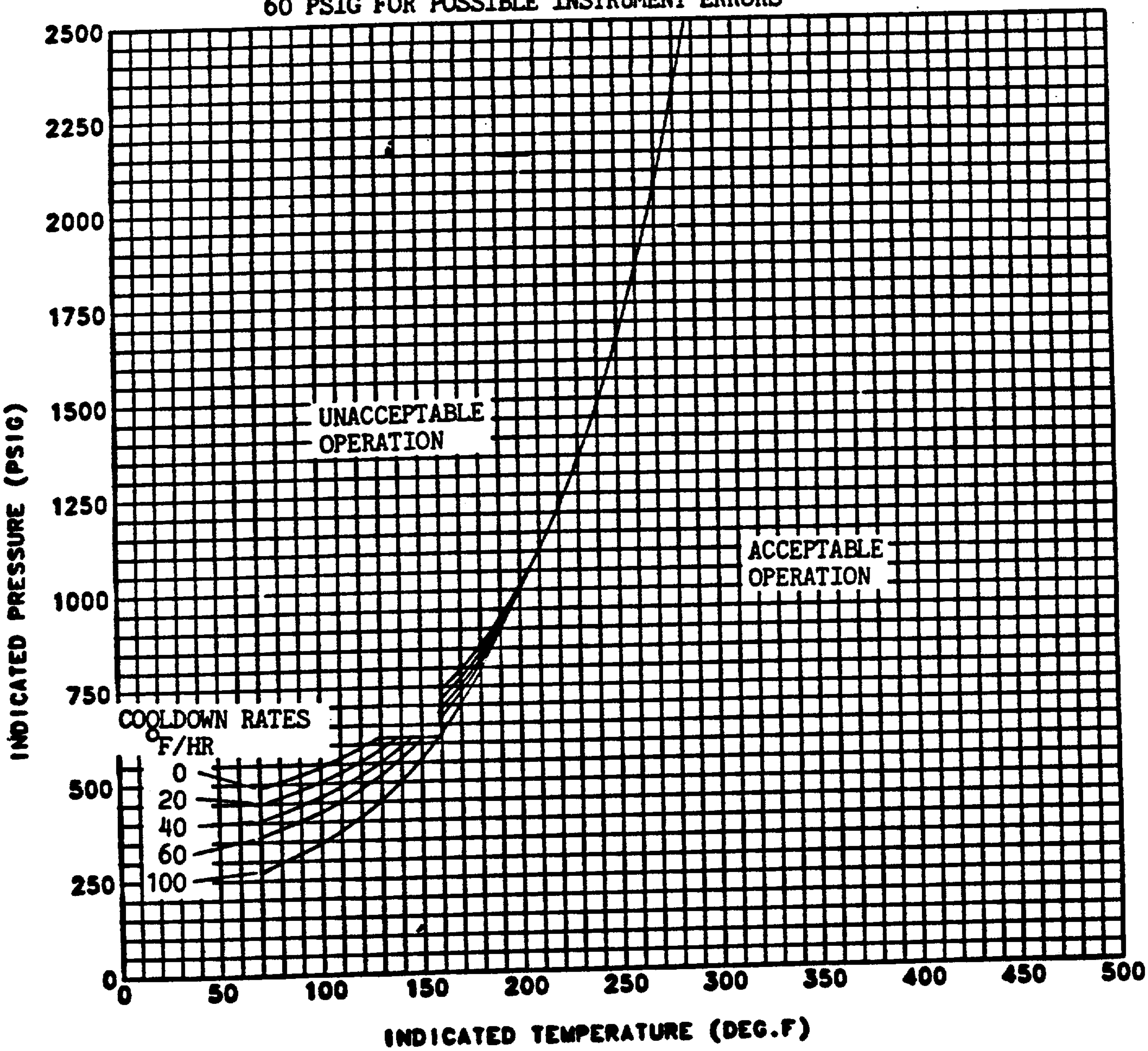
REACTOR COOLANT SYSTEM HEATUP LIMITATIONS
APPLICABLE FOR THE FIRST 9 EFPY.
UPRATED TO 3565 MWt.

Figure 3.4-2

MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL:	R.V. LOWER SHELL
COPPER CONTENT:	0.07 WT%
NICKEL CONTENT:	0.59 WT%
INITIAL RT_{NDT}:	$50^{\circ}F$
RT_{NDT} AFTER 9 EFPY:	1/4T, $116.4^{\circ}F$ 3/4T, $108.3^{\circ}F$

CURVES APPLICABLE FOR COOLDOWN RATES UP TO $100^{\circ}F$/HR FOR THE SERVICE PERIOD UP TO 9 EFPY AND CONTAINS MARGINS OF $10^{\circ}F$ AND 60 PSIG FOR POSSIBLE INSTRUMENT ERRORS



REACTOR COOLANT SYSTEM COOLDOWN LIMITATIONS
APPLICABLE FOR THE FIRST 9 EFPY.
UPRATED TO 3565 MWt.

FIGURE 3.4-3

TABLE 4.4-5

REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM - WITHDRAWAL SCHEDULE

CAPSULE NUMBER	VESSEL LOCATION	LEAD FACTOR	WITHDRAWAL TIME (EFPY)
U	58.5°	4.06	1st Refueling
Y	241°	3.70	5
V	61°	3.70	9
X	238.5°	4.06	15
W	121.5°	4.06	Standby
Z	301.5°	4.06	Standby

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:

1) By verifying at least once per 31 days that RHR RCS Suction Isolation Valve (RRSIV) 8701B is open with power to the valve operator removed, and

2) By verifying at least once per 12 hours that RRSIV 8702B is open.

b. For RHR suction relief valve 8708A:

1) By verifying at least once per 31 days that RRSIV 8702A is open with power to the valve operator removed, and

2) By verifying at least once per 12 hours that RRSIV 8701A is 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.

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*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.

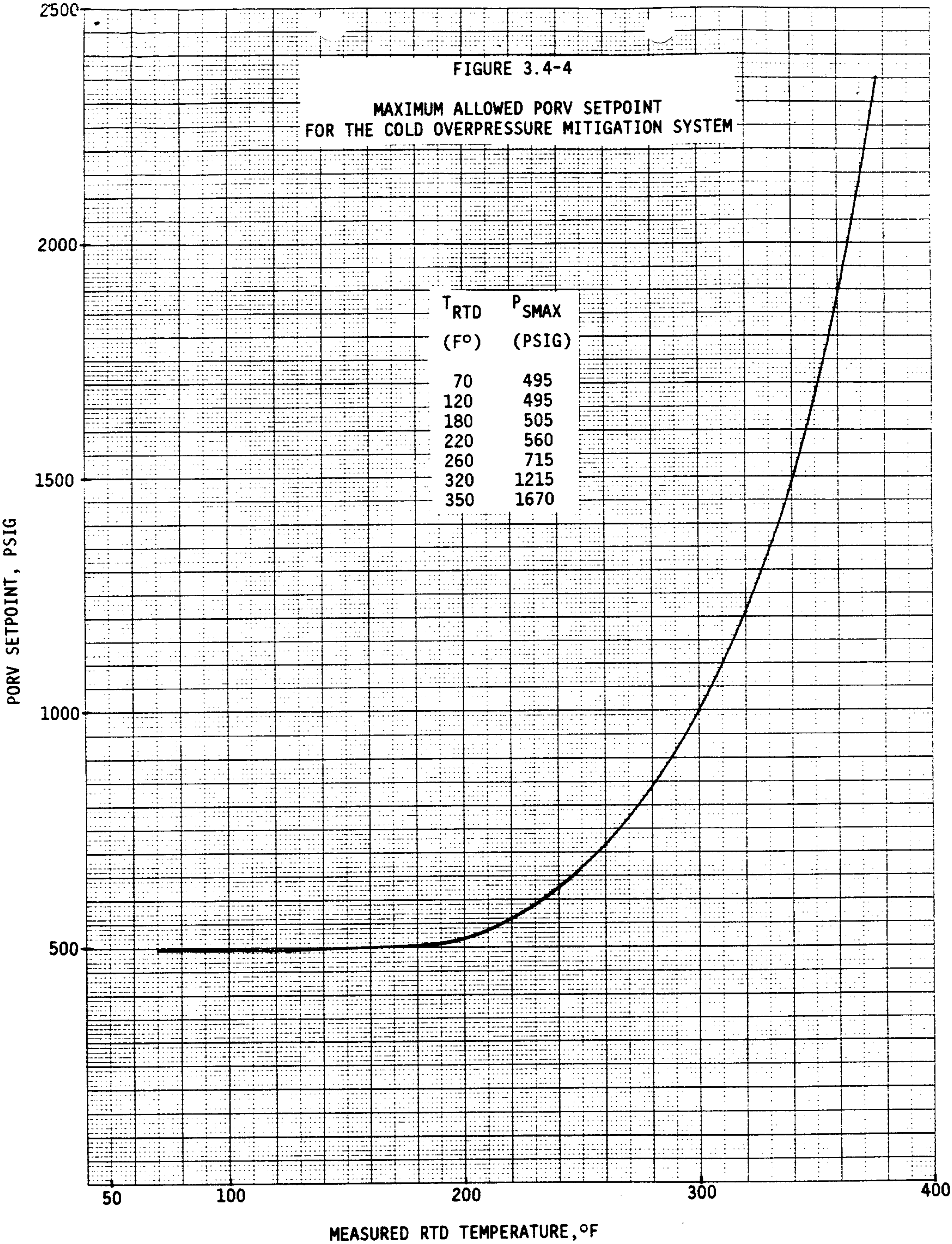

FIGURE 3.4-4

MAXIMUM ALLOWED PORV SETPOINT
FOR THE COLD OVERPRESSURE MITIGATION SYSTEM

T_{RTD} (F°)	P_{SMAX} (PSIG)
70	495
120	495
180	505
220	560
260	715
320	1215
350	1670

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 9 effective full power years (EFPY) of service life. The 9 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 9 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 surveillance specimen withdrawal

PRESSURE/TEMPERATURE LIMITS (Continued)

schedule is shown in Table 4.4-5. The lead factor represents the 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,

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION RELATED TO AMENDMENT NO. 36 TO FACILITY OPERATING LICENSE NO. NPF-30

UNION ELECTRIC COMPANY
CALLAWAY PLANT, UNIT I
DOCKET NO. STN 50-483

INTRODUCTION

By letter dated July 31, 1987, as supplemented by letter dated February 19, 1988, the Union Electric Company requested revision of the pressure-temperature limits in Callaway Plant Technical Specifications Section 3/4.4.9. The proposed limits will incorporate the updated pressure vs. temperature curves through 9 effective full power years (EFPY) to account for irradiation effects on the reactor vessel material. The proposed limits were based on the Westinghouse analysis (WCAP-11374, Rev. 1) of irradiation data taken from surveillance specimen Capsule U. The limits provide maximum permissible pressure at temperature for three reactor operations--system hydrostratic and leakage test, heatup or cooldown, and core critical condition. The pressure-temperature limits have to satisfy the requirements in Appendix G of 10 CFR Part 50. The licensee also requested revision of the maximum allowed power-operated relief valve (PORV) curve for the cold overpressure mitigation system, and a revision to the surveillance capsule withdrawal schedule based on the neutron fluence data taken from Capsule U.

DISCUSSION

Part of the NRC's effort to ensure integrity of the reactor vessel is to periodically evaluate the reduction in fracture toughness of the vessel material due to neutron irradiation damage. The effort consists of three steps. First, the licensee is required to establish a a surveillance program in accordance with Appendix H of 10 CFR Part 50, which requires periodic withdrawal of surveillance capsules from the reactor vessel. The capsules are installed in the vessel prior to startup and they should contain test specimens that were made from the plate, weld, and heat-affected zone materials of the reactor beltline. Secondly, the licensee is required to perform Charpy impact tests, tensile tests, and neutron fluence measurements of the specimens. These tests provide data for the actual neutron irradiation damage to the reactor vessel in terms of the reference temperature, RT_{NDT}, and the upper shelf energy (USE). The neutron damage is indicated by the decrease in USE and temperature shift in RT_{NDT}. The USE is the average energy value for all specimens whose test temperature is above the upper end of the transition temperature region. The USE decreases as a function of neutron fluence and copper content in the irradiated material. The shift of the adjusted reference temperature is the temperature shift in the Charpy curve for the irradiated material relative to that for the unirradiated material. According to Appendix G of 10 CFR Part 50, the USE must not be less than 50 ft-lb and the adjusted RT_{NDT} not more than 200°F. Thirdly, the licensee is required to construct pressure-temperature limit curves in the Technical Specifications. To construct the curves, the licensee may use the guidelines delineated in Reg. Guide 1.99, Revision 2, and Appendix G, Section III of the ASME Code.

EVALUATION

The Callaway Plant has six surveillance capsules for monitoring the effects of neutron irradiation on the reactor vessel material properties and they are located in the reactor vessel between the core barrel and vessel wall. Capsule U was removed after 1.05 EFPY and the withdrawal schedule for the next three capsules will be at 5, 9, and 15 EFPY. Capsule U contained specimens taken from base metal, weld metal, and heat-affected zone metal. The controlling material, which is the base plate, was included in the capsule. The withdrawal schedule for the capsules and the testing of Capsule U were performed in accordance with ASTM standard E185-82, and 10 CFR Part 50, Appendix H.

Capsule U experienced an average fast neutron fluence of 3.27×10^{18} n/cm². The lower shell plate material, R2708-1, has the highest initial RT_{NDT}, 50°F, the highest percentage of copper, 0.07%, and the highest percentage of nickel 0.59%. Using Regulatory Guide 1.99, Rev. 2, the adjusted RT_{NDT} of the plate was calculated to be 122°F and 112.5°F at the 1/4 T and 3/4 T vessel-wall-thickness locations, respectively. These results show that the predicted transition temperature shift based on Regulatory Guide 1.99, Rev. 2 is higher than the shift obtained directly from the surveillance capsule. Therefore, it is conservative and acceptable using the results of the Regulatory Guide to develop the pressure-temperature curves.

After irradiation, the average USE of the plate material decreased from 104 to 93 ft-lb. and the limiting weld decreased from 112 to 101 ft-lb. The transition temperature increase for the plate and weld are 0°F and 70°F, respectively. Based on these data, the USE and adjusted RT_{NDT} of both materials should be within the 50 ft-lb limit and the 200°F limit through the proposed 9 EFPY. The proposed pressure-temperature curves have included the safety margins required by 10 CFR Part 50, Appendix G.

The staff has used the method of calculating pressure-temperature limits in USNRC Standard Review Plan 5.3.2, NUREG-0800, to evaluate the proposed pressure-temperature limits. The amount of neutron irradiation damage was calculated based on Regulatory Guide 1.99, Rev. 2. The staff concludes that the proposed pressure-temperature limits meet the requirements of 10 CFR Part 50, Appendix G for 9 EFPY and may be incorporated into the Callaway Plant's Technical Specification. Also, the revised surveillance capsule withdrawal schedule satisfies the requirements of 10 CFR Part 50, Appendix H, and is acceptable.

The Callaway low temperature overpressure protection is accomplished with the Cold Overpressure Mitigation System (COMS), which consists of three subsystems, namely, (1) two residual heat removal suction relief valves, (2) two pressurizer PORV's, and (3) a reactor coolant system (RCS) vent of greater than or equal to 2 square inches. The limiting conditions for operation in TS 3.4.9.3 requires at least one of these three subsystems to be operable during modes 3, 4, 5 and 6. The TS also requires that the PORV setpoints do not exceed the maximum allowable limits established in Figure 3.4-4. This restriction is necessary to maintain the RCS pressure within the pressure-temperature limits specified in Figures 3.4-2 and 3.4-3 during low temperature heatup and cooldown operations. Therefore, a reevaluation of the PORV setpoint limits is necessary any time the pressure-temperature limit curves are revised.

The analysis must be done with consideration of the single failure criteria such that operation of one of the two PORV's is sufficient to prevent the RCS pressure from exceeding the Appendix G pressure limits which are determined from the heatup and cooldown pressure-temperature limit curves specified in Figures 3.4-2 and 3.4-3. The analysis is performed with the limiting transients having maximum mass and energy additions to the reactor coolant system.

At the staff's request, the licensee provided the analyses, including the limiting mass and heat addition transients analyzed, the initial conditions, major assumptions and the results of the analyses, to support the maximum allowable PORV setpoints in the revised Figure 3.4-4.

The limiting mass addition transient involves the operation of a single charge pump at the maximum flow with inadvertent isolation of letdown flow and residual heat removal system (RHRS) relief valves. The limiting heat addition transient is an inadvertent RCS coolant pump startup in one loop with a temperature asymmetry in the RCS, whereby the steam generator is at a temperature 50°F higher than the rest of the RCS. The initial conditions and major assumptions include the pressurizer being water solid to maximize the pressure surge, the failure of one PORV, and the time delay in the PORV operation. As indicated in the licensee's response dated February 19, 1988, these transients and assumptions are identical to those used in the previous analyses in the Callaway FSAR Section 5.2.2.10 except for the use of the revised Figure 3.4-4 for the PORV setpoints.

The results of analyses indicate that the peak RCS pressures for the transients analyzed fall below the nominal Appendix G pressure limits. The staff, therefore, concludes that the revised Figure 3.4-4 is also acceptable.

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. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

CONCLUSION

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; 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.

ACKNOWLEDGEMENT

Principal Contributors: J. Tsao, EMTB
Y. Hsii, SRXB

Dated: May 24, 1988