



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

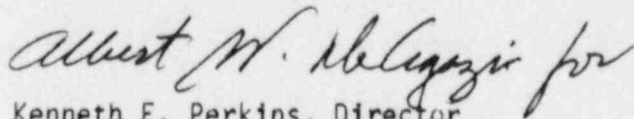
1. The Nuclear Regulatory Commission (the Commission) has found that:
  - A. The application for amendment filed by Union Electric Company (UE, the licensee) dated July 31, 1987, supplemented February 19, 1988, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (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



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

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

INSERT

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

## REACTOR COOLANT SYSTEM

### 3/4.4.9 PRESSURE/TEMPERATURE LIMITS

## REACTOR COOLANT SYSTEM

### LIMITING CONDITION FOR OPERATION

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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

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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.

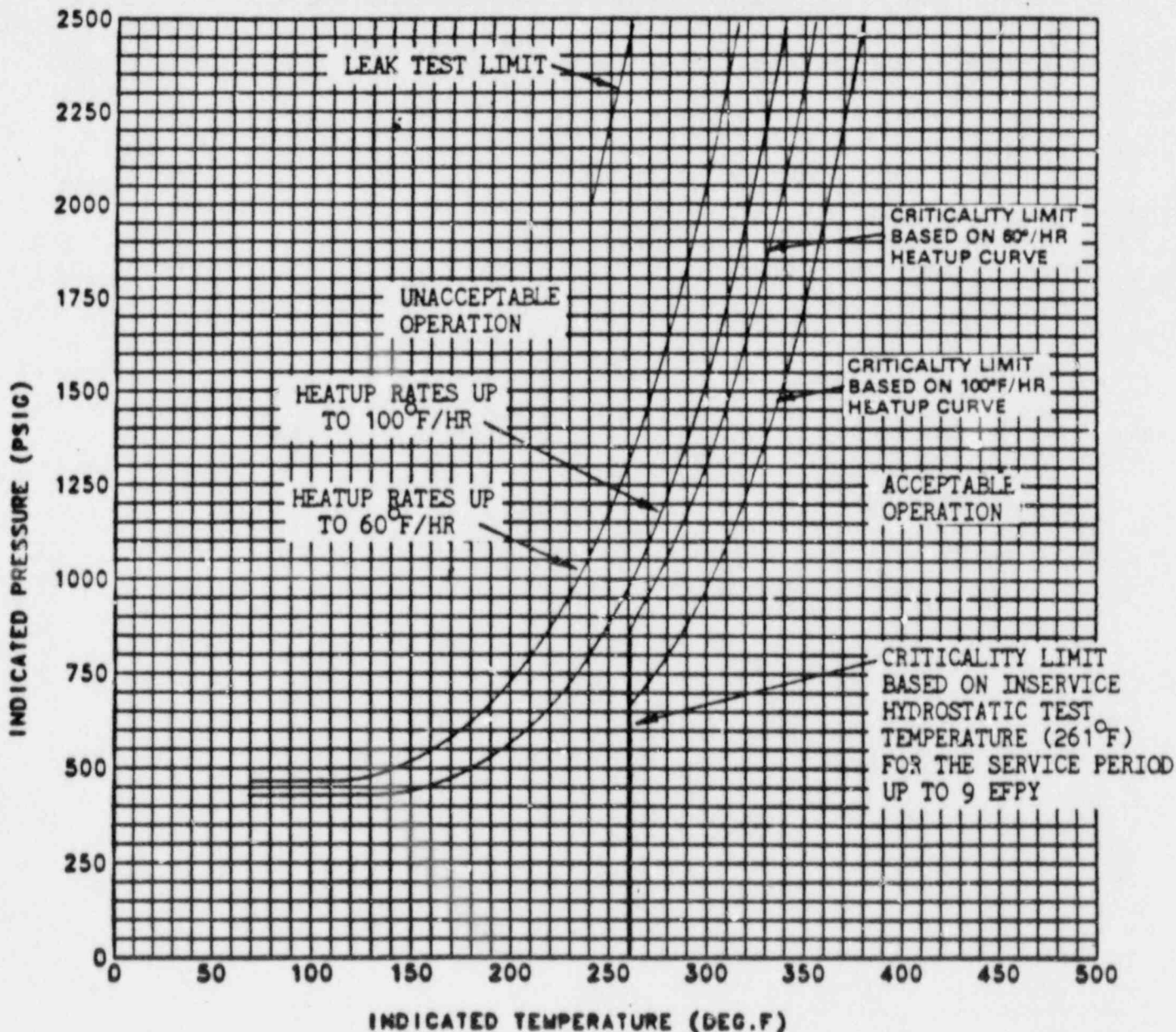
4.4.9.1.2 The reactor vessel material irradiation surveillance specimens shall be removed and examined, to determine changes in material properties, as required by 10 CFR Part 50, Appendix H in accordance with the schedule in Table 4.4-5. The results of these examinations shall be used to update Figures 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<sub>NDT</sub>: 50°F

RT<sub>NDT</sub> AFTER 9 EPFY: 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 EPFY AND CONTAINS MARGINS OF 10°F AND 60 PSIG FOR POSSIBLE INSTRUMENT ERRORS



REACTOR COOLANT SYSTEM HEATUP LIMITATIONS  
 APPLICABLE FOR THE FIRST 9 EPFY.  
 UPDATED TO 3565 MWt.

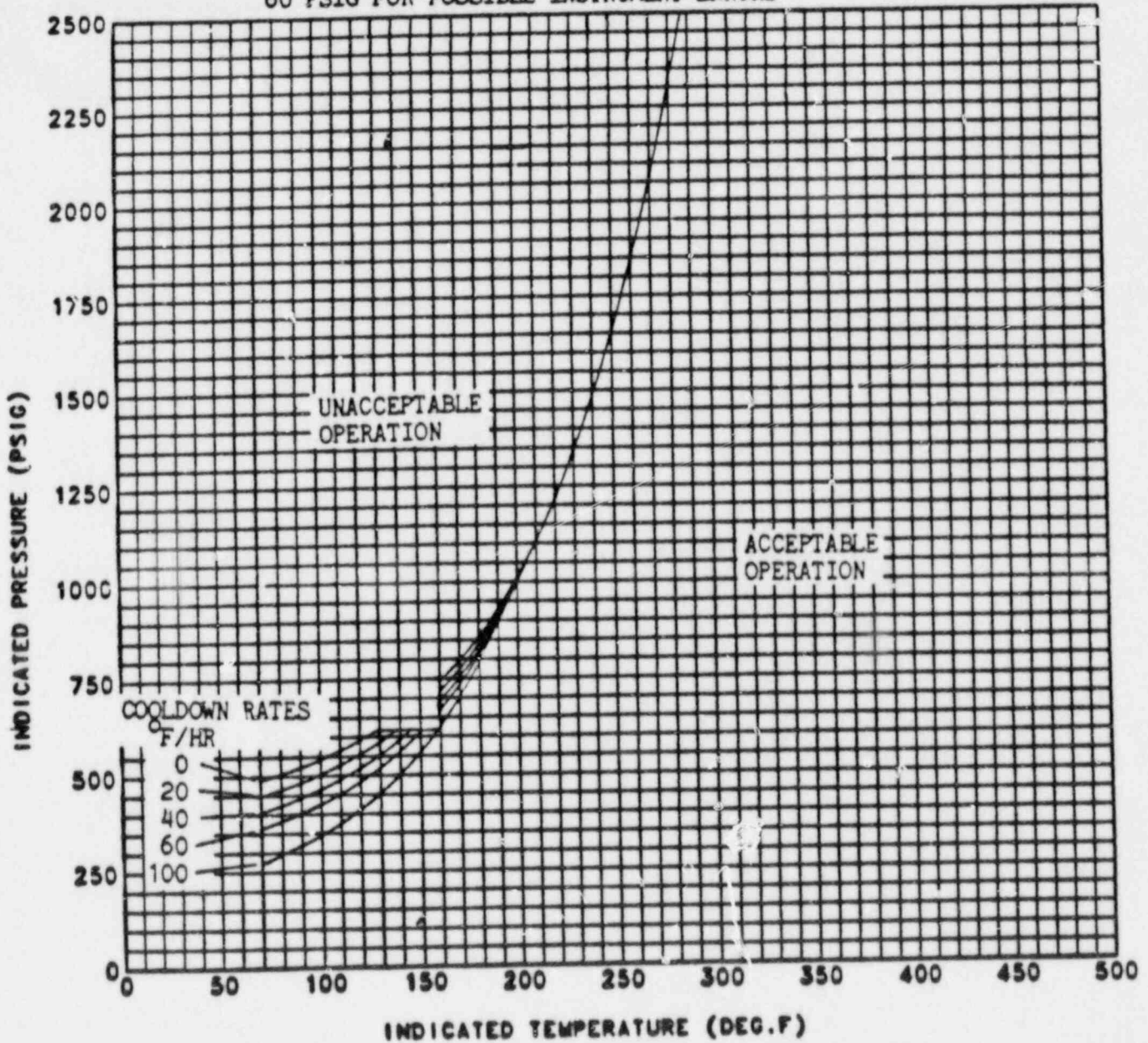
Figure 3.4-2

MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL: R.V. LOWER SHELL  
COPPER CONTENT: 0.07 WT%  
NICKEL CONTENT: 0.59 WT%  
INITIAL RT<sub>NDT</sub>: 50°F

RT<sub>NDT</sub> AFTER 9 EFPY: 1/4T, 116.4°F  
3/4T, 108.3°F

CURVES APPLICABLE FOR COOLDOWN 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 COOLDOWN LIMITATIONS  
APPLICABLE FOR THE FIRST 9 EFPY.  
UPRATED TO 3565 Mwt.

FIGURE 3.4-3

CALLAWAY - UNIT 1

3/1 4-32

Amendment No. 36

TABLE 4.4-5

REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM - WITHDRAWAL SCHEDULE

<u>CAPSULE NUMBER</u>	<u>VESSEL LOCATION</u>	<u>LEAD FACTOR</u>	<u>WITHDRAWAL TIME (EFPY)</u>
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

## REACTOR COOLANT SYSTEM

### SIIRVEILLANCE REQUIREMENTS

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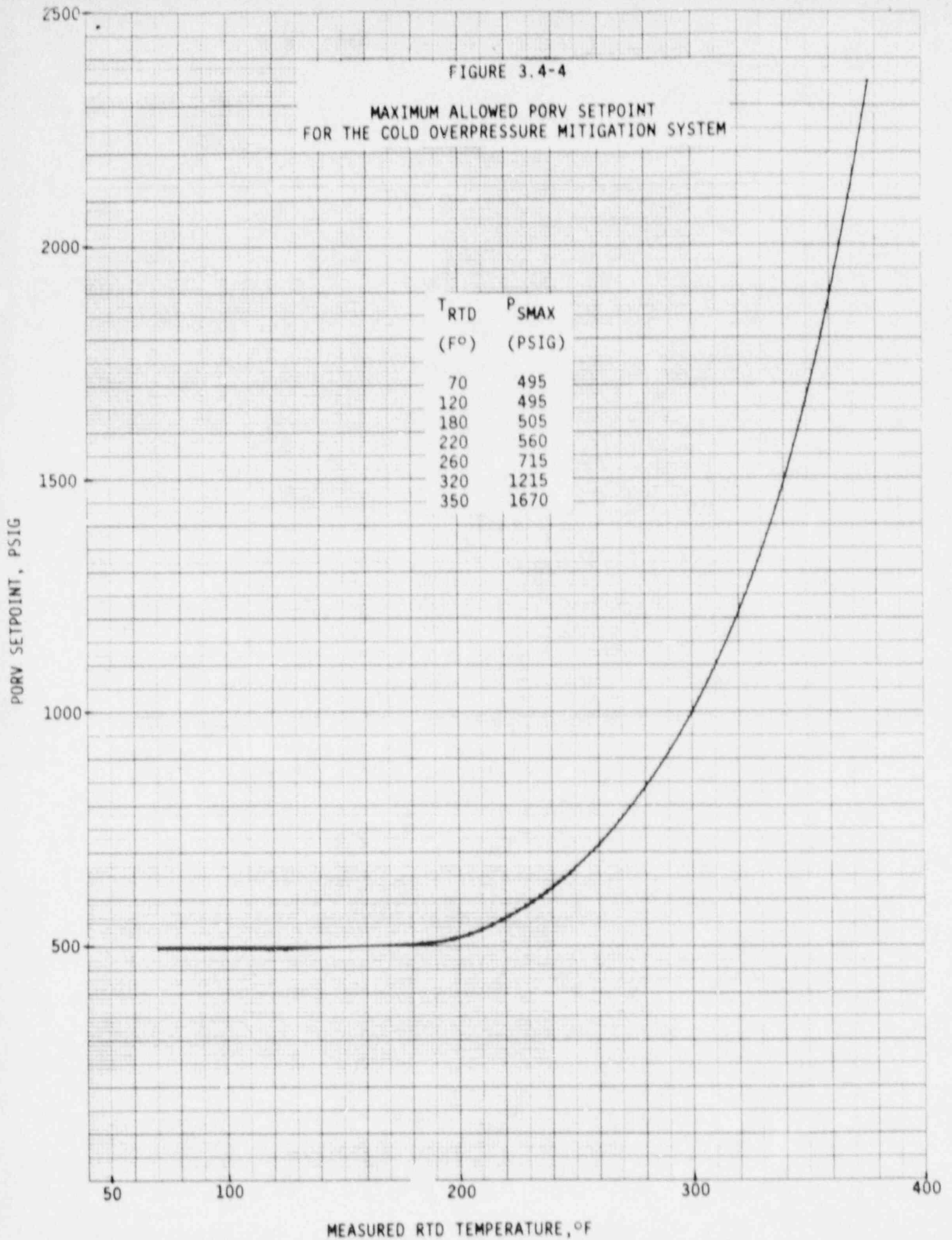
\*Except when the vent pathway is provided with a valve which is locked, sealed, or otherwise secured in the open position, then verify these valves open at least once per 31 days.



FIGURE 3.4-4

MAXIMUM ALLOWED PORV SETPOINT  
FOR THE COLD OVERPRESSURE MITIGATION SYSTEM

T <sub>RTD</sub> (F°)	P <sub>S MAX</sub> (PSIG)
70	495
120	495
180	505
220	560
260	715
320	1215
350	1670



## REACTOR COOLANT SYSTEM

### BASES

#### 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  $\Delta RT_{NDT}$  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  $\Delta RT_{NDT}$  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

## REACTOR COOLANT SYSTEM

### BASES

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#### 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  $\Delta RT_{NDT}$  determined from the surveillance capsule exceeds the calculated  $\Delta RT_{NDT}$  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,  $\Delta RT_{NDT}$ , 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:

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

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,