ATTACHMENT B

Proposed changes to Technical Specifications of Facility Operating License NPF 72 and 77:

1. Braidwood Station Onsite Review

2. BwAP 1205-3T3, Request for Offsite Review

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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 3.4-2a and 3.4-3a for Unit 1 (Figures 3.4-2b and 3.4-3b for Unit 2) 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 3.4-4a for Unit 1 (Figure 3.4-4b for Unit 2).

3.4-2a and 3.4-3a for Unit 1 (Figures 3.4-2b and 3.4-3b for Unit 2),

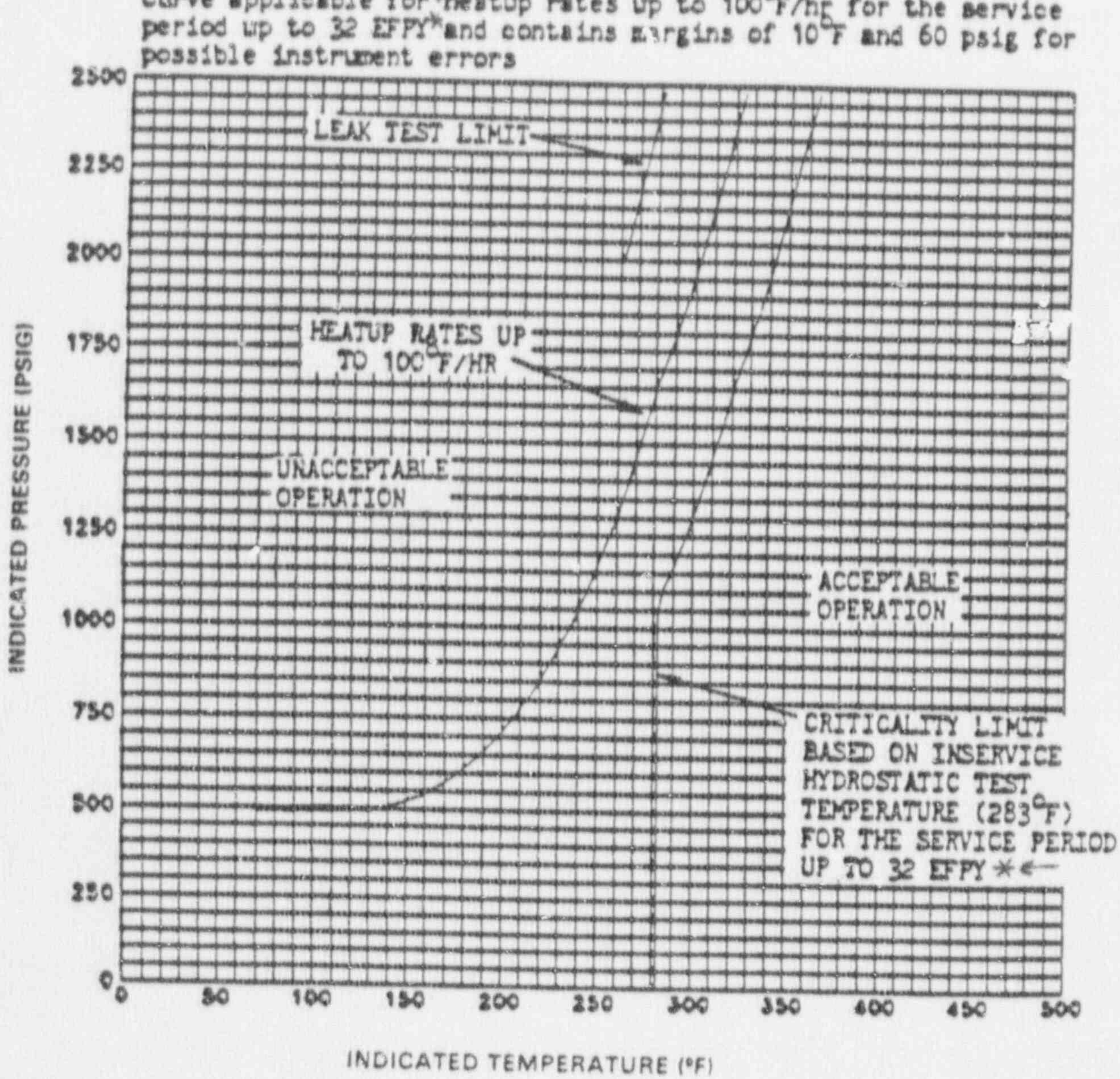

FIGURE 3.4-2a
REACTOR COOLANT SYSTEM HEATUP LIMITATIONS
APPLICABLE UP TO 32 EFPY* (UNIT 1)

* applicability date has been reduced per Regulatory Guide 1.99 Revision 2 to 4.5 EFPY. The calculation to determine applicability utilized actual copper content of 0.05 Wt %.

Replace with new figure 3.4-2b.

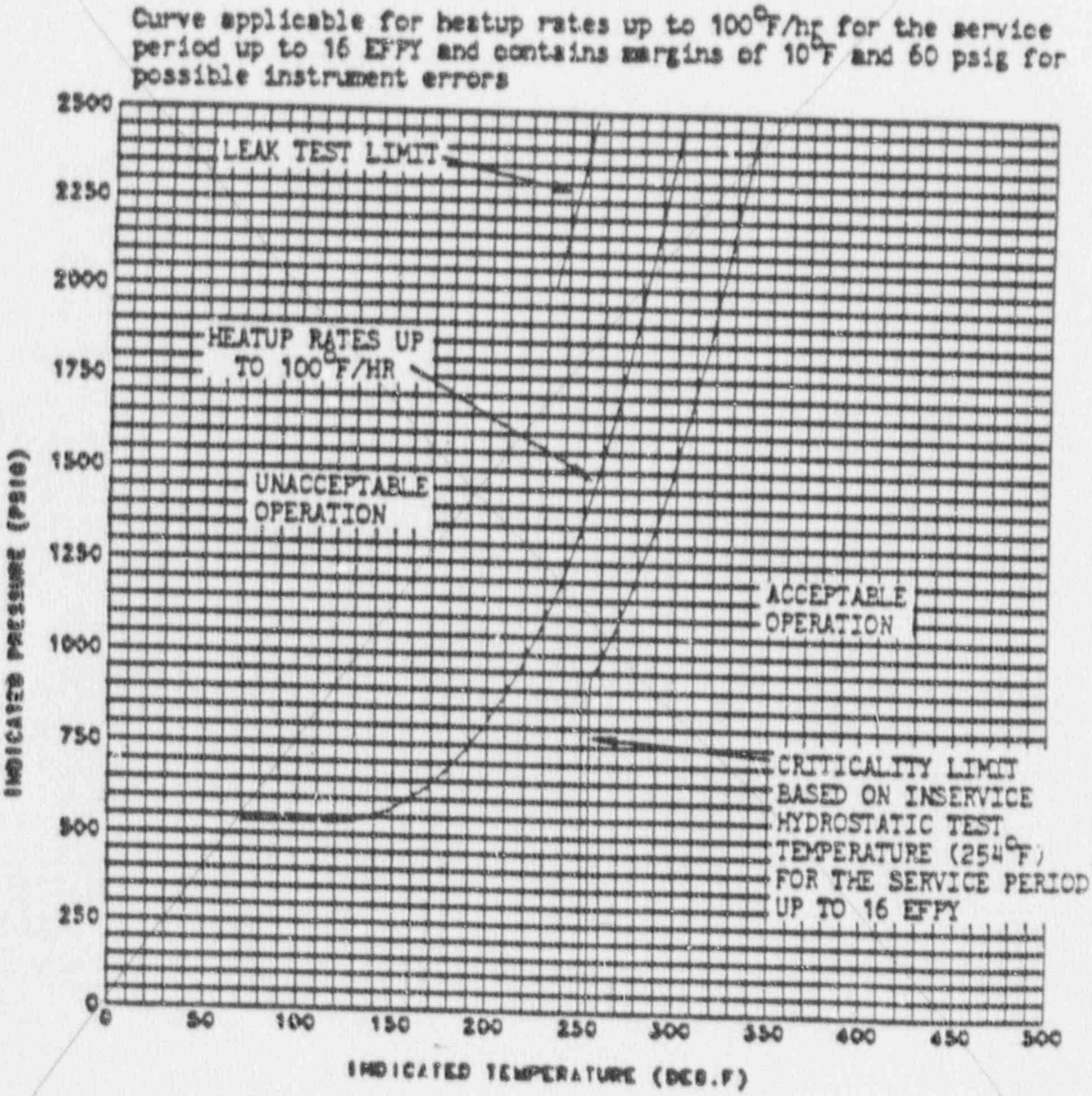

FIGURE 3.4-2b

REACTOR COOLANT SYSTEM HEATUP LIMITATIONS
APPLICABLE UP TO 16 EFPY (UNIT 2)

New Figure
3.4-2b

CURVES APPLICABLE FOR HEATUP RATES UP TO 100°F/HR FOR THE SERVICE PERIOD UP TO 16 EFPY. CONTAINS MARGIN OF 10°F AND 60 PSIG FOR POSSIBLE INSTRUMENT ERRORS.

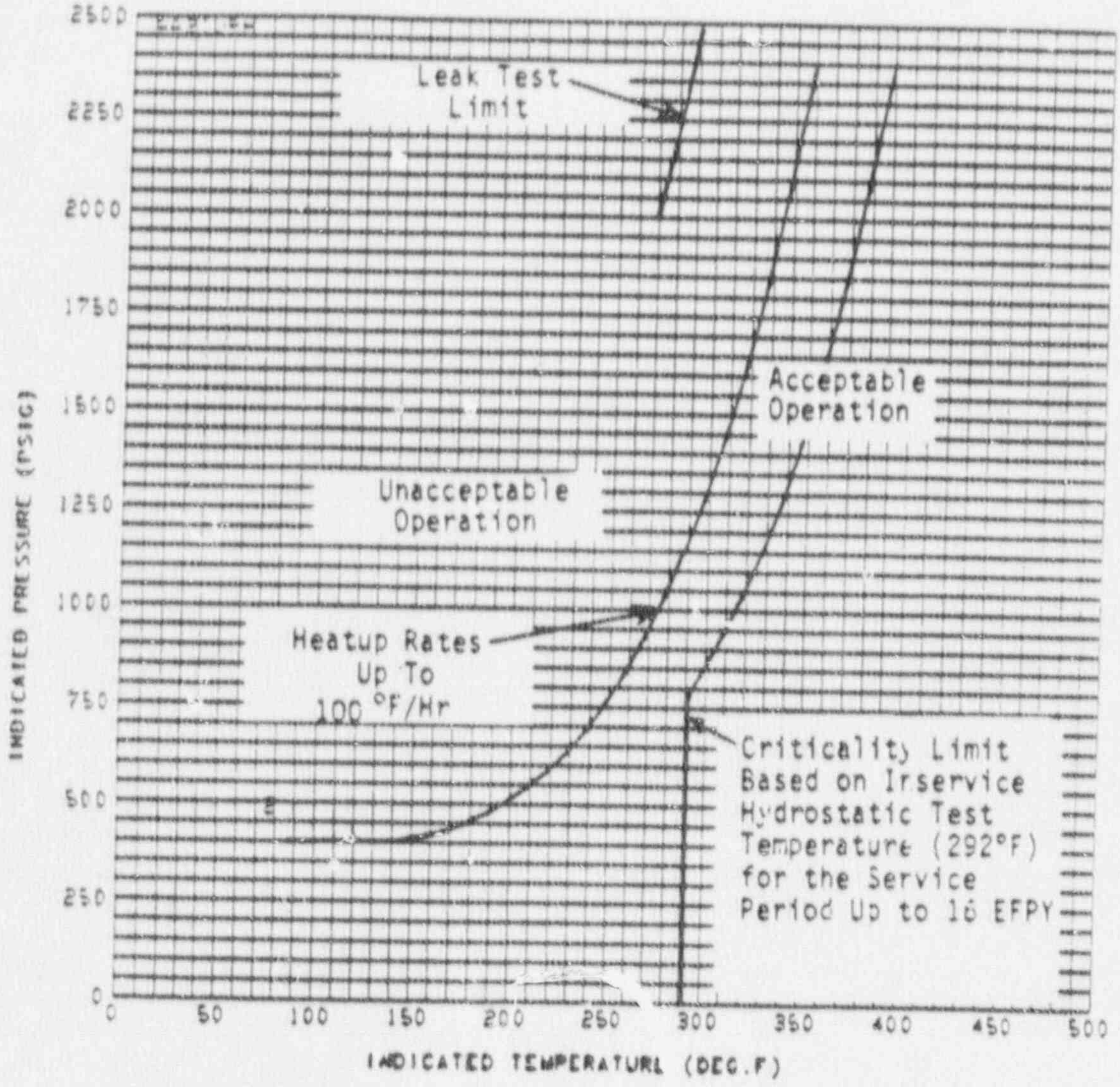

FIGURE 3.4-2b

REACTOR COOLANT SYSTEM HEATUP LIMITATIONS
APPLICABLE UP TO 16 EFPY (UNIT 2)

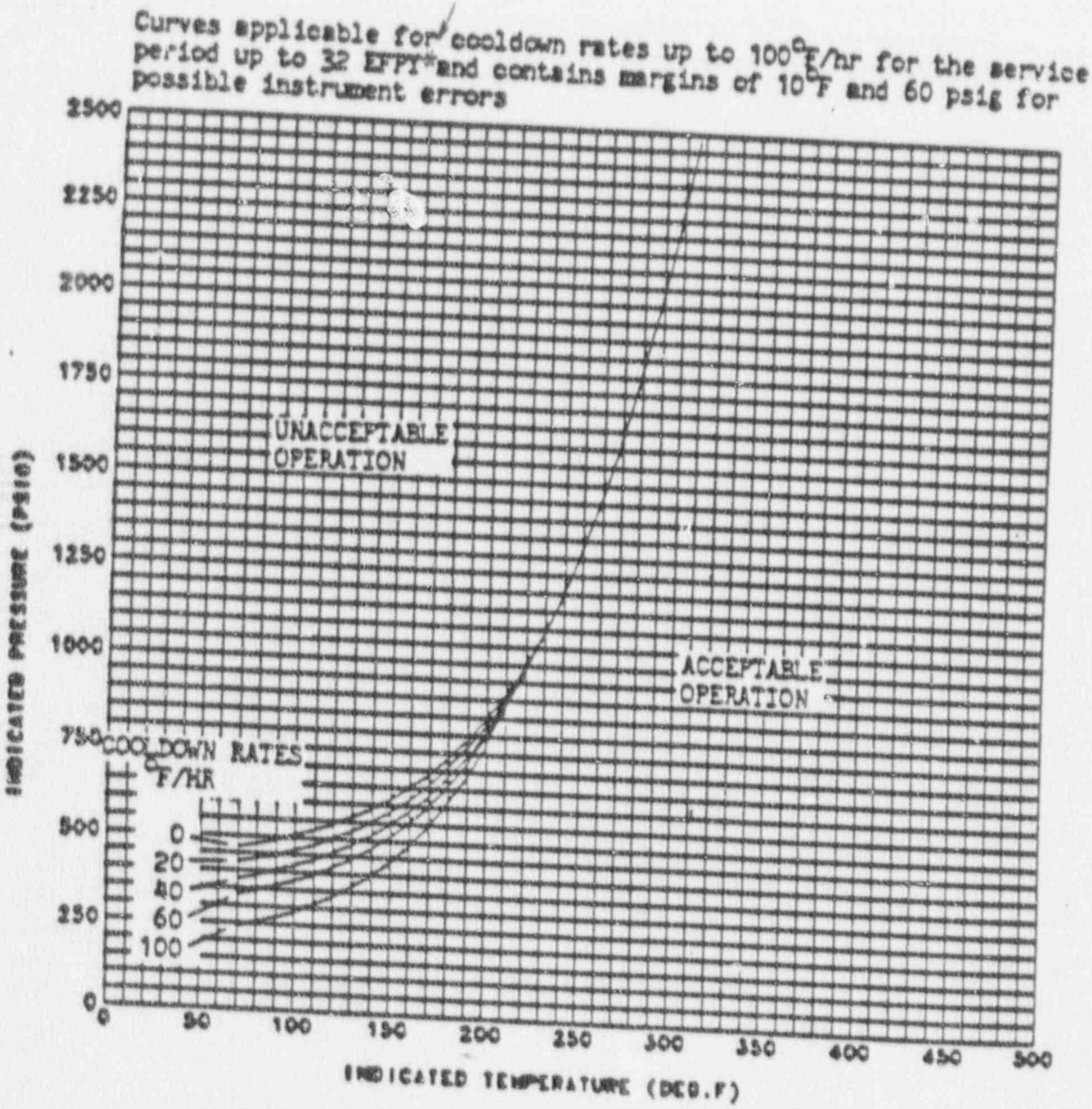

FIGURE 3.4-3a

REACTOR COOLANT SYSTEM COOLDOWN LIMITATIONS
APPLICABLE UP TO 32 EFPY* (UNIT 1)

* applicability has been reduced per Regulatory Guide 1.99 Revision 2 to 12 EFPY. The calculation to determine applicability utilized actual copper content of 0.05 Wt %.

Replace with new figure 3.4-3b

Curves applicable for cooldown rates up to 100°F/hr for the service period up to 16 EFPY and contains margins of 10°F and 60 psig for possible instrument errors

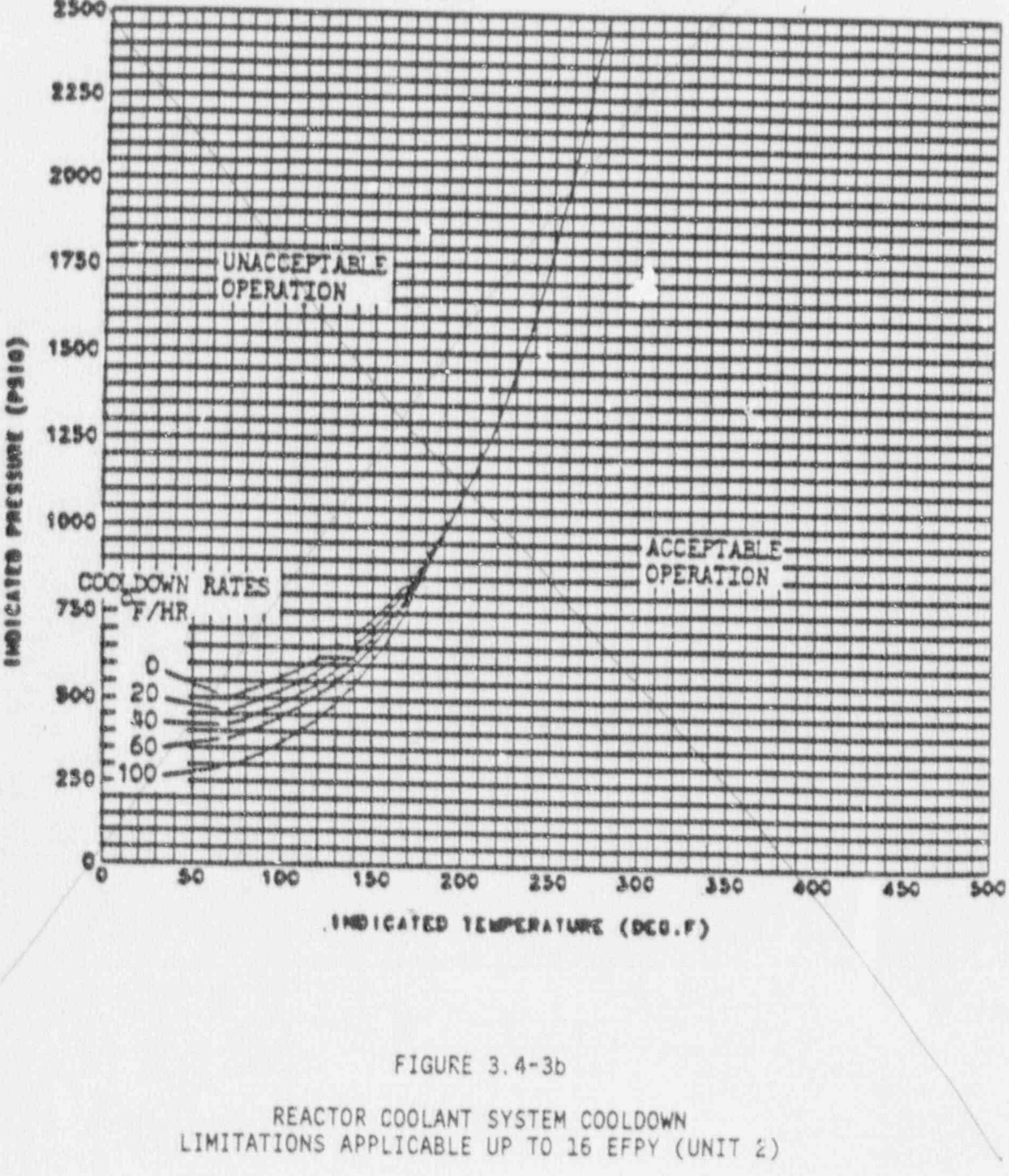

FIGURE 3.4-3b

REACTOR COOLANT SYSTEM COOLDOWN
LIMITATIONS APPLICABLE UP TO 16 EFPY (UNIT 2)

New Figure 3.4-3.b

CURVES APPLICABLE FOR COOLDOWN RATES UP TO 100°F/HR FOR THE SERVICE PERIOD UP TO 16 EFPY. CONTAINS MARGIN OF 10°F AND 60 PSIG FOR POSSIBLE INSTRUMENT ERRORS.

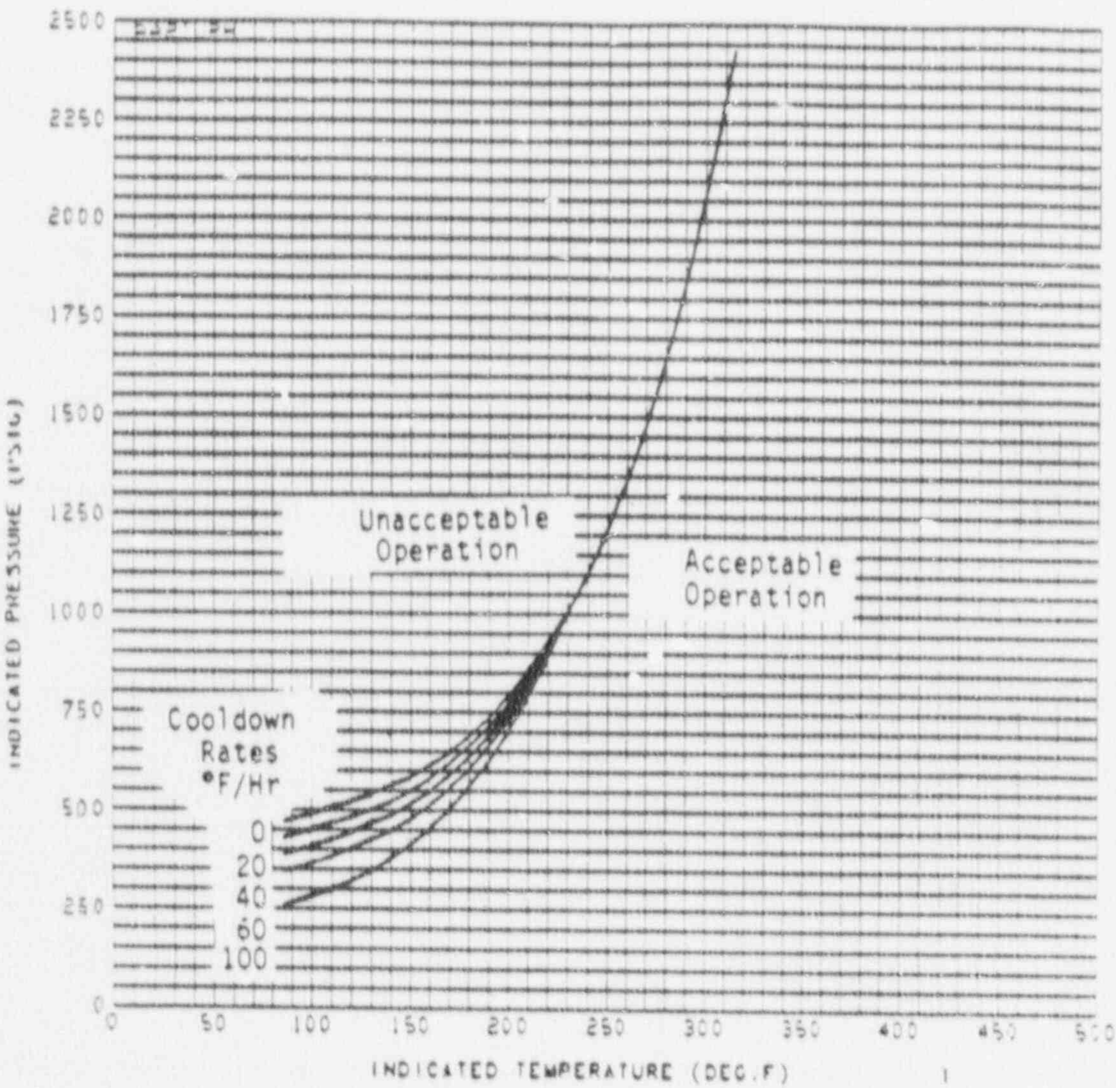

FIGURE 3.4-3b

REACTOR COOLANT SYSTEM COOLDOWN
LIMITATIONS APPLICABLE UP TO 16 EFPY (UNIT 2)

OVERPRESSURE PROTECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.9.3 At least one of the following Overpressure Protection Systems shall be OPERABLE:

a. Two residual heat removal (RHR) suction relief valves each with a Setpoint of 450 psig ± 1%, or

b. Two power-operated relief valves (PORVs) with lift Setpoints that vary with RCS temperature which do not exceed the limit established in Figure ~~3.4-4~~ 3.4-4a for Unit 1 (3.4-4b for Unit 2), or

c. The Reactor Coolant System (RCS) depressurized with an RCS vent of greater than or equal to 2 square inches.

APPLICABILITY: MODES 4 and 5, and MODE 6 with the reactor vessel head on.

ACTION:

a. With one PORV and one RHR suction relief valve inoperable, either restore two PORVs or two RHR suction relief valves to OPERABLE status within 7 days or depressurize and vent the RCS through at least a 2 square inch vent within the next 8 hours.

b. With both PORVs and both RHR suction relief valves inoperable, depressurize and vent the RCS through at least a 2 square inch vent within 8 hours.

c. In the event the PORVs, or the RHR suction relief valves, or the RCS vent(s) are used to mitigate an RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the PORVs, or the RHR suction relief valves, or RCS vent(s) on the transient, and any corrective action necessary to prevent recurrence.

d. The provisions of Specification 3.0.4 are not applicable.

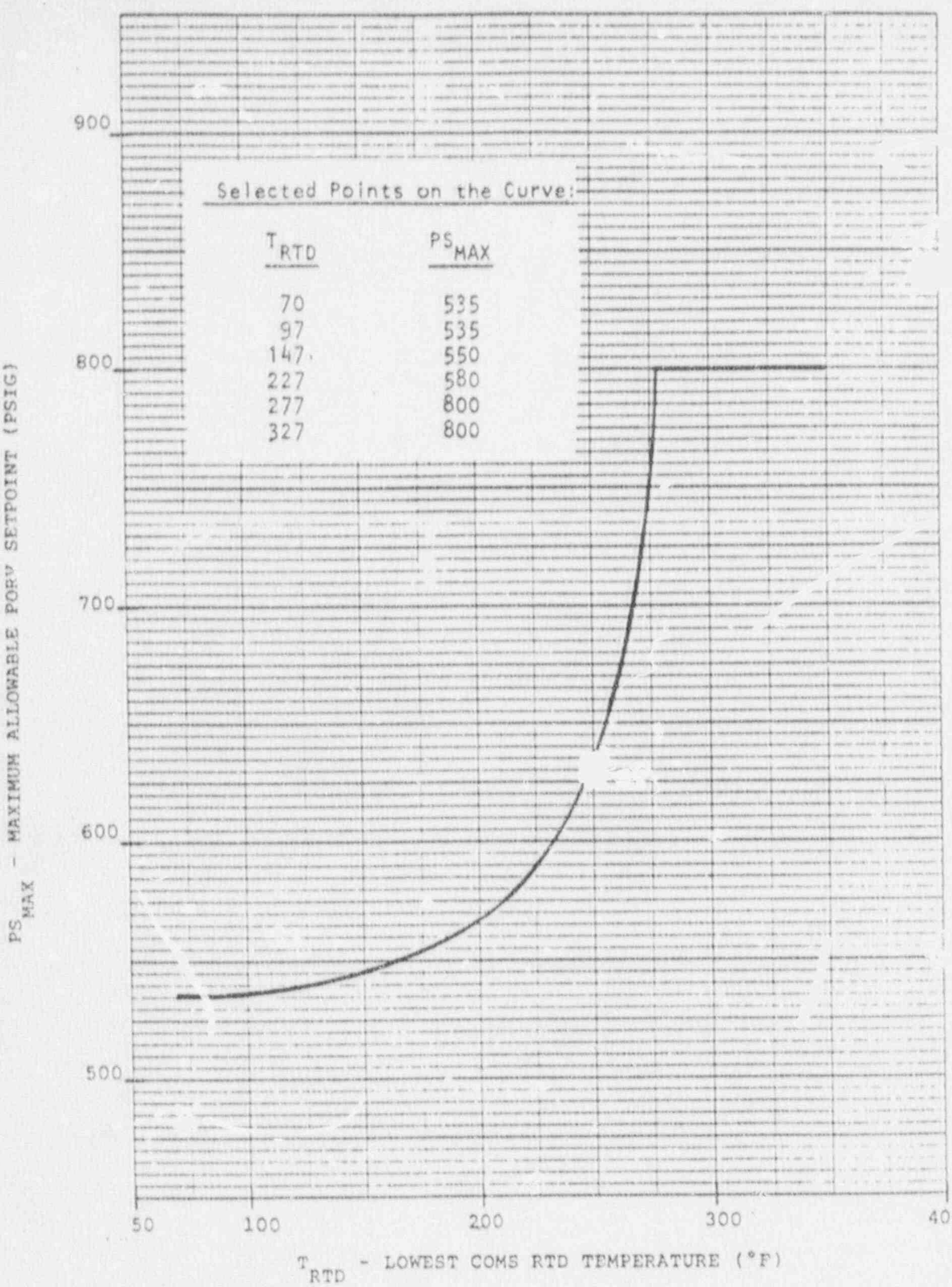

FIGURE 3.4-4a

NOMINAL PORV PRESSURE RELIEF SETPOINT VERSUS
RCS TEMPERATURE FOR THE COLD OVERPRESSURE PROTECTION SYSTEM
APPLICABLE UP TO 10 EFPY (Unit 1)

* applicability has been reduced per Regulatory Guide 1.99 Revision 2 to 4.5 EFPY. The calculation to determine applicability utilized actual copper co[illegible]t of .05 %.

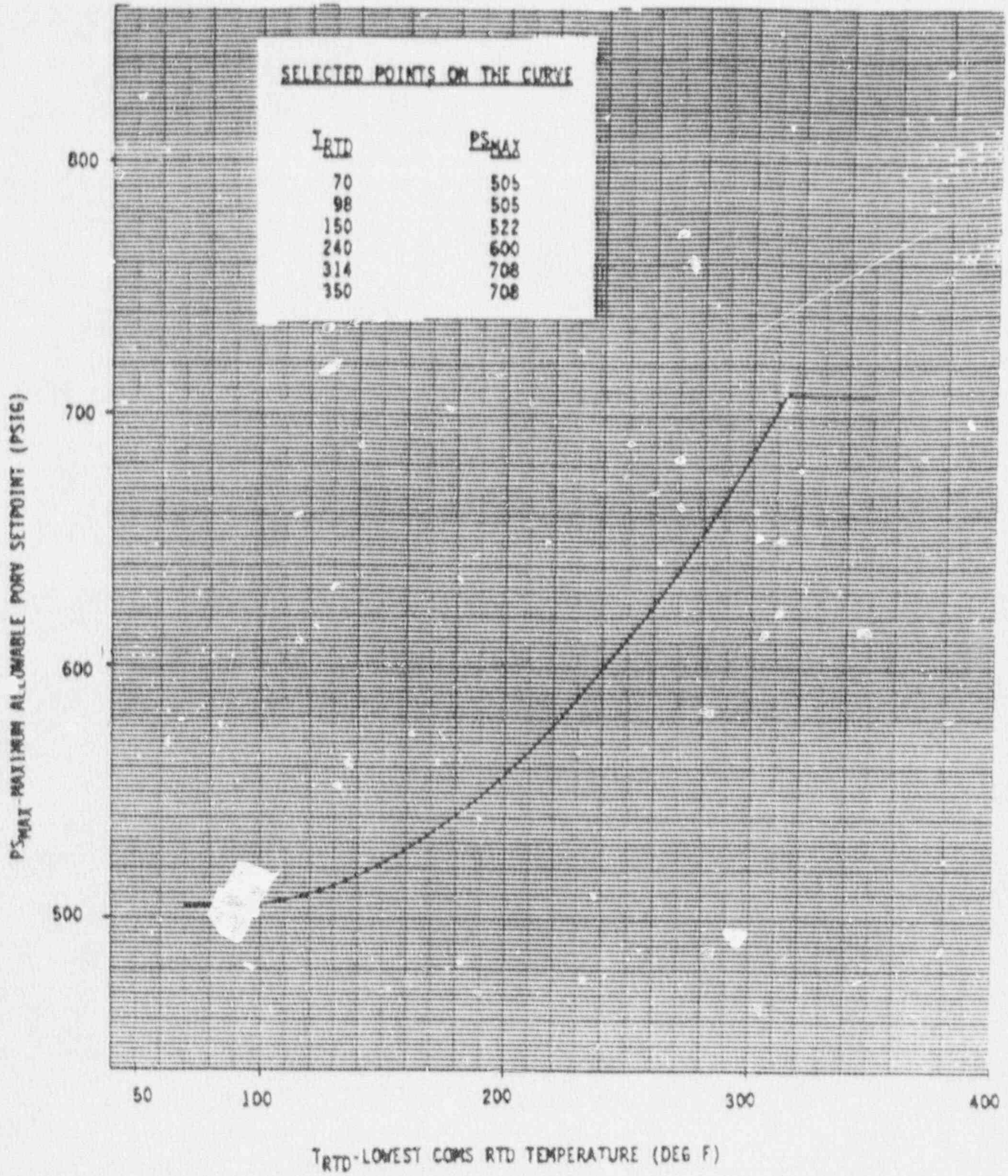

FIGURE 3.4-4b

NOMINAL PORV PRESSURE RELIEF SETPOINT VERSUS
RCS TEMPERATURE FOR THE COLD OVERPRESSURE PROTECTION SYSTEM
(UNIT 2)

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:

1) By verifying at least once per 31 days that RHR RCS Suction Isolation Valve RH8702A is open with power to the valve operator removed, and

2) By verifying at least once per 12 hours that RH8702B is open.

b. For RHR suction relief valve 8708A:

1) By verifying at least once per 31 days that RH8701B is open with power to the valve operator removed, and

2) By verifying at least once per 12 hours that RH8701A 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.

~~#The specified 18 month interval may be extended to 32 months for cycle 1 only.~~

SPECIFIC ACTIVITY (Continued)

take corrective action. Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomenon. 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~~ 3.4-2a (3.4-2b) and ~~3.4-3~~ 3.4-3a (3.4-3b) for the service period specified thereon:

    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; and

    b. Figures ~~3.4-2~~ 3.4-2a (3.4-2b) and ~~3.4-3~~ 3.4-3a (3.4-3b) 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° 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, and

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 1973 Summer Addenda to Section III of the ASME Boiler and Pressure Vessel and Code.

PRESSURE/TEMPERATURE LIMITS (Continued)

Heatup and cooldown limit curves are calculated using the most limiting value of the nil-ductility reference temperature, RT_{NDT}, at the end of 32* effective full power years for Unit 1 (16 effective full power years for Unit 2) of service life. The 32* EFPY for Unit 1 (16 EFPY for Unit 2) 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 3/4.4-1a for Unit 1 (Table B 3/4.4-1b for Unit 2). 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 copper content and ~~phosphorus~~ nickel 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 ~~1, "Effects of Residual Elements on Predicted Radiation Damage to~~ 2, "Radiation Embrittlement of 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~~ 3.4-2a (3.4-2b) and 3.4-3a (3.4-3b) include predicted adjustments for this shift in RT_{NDT} at the end of 32* EFPY for Unit 1 (16 EFPY for Unit 2) as well as adjustments for possible errors in the pressure and temperature sensing instruments. Insert 1

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 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, "Basis for Heatup and Cooldown Limit Curves," April 1975.

* For Unit 1 applicability dates have been revised in accordance with Regulatory Guide 1.99 Revision 2, to 4.5 EFPY for heatup and 12.0 EFPY for cooldown.

INSERT 1

Revised heatup and cooldown curves have been generated for Unit 2 in accordance with Regulatory Guide 1.99 Revision 2. For Unit 1 the curves remain the same; However, the applicability date has been reduced per Regulatory Guide 1.99 Revision 2 to 4.5 EFPY for heatup and 12.0 EFPY for cooldown.

TABLE B 3/4.4-1a

REACTOR VESSEL TOUGHNESS
(UNIT 1)

COMPONENT	Heat No.	MATERIAL SPEC.	Cu %	P %	T_{NDT} °F	RT_{NDT} F°	Average Shelf Energy NMWD* ft-lbs	MWD** ft-lbs
Closure Head Dome	D1398-1	A533B, Cl. 1	.06	.009	-30	-30	129	-
Closure Head Ring	49C1126-1-1	A508, Cl. 3	.02	.009	-20	-20	123	-
Closure Head Flange	2030-V-1	A508, Cl. 2	.11	.009	-20	-20	163	-
Vessel Flange	122N357VA1	A508, Cl. 2	-	.010	-10	-10	106	-
Inlet Nozzle	21-3257	A508, Cl. 2	.09	.008	-20	-20	144	-
Inlet Nozzle	21-3257	A508, Cl. 2	.09	.010	-10	-10	144	-
Inlet Nozzle	22-3313	A508, Cl. 2	.07	.008	-10	-10	130	-
Inlet Nozzle	22-3313	A508, Cl. 2	.07	.010	0	0	115	-
Outlet Nozzle	22-3025	A508, Cl. 2	.13	.013	-10	-10	125	-
Outlet Nozzle	4-3329	A508, Cl. 2	.08	.009	-20	-20	156	-
Outlet Nozzle	4-3383	A508, Cl. 2	.08	.008	-20	-20	147	-
Outlet Nozzle	11-5226	A508, Cl. 2	.09	.007	-10	-10	125	-
Nozzle Shell	5P7016	A508, Cl. 2	.04	.008	10	10	155	-
Upper Shell***	49D383/ 49C344-1-1	A508, Cl. 3	.05	.008 (.73)	-30	-30	122	173
Lower Shell***	49D867/ 49C813-1-1	A508, Cl. 3	.03	.007 (.73)	-20	-20	135	151
Bottom Head Ring	49D148-1-1	A508, Cl. 3	.05	.008	-50	-50	147	-
Bottom Head Dome	C4882-1	A533B, Cl. 1	.14	.010	-20	-20	123	-
Upper Shell to*** Lower Shell Girth Weld	WF-562		.04	.015 (.67)	40	40	80	-
Weld HAZ					-70	<-10	151	-

*Normal to major working direction.
**Major working direction.
*** Calculations per Regulatory Guide 1.99 Revision 2 use the Nickel content shown in parentheses

TABLE B 3/4.4-1b

REACTOR VESSEL TOUGHNESS
(UNIT 2)

COMPONENT	HEAT NO.	MATERIAL SPEC.	Cu %	P %	T_{NDT} °F	RT_{NDT} F°	Average Shelf Energy NMWD* ft-lbs	MWD** ft-lbs
Closure Head Dome	B9754-1	A533B, Cl. 1	.16	.005	-60	-60	151	-
Closure Head Ring	50C478-1-1	A508, Cl. 3	.05	.006	-30	-30	128	-
Closure Head Flange	2031-V-1	A508, Cl. 2	-	.009	20	20	135	-
Vessel Flange	124P455	A508, Cl. 2	.07	.010	20	20	128	-
Inlet Nozzle	41-5414	A508, Cl. 2	.07	.008	-10	-10	137	-
Inlet Nozzle	41-5414	A508, Cl. 2	.07	.009	-10	-10	140	-
Inlet Nozzle	42-5417	A508, Cl. 2	.09	.011	-10	-10	122	-
Inlet Nozzle	42-5417	A508, Cl. 2	.09	.009	-10	-10	116	-
Outlet Nozzle	4-3502	A508, Cl. 2	.09	.012	-10	-10	155	-
Outlet Nozzle	11-5226	A508, Cl. 2	.09	.009	-10	-10	116	-
Outlet Nozzle	4-3481	A508, Cl. 2	.07	.008	-10	-10	163	-
Outlet Nozzle	11-5266	A508, Cl. 2	.09	.010	10	10	117	-
Nozzle Shell	5P7056	A508, Cl. 2	.04	.005	30	30	115	-
Upper Shell ***	49D963/ 49C904-1-1	A508, Cl. 3	.03	.007 (.71)	-30	-30	119	147
Lower Shell ***	50D102/ 50C97-1-1	A508, Cl. 3	.06	.006 (.75)	-30	-30	144	168
Bottom Head Ring	49 ~~47~~D1066-1-1	A508, Cl. 3	.07	.008	-30	-30	156	-
Bottom Head Dome	D1429-1	A533B, Cl. 1	.11	.010	-20	-20	120	-
Upper Shell to *** Lower Shell Girth Weld	WF-562		.04	.015 (.6[illegible])	40	40	80	-
Weld HAZ					-30	-30	145	-

*Normal to major working direction.
**Major working direction.

*** Calculations per Regulatory Guide 1.99 Revision 2 use Nickel content shown in parentheses

PRESSURE/TEMPERATURE LIMITS (Continued) 3.4-3a for Unit 1 (3.4-3b for Unit 2) may be present

A ~~The~~ notch in the cooldown curve of Figure ~~3.4-3 is~~ due to the added constraint on the vessel closure flange given in Appendix G of 10 CFR 50. This constraint requires that, at pressures greater than 20% of the preservice system hydrostatic test pressure, the flange regions that are highly stressed by the bolt preload must exceed the RT_{NDT} of the material by at least 120°F. The flange RT_{NDT} + 120°F may impinge~~s~~ on the cooldown curves and therefore the notch is required. If no notch is present, this indicates that the vessel closure flange region has been determined to be not limiting

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.

PRESSURE/TEMPERATURE LIMITS (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 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 350°F. Either PORV 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.

These two scenarios are analyzed to determine the resulting overshoots assuming a single PORV actuation with a stroke time of 2.0 seconds from full closed to full open. Figure ~~3.4-4 is~~ 3.4-4a (3.4-4b) are based upon this analysis and represents the maximum allowable PORV variable setpoint such that, for the two overpressurization transients noted, the resulting pressure will not exceed the ~~nominal 10 effective full power years (EFPY)~~ Appendix G reactor vessel NDT limits.

RHR RCS suction isolation valves 8701A and 8702A are interlocked with an "A" train wide range pressure transmitter and valves 8701B and 8702B are interlocked with a "B" train wide range pressure transmitter. Removing power from valves 8701B and 8702A, prevents a single failure from inadvertently isolating both RHR suction relief valves while maintaining RHR isolation capability for both RHR flow paths.

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

ATTACHMENT C

EVALUATION OF SIGNIFICANT HAZARDS CONSIDERATIONS

Commonwealth Edison has evaluated this proposed amendment and determined that it involves no significant hazards considerations. According to 10 CFR 50.92(c), a proposed amendment to an operating license involves no significant hazards considerations if operation of the facility in accordance with the proposed amendment would not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated; or

2. Create the possibility of a new or different kind of accident from any accident previously evaluated; or

3. Involve a significant reduction in a margin of safety.

The proposed change does not result in a significant increase in the probability or consequences of accidents previously evaluated. Several of the changes involved are administrative in nature, and as such have no impact on the probability or consequences of accidents. The changes to the Unit 2 heatup, cooldown and cold overpressure protection figures are the result of reanalysis performed in accordance with Regulatory Guide 1.99 Revision 2. This revision effectively resulted in a shifting of the above curves in a more conservative direction. The shifting of these curves has no effect on the probability for occurrence of any accidents. The consequences for accidents would remain unchanged, the opening setpoint for cold overpressure protection will be at a lower value, thus ensuring that the Appendix G limits of 10 CFR 50 will continue to be met. The reduction in Effective Full Power Years (EFPY) for the Unit 1 vessel will ensure that the Appendix G limits will continue to be met until new curves are generated.

The proposed change does not create the possibility for a new or different kind of accident from any accident previously evaluated. The proposed change does not introduce any new equipment or change the fashion in which the installed equipment will be operated. The revised setpoints for the cold overpressure protection setpoints are still high enough to allow normal heatup and cooldown operations without requiring programmatic changes. The changes involved will place Unit 2 in compliance with the new methodology for the calculation of heatup, and cooldown curves outlined in Regulatory Guide 1.99 Revision 2. This change also addresses the impact of this reanalysis on the cold overpressurization systems setpoints. The Unit 1 curves have had their effective dates revised to ensure compliance until new curves are generated.

The proposed change does not involve a significant reduction in a margin of safety. The margin to safety will remain unchanged. The reduced applicability date for the Unit 1 curve will ensure all current limitations are met up to and including that date. The changes made to the Unit 2 curves are in accordance with the new methodology outlined in Regulatory Guide 1.99 Revision 2. The Unit 2 curves are effectively being shifted in the more conservative direction, and as such will not reduce the margin to safety.

ATTACHMENT D

ENVIRONMENTAL ASSESSMENT STATEMENT

Braidwood Station has evaluated the proposed amendment against the criteria for and identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.21. It has been determined that the proposed change meets the criteria for a categorical exclusion as provided for under 10 CFR 51.22(c)(9). This determination is based on the fact that this change is being proposed as an amendment to a license issued pursuant to 10 CFR 50, and the change involves no significant hazards considerations. There is no change in the amount or type of releases made offsite, and there is no significant increase in individual or cumulative occupational radiation exposure

ATTACHMENT E

ADDITIONAL INFORMATION

The following was asked by the NRC with respect to the Byron submittal. This information is being provided for the Braidwood submittal.

1. What is the vessel inside radius at beltline?

   Unit 1 - 86.625"

   Unit 2 - 86.5"

2. What is the fluence rate for Braidwood Unit 1 @ 4.5 EFPY (beltline)?

   5×10^{18} n/cm^2

3. What is the Braidwood vessels fabricator?

   Babcock & Wilcox

4. What is the Braidwood 1 thickness a. beltlin·

   8.5"

5. When were the irradiation coupons taken from Braidwood Unit 1?

   During A1R01, September, 1989.

6. What is the Braidwood 1 new adjusted RT_{NDT_O} at 1/4T and 3/4T for 4.5 EFPY?

   1/4T = 109°F
   3/4T = 84°F.