June 27, 1991

Docket Nos. STN 50-456
and STN 50-457

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Mr. Thomas J. Kovach
Nuclear Licensing Manager
Commonwealth Edison Company-Suite 300
OPUS West III
1400 OPUS Place
Downers Grove, Illinois 60515

Dear Mr. Kovach:

SUBJECT: ISSUANCE OF AMENDMENTS (TAC NOS. 79440 AND 79441)

The Commission has issued the enclosed Amendment No. 30 to Facility Operating License No. NPF-72 and Amendment No. 30 to Facility Operating License No. NPF-77 for the Braidwood Station, Unit Nos. 1 and 2, respectively. The amendments are in response to your application dated December 19, 1990.

The amendments revise the Braidwood Station, Unit 2, Technical Specifications (TS) by changing the heatup and cooldown curves, the power operated relief valve Low Temperature Overpressure Protection (LTOP) setpoints and their bases. These amendments also shorten the applicability of the Braidwood, Unit 1, heatup and cooldown curves.

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

Sincerely,

Original Signed By:

Robert M. Pulsifer, Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 30 to NPF-72
2. Amendment No. 30 to NPF-77
3. Safety Evaluation

cc w/enclosures:
See next page

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PDR ADOCK 05000456
P PDR

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OFC	:LA:PDIII-2	:PM:PDIII-2	:BC/EMCB	:BC/SRXB	:D:PDIII-2	:OGC
NAME	:CMOORE	:RPULSIFER:ta	:CYCHENG	:RJONES	:RBARRETT	: Scott
DATE	:6/13/91	:6/13/91	:6/14/91	:6/13/91	:6/26/91	:6/18/91

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Document Name: AMENDMENT 79440/41

380067

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

COMMONWEALTH EDISON COMPANY

DOCKET NO. STN 50-456

BRAIDWOOD STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 30
License No. NPF-72

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

   A. The application for amendment by Commonwealth Edison Company (the licensee) dated December 19, 1990, 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-72 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. 30 and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. The licensee 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

Richard J. Barrett, Director
Project Directorate III-2
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: June 27, 1991

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

COMMONWEALTH EDISON COMPANY

DOCKET NO. STN 50-457

BRAIDWOOD STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 30
License No. NPF-77

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

   A. The application for amendment by Commonwealth Edison Company (the licensee) dated Decemner 19, 1990, 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 1;

   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-77 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. 30 and the Environmental Protection Plan contained in Appendix B, both of which were attached to License No. NPF-72, dated July 2, 1987, are hereby incorporated into this license. The licensee 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 if its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

R J Barrett

Richard J. Barrett, Director
Project Directorate III-2
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: June 27, 1991

ATTACHMENT TO LICENSE AMENDMENT NOS. 30 AND 30

FACILITY OPERATING LICENSE NOS. NPF-72 AND NPF-77

DOCKET NOS. STN 50-456 AND STN 50-457

Replace the following pages of the Appendix "A" Technical Specifications with the attached pages. The revised pages are identified by amendment number and contain vertical lines indicating the area of change. Overleaf pages identified by an asterisk are provided for convenience.

Remove Pages	Insert Pages
VII	VII
VIII	VIII
*3/4 4-31	*3/4 4-31
3/4 4-32	3/4 4-32
3/4 4-33	3/4 4-33
3/4 4-34	3/4 4-34
3/4 4-35	3/4 4-35
3/4 4-36	3/4 4-36
3/4 4-39	3/4 4-39
3/4 4-40	3/4 4-40a
	3/4 4-40b
3/4 4-41	3/4 4-41
B 3/4 4-7	B 3/4 4-7
B 3/4 4-8	B 3/4 4-8
B 3/4 4-9	B 3/4 4-9
*B 3/4 4-10	*B 3/4 4-10
B 3/4 4-11	B 3/4 4-11
B 3/4 4-12	B 3/4 4-12
B 3/4 4-15	B 3/4 4-15
B 3/4 4-16	B 3/4 4-16

*overleaf pages provided for convenience.

TABLE 4.4-4 (Continued)

TABLE NOTATIONS

#Until the specific activity of the Reactor Coolant System is restored within its limits.

*Sample to be taken after a minimum of 2 EFPD and 20 days of POWER OPERATION have elapsed since reactor was last subcritical for 48 hours or longer.

**A gross radioactivity analysis shall consist of the quantitative measurement of the total specific activity of the reactor coolant except for radionuclides with half-lives less than 10 minutes and all radioiodines. The total specific activity shall be the sum of the degassed beta-gamma activity and the total of all identified gaseous activities in the sample within 2 hours after the sample is taken and extrapolated back to when the sample was taken. Determination of the contributors to the gross specific activity shall be based upon those energy peaks identifiable with a 95% confidence level. The latest available data may be used for pure beta-emitting radio-nuclides.

***A radiochemical analysis for $\bar{E}$ shall consist of the quantitative measurement of the specific activity for each radionuclide, except for radionuclides with half-lives less than 10 minutes and all radio-iodines, which is identified in the reactor coolant. The specific activities for these individual radionuclides shall be used in the determination of $\bar{E}$ for the reactor coolant sample. Determination of the contributors to $\bar{E}$ shall be based upon these energy peaks identifiable with a 95% confidence level.

REACTOR COOLANT SYSTEM

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

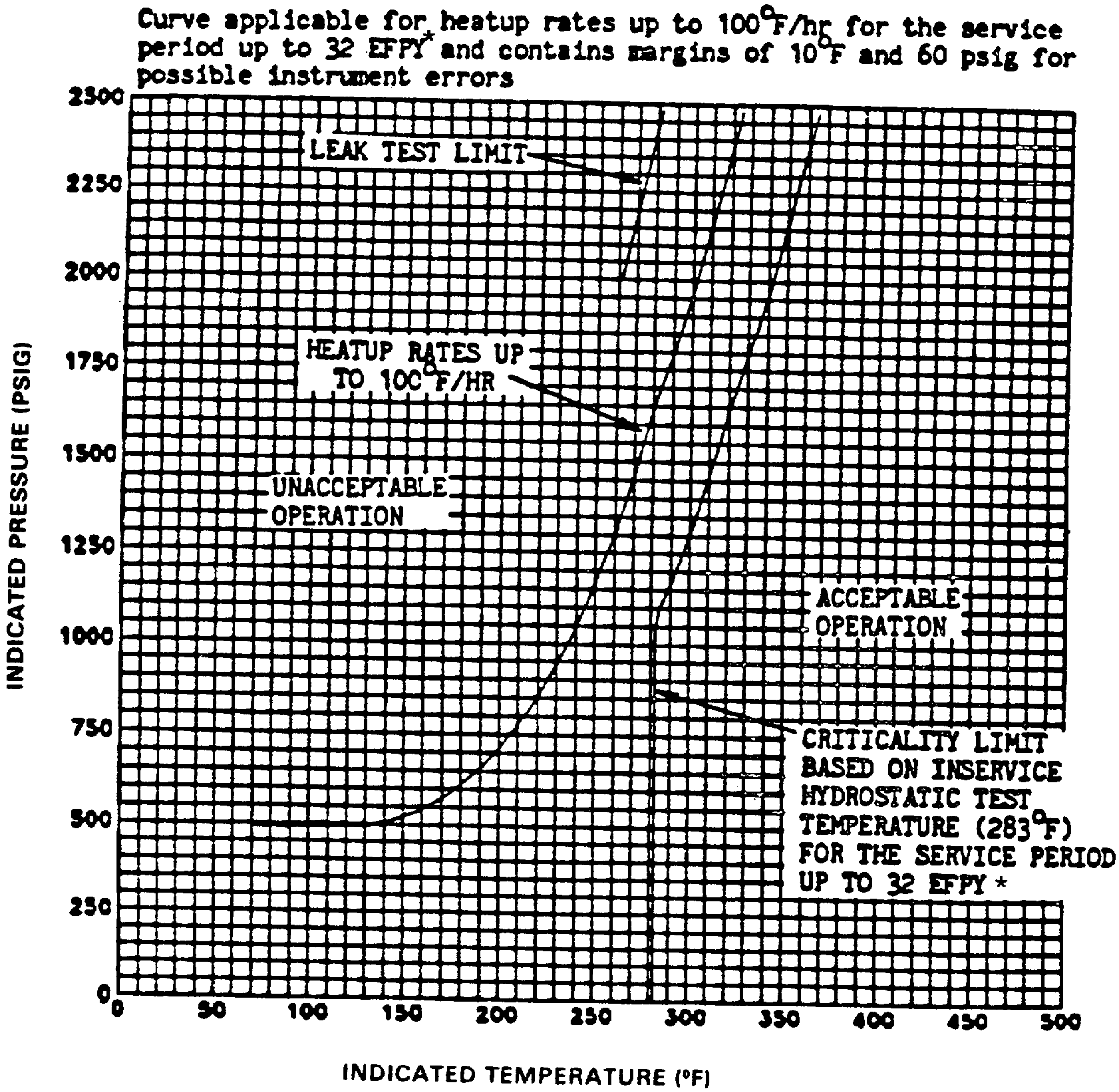


FIGURE 3.4-2a
REACTOR COOLANT SYSTEM HEATUP LIMITATIONS
APPLICABLE UP TO 32 EFPY*(UNIT 1)

*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%.

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.

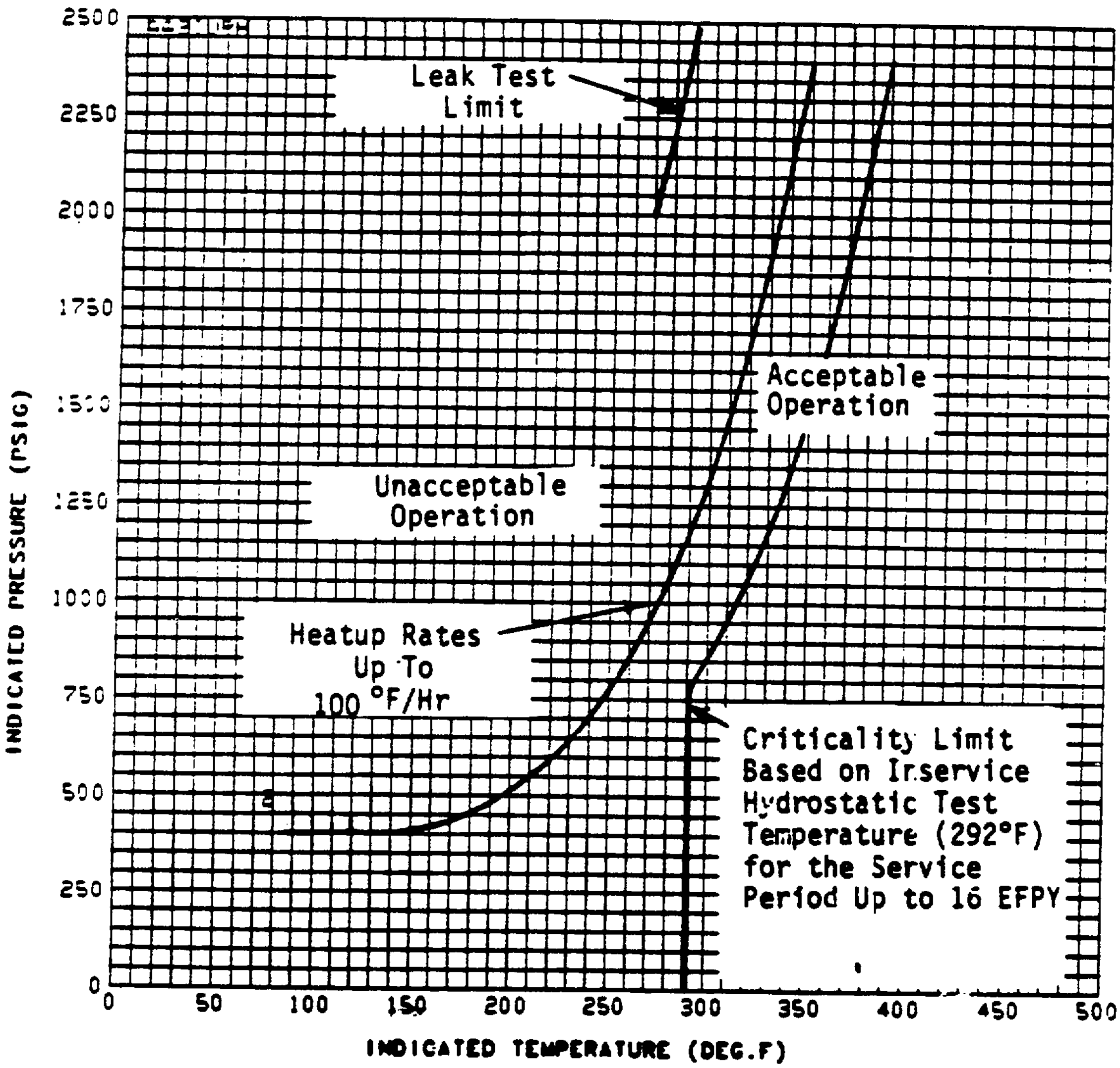


FIGURE 3.4-2b

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

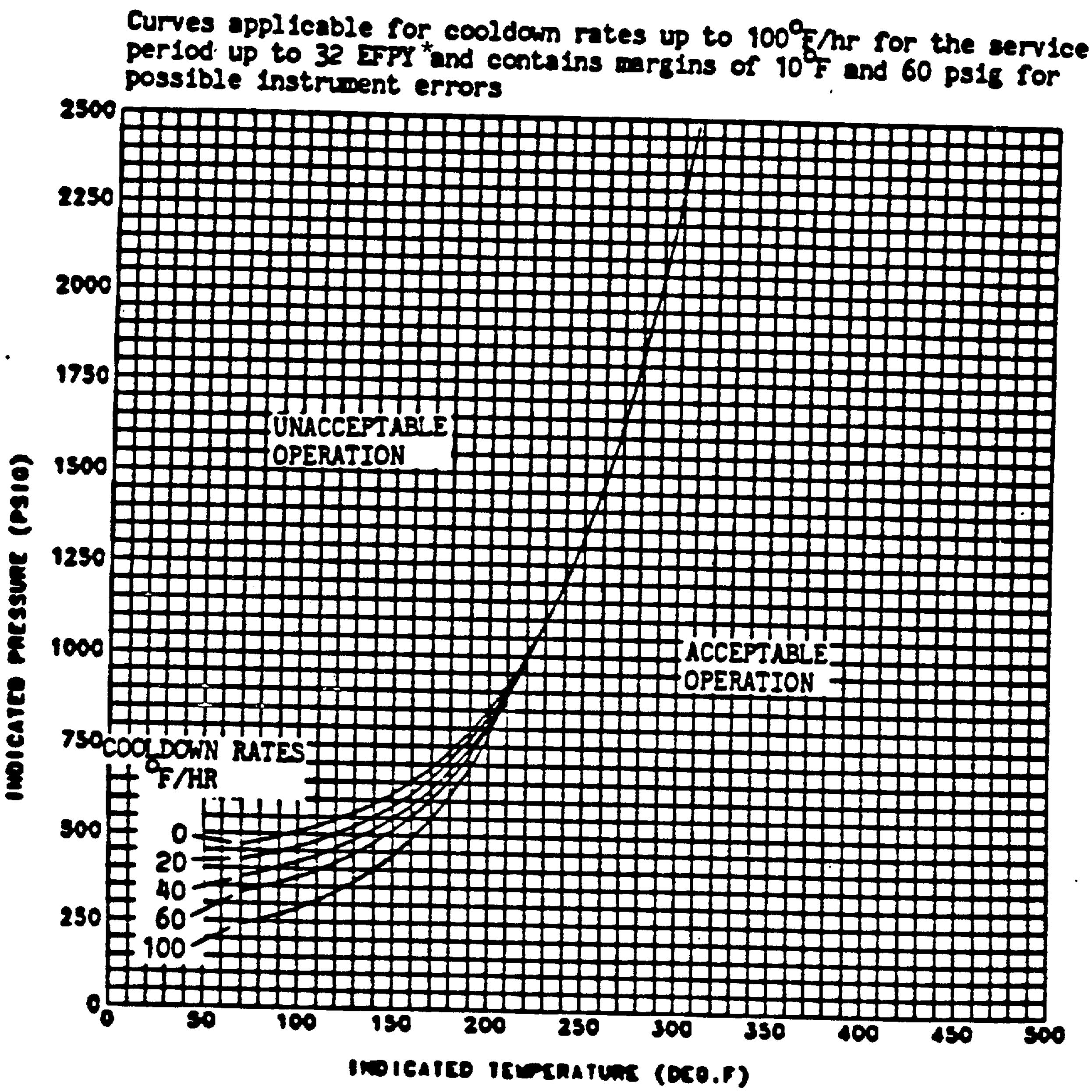


FIGURE 3.4-3a

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%.

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.

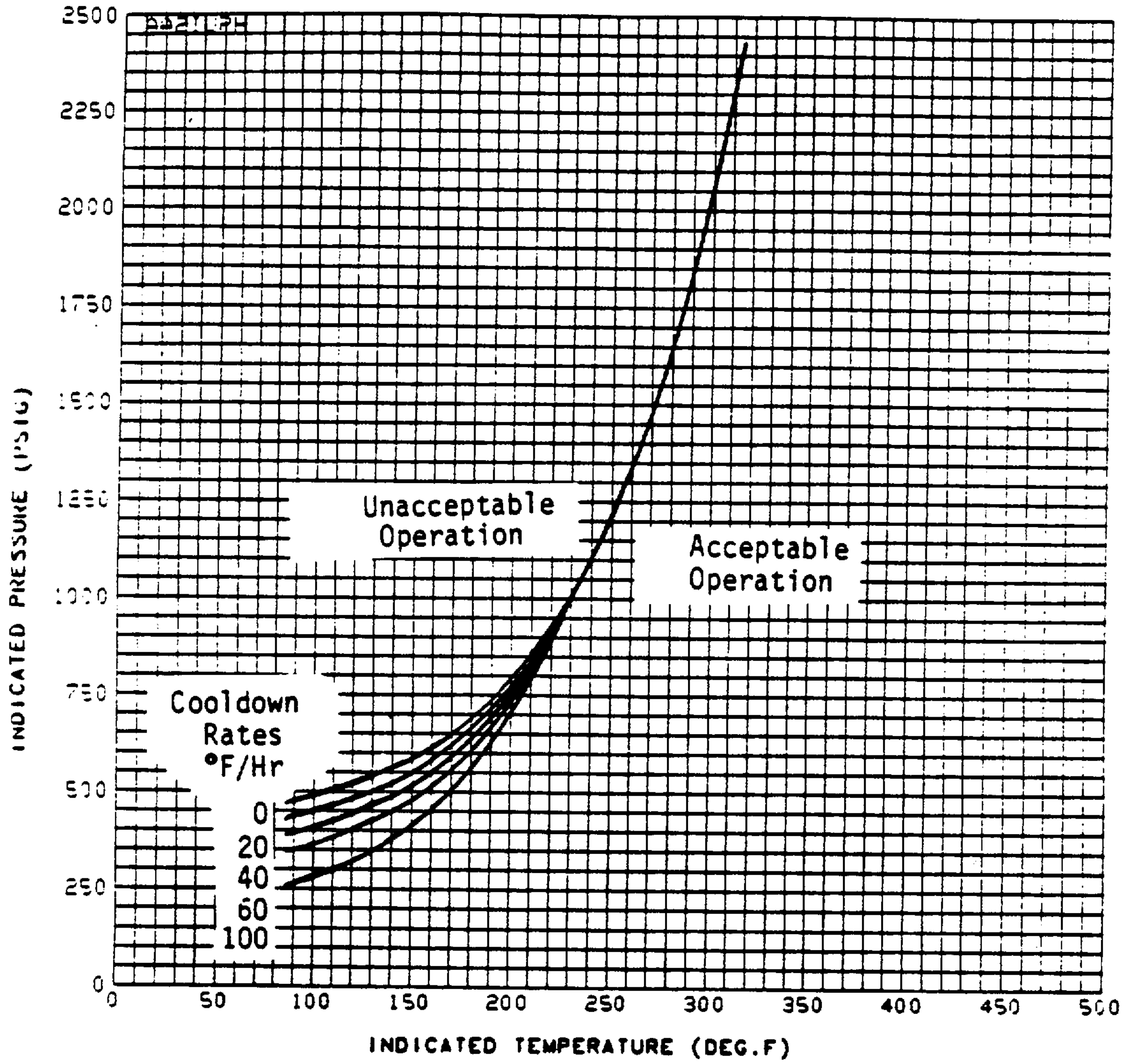


FIGURE 3.4-3b

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

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 3.4-4a 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.

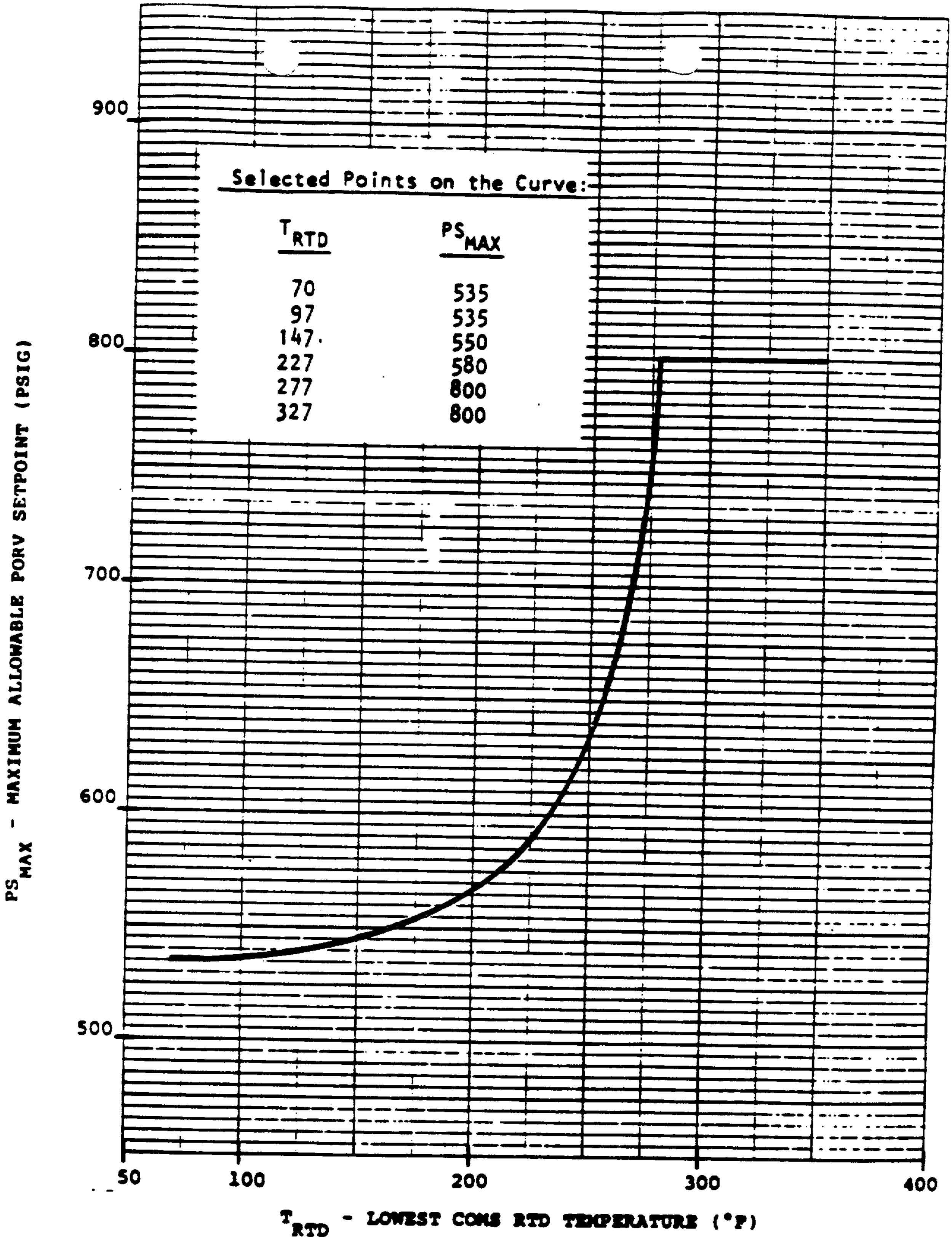
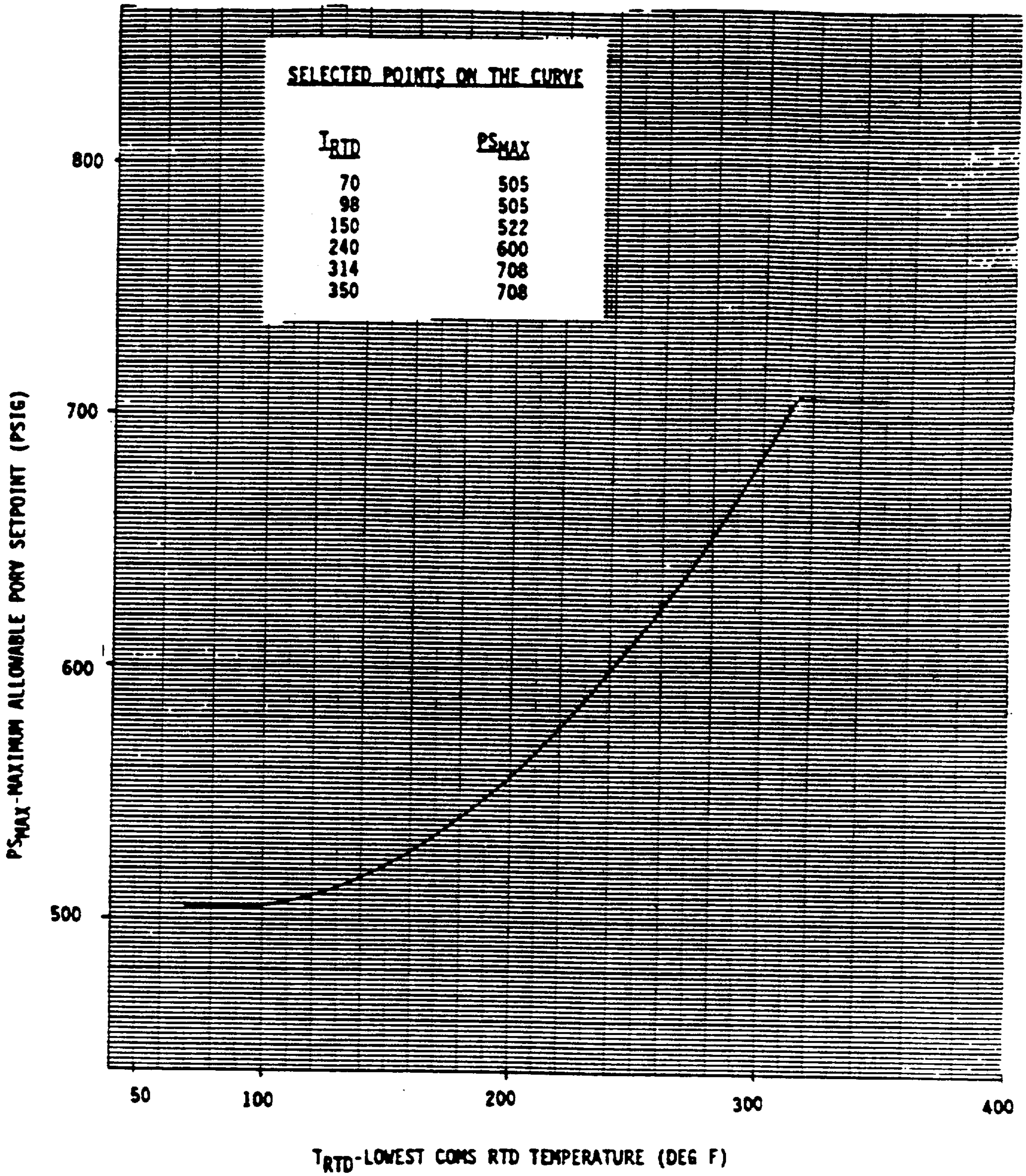


FIGURE 3.4-4a

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 content of 0.05 wt%.



T_{RTD}	PS_{MAX}
70	505
98	505
150	522
240	600
314	708
350	708

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 RH8708B verify at least once per 72 hours that valves RH8702A and RH8702B are open.

b. For RHR suction relief valve RH8708A verify at least once per 72 hours that valves RH8701A and RH8701B 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.

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 3.4-2a (3.4-2b) and 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 3.4-2a (3.4-2b) and 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 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 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 3.4-2a (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. 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.

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.

*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.

PRESSURE/TEMPERATURE LIMITS (Continued)

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.

The general method for calculating heatup and cooldown limit curves is based upon the principles of the linear elastic fracture mechanics (LEFM) technology. In the calculation procedures a semi-elliptical surface defect with a depth of one-quarter of the wall thickness, T, and a length of 3/2T is assumed to exist at the inside of the vessel wall as well as at the outside of the vessel wall. The dimensions of this postulated crack, referred to in Appendix G of ASME Section III as the reference flaw, amply exceed the current capabilities of inservice inspection techniques. Therefore, the reactor operation limit curves developed for this reference crack are conservative and provide sufficient safety margins for protection against non-ductile failure. To assure that the radiation embrittlement effects are accounted for in the calculation of the limit curves, the most limiting value of the nil-ductility reference temperature, RT_{NDT}, is used and this includes the radiation-induced shift, ΔRT_{NDT}, corresponding to the end of the period for which heatup and cooldown curves are generated.

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, the governing equation for the heatup-cooldown analysis is defined in Appendix G of the ASME Code as follows:

$$C\ K_{IM} + K_{It} \leq K_{IR} \qquad (2)$$

Where: K_{IM} = the stress intensity factor caused by membrane (pressure) stress,

K_{It} = the stress intensity factor caused by the thermal gradients,

PRESSURE/TEMPERATURE LIMITS (Continued)

K_{IR} = constant provided by the code as a function of temperature relative to the RT_{NDT} of the material,

C = 2.0 for level A and B service limits, and

C = 1.5 for inservice hydrostatic and leak test operations.

At any time during the heatup or cooldown transient, K_{IR} is determined by the metal temperature at the tip of the postulated flaw, the appropriate value for RT_{NDT}, and the reference fracture toughness curve. The thermal stresses resulting from temperature gradients through the vessel wall are calculated and then the corresponding thermal stress intensity factor, K_{It}, for the reference flaw is computed. From Equation (2) the pressure stress intensity factors are obtained and, from these, the allowable pressures are calculated.

COOLDOWN

For the calculation of the allowable pressure versus coolant temperature during cooldown, the Code reference flaw is assumed to exist at the inside of the vessel wall. During cooldown, the controlling location of the flaw is always at the inside of the wall because the thermal gradients produce tensile stresses at the inside, which increase with increasing cooldown rates. Allowable pressure-temperature relations are generated for both steady-state and finite cooldown rate situations. From these relations, composite limit curves are constructed for each cooldown rate of interest.

The use of the composite curve in the cooldown analysis is necessary because control of the cooldown procedure is based on measurement of reactor coolant temperature, whereas the limiting pressure is actually dependent on the material temperature at the tip of the assumed flaw. During cooldown, the 1/4T vessel location is at a higher temperature than the fluid adjacent to the vessel ID. This condition, of course, is not true for the steady-state situation. It follows that at any given reactor coolant temperature, the ΔT developed during cooldown results in a higher value of K_{IR} at the 1/4T location for finite cooldown rates than for steady-state operation. Furthermore, if conditions exist such that the increase in K_{IR} exceeds K_{It}, the calculated allowable pressure during cooldown will be greater than the steady-state value.

The above procedures are needed because there is no direct control on temperature at the 1/4T location; therefore, allowable pressures may unknowingly be violated if the rate of cooling is decreased at various intervals along a cooldown ramp. The use of the composite curve eliminates this problem and assures conservative operation of the system for the entire cooldown period.

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	Average Shelf Energy 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	Average Shelf Energy 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	49D1066-1-1	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(.67)	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)

A notch in the cooldown curve of Figure 3.4-3a for Unit 1 (3.4-3b for Unit 2) may be present 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 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

PRESSURE/TEMPERATURE LIMITS (Continued)

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.

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 3.4-4a (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 Appendix G reactor vessel NDT limits.

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).

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 30 TO FACILITY OPERATING LICENSE NO. NPF-72

AND AMENDMENT NO. 30 TO FACILITY OPERATING LICENSE NO. NPF-77

COMMONWEALTH EDISON COMPANY

BRAIDWOOD STATION, UNIT NOS. 1 AND 2

DOCKET NOS. STN 50-456 AND STN 50-457

1.0 INTRODUCTION

In response to Generic Letter (GL) 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Effect on Plant Operations," the Commonwealth Edison Company (the licensee) requested permission to revise the pressure/temperature (P/T) limits in the Braidwood Station, Unit Nos. 1 and 2, Technical Specifications (TS) Section 3.4. The request was documented in a letter from the licensee dated December 19, 1990. The proposed P/T limits requested were for 4.5 effective full power years (EFPY) for heatup, leak test, and criticality and 12 EFPY for cooldown for Unit 1 and 16 EFPY for heatup, leak test, criticality, and cooldown for Unit 2. The current P/T limits were developed using Regulatory Guide (RG) 1.99, Revision 2. GL 88-11 recommends RG 1.99, Revision 2, be used in calculating P/T limits, unless the use of different methods can be justified.

To evaluate the P/T limits, the staff uses the following NRC regulations and guidance: Appendices G and H of 10 CFR Part 50; the American Society for Testing and Materials (ASTM) Standards and the American Society of Mechanical Engineers (ASME) Code, which are referenced in Appendices G and H; 10 CFR 50.36(c)(2); RG 1.99, Revision 2; Standard Review Plan (SRP) Section 5.3.2; and GL 88-11.

Each licensee authorized to operate a nuclear power reactor is required by 10 CFR 50.36 to provide TS for the operation of the plant. In particular, 10 CFR 50.36(c)(2) requires that limiting conditions for operation be included in the TSs. The P/T limits are among the limiting conditions for operation in the TSs for all commercial nuclear plants in the United States. 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). GL 88-11 requested that licensees and permittees use the methods in RG 1.99, Revision 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

In a CECo submittal dated January 17, 1986, which was reviewed by the staff in connection with the 10 CFR 50.61 requirements, we found the peak inside surface pressure vessel fluence value of 2.8×10^{19} n/cm² acceptable for 32 EFPY of operation. This value was found acceptable on the basis of the methodology, cross sections and neutron source distribution data used for the calculation. In particular, the source distribution was derived from statistical studies of long term operation of Westinghouse four-loop plants. However, the removal and measurement of the material surveillance capsule U from the sister plant (Byron Unit 1) yielded a more precise value for the source distribution (Reference 4). This combined with improved calculational techniques and improved iron cross sections yielded a new peak azimuthal value of 3.17×10^{19} n/cm² for 32 EFPYs of operation. The methodology satisfies all of the staff's requirements for such estimates and is acceptable.

In addition, we examined the surveillance capsule U reports for Braidwood, Unit Nos. 1 and 2 (References 5 and 6) and found that both yielded a projected peak fluence value of 2.83×10^{19} n/cm². Thus, the licensee chose the more conservative value of 3.17×10^{19} m/cm² from the Byron 1 plant.

The staff also evaluated the effect of neutron irradiation embrittlement on each beltline material in the Braidwood 1 and 2 reactor vessels. The amount of irradiation embrittlement was calculated in accordance with RG 1.99, Revision 2.

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 initial USE is circumferential weld WF562 with 80 ft-lb. Using the method in RG 1.99, Revision 2, the predicted Charpy USE of the weld metal at the end-of-life will be 60 ft-lb. This is greater than 50 ft-lb at Braidwood, Unit Nos. 1 and 2, and, therefore, is acceptable.

2.1 Braidwood Unit 1

The staff has determined that the material with the highest ART at 4.5 and 12 EFPY was the circumferential weld between the upper and lower shells (WF562) with 0.04% copper (Cu), 0.67% (Ni), and an initial RT_{ndt} of 40°F.

The licensee removed one surveillance capsule from Braidwood Unit 1. The results were published in Westinghouse report WCAP-12685. The surveillance capsule contained Charpy impact specimens and tensile specimens made from base metal, weld metal, and HAZ metal.

For the limiting beltline, circumferential weld (WF562), the staff calculated the ART to be 109.2°F at 1/4T (T = reactor vessel beltline thickness) and 84.2°F for 3/4T at 4.5 EFPY. The staff used a neutron fluence of 2.67E18 n/cm² at 1/4T and 9.6E17 n/cm² at 3/4T. For the same weld, the staff calculated the ART to be 137.8°F at 1/4T and 108.2°F for 3/4T at 12 EFPY. The staff used a neutron fluence of 7.13E18 n/cm² at 1/4T and 2.57E18 n/cm² at 3/4T for 12 EFPY. The ART was determined using Section 1 of RG 1.99, Revision 2, because only one surveillance capsule has been withdrawn from Braidwood Unit 1.

The licensee used the method in RG 1.99 Revision 2, to calculate an ART of 109°F at 4.5 EFPY at 1/4T for the same limiting weld metal. The staff judges that a difference of 0.2°F is acceptable. Substituting the ART of 109.2°F into equations in SRP 5.3.2, the staff verified that the proposed P/T limits for heatup and hydrotest meet the beltline material requirements in Appendix G of 10 CFR Part 50.

For the cooldown limits, the licensee calculated an ART of 138°F at 12 EFPY at 1/4T for the same limiting weld metal. The staff judges that the licensee's ART of 138°F is more conservative than the staff's ART of 137.8°F, and it is acceptable. Substituting the ART of 138°F into equations in SRP 5.3.2, the staff verified that the proposed P/T limits for cooldown 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 -10°F, the staff has determined that the proposed P/T limits satisfy Section IV.2 of Appendix G.

The staff concludes for Braidwood Unit 1 that the proposed heatup, leak test, and criticality limits are valid up to 4.5 EFPY and cooldown limits are valid up to 12 EFPY because the limits conform to the requirements of

Appendices G and H of 10 CFR Part 50. The limits also satisfy GL 88-11 because the licensee used the method in RG 1.99, Revision 2 to calculate the ART. Hence, the proposed P/T limits may be incorporated into the Braidwood Unit 1 TSs.

2.2 Braidwood Unit 2

The staff has determined that the material with the highest ART at 16 EFPY was the circumferential weld between the upper and lower shells (WF562) with 0.04% copper (Cu), 0.67% nickel (Ni), and an initial RT_{ndt} of 40°F.

The licensee removed one surveillance capsule from Braidwood Unit 2. The results were published in Westinghouse report WCAP-12845.

For the limiting beltline material, circumferential weld WF562, the staff calculated the ART to be 146.6°F at 1/4T (T = reactor vessel beltline thickness) and 116.2°F for 3/4T at 16 EFPY. The staff used a neutron fluence of 9.54E18 n/cm^2 at 1/4T and 3.44E18 n/cm^2 at 3/4T.

The licensee used the method in RG 1.99, Revision 2, to calculate an ART of 146.5°F at 16 EFPY at 1/4T for the same limiting weld metal. Substituting the ART of 146.6°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 20°F, the staff has determined that the proposed P/T limits satisfy Section IV.2 of Appendix G.

The staff concludes that for Braidwood Unit 2 the proposed P/T limits for the reactor coolant system for heatup, cooldown, leak test, and criticality are valid up to 16 EFPY because the limits conform to the requirements of Appendices G and H of 10 CFR Part 50. The limits also satisfy GL 88-11 because the licensee used the method in RG 1.99, Revision 2 to calculate the ART. Hence, the proposed P/T limits may be incorporated into the Braidwood Unit 2 TSs.

2.3 Braidwood Unit 1 and 2 - Common

Two administrative changes were made in this amendment. A note was deleted from page 3/4 4-41 because it deals with Cycle 1 and is no longer applicable. Table B 3/4.4-1b also corrects two Heat numbers. These changes are acceptable.

3.0 STATE CONSULTATION

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

4.0 ENVIRONMENTAL CONSIDERATION

The amendments change 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. The NRC staff has determined that the amendments involve 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 the amendments involve no significant hazards consideration, and there has been no public comment on such finding (56 FR 11774). Accordingly, the amendments meet 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 the amendments.

5.0 CONCLUSION

The Commission 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 the amendments will not be inimical to the common defense and security or to the health and safety of the public.

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Date: June 27, 1991

6.0 REFERENCES

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

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

3. Letter from A. R. Checca (CECo) to T. E. Murley (USNRC), Subject: Braidwood Station, Units 1 and 2, Application for Amendment to Facility Operating License NPF-72 and NPF-77, NRC Docket No. 50-456 and 50-457, December 19, 1990.

4. WCAP-11651, "Analysis of Capsule U from the Commonwealth Edison Company Byron Unit 1 Reactor Vessel Radiation Surveillance Program," November 1987.

5. WCAP-12685, "Analysis of Capsule U from the Commonwealth Edison Company Braidwood Unit 1, Reactor Vessel Radiation Surveillance Program," August 1990.

6. WCAP-12845, "Analysis of Capsule U from the Commonwealth Edison Company Braidwood Unit 2, Reactor Vessel Radiation Surveillance Program," March 1991.