ATTACHMENT B

MARKED UP PAGES FOR
PROPOSED CHANGES TO APPENDIX A
TECHNICAL SPECIFICATIONS OF
FACILITY OPERATING LICENSES
NPF-72 AND NPF-77

BRAIDWOOD STATION UNITS 1 & 2
REVISED PAGES:

VIII

3/4 4-32*

3/4 4-33 (Figure 3.4-2a)

3/4 4-35 (Figure 3.4-3a)

3/4 4-37 (Table 4.4-5)

3/4 4-39

3/4 4-40 (New page)

3/4 4-40a (Figure 3.4-4a) (New 3/4 4-40 page)

3/4 4-40b (Figure 3.4-4b) (eliminated, new 3/4 4-40a page)

B 3/4 4-8

*NOTE: THESE PAGES HAVE NO CHANGES BUT ARE INCLUDED FOR CONTINUITY.

k:nla:brdwd:ltop:18

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).

REPLACE THIS PAGE WITH ATTACHED PAGE

Curve applicable for heatup rates up to 100°F/hr for the service period up to 32 EFPY and contains margins of 10°F and 60 psig for possible instrument errors

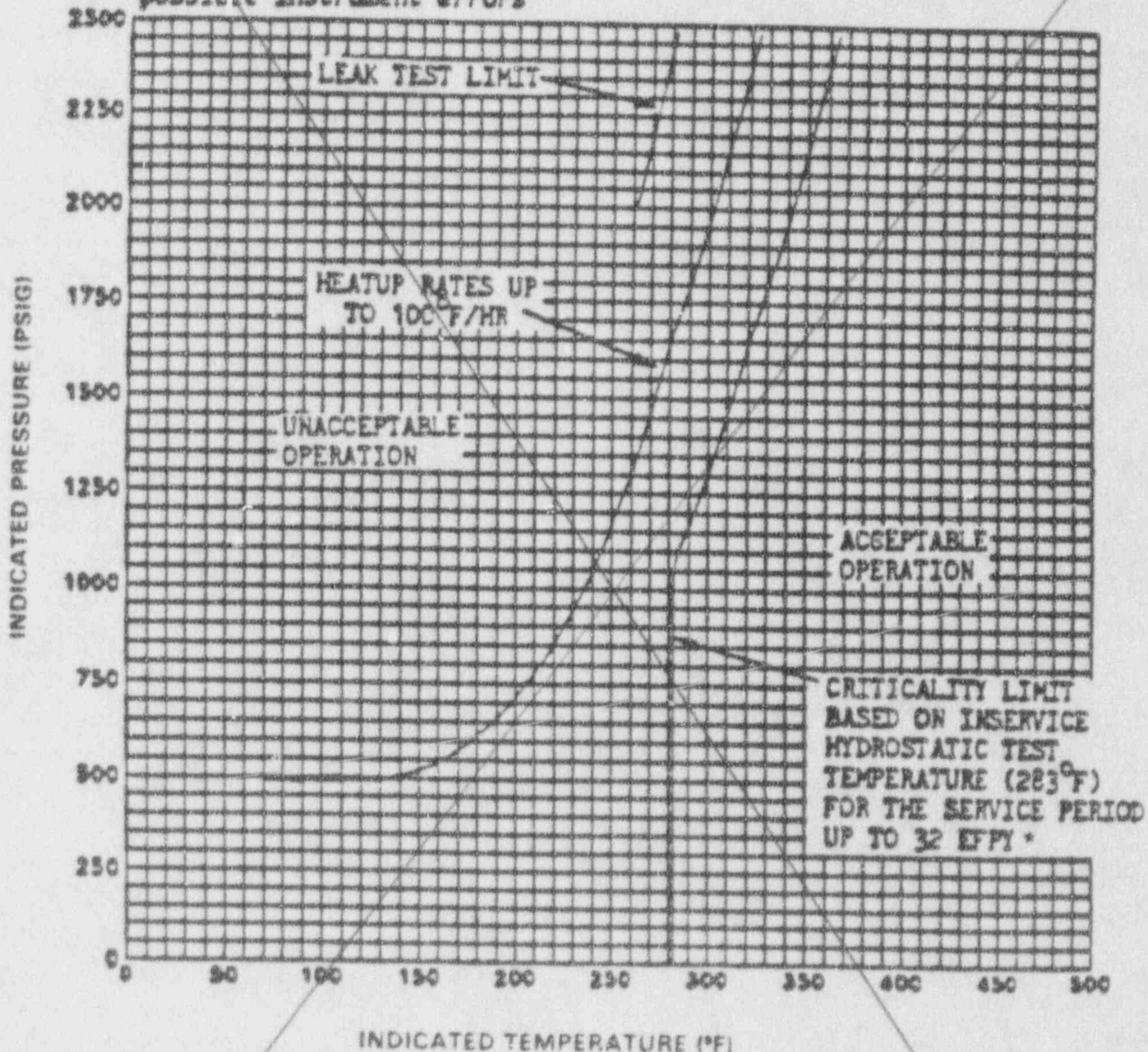

FIGURE 3.4-2a
REACTOR COOLANT SYSTEM HEATUP LIMITATIONS
APPLICABLE UP TO 32 EFPY*(UNIT 1)

*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%.

MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL: CIRCUFERENTIAL WELD
INITIAL RT_{NDT}: 40°F
ART AFTER 32 EFPY: 1/4T, 159°F
3/4T, 135°F

These curves are applicable for heatup rates up to 100°F/hr for the service period up to 32 EFPY and contain margins of 10°F and 60 psig for possible instrument errors.

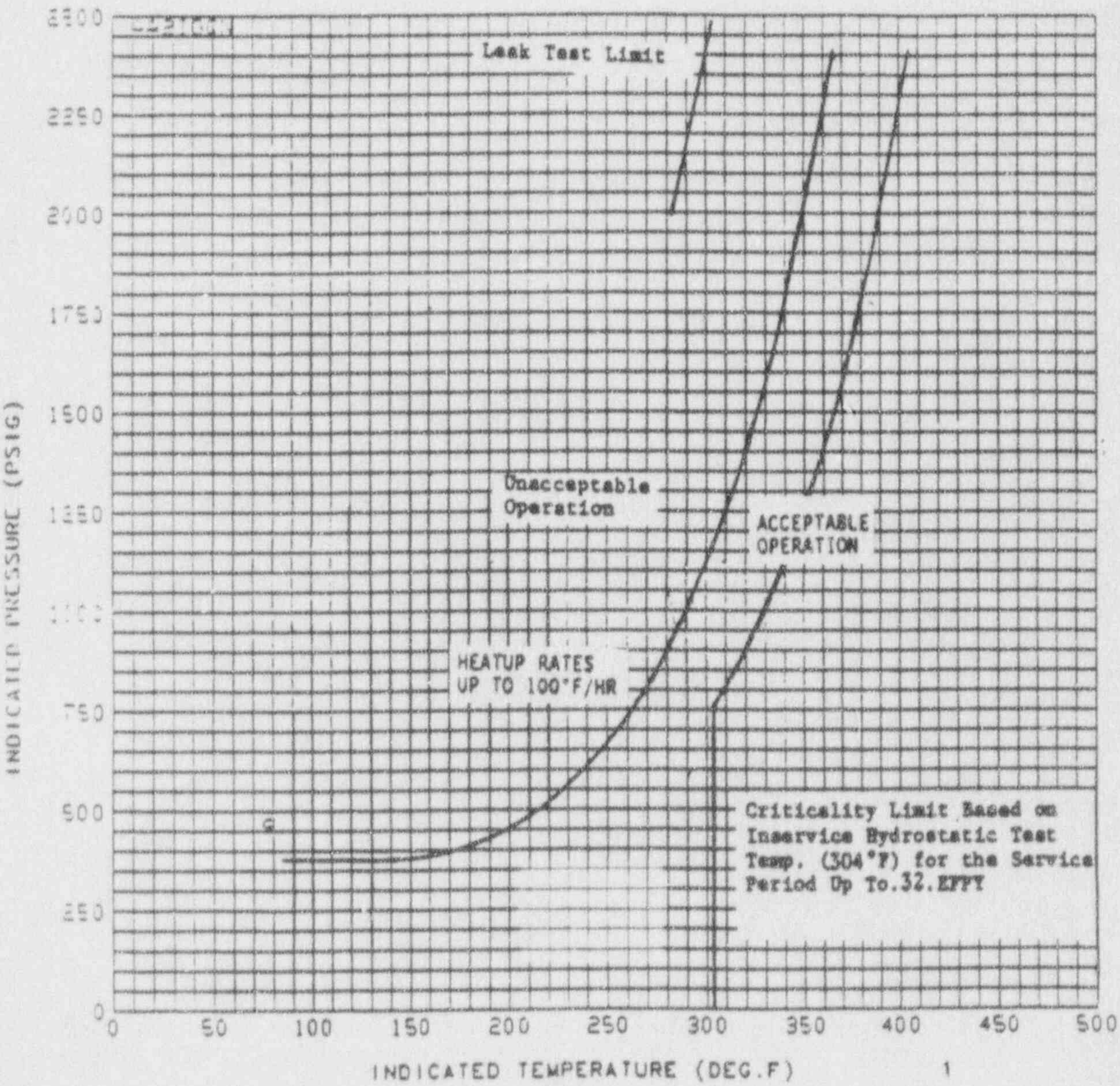

FIGURE 3.4-2a
REACTOR COOLANT SYSTEM HEATUP LIMITATIONS
APPLICABLE UP TO 32 EFPY (UNIT 1)

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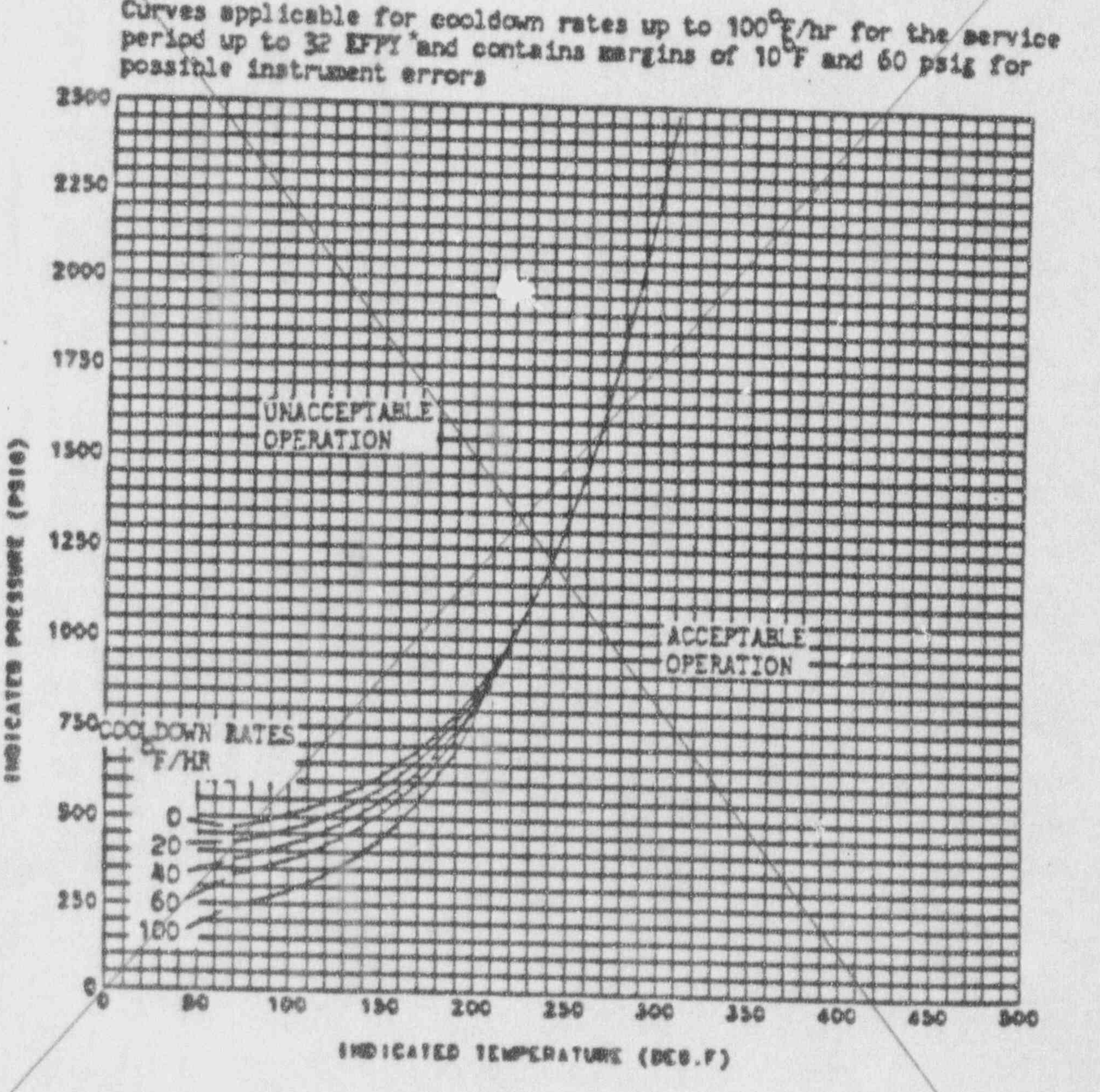

FIGURE 3.4-3a

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%.

MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL:	CIRCUMFERENTIAL WELD
INITIAL RT_{NDT}:	40^0F
ART AFTER 32 EFPY:	1/4T, 159^0F
	3/4T, 135^0F

These curves are applicable for cooldown rates up to 100^0F/hr for the service period up to 32 EFPY and contain margins of 10^0F and 60 psig for possible instrument errors.

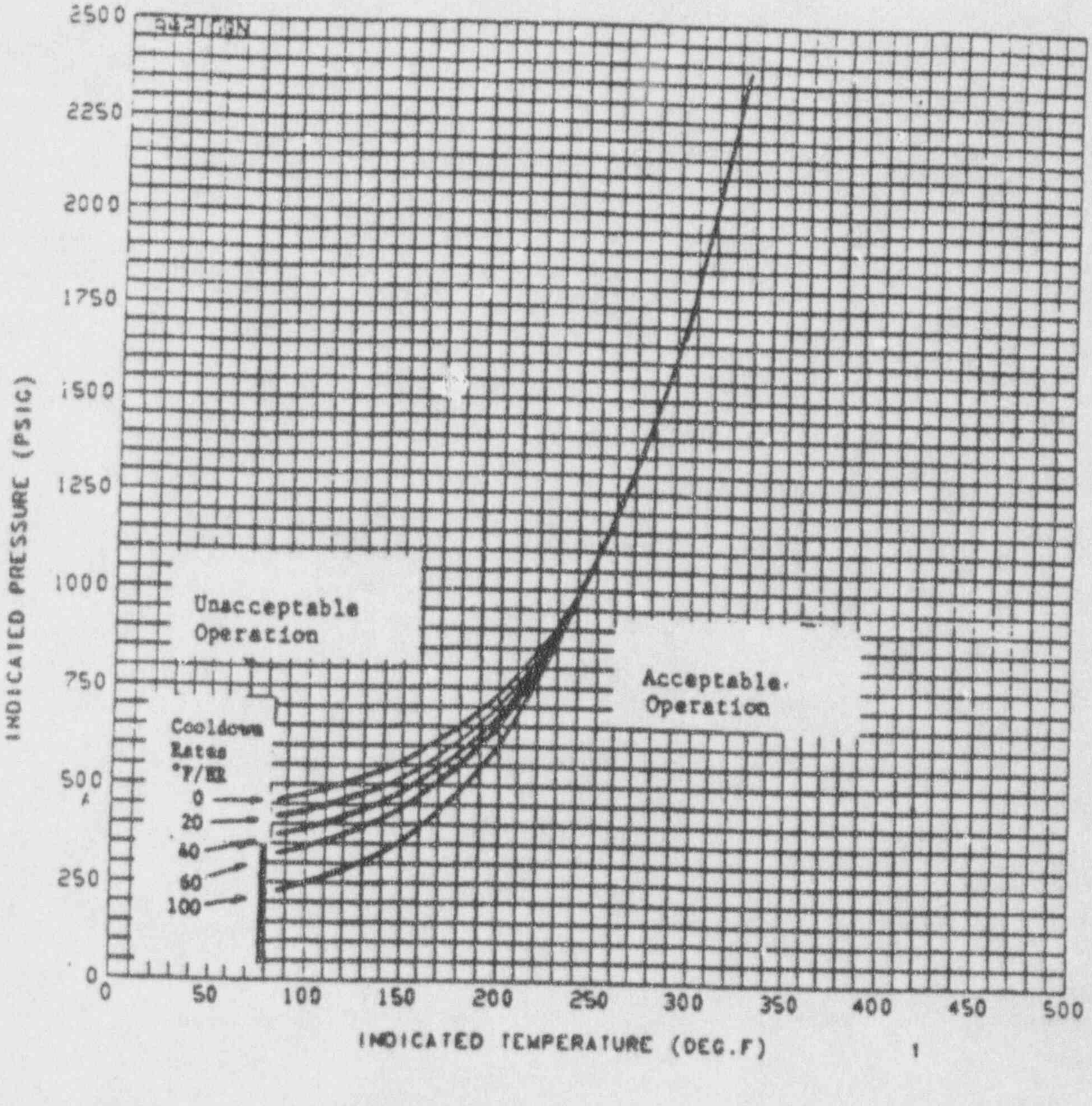

FIGURE 3.4-3a
REACTOR COOLANT SYSTEM COOLDOWN LIMITATIONS
APPLICABLE UP TO 32 EFPY (UNIT 1)

TABLE 4.4-5

REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM - WITHDRAWAL SCHEDULE

UNIT-1

CAPSULE ~~NUMBER~~ DESIGNATION	VESSEL LOCATION	LEAD FACTOR	WITHDRAWAL TIME (EFPY)*
U	58.5°	~~3.6~~ 4.00	~~1st Refueling~~ 1.10 (Removed)
X	238.5°	~~3.6~~ 4.02	~~6~~ 4.50
V	61°	~~3.0~~ 3.75	~~10~~ 9.00
Y	241°	~~3.0~~ 3.75	15.00
W	121.5°	~~3.6~~ 4.02	Standby
Z	301.5°	~~3.6~~ 4.02	Standby

UNIT-2

U	58.5°	4.05	~~1st Refueling~~ 1.15 (Removed)
~~W~~	~~121.5°~~	~~4.05~~	~~4.5~~
X	238.5°	~~4.05~~ 4.02	~~8.0~~ 4.50
V	61°	~~3.37~~ 3.75	~~15.0~~ 9.00
Y	241°	~~3.37~~ 3.75	~~Standby~~ 15.00
Z	301.5°	~~4.05~~ 4.02	Standby
W	121.5°	4.02	Standby

INSERT

*Withdrawal time may be modified to coincide with those refueling outages or reactor shutdowns most closely approaching the withdrawal schedule.

LIMITING CONDITION FOR OPERATION

3.4.9.3 At least two overpressure protection devices shall be OPERABLE, and each device shall be either:

a. A residual heat removal (RHR) suction relief valve with a lift setting of less than or equal to 450 psig, or

b. A power operated relief valve (PORV) with a lift setpoint that varies with RCS temperature which does not exceed the limit established in Figure ~~3.4-4.~~ 3.4-4a for Unit 1 (Figure 3.4-4b for Unit 2)

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

ACTION:

a. With one of the two required overpressure protection devices inoperable in MODE 4, restore two overpressure protection devices 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 one of the two required overpressure protection devices inoperable in MODES 5 or 6, restore two overpressure protection devices to OPERABLE status within 24 hours or vent the RCS through at least a 2 square inch vent within the next 8 hours.

c. With both of the required overpressure protection devices inoperable, depressurize and vent the RCS through at least a 2 square inch vent within 8 hours.

d. With the RCS vented per ACTIONS a, b, or c, verify the vent pathway at least once per 31 days when the pathway is provided by a valve(s) that is locked, sealed, or otherwise secured in the open position; otherwise, verify the vent pathway every 12 hours.

e. In the event either the PORVs, RHR suction relief valves, or the RCS vents 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, RHR suction relief valves, or RCS vents on the transient, and any corrective action necessary to prevent recurrence.

f. The provisions of Specification 3.0.4 are not applicable.

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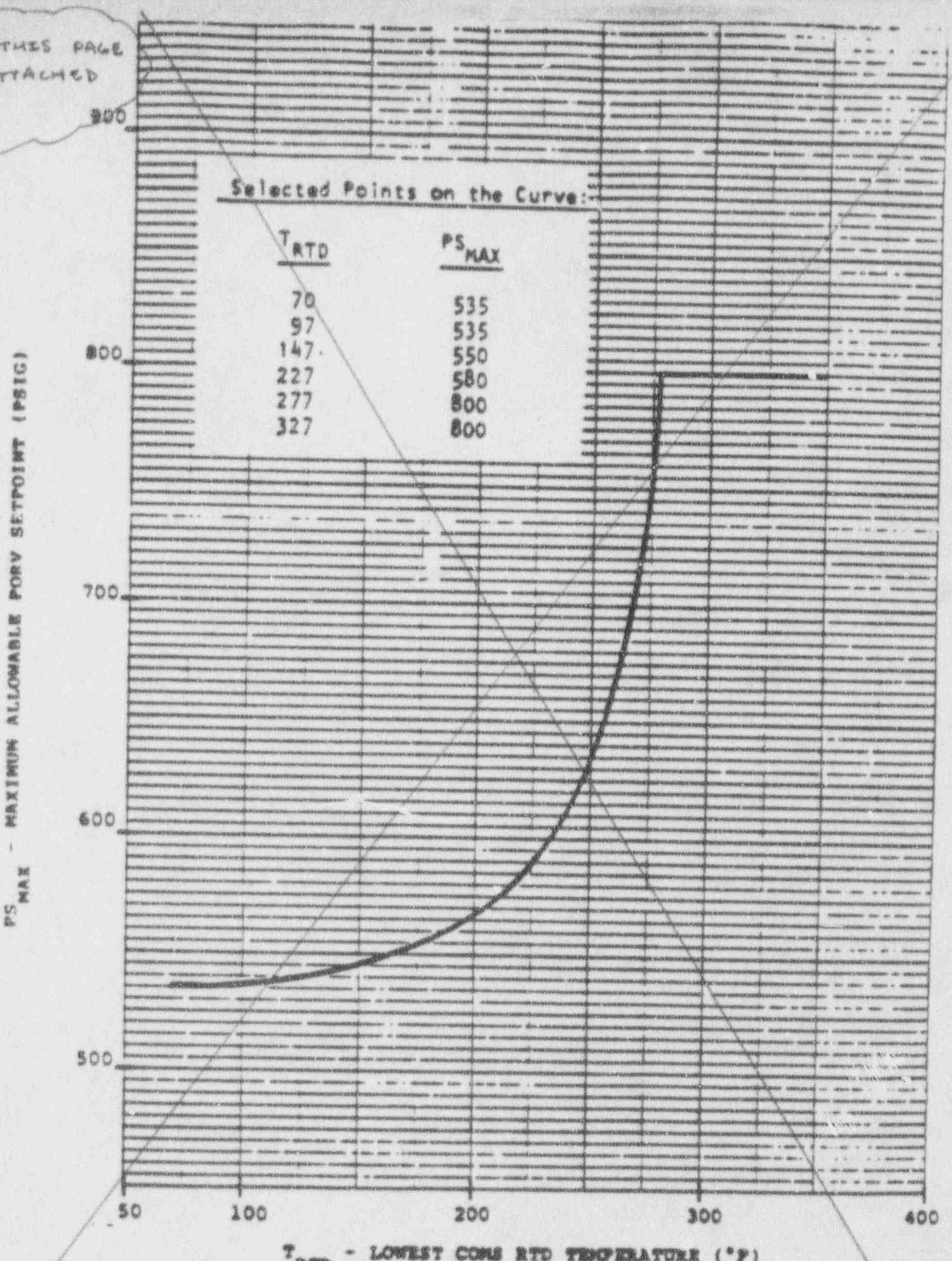

FIGURE 3.4-4a

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%.

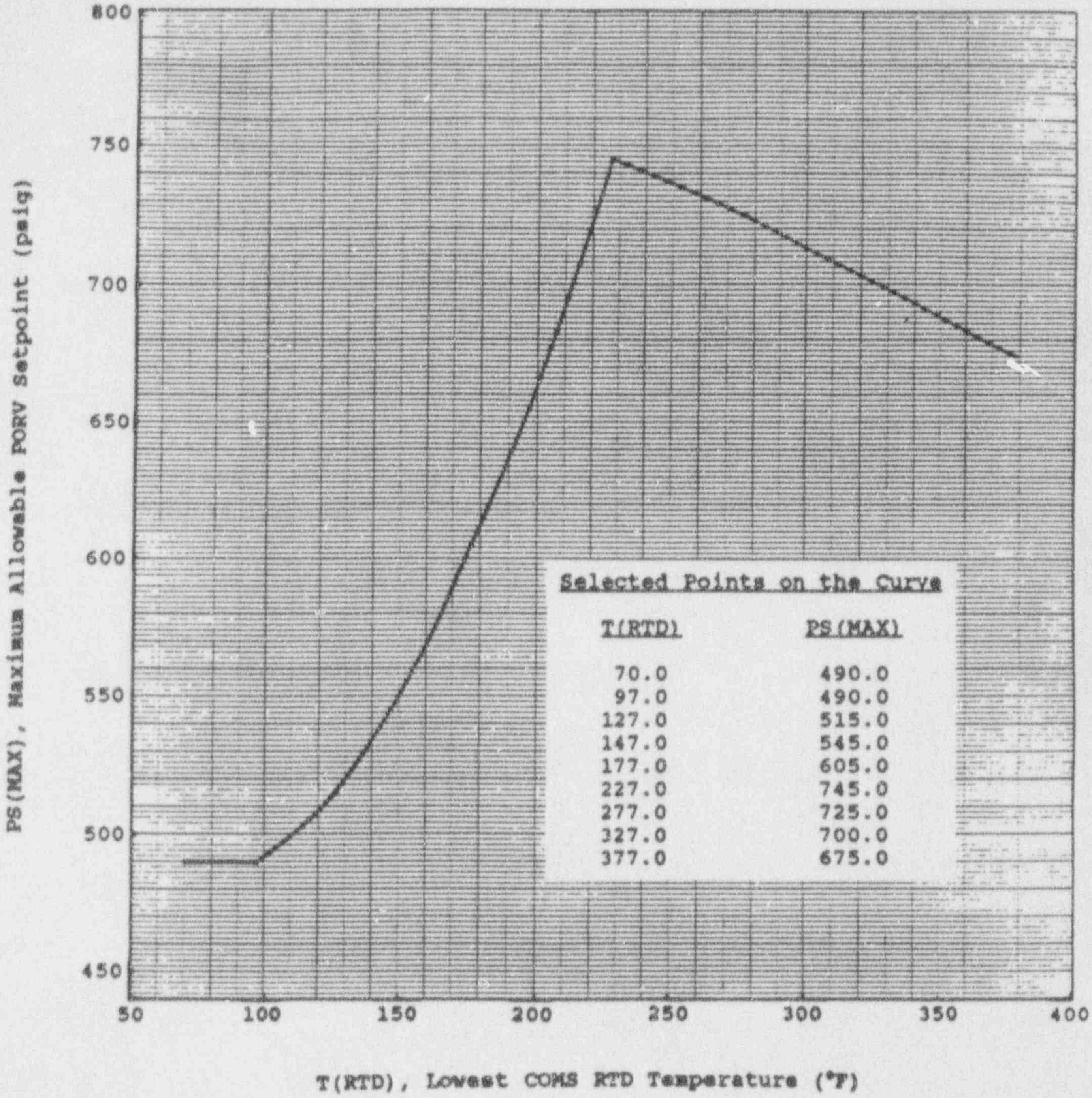

FIGURE 3.4-4a
NOMINAL PORV PRESSURE RELIEF SETPOINT VERSUS
RCS TEMPERATURE FOR THE COLD OVERPRESSURE PROTECTION SYSTEM
APPLICABLE UP TO 32 EFPY (UNIT 1)

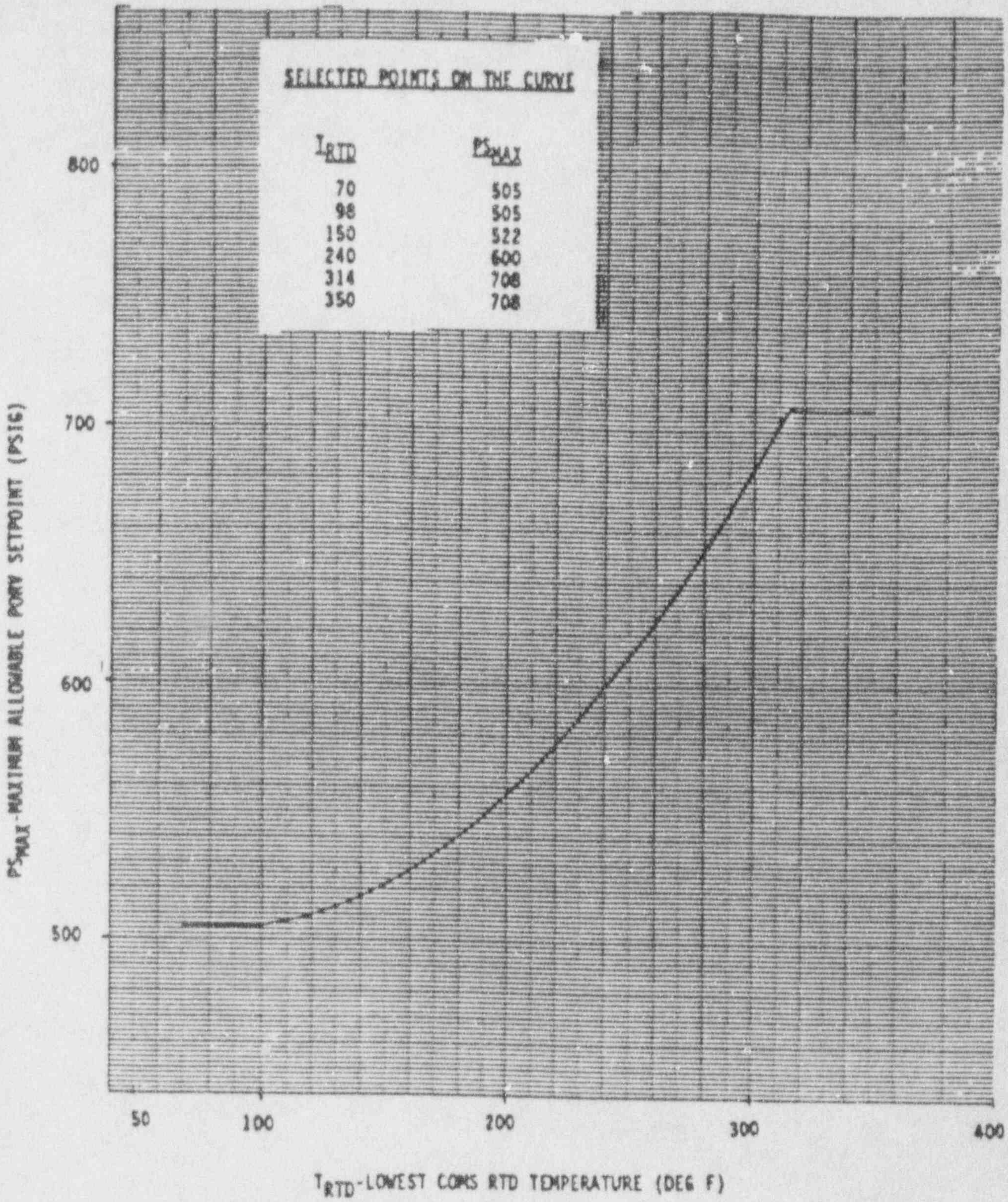

FIGURE 3.4-4b
NOMINAL PORV PRESSURE RELIEF SETPOINT VERSUS
RCS TEMPERATURE FOR THE COLD OVERPRESSURE PROTECTION SYSTEM
APPLICABLE UP TO 16 EFPY (UNIT 2)

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.~~

ATTACHMENT C

EVALUATION OF SIGNIFICANT HAZARDS CONSIDERATIONS FOR
PROPOSED CHANGES TO APPENDIX A
TECHNICAL SPECIFICATIONS OF
FACILITY OPERATING LICENSES
NPF-72 AND NPF-77

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 changes will incorporate new pressure-temperature curves and new low temperature overpressure protection curves for Braidwood Unit 1 covering plant operations through 32 effective full power years (EFPY). The changes will also revise the specimen removal table.

A. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The use of new pressure-temperature limit curves and low temperature overpressure protection curves will not change any postulated accident scenarios. The revised curves were developed using industry standards and regulations which are recognized as being inherently conservative. The pressure-temperature low temperature overpressure curves provide reactor coolant system (RCS) limits to protect the reactor pressure vessel from brittle fracture by clearly separating the region of normal operations from the region where the vessel is subject to brittle fracture. The heatup and cooldown limits are designed to ensure that the 10 CFR 50 Appendix G Pressure Temperature limits for the RCS are not exceeded during any condition of normal operation including anticipated operational occurrences.

General Design Criterion 32 of 10 CFR 50 Appendix A requires that the reactor coolant boundary shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident condition, (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized.

10 CFR 50 Appendix G, "Fracture Toughness Requirements," requires that the effects of changes in the fracture toughness of reactor vessel materials caused by neutron radiation throughout the service life of nuclear reactor be considered in the pressure-temperature limits. The change is used in conjunction with the material initial reference temperature (RT_{NDT}) to establish the limiting pressure-temperature curves. Regulatory Guide 1.99, Rev. 2, contains procedures for calculating the effects of neutron radiation embrittlement of the low-alloy steels currently used for light-water-cooled reactor vessels.

Using the Regulatory Guide 1.99, Revision 2, Braidwood Unit 1 Surveillance Capsule U results, and Appendix G to 10 CFR 50, new Pressure-Temperature curves prepared for the projected reactor vessel exposure at 32 EFPY of operation. These new curves, in conjunction with the heatup and cooldown ranges and the revised Low-Temperature Overpressure Protection System setpoints, provide the required assurance that the reactor pressure vessel is protected from brittle fracture up 32 EFPY of operation. No changes to the design of the facility have been made and no new equipment has been added or removed. The revised analysis and resultant adjustment of the operating limitations provide assurance that the Reactor Coolant System is protected from brittle fracture.

Revising the Reactor Vessel Material Surveillance Program Withdrawal Schedule does not result in the addition or removal of any equipment, or any design changes to the facility. Capsule lead times are revised and, for Braidwood Unit 2, Capsule X will be removed next vice Capsule W. The proposed removal schedules remain consistent with ASTM
185-82.

Therefore, the proposed amendment to the pressure temperature limitations does not involve a significant increase in the probability or consequences of an accident previously evaluated.

B. The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The use of the new pressure-temperature operating limits and the new low temperature overpressure protection curve does not change any postulated accident scenarios. The new curves do not represent any appreciable change in the current methodologies; they merely provide assurance that the Reactor Coolant System is protected from brittle fracture. No new accident or malfunction mechanism is introduced by the amendment and no physical plant changes will result from this amendment.

Revision of the Reactor Vessel Material Surveillance Program Withdrawal Schedule does not introduce a new accident or malfunction mechanism. Capsule lead times are revised, and, other than changing the order of specimen removal, consistent with ASTM 185-82, no physical plant changes will result from this revised schedule.

Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

C. The proposed change does not involve a significant reduction in a margin of safety.

The new pressure-temperature operating limits low temperature overpressure protection curves were generated with the currently accepted conservative methodology using capsule surveillance data. The new pressure-temperature curves were developed using industry standards and regulations (ASME Code Section III, and NRC Regulatory Guide 1.99, Revision 2) which are recognized as being inherently conservative. The use of the new pressure-temperature operating limits and low temperature overpressure protection limits would not change postulated accident scenarios.

The proposed revision to the Reactor Vessel Material Surveillance Program Withdrawal Schedule would not change postulated accident scenarios. Capsule lead times are revised, and, other than changing the order of specimen removal, consistent with ASTM 185-82, no physical plant changes will result from this revised schedule. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

SUMMARY

Based upon the above evaluation, CECo has concluded that these changes involve no significant hazards considerations.

ATTACHMENT D

ENVIRONMENTAL ASSESSMENT FOR
PROPOSED CHANGES TO APPENDIX A
TECHNICAL SPECIFICATIONS OF
FACILITY OPERATING LICENSES
NPF-72 AND NPF-77

Commonwealth Edison has evaluated the proposed amendment and determined that it meets the criteria for a categorical exclusion set forth in 10 CFR 51.22(c)(9). This determination is based upon the following: The proposed amendment changes requirements regarding the installation and use of facility components located within the restricted area (as defined in 10 CFR 20) and surveillance requirements; and the proposed amendment involves no significant hazards considerations, no change in the amount or type of any effluent that may be released offsite, and no increase in individual or cumulative occupational radiation exposure. Pursuant to 10 CFR 51.22(b), neither an environmental impact statement nor an environmental assessment is necessary for the proposed amendment.

APPENDIX 1
FIGURES

CONTENTS

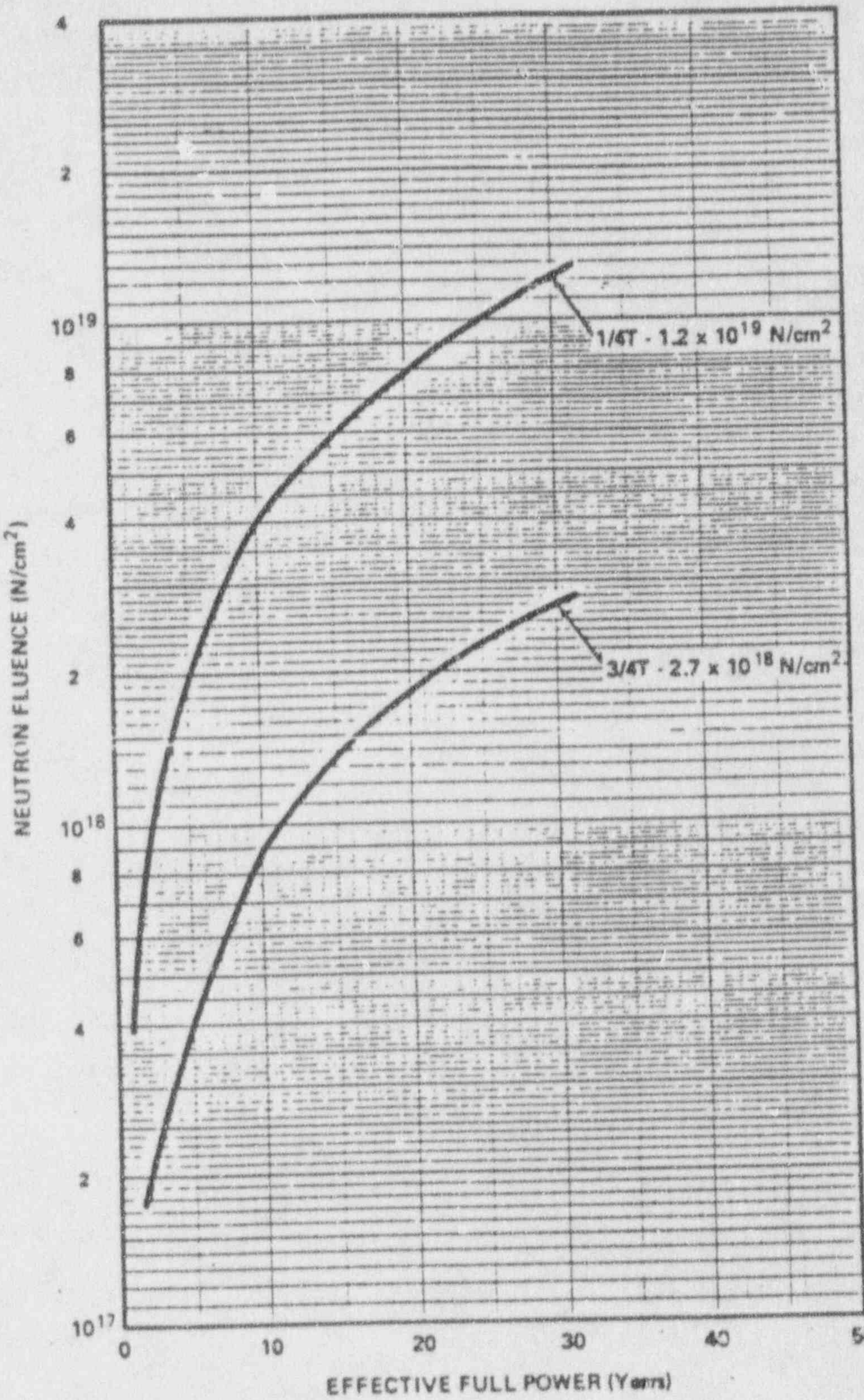

FIGURE B 3/4.4-1

FAST NEUTRON FLUENCE (E > 1 MeV) AS A FUNCTION OF FULL POWER SERVICE LIFE

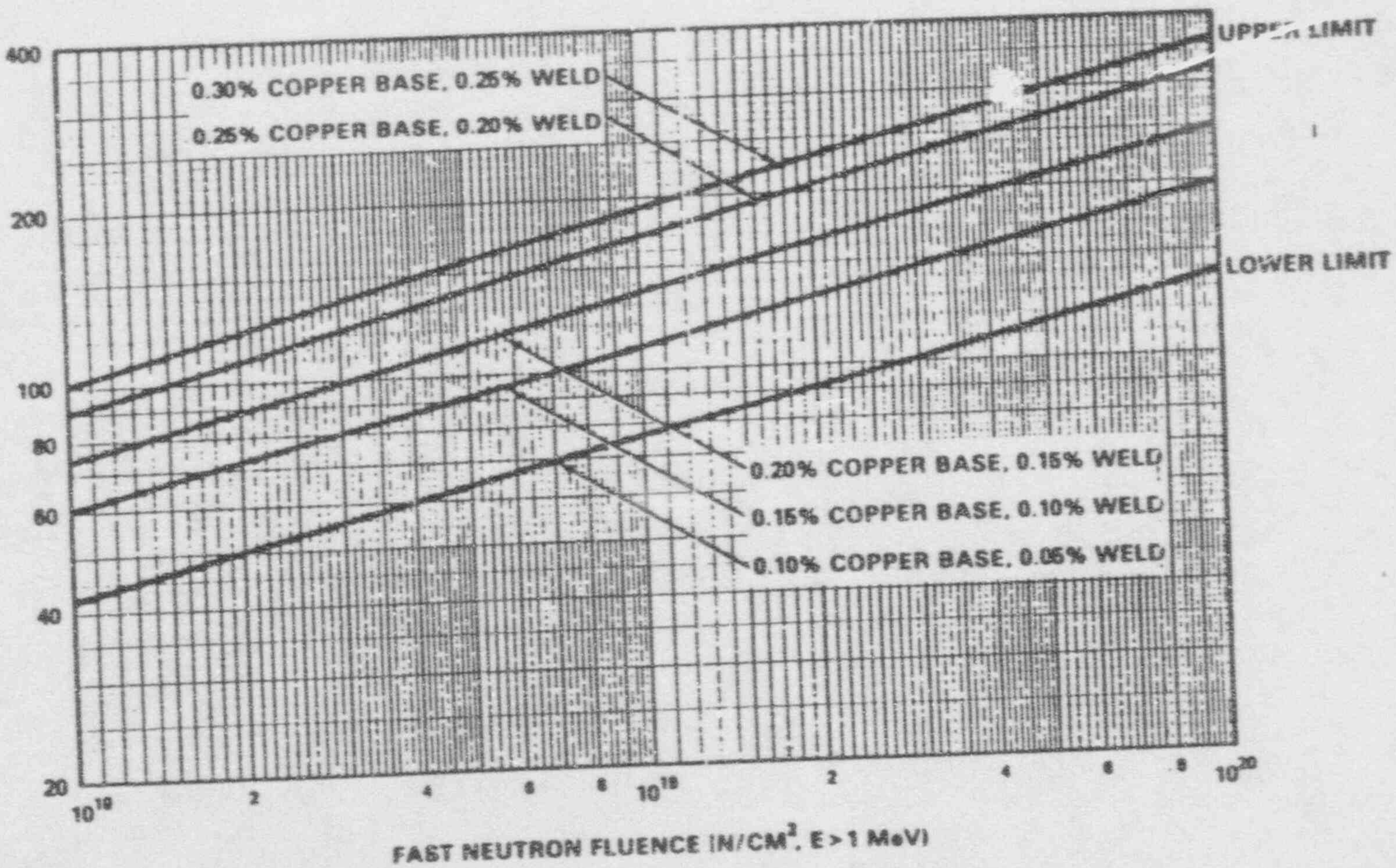

FIGURE B 3/4.4-2

EFFECT OF FLUENCE AND COPPER ON SHIFT OF RT_{NDT} FOR REACTOR VESSEL STEELS EXPOSED TO IRRADIATION AT 550°F

RECEIVED
DEC 10 1990
BRAIDWOOD STATION

Dec.7,1990
In reply refer to
CHRON No. 160070

To: K.L. Kofron
Braidwood Station Manager

Subject: Heatup and Cooldown Curves for Braidwood Unit 1

Reference: NED letter dated Nov. 27,1990 from E.D. Swartz to K.L. Kofron

The reference letter transmitted revised heatup/cooldown curves for Braidwood Unit 1 based on fluence data gather from the analysis of Surveillance Capsule U. It was noted in the letter that there were two (2) typographical errors on page 10 of the Westinghouse transmittal. Westinghouse has supplied a corrected page which has been reviewed by NED with no comment, and is attached to this letter.

Should you have any question or comments, please call Bob Waninski at x-7387, Downers Grove.

12/7/90
R.E. Waninski
PWR System Design Engineer

12/7/90
E.D. Swartz
PWR System Design Supervisor

Attachment
REW/

cc: M.E. Lohnmann (1/0)
G.S. Gerzen (1/0)
A.R. Checca (1/1)
M.F. Sears (1/1)
J. Feimster (Westinghouse-Braidwood) (1/0)

Westinghouse Electric Corporation

Energy Systems

Box 355
Pittsburgh Pennsylvania 15230-0355

CCE-90-324
November 30, 1990

Ref. SM&RT-190(90), dated 11/6/90 (CCE-90-317)

Mr. E. D. Swartz
Commonwealth Edison Company
1400 Opus Place, Suite 400
Downers Grove, Illinois 60515

Commonwealth Edison Company
Braidwood Unit 1
Heatup/Cooldown Final Report (Capsule U)

Dear Mr. Swartz:

Braidwood Unit 1 Heatup/Cooldown Final Report (Capsule U) was transmitted to you via Reference 1.

In response to Robert Waninski's request, the following changes were made to the attached page:

- o The heatup curve is now identified as $100^{o}F/Hr$. It was inadvertently identified as $60^{o}F/Hr$.
- o *"Acceptable operation"* tag has been moved to a different location. This is done as per the customer's request.

If you have any questions, please do not hesitate to call.

Very truly yours,

WESTINGHOUSE ELECTRIC CORPORATION

S. A. Pusadas GY

G. P. Toth, Manager
Commonwealth Edison Projects
Domestic Customer Projects

/cem

cc: G. Gerzen
R. Waninski
K. Kofron
M. Lohmann
W. Feimster

MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL: CIRCUMFERENTIAL WELD

INITIAL RT_{NDT}: 40°F

ART AFTER 32 EFPY: 1/4T, 159°F

3/4T, 135°F

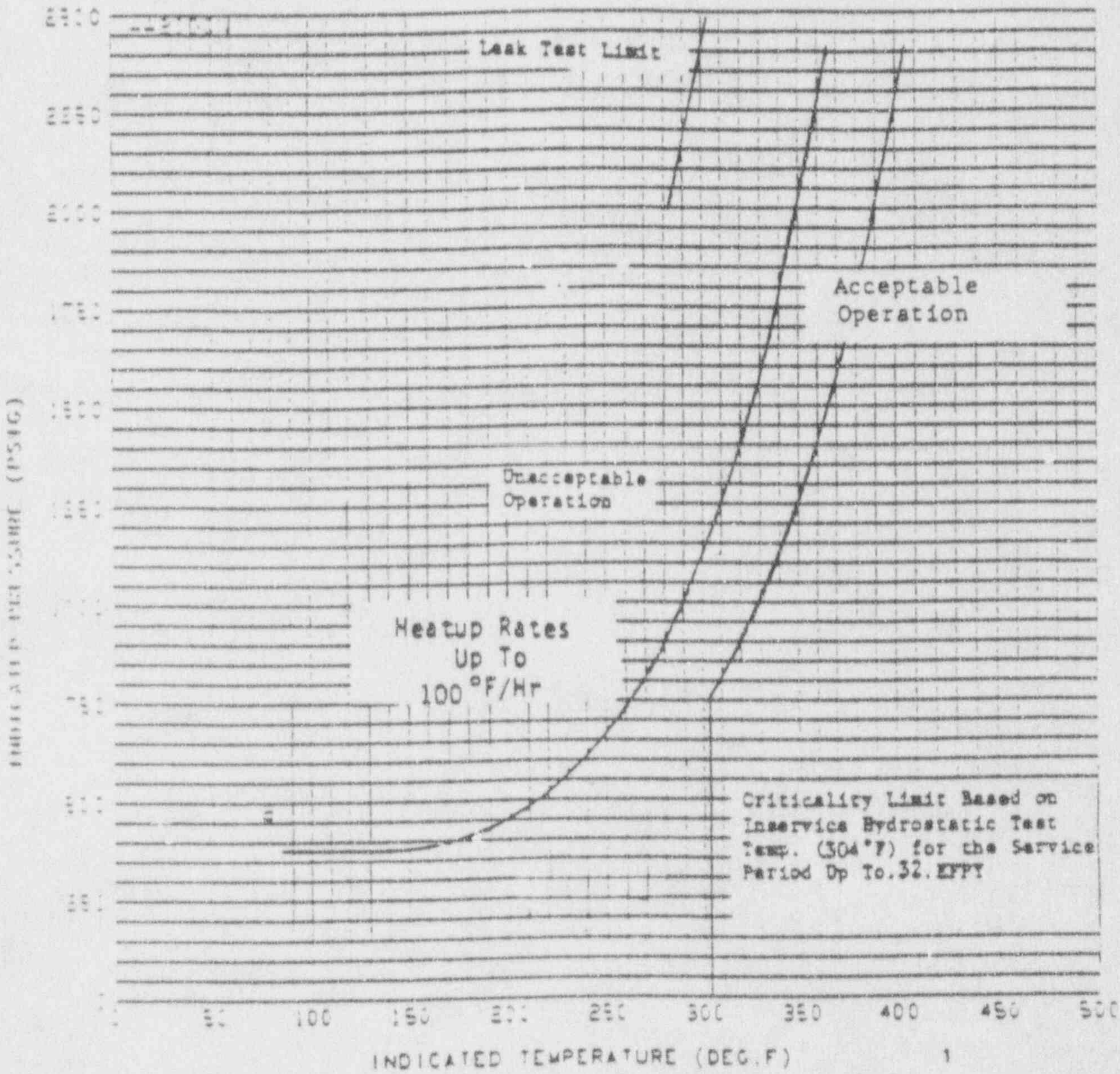

Figure 1. Braidwood Unit 1 Reactor Coolant System Heatup Limitations (Heat up rate up to 100°F/hr) Applicable for the First 32 EFPY (With Margins 10°F and 60 psig for Instrumentation Errors)

Nov.27,1990
In reply refer to
CHRON No. **159714**

To: K.L. Kofron
Braidwood Station Manager

Subject: Heatup and Cooldown Curves for Braidwood Unit 1

Reference: Westinghouse letter CCE-90-317 dated Nov.19,1990

Westinghouse has recently completed the analysis of Unit 1 Vessel Surveillance Capsule U. As part of their contract, they have provided heatup/cooldown curves to reflect the fluence data gathered from the capsule analysis. Prior to issuing the curves, NED-PWR Systems Design Group discussed with the Station Operating Engineer and Tech Staff the proposed curves and agreed that the heatup/cooldown rates should remain the same as noted on current Tech Spec curves.

The reference heatup/cooldown curves bound those currently in the Tech Spec, and therefore the Station has the option of submitting these curves for Tech Spec revision or continuing to use the curves currently in the Tech Spec. However, the new curves also comply with NRC Regulatory Guide 1.99 rev.2 for determining RT_{NDT} values at 1/4T and 3/4T locations. It must be noted that the applicability date of the present Tech Spec curves to Reg. Guide 1.99 rev.2 is 4.5 EFPY. This date will approximately be spring of 1993. By that time, revised Unit 1 curves <u>must</u> be incorporated into the Tech Spec. The Unit 2 curves, which have an applicability date of 2.2 EFPY, have already been revised to comply with rev.2 of Reg. Guide 1.99 and are currently being processed by the Nuclear Licensing Department for Tech Spec incorporation.

NED recommends the attached curves be submitted as a Tech Spec change in order to bring the Unit 1 curves into compliance with revision 2 of Reg.Guide 1.99. The schedule for submittal should be determined by Station/Licensing. It should be noted that an evaluation of low temperature overpressure protection (LTOP's) must be performed prior to implementing revised heatup/cooldown curves. This is a separate cost, not associated with the generation of heatup/cooldown curves as part of the capsule analysis. Westinghouse has quoted a cost of $ 27,650.00 to perform an LTOP evaluation. The evaluation would be complete approximately one month from issuance of a work release.

DSN-3

It should be noted that there are two (2) typographical errors on page 10 of the Westinghouse report.

1. $60^{0}F/Hr$ shown on the graph shall be changed to $100^{0}F/Hr$, and
2. The words "Acceptable Operation" shall be relocated across the Criticality Limit line consistent with the revised Braidwood Unit 2 heatup/cooldown curves.

Westinghouse has been advised of these comments and will submit a revised page accordingly.

Should you have any question or comments, please call Bob Waninski at x-7387, Downers Grove.

11/27/90
R.E. Waninski
PWR System Design Engineer

11/29/90
E.D. Swartz
PWR System Design Supervisor

Attachment
REW/

cc: M.E. Lohnmann (1/0)
G.S. Gerzen (1/0)
A.R. Checca (1/1)
M.F. Sears (1/1)
J. Feimster (Westinghouse-Braidwood) (1/0)
NEDCC (1/1) REW 11/27/90

REV
[illegible]

NOV 26 REC'D

Westinghouse Electric Corporation

Energy Systems

Box 355
Pittsburgh Pennsylvania 15230 0355

CCE-90-317
November 19, 1990

Mr. E. D. Swartz
Commonwealth Edison Company
1400 Opus Place, Suite 400
Downers Grove, Illinois 60515

Commonwealth Edison Company
Braidwood Unit 1
Heatup/Cooldown Final Report (Capsule U)

Dear Mr. Swartz:

Attached is the final report on *"Heatup and Cooldown Curves for the Braidwood Unit 1 Reactor Vessel (Capsule U)"*. Comments from Commonwealth Edison Company on the draft report have been incorporated in the final report.

If you have any questions, please do not hesitate to call.

Very truly yours,

WESTINGHOUSE ELECTRIC CORPORATION

S. A. Pujadas for
G. P. Toth, Manager
Commonwealth Edison Projects
Customer Projects Department

SAP/cem

cc: R. Waninski
G. Gerzen
K. Kofron
M. Lohmann
W. Feimster