ENCLOSURE

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of PACIFIC GAS AND ELECTRIC COMPANY Diablo Canyon Power Plant Units 1 and 2	Docket No. 50-275 Facility Operating License No. DPR-80 Docket No. 50-323 Facility Operating License No. DPR-82 License Amendment Request No. 89-07

Pursuant to 10 CFR 50.90, Pacific Gas and Electric Company (PG&E) hereby applies to amend its Diablo Canyon Power Plant (DCPP) Facility Operating License Nos. DPR-80 and DPR-82 (Licenses).

The proposed changes amend the Units 1 and 2 Technical Specifications (Appendix A of the Licenses) regarding Technical Specifications Figures 3.4-2 and 3.4-3 and associated Bases, and deletion of Technical Specification 4.4.9.1.2 and surveillance capsule withdrawal schedule.

Information on the proposed changes is provided in Attachments A and B.

These changes have been reviewed and are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92 or an unreviewed environmental question. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

Subscribed to in San Francisco, California this 7th day of July 1989.

Respectfully submitted,

Pacific Gas and Electric Company

By _____
W. H. Wallace
Vice President
Engineering

Howard V. Golub
Richard F. Locke
Attorneys for Pacific
Gas and Electric Company

By _____
Richard F. Locke

Subscribed and sworn to before me
this 7th day of July 1989

_____
Adriane D. Tolefree, Notary Public
for the County of Alameda,
State of California

My commission expires December 22, 1992.

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2336S/0069K

8 pp.

OFFICIAL SEAL
ADRIANE D. TOLEFREE
NOTARY PUBLIC - CALIFORNIA
ALAMEDA COUNTY
My Comm. Expires Dec. 22, 1992

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Attachment A

TECHNICAL SPECIFICATIONS FIGURES 3.4-2 AND 3.4-3 AND ASSOCIATED BASES AND DELETION OF TS 4.4.9.1.2 AND SURVEILLANCE CAPSULE WITHDRAWAL SCHEDULE

A. DESCRIPTION OF AMENDMENT REQUEST

This License Amendment Request (LAR) proposes to revise Technical Specifications (TS) regarding revised heatup and cooldown curves and to delete TS 4.4.9.1.2. The proposed changes to the TS are:

1. Figure 3.4-2, "Reactor Coolant System Heatup Limitations," and Figure 3.4-3, "Reactor Coolant System Cooldown Limitations," are being revised to update the controlling materials chemical composition and Pressure-Temperature (P/T) curves based on NRC Regulatory Guide 1.99, Revision 2, calculations.

2. TS 4.4.9.1.2 and Table 4.4-5, "Reactor Vessel Material Surveillance Program Withdrawal Schedule," are being deleted.

3. TS Bases 3/4.4.9 is being revised to update information and to delete figures and tables that have been included in the DCPP FSAR Update, Revision 4.

Proposed changes to the TS of Operating License Nos. DPR-80 and DPR-82 are noted in the marked-up copy of the applicable TS (See Attachment B).

B. BACKGROUND

The present DCPP Pressure/Temperature (P/T) curves were based on the adjusted reference temperature (ART) predicted by the most limiting method of either NRC Regulatory Guide (RG) 1.99, Revision 1, or the Westinghouse Trend Curves. The present P/T curves are applicable to 6 effective full power years (EFPY). The expected exposures for the Units 1 and 2 reactor vessels at the end of July 1989 are 3.27 and 2.64 EFPY, respectively. The existing operation limits are therefore valid for a minimum of 2.73 EFPY beyond the July 1989 exposure.

On July 12, 1988, the NRC issued Generic Letter (GL) 88-11, "NRC Position On Radiation Embrittlement of Reactor Vessel Material And Its Impact on Plant Operations." GL 88-11 recommends the use of Revision 2 to RG 1.99 to predict the effect of neutron radiation on reactor vessel materials. The proposed P/T limits for Diablo Canyon Units 1 and 2 are based on the methods described in RG 1.99, Revision 2, and the requirements of 10 CFR 50, Appendix G. The proposed P/T limits are applicable through 8 EFPY. PG&E reviewed the impact of the proposed new Appendix G heatup and cooldown curves on the LTOP pressure setpoint and the LTOP enable temperature in accordance with BTP RSB 5-2 and has concluded that the present setpoints are conservative.

C. JUSTIFICATION

The proposed changes are in accordance with NRC Generic Letter 88-11, which advised Licensees that use of NRC RG 1.99, Revision 2, is an acceptable method to predict the effect of neutron radiation on reactor vessel materials. PG&E has used the information provided in NRC RG 1.99, Revision 2, to calculate the new P/T limits for Units 1 and 2.

The proposed removal of TS Tables B 3/4.4-1a and B 3/4 4-1b, and TS Figures B 3/4.4-1 and B 3/4.4-2 and relocation to the FSAR Update will simplify future TS revisions. The information being removed from the TS is in the current DCPP FSAR Update. This data is not needed for reference by the Operations staff and will be accessible in the FSAR Update for engineering personnel.

TS 4.4.9.1.2 and TS Table 4.4-5 are also proposed to be deleted from the TS. The surveillance schedule will be included in the next FSAR Update revision. The justification for these changes is that 10 CFR 50, Appendix H, requires a surveillance capsule removal program, and also requires that P/T limits in TS be modified if necessary. Table A lists PG&E's proposed capsule withdrawal schedule in accordance with 10 CFR 50, Appendix H. This schedule meets the requirements of ASTM E 185-82, which is referenced in Appendix H.

Deletion of the surveillance capsule removal program from the Technical Specifications was previously approved by the NRC when the Vogtle Units 1 and 2 TS were issued.

D. SAFETY EVALUATION

Based on the analysis of commercial power reactor surveillance capsule tests results, the NRC has determined that the methodology of RG 1.99, Revision 2, provides a better prediction of neutron radiation embrittlement than previous revisions. The calculational method of RG 1.99, Revision 2, based on copper and nickel content and neutron fluence at the most limiting location, was used to determine the ART at the end of 8 EFPY. In accordance with RG 1.99 Revision 2, the DCPP surveillance capsule test results were not used in determining an ART since data from only one capsule per unit was available.

As required by 10 CFR 50, Appendix G, PG&E used the methodology of ASME Section III, Appendix G, "Protection Against Nonductile Failure," in calculating the proposed P/T curves. These standards define the methodology for calculating P/T limits which prevent brittle fracture of the vessel and closure head flanges during both normal operation and leak testing. In addition, specific rules, provided in 10 CFR 50, Appendix G, for calculating the minimum allowable temperatures for reactor criticality, were followed. Both the existing and the proposed P/T curves comply with Appendix G requirements.

TABLE A

REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM - WITHDRAWAL SCHEDULE

UNIT 1

Capsule	Location	Lead Factor	Removal Time (EFPY)
S	320°	3.57	1.26 (Removed)
Y	40°	3.57	5.2
V	184°	1.17	15.0
U	356°	1.17	Standby
X	176°	1.17	Standby
W	4°	1.17	Standby
T	140°	3.57	Standby
Z	220°	3.57	Standby

UNIT 2

Capsule	Location	Lead Factor	Removal Time (EFPY)
U	56°	5.05	.99 (Removed)
X	236°	5.05	3.3
Y	238.5°	4.21	7.2
W	124°	5.05	9.1
V	58.5°	4.21	Standby
Z	304°	5.05	Standby

The proposed P/T curves are based on more refined vessel neutron fluence predictions, which reflect the advances made in achieving lower leakage core loading patterns. Both Units 1 and 2 Cycle 1 fast neutron fluences were determined from surveillance capsule and reactor cavity dosimetry measurements. Flux reduction factors were then estimated for subsequent cycles out to 8 EFPY for actual and projected core loading patterns. For the limiting Unit 2 shell plate, the ARTs are based on the projected maximum 8 EFPY fast neutron fluence values of 0.2561×10^{19} n/cm^2 at 1/4T and 0.0910×10^{19} n/cm^2 at 3/4T.

The present P/T limits are based on ARTs of 154°F and 122°F at 1/4T and 3/4T, respectively. The ART calculations for Units 1 and 2 show that the Unit 2 shell plate B5454-2 is the limiting vessel material up to 8 EFPY. The chemical composition of this shell plate is 0.14 weight percent copper and 0.59 weight percent nickel. The revised ART for this material at the end of 8 EFPY is 164°F at 1/4T and 141°F at 3/4T (these temperatures are labeled as projected RT_{NDT} on the proposed P/T limit graphs). In general, the proposed P/T heatup and cooldown limits are slightly more restrictive than the present limits except in a narrow low temperature band on the heatup curve.

PG&E has reviewed the impact of the new Appendix G curves on the LTOP pressure setpoint and enable temperature in accordance with BTP RSB 5-2 and has concluded that the present setpoints are conservative.

Therefore, operating within the proposed P/T limits will ensure compliance with the requirements of 10 CFR 50, Appendix G, and maintain the present margin of safety with regard to brittle fracture through 8 EFPY. Based upon the information provided above, PG&E believes that there is reasonable assurance that the health and safety of the public will not be adversely affected by revising the P/T limits.

E. NO SIGNIFICANT HAZARDS EVALUATION

PG&E has evaluated the hazards considerations involved with the proposed amendment, focusing on the three standards set forth in 10 CFR 50.92(c) as quoted below:

> The Commission may make final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under paragraph 50.21(b) or paragraph 50.22 or for a testing facility involves no significant hazards consideration, 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 accidents previously evaluated; or
>
> (3) Involve significant reduction in a margin of safety.

The following evaluation is provided for the three categories of the no significant hazards consideration standards.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed changes to TS 3/4.4.9, "Reactor Coolant System - Pressure/Temperature (P/T) Limits" and associated Bases revise the heatup/cooldown curves in accordance with NRC Regulatory Guide 1.99, Revision 2. The proposed P/T limits are generally more restrictive than the present P/T limits except in a narrow low temperature band on the heatup curve. The present LTOP pressure setpoint and the enable temperature were reviewed and found to be acceptable for the new P/T limits.

   The P/T limits are prescribed as guidance used during normal operation to avoid encountering pressure, temperature and temperature rate-of-change conditions which might cause undetected flaws to propagate resulting in non-ductile failure of the reactor coolant pressure boundary. The P/T limits are not derived from design bases accident events presented in the Diablo Canyon FSAR Update. As such the revision to the P/T limits does not affect the probability or consequences of a previously analyzed accident.

   The proposed deletion of TS 4.4.9.1.2 and the associated withdrawal schedule are administrative changes. 10 CFR 50, Appendix H, requires a withdrawal schedule to be established, and repeating the requirement in the Technical Specifications is unnecessary. In addition, deletion of the tables and figures from the Bases is an administrative change. This information will be provided in the Diablo Canyon FSAR Update. These changes are purely administrative.

   Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Using more conservative vessel embrittlement projections, the proposed revision merely updates the existing pressure/temperature limits. The proposed P/T curves are in general slightly more restrictive than the present P/T limits except in a narrow low temperature band on the heatup curve. The LTOP pressure setpoint and enable temperature are not being changed and have been shown to be set conservatively in accordance with analysis and NUREG-0800 methodology, respectively. The removal of the capsule withdrawal schedule and surveillance requirement is purely administrative. None of the proposed revisions to the TS and associated Bases result in any physical alteration to any plant system, nor is there a change in the method by which any safety-related system performs its function.

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

3. Does the change involve a significant reduction in a margin of safety?

The proposed updates to the P/T limits are based on the conservative methodology of Revision 2 to NRC Regulatory Guide 1.99. The revisions to TS 3/4.4.9 and associated Bases meet all of the requirements of 10 CFR 50, Appendix G. The removal of the capsule withdrawal schedule and surveillance requirement from the TS does not affect the requirements of 10 CFR 50, Appendix H. The FSAR Chapter 15 safety analyses are unaffected by the proposed changes.

Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

E. NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

In conclusion, based on the above evaluation, PG&E submits that the activities associated with this LAR satisfy the no significant hazards consideration standards of 10 CFR 50.92(c) and, accordingly, a no significant hazards consideration finding is justified.

F. ENVIRONMENTAL EVALUATION

PG&E has evaluated the proposed changes and determined that the changes do not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed changes meet the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), an environmental assessment of the proposed changes is not required.

Attachment B

MARKED-UP TECHNICAL SPECIFICATIONS

Remove Page	Insert Page
viii	viii
xv	xv
3/4 4-30	3/4 4-30
3/4 4-31	3/4 4-31
3/4 4-32	3/4 4-32
3/4 4-33	---
B 3/4 4-7	B 3/4 4-7
B 3/4 4-8	---
B 3/4 4-9	---
B 3/4 4-10	---
B 3/4 4-11	---
B 3/4 4-12	B 3/4 4-12
B 3/4 4-13	B 3/4 4-13

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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

BASES

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

LIMITING CONDITION FOR OPERATION

3.4.9.1 The Reactor Coolant System (except the pressurizer) temperature and pressure shall be limited in accordance with the limit lines shown on Figures 3.4-2 and 3.4-3 during heatup, cooldown, criticality, and inservice leak and hydrostatic testing with:

a. A maximum heatup of 100°F in any 1-hour period.

b. A maximum cooldown of 100°F in any 1-hour period.

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.

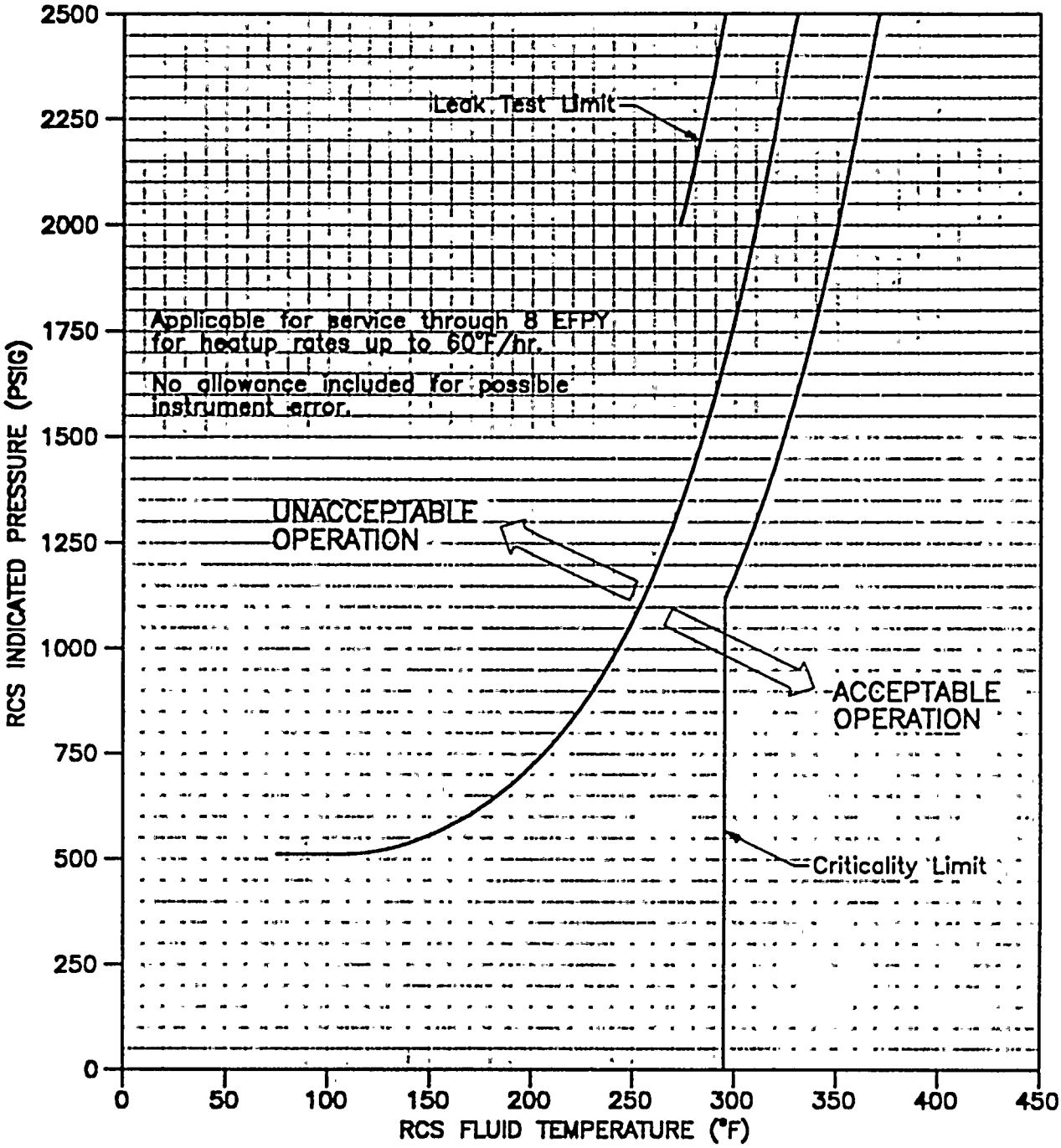
ACTION:

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 hour 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 and 3.4-3, and the setpoint of Technical Specification 3.4.9.3.a.~~

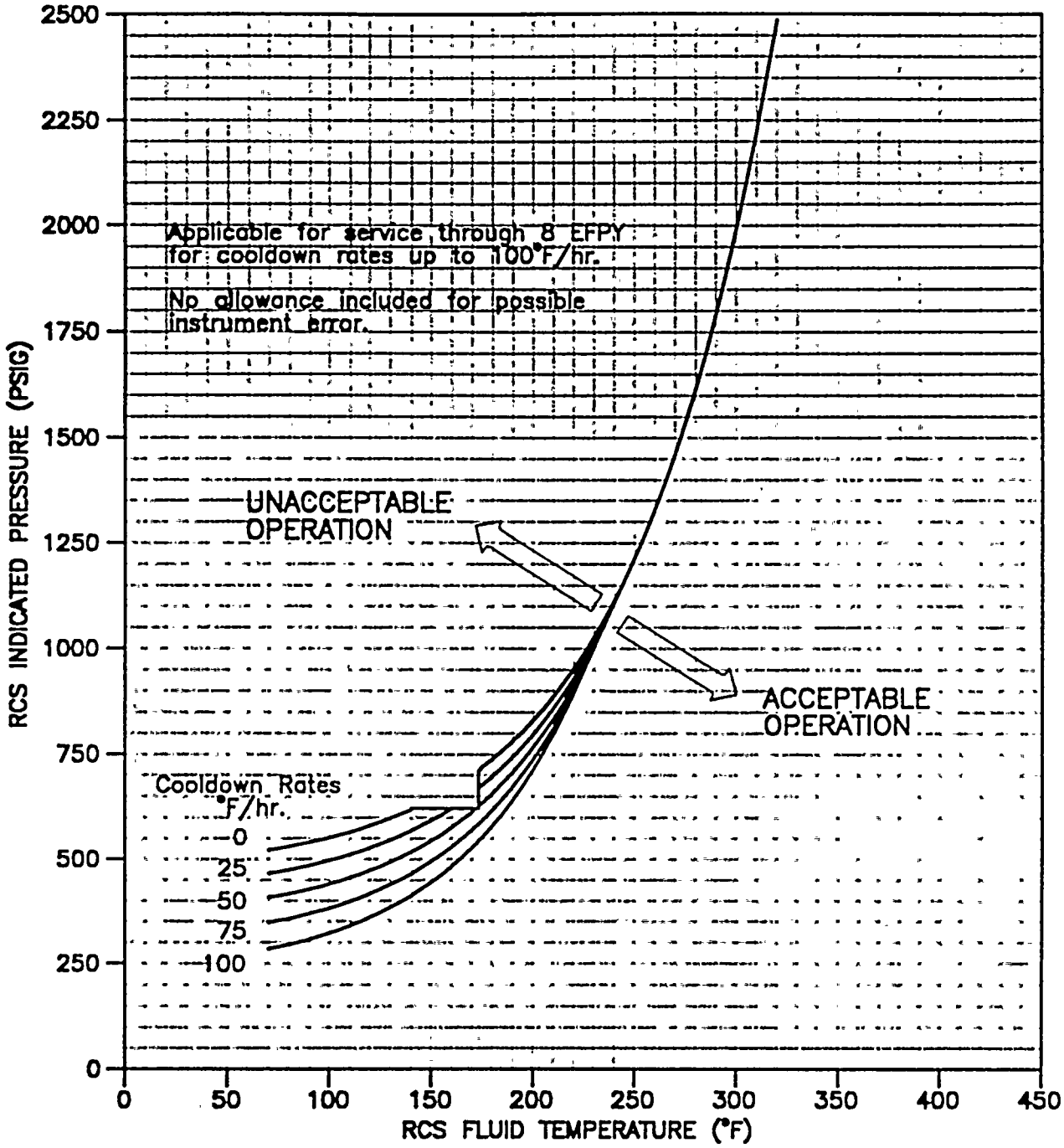


Controlling Material:

Unit 2 Intermediate Shell Plate B5454-2 0.14wt.% Cu 0.59wt.% Ni

Initial RT_{NDT}=67°F Projected RT_{NDT} 1/4T = 164°F 3/4T = 141°F

Figure 3.4-2
REACTOR COOLANT SYSTEM HEATUP LIMITATIONS - APPLICABLE UP TO 8 EFPY



Controlling Material:

Unit 2 Intermediate Shell Plate B5454-2 0.14wt.% Cu 0.59wt.% Ni

Initial RT_{NDT}=67°F Projected RT_{NDT} 1/4T = 164°F 3/4T = 141°F

Figure 3.4-3
REACTOR COOLANT SYSTEM COOLDOWN LIMITATIONS - APPLICABLE UP TO 8 EYPY

DELETE

TABLE 4.4-5

REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM - WITHDRAWAL SCHEDULE

UNIT 1

CAPSULE NUMBER	VESSEL LOCATION	LEAD FACTOR	WITHDRAWAL TIME (EFPY)
S	320°	3.7	First refueling outage
T	140°	3.7	Standby
U	356°	1.1	12
V	184°	1.1	24
W	4°	1.1	38
X	176°	1.1	50
Y	40°	3.7	Standby
Z	220°	3.7	Standby

UNIT 2

CAPSULE NUMBER	VESSEL LOCATION	LEAD FACTOR	WITHDRAWAL TIME (EFPY)
U	56°	4.8	First refueling outage
X	236°	4.8	3
V	58.5°	4.0	6
Y	238.5°	4.0	10
W	124°	4.8	15
Z	304°	4.8	Standby

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 Presure 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 and 3.4-3 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 and 3.4-3 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 560°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. Insert

The fracture toughness testing of the ferritic materials in the reactor vessel was performed in accordance with the 1966 Edition for Unit 1 and the 1968 Edition for Unit 2 of the ASME Boiler and Pressure Vessel Code, Section III. These properties are then evaluated in accordance with the NRC Standard Review Plan.

Heatup and cooldown limit curves are calculated using the most limiting value of the nil ductility reference temperature, RT_{NDT}, at the end of ~~8~~ 8 effective full power years (EFPY) of service life. The ~~8~~ 8 EFPY service life period is chosen such that the limiting RT_{NDT} at the 1/4T location in the core region

INSERT, TS BASES 3/4.4.9 (Page B 3/4 4-7)

Allowable pressures and temperatures for inservice leak and hydrostatic tests are given in Figure 3.4-2.

6. The criticality limit on Figure 3.4-2 is based on the minimum allowable temperature of 295°F for an inservice hydrostatic test of 110% of operating pressure.

TABLE B 3/4.4-1a

REACTOR VESSEL TOUGHNESS DATA-UNIT 1

DELETE

COMPONENT	PLATE NO.	MATERIAL TYPE	Cu (Wt%)	P (Wt%)	NDTT °F	MINIMUM 50 FT-LB/35 Mil TEMP °F LONG	MINIMUM 50 FT-LB/35 Mil TEMP °F TRANS	RT_{NDT} °FT	AVERAGE UPPER SHELF FT-LB LONG	AVERAGE UPPER SHELF FT-LB TRANS
Cl.Hd.Dome	B4108	A533B1			-30	51	71*	11		72*
Cl.Hd.Seg.	B4109-1	A533B1			0	53	73*	13		81*
Cl.Hd.Seg.	B4109-2	A533B1			-10	50	70*	10		89
Cl.Hd.Seg.	B4109-3	A533B1			-20	50	70*	10		79*
Cl.Hd.Flg.	B4102	A508,2			53*	30	50*	53		103*
Ves.Sh.Flg.	B4101	A508,2			35*	-5	15*	35		99*
Inlet Noz.	B4103-1	A508,2			60*	17	37*	60		77*
Inlet Noz.	B4103-2	A508,2			60*	27	47*	60		75*
Inlet Noz.	B4103-3	A508,2			43*	10	30*	43		108*
Inlet Noz.	B4103-4	A508,2			48*	2	22*	48		106*
Outlet Noz.	B4104-1	A508,2			60*	-13	7*	60		77*
Outlet Noz.	B4104-2	A508,2			43*	-3	17*	43		74*
Outlet Noz.	B4104-3	A508,2			54*	-12	8*	54		86*
Outlet Noz.	B4104-4	A508,2			60*	30	50*	60		84*
Upper Shl.	B4105-1	A533B1	0.12	0.010	10	68	88*	28		80*
Upper Shl.	B4105-2	A533B1	0.12	0.008	0	49	69*	9		74*
Upper Shl.	B4105-3	A533B1	0.14	0.010	0	54	74*	14		81*
Inter.Shl.	B5106-1	A533B1	0.14	0.013	-10	57	40	-10	134	116
Inter.Shl.	B4106-2	A533B1	0.13	0.013	-10	36	57	- 3	132	114
Inter.Shl.	B4106-3	A533B1	0.10	0.011	10	70	90*	30	119	77
Lower Shl.	B4107-1	A533B1	0.13	0.011	-10	59	75	15	127	110
Lower Shl.	B4107-2	A533B1	0.12	0.010	-10	64	80	20	127	108
Lower Shl.	B4107-3	A533B1	0.12	0.010	-50	52	38	-22	135	116
Bot.Hd.Seg.	B4111-1	A533B1			-20	33	53*	- 7		82*
Bot.Hd.Seg.	B4111-2	A533B1			-40	16	36*	-14		90*
Bot.Hd.Seg.	B4111-3	A533B1			-40	21	41*	-19		85*
Bot.Hd.Seg.	B4110	A553B1			-10	60	80*	20		75*

*Estimated per NRC Standard Review Plan Section 5.3.2.

TABLE B 3/4.4-1b

REACTOR VESSEL TOUGHNESS DATA-UNIT 2

DELETE

COMPONENT	PLATE NO.	MATERIAL TYPE	Cu (Wt%)	P (Wt%)	NDTT °F	MINIMUM 50 FT-LB/35 Mil TEMP °F LONG	MINIMUM 50 FT-LB/35 Mil TEMP °F TRANS	RT_{NDT} °F	AVERAGE UPPER SHELF FT-LB LONG	AVERAGE UPPER SHELF FT-LB TRANS
Cl.Hd.Dome	B5457	A533BCL1			-20	35	55*	-5	135	88*
Cl.Hd.Seg.	B5456-1	A533BCL1			-50	23	43*	-17	134	[illegible]
Cl.Hd.Seg.	B5456-2	A533BCL1			-20	62	82*	22	131	88*
Cl.Hd.Seg.	B5456-3	A533BCL1			-20	15	35*	-20	124	81*
Cl.Hd.Flg.	B5452	A508CL2			20	15	35*	20	151	98*
Ves.Sh.Flg.	B5451	A508CL2			-10	-10	10*	-10	158	103*
Inlet Noz.	B5461-1	A508CL2			-20	23	43*	-17	116	75
Inlet Noz.	B5461-2	A508CL2			-20	- 2	18*	-20	119	77*
Inlet Noz.	B5461-3	A508CL2			-40	-45	-25*	-40	127	83*
Inlet Noz.	B5461-4	A508CL2			-40	-48	-28*	-40	129	84*
Outlet Noz.	B5462-1	A508CL2			-50	- 4	16*	-44	145	94
Outlet Noz.	B5462-4	A508CL2			-40	-10	10*	-40	137.5	89*
Outlet Noz.	B5462-2	A508CL2			-40	14	34*	-26	135.5	88*
Outlet Noz.	B5462-3	A508CL2			-50	17	37*	-23	131.5	85*
Upper Shl.	B5453-1	A533BCL1			0	85	88	28	92	82
Upper Shl.	B5453-3	A533BCL1			-10	45	65*	5	136.5	89*
Upper Shl.	B5011-1	A533BCL1			10	40	60*	0	110	72*
Inter.Shl.	B5454-1	A533BCL1	0.15	0.010	-40	14	112	52	128	91
Inter.Shl.	B5454-2	A533BCL1	0.14	0.012	0	60	127	67	113	[illegible]
Inter.Shl.	B5454-3	A533BCL1	0.15	0.012	-40	30	93	33	129	90
Lower Shl.	B5455-1	A533BCL1	0.14	0.010	-20	42	45	-15	134	112
Lower Shl.	B5455-2	A533BCL1	0.14	0.011	0	25	45	0	137	122
Lower Shl.	B5455-3	A533BCL1	0.10	0.010	0	55	75	15	128	100
Bot.Hd.Seg.	B5009-2	A533BCL1			-10	110	130*	70	85	55*
Bot.Hd.Seg.	B5009-3	A533BCL1			-20	-12	8*	-20	131	84
Bot.Hd.Seg.	B5009-1	A533BCL1			0	88	108*	48	95	62*
Bot.Hd.Seg.	B5010	A533BCL1			-30	20	40*	-20	114	74

*Estimated per NRC Standard Review Plan Section 5.3.2.

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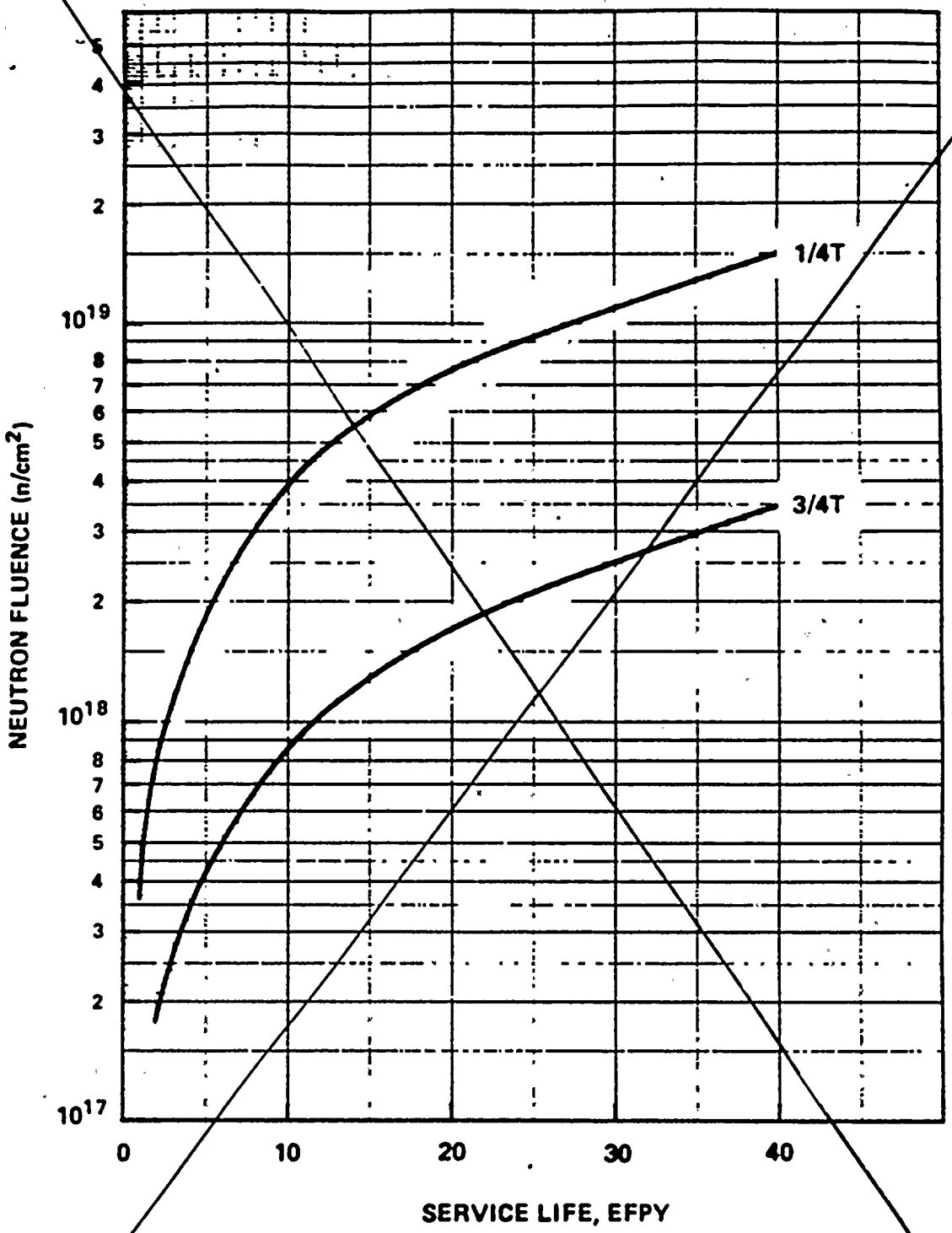

FIGURE B 3/4.4-1

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

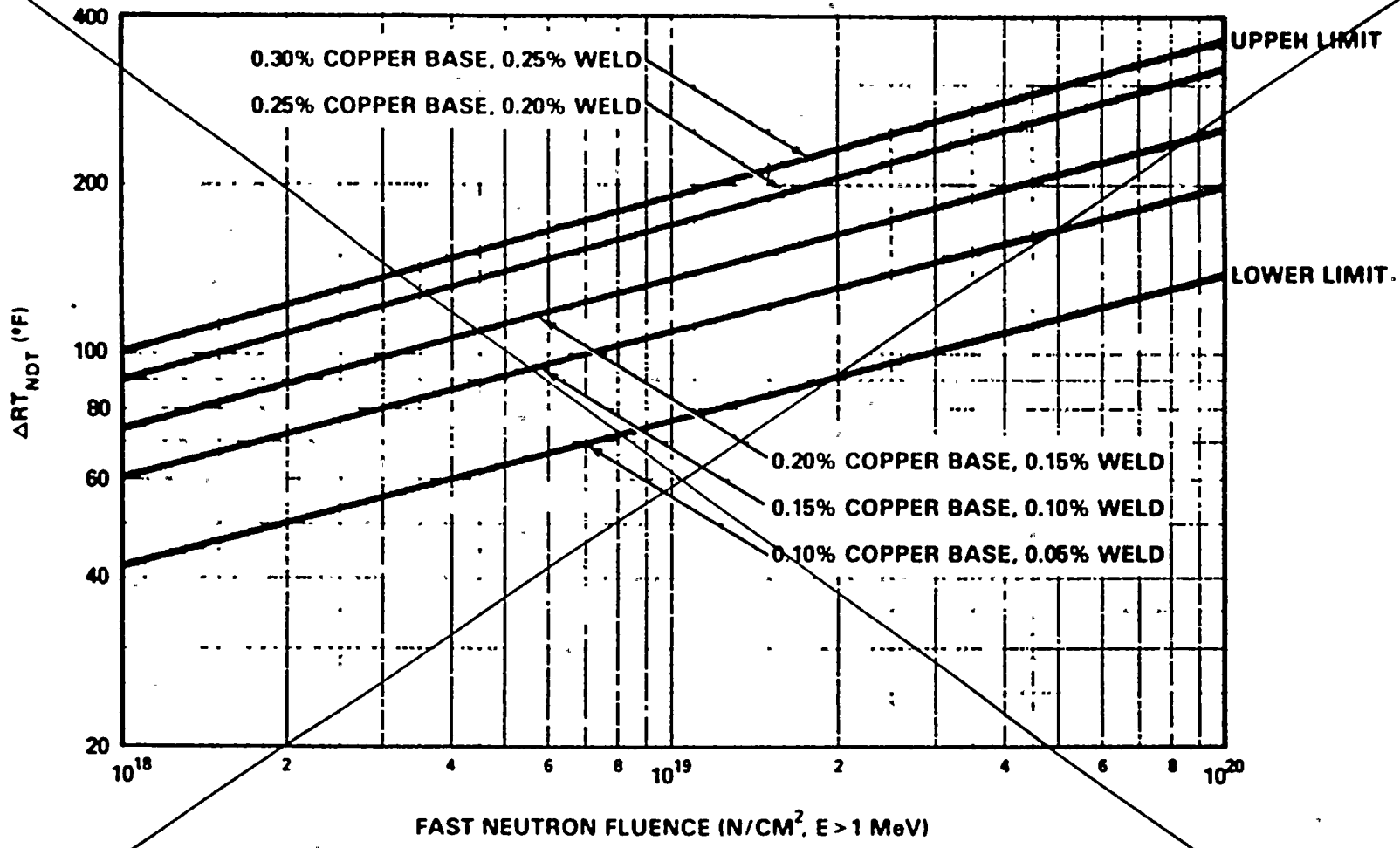

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FIGURE B 3/4.4-2

EFFECT OF FLUENCE AND COPPER CONTENT ON SHIFT OF RT_{NDT} FOR REACTOR VESSELS EXPOSED TO 550°F

PRESSURE/TEMPERATURE LIMITS (Continued)

for the maximum neutron fluence at the locations of interest.

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 and Table B 3/4.4-1b for Unit 2.~~ the FSAR Update. 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 ~~phosphorous~~ 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~~ 2, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials," ~~or the Westinghouse Copper Trend Curves shown in Figure B 3/4.4-2.~~ The heatup and cooldown limit curves of Figures 3.4-2 and 3.4-3 include predicted adjustments for this shift in RT_{NDT} at the end of ~~6~~ 8 EFPY.

Values of ΔRT_{NDT} determined in this manner ~~may~~ will be used until the results from the material surveillance program, evaluated according to ASTM E185-~~73~~ 82, can be used ~~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. 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 the following paragraphs.

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 will be maintained in the FSAR Update.

PRESSURE/TEMPERATURE LIMITS (Continued)

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:

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

Where: K_{IR} is the reference stress intensity factor as a function of the metal temperature T and the metal nil ductility reference temperature RT_{NDT}. Thus, the governing equation for the heatup-cooldown analysis is defined in Appendix G of the ASME Code as follows:

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

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

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

K_{IR} = ~~constant~~ reference stress intensity factor 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 of 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 factors, K_{It}, for the reference flaw ~~is~~ are computed. From Equation (2) the pressure stress intensity factors are obtained and, from these, the allowable pressures are calculated.