2.0 LIMITING CONDITIONS FOR OPERATION
2.1 Reactor Coolant System (Continued)
2.1.2 Heatup and Cooldown Rate (Continued)

(a) The curve in Figure 2-3 shall be used to predict the increase in transition temperature based on integrated fast neutron flux. If measurements on the irradiation specimens indicate a deviation from this curve, a new curve shall be constructed.

(b) The limit line on the figures shall be updated for a new integrated power period as follows: the total integrated reactor thermal power from startup to the end of the new period shall be converted to an equivalent integrated fast neutron exposure ($E \geq 1$ MeV). For this plant, based upon surveillance materials tests, weld chemical composition data, and the effect of a reduced vessel fluence rate provided by core load designs beginning with fuel Cycle 8, the predicted surface fluence at the critical reactor vessel beltline weld material for 40 years at 1500 MWt and an 80% load factor is 2.9×10^{19} n/cm^2. The flux reduction applied to the fluence calculations was based on a Cycle 1-9 average azimuthal flux distribution plot generated using DOT 4.3. The predicted transition temperature shift to the end of the new period shall then be obtained from Figure 2-3.

(c) The limit lines in Figures 2-1A and 2-1B shall be moved parallel to the temperature axis (horizontal) in the direction of increasing temperature a distance equivalent to the transition temperature shift during the period since the curves were last constructed. The boltup temperature limit line shall remain at 82°F as it is set by the NDTT of the reactor vessel flange and not subject to fast neutron flux. The lowest service temperature shall remain at 182°F because components related to this temperature are also not subject to fast neutron flux.

(d) The Technical Specification 2.3(3) shall be revised each time the curves of Figure 2-1A and 2-1B are revised.

Basis

All components in the reactor coolant system are designed to withstand the effects of cyclic loads due to reactor coolant system temperature and pressure changes.(1) These cyclic loads are introduced by normal unit load transients, reactor trips and startup and shutdown operation.

During unit startup and shutdown, the rates of temperature and pressure changes are limited. The design number of cycles for heatup and cooldown is based upon a rate of 100°F in any one hour period and for cyclic operation.

The maximum allowable reactor coolant system pressure at any temperature is based upon the stress limitations for brittle fracture considerations. These limitations are derived by using the rules contained in Section III[2] of the ASME Code including Appendix G, Protection Against Nonductile Failure, and the rules contained in 10 CFR 50, Appendix G, Fracture Toughness Requirements. This ASME Code assumes that a crack 10-11/16 inches long and 1-25/32 inches deep exists on the inner surface of the vessel. Furthermore, operating limits on pressure and temperature assure that the crack does not grow during heatups and cooldowns.

The reactor vessel belt-line material consists of six plates. The nilductility transition temperature (T_{NDT}) of each plate was established by drop weight tests. Charpy tests were then performed to determine at what temperature the plates exhibited 50 ft-lbs. absorbed energy and 35 mils lateral expansion for the longitudinal direction. NRC technical position MTEB 5-2 was used to establish a reference temperature for transverse direction (RT_{NDT}) of -12°F.

The mean initial RT_{NDT} value for the Fort Calhoun submerged arc vessel weldments was determined to be -56°F with a standard deviation of 17°F. By applying the shift prediction methodology of the proposed Regulatory Guide 1.99, Revision 2, a weld material adjusted reference temperature (RT_{NDT}) was established at 10°F based on the mean value plus two standard deviations. The standard deviation was determined by using the root-mean-squares method to combine the margin of 28°F for uncertainty in the shift equation with the margin of 17°F, for uncertainty in the initial RT_{NDT} value.

Similar testing was not performed on all remaining material in the reactor coolant system. However, sufficient impact testing was performed to meet appropriate design code requirements[3] and a conservative RT_{NDT} of 50°F has been established.

As a result of fast neutron irradiation in the region of the core, there will be an increase in the T_{NDT} with operation. The techniques used to predict the integrated fast neutron ($E \geq 1$ Mev) fluxes of the reactor vessel are described in Section 3.4.6 of the USAR, except that the integrated fast neutron flux ($E \geq 1$ MeV) is 2.9×10^{19} n/cm^2, including tolerance at the inside surface of the critical reactor vessel beltline weld material, over the 40 year design life of the vessel.[5]

Since the neutron spectra and the flux measured at the samples and reactor vessel inside radius should be nearly identical, the measured transition shift for a sample can be applied to the adjacent section of the reactor vessel for later stages in plant life equivalent to the difference in calculated flux magnitude. The maximum exposure of the reactor vessel will be obtained from the measured sample exposure by application of the calibrated azimuthal neutron flux variation. The

maximum integrated fast neutron ($E \geq 1$ MeV) exposure of the reactor vessel at the critical reactor vessel beltline location, including tolerance, is computed to be 2.9×10^{19} n/cm^2 at the vessel inside surface for 40 years operation at 1500 MWt and 80% load factor. The predicted shift at this location at the 1/4t depth from the inner surface is 332°F, including margin and was calculated using the shift prediction equation of the proposed Regulatory Guide 1.99, Revision 2. The actual shift in T_{NDT} will be reestablished periodically during the plant operation by testing of reactor vessel material samples which are irradiated cumulatively by securing them near the inside wall of the reactor vessel as described in Section 4.5.3 and Figure 4.5-1 of the USAR. To compensate for any increase in the T_{NDT} caused by irradiation, limits on the pressure-temperature relationship are periodically changed to stay within the stress limits during heatup and cooldown. Analysis of the second removed irradiated reactor vessel surveillance specimen[8], combined with weld chemical composition data and reduced fluence core loading designs initiated in Cycle 8, indicates that the fluence at the end of 15 Effective Full Power Years (EFPY) at 1500 MWt will be less than 1.4×10^{19} n/cm^2 on the inside surface of the reactor vessel. This results in a total shift to the initial RT_{NDT} of 285°F shift, including margin, for the area of greatest sensitivity (weld metal) at the 1/4t location as determined from Figure 2-3. Operation through fuel Cycle 16 will result in less than 15 EFPY.

The limit lines in Figures 2-1A and 2-1B are based on the following:

A. Heatup and Cooldown Curves - From Section III of the ASME Code, Appendix G-2215.

$$K_{IR} = 2\,K_{IM} + K_{IT}$$

K_{IR} = Allowance stress intensity factor at temperatures related to RT_{NDT} (ASME III Figure G-2100.1).

K_{IM} = Stress intensity factor for membrane stress (pressure). The 2 represents a safety factor of 2 on pressure.

K_{IT} = Stress intensity factor radial thermal gradient.

The above equation is applied to the reactor vessel belt-line. For plant heatup the thermal stress is opposite in sign from the pressure stress and consideration of a heatup rate would allow for a higher pressure. For heatup it is, therefore, conservative to consider an isothermal heatup or $K_{IT} = 0$.

2.0 LIMITING CONDITIONS FOR OPERATION
2.1 Reactor Coolant System (Continued)
2.1.2 Heatup and Cooldown Rate (Continued)

For plant cooldown thermal and pressure stress are additive.

$$K_{IM} = M_M \frac{PR}{t}$$

M_M = ASME III, Figure G-2214-1

P = Pressure, psia

R = Vessel Radius - in.

t = Vessel Wall Thickness - in.

$$K_{IT} = M_T \Delta T_W$$

M_T = ASME III, Figure G-2214-2

ΔT_W = Highest Radial Temperature Gradient Through Wall at End of Cooldown

K_{IT} is, therefore, calculated at a maximum gradient and is considered a constant = A for cooldown and zero for heatup. $\frac{M_M}{t}$ R is also a constant = B.

Therefore:

$$K_{IR} = AP + B$$
$$P = \frac{K_{IR} - B}{A}$$

K_{IR} is then varied as a function of temperature from Figure G-2110-1 of ASME III and the allowable pressure calculated. Hydrostatic head (48 psi) and instrumentation errors (12°F and 32 psi) are considered when plotting the curves.

B. System Hydrostatic Test - The system hydrostatic test curve is developed in the same manner as in A above with the exception that a safety factor of 1.5 is allowed by ASME III in lieu of 2.

C. Lowest Service Temperature = 50°F + 120°F + 12°F = 182°F. As indicated previously, an RT_{NDT} for all material with the exception of the reactor vessel belt-line was established at 50°F. 10 CFR 50, Appendix G, IV.a.2, requires a lowest service temperature of RT_{NDT} + 120°F for piping, pumps and valves. Below this temperature a pressure of 20 percent of the system hydrostatic test pressure (.20)(3125) - 48 - 32 psi = 545 psia cannot be exceeded.

2.0 LIMITING CONDITIONS FOR OPERATION
2.1 Reactor Coolant System (Continued)
2.1.2 Heatup and Cooldown Rate (Continued)

D. Boltup Temperature = 10°F + 60°F + 12°F = 82°F. At pressure below 545 psia, a minimum vessel temperature must be maintained to comply with the manufacturer's specifications for tensioning the vessel head. This temperature is based on previous NDTT methods. This temperature corresponds to the measured 10°F NDTT of the reactor vessel flange, which is not subject to radiation damage, plus 60°F data scatter in NDTT measurements, plus 12°F instrument error.

References

(1) USAR, Section 4.2.2

(2) ASME Boiler and Pressure Vessel Code, Section III

(3) USAR, Section 4.2.4

(4) USAR, Section 3.4.6

(5) Omaha Public Power District, Fort Calhoun Station Unit No. 1, Evaluation of Irradiated Capsule W-225, Revision 1, August, 1980

(6) Technical Specification 2.3(3)

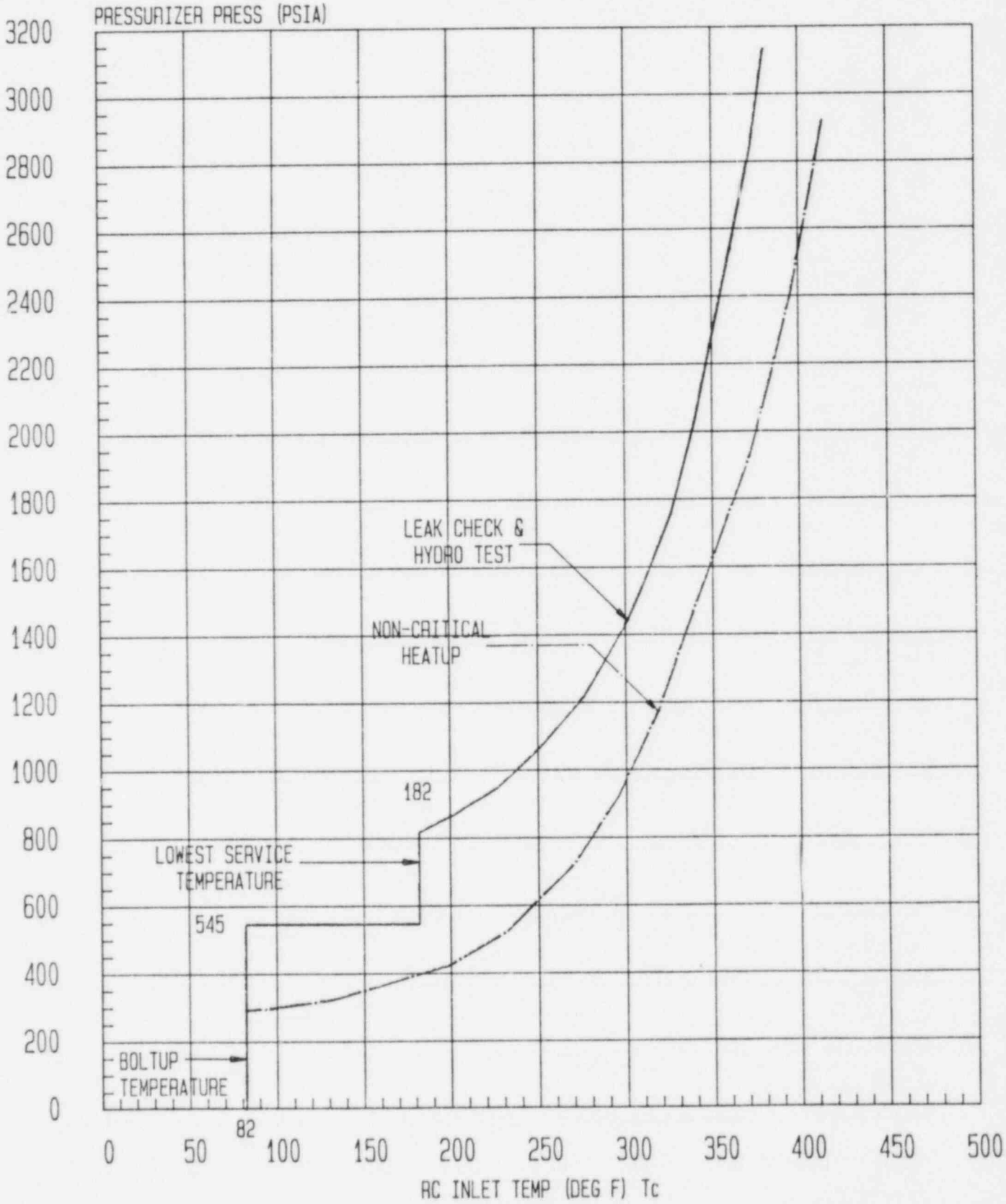
(7) Article IWB-5000, ASME Boiler and Pressure Vessel Code, Section XI

(8) Omaha Public Power District, Fort Calhoun Station Unit No. 1, Evaluation of Irradiated Capsule W-265, March 1984

RCS PRESS-TEMP LIMITS HEATUP 15 EFPY

REACTOR NOT CRITICAL

1500 MWt



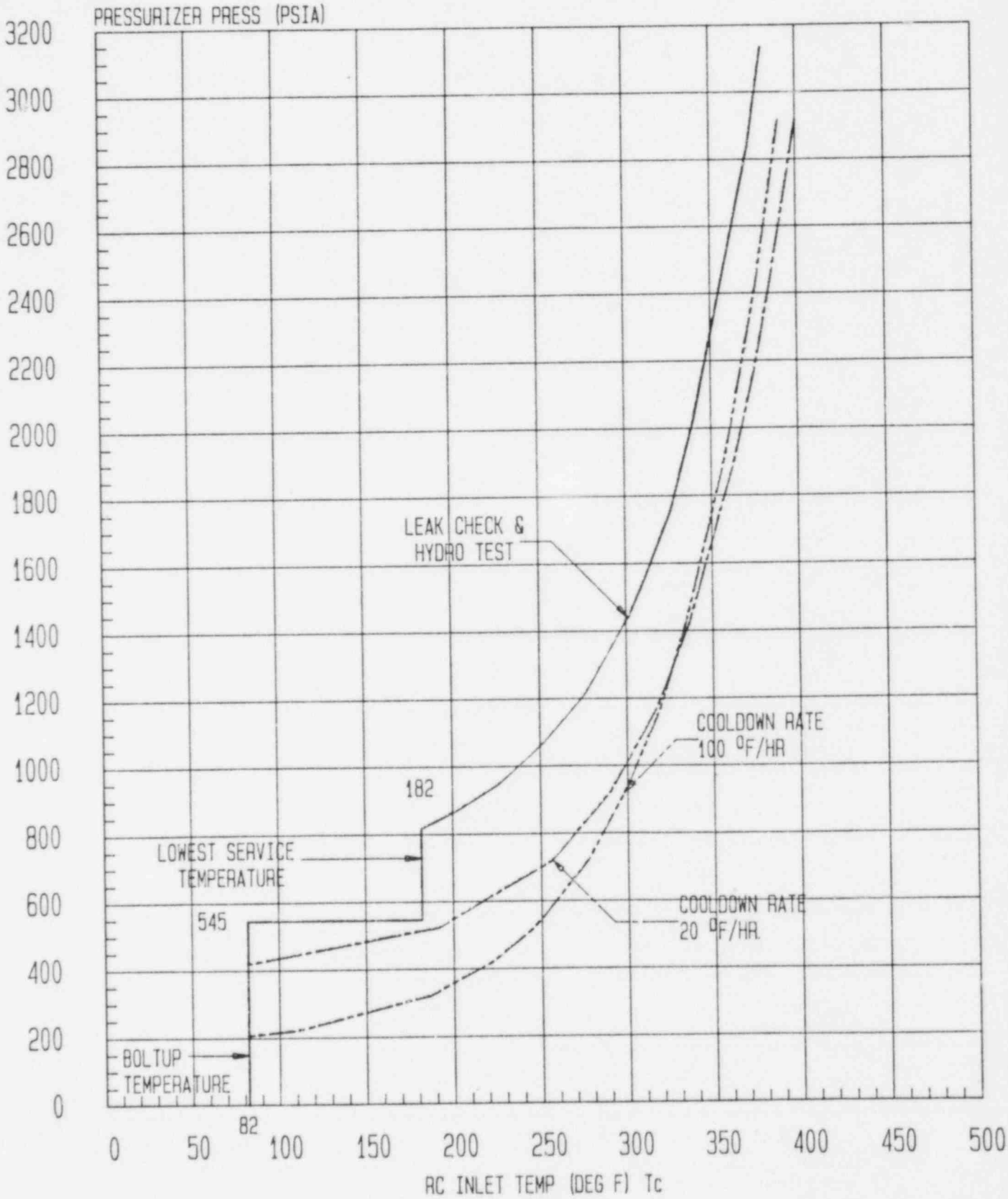
FORT CALHOUN
TECHNICAL SPECIFICATIONS

FIGURE
2-1A

RCS PRESS-TEMP LIMITS COOLDOWN 15 EFPY

REACTOR NOT CRITICAL

1500 MWt

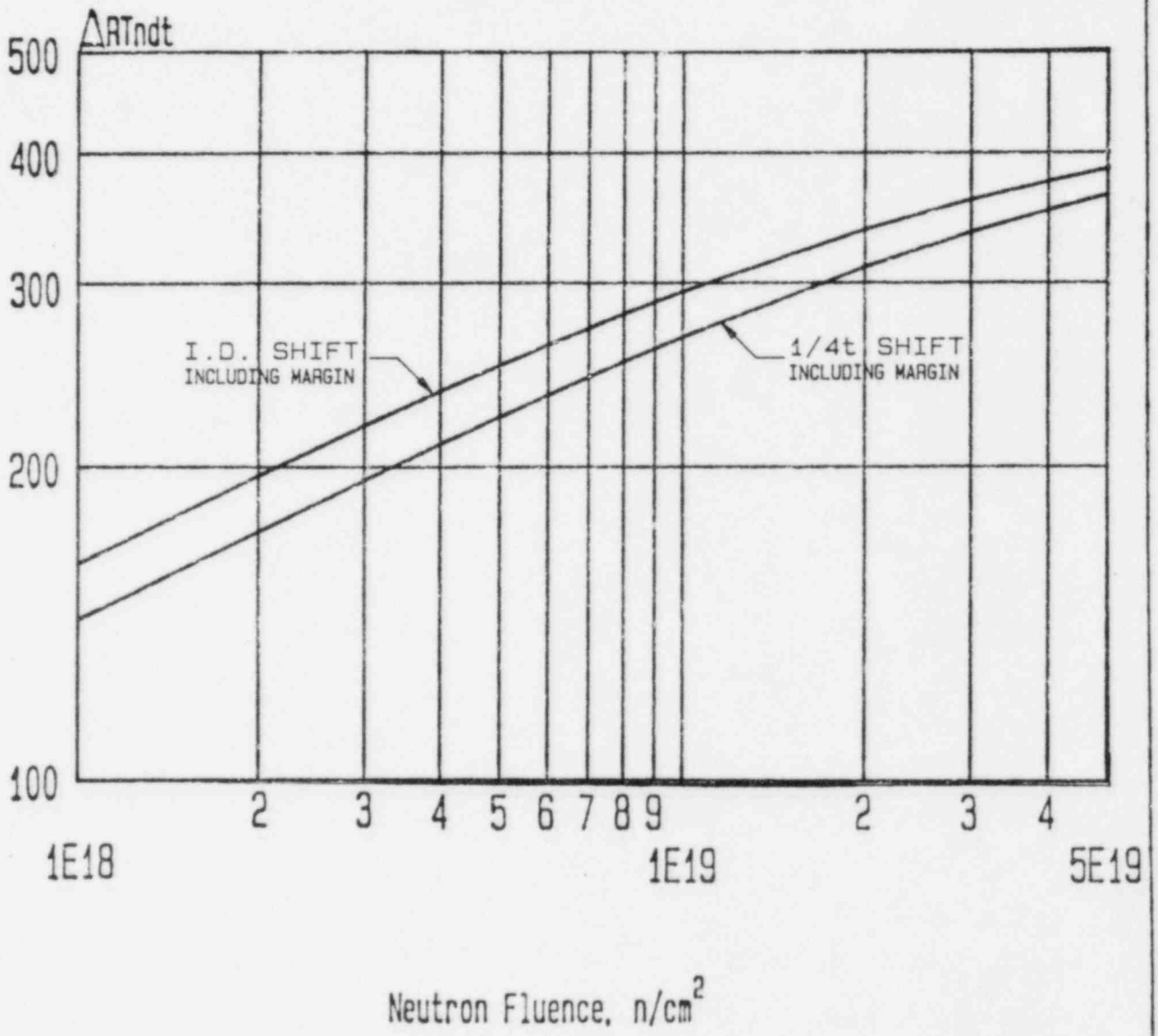


FORT CALHOUN
TECHNICAL SPECIFICATIONS

FIGURE
2-1B

PREDICTED RADIATION INDUCED NDTT SHIFT

FORT CALHOUN REACTOR VESSEL BELTLINE



FORT CALHOUN
TECHNICAL
SPECIFICATIONS

FIGURE
2-3

(3) Protection Against Low Temperature Overpressurization

The following limiting conditions shall be applied during scheduled heatups and cooldowns. Disabling of the HPSI pumps need not be required if the reactor vessel head, a pressurizer safety valve, or a PORV is removed.

Whenever the reactor coolant system cold leg temperature is below 320°F, at least one (1) HPSI pump shall be disabled.

Whenever the reactor coolant system cold leg temperature is below 312°F, at least two (2) HPSI pumps shall be disabled.

Whenever the reactor coolant system cold leg temperature is below 271°F, all three (3) HPSI pumps shall be disabled.

In the event that no charging pumps are operable, a single HPSI pump may be made operable and utilized for boric acid injection to the core.

Basis

The normal procedure for starting the reactor is to first heat the reactor coolant to near operating temperature by running the reactor coolant pumps. The reactor is then made critical by withdrawing CEA's and diluting boron in the reactor coolant. With this mode of start-up, the energy stored in the reactor coolant during the approach to criticality is substantially equal to that during power operation and therefore all engineered safety features and auxiliary cooling systems are required to be fully operable. During low power physics tests at low temperatures, there is a negligible amount of stored energy in the reactor coolant; therefore, an accident comparable in severity to the design basis accident is not possible and the engineered safeguards systems are not required.

The SIRW tank contains a minimum of 283,000 gallons of usable water containing at least 1700 ppm boron.(1) This is sufficient boron concentration to provide a shutdown margin of 5%, including allowances for uncertainties, with all control rods withdrawn and a new core at a temperature of 60°F.(2)

The limits for the safety injection tank pressure and volume assure the required amount of water injection during an accident and are based on values used for the accident analyses. The minimum 116.2 inch level corresponds to a volume of 825 ft^3 and the maximum 128.1 inch level corresponds to a volume of 895.5 ft^3.

Prior to the time the reactor is brought critical, the valving of the safety injection system must be checked for correct alignment and appropriate valves locked. Since the system is used for shut down cooling, the valving will be changed and must be properly aligned prior to start-up of the reactor.

be available for emergency core cooling, but the contents of one of the tanks is assumed to be lost through the reactor coolant system. In addition, of the three high-pressure safety injection pumps and the two low-pressure safety injection pumps, for large break analysis it is assumed that two high pressure and one low pressure operate while only one of each type is assumed to operate in the small break analysis[5]; and also that 23% of their combined discharge rate is lost from the reactor coolant system out of the break. The transient hot spot fuel clad temperatures for the break sizes considered are shown on FSAR, Appendix K, Figures 1-19 (Amendment No. 34).

Inadvertent actuation of three (3) HPSI pumps and three (3) charging pumps, coincident with the opening of one of the two PORV's, would result in a peak primary system pressure of 1190 psia. 1190 psia corresponds with a minimum permissible temperature of 320°F on Figure 2-1B. Thus, at least one HPSI pump is disabled at 320°F.

Inadvertent actuation of two (2) HPSI pumps and three (3) charging pumps, coincident with the opening of one of the two PORV's, would result in a peak primary system pressure of 1040 psia. 1040 psia corresponds with a minimum permissible temperature of 312°F on Figure 2-1B. Thus, at least two HPSI pumps will be disabled at 312°F.

Inadvertent actuation of one (1) HPSI and three (3) charging pumps, coincident with opening of one of the two PORV's, would result in a peak primary system pressure of 685 psia. 685 psia corresponds with a minimum allowable temperature of 271°F on Figure 2-1B. Thus all three HPSI pumps will be disabled at 271°F.

Inadvertent actuation of three (3) charging pumps, coincident with the opening of one of the two PORV's, would result in a peak primary system pressure of 160 psia. 160 psia would correspond with a minimum allowable temperature that is less than the 82°F boltup temperature limit on Figure 2-1B. Therefore, operation of the charging pumps need not be restricted.

Removal of the reactor vessel head, one pressurizer safety valve, or one PORV provides sufficient expansion volume to limit any of the design basis pressure transients. Thus, no additional relief capacity is required.

Technical Specification 2.2(1) specifies that, when fuel is in the reactor, at least one flow path shall be provided for boric acid injection to the core. Should boric acid injection become necessary, and no charging pumps are operable, operation of a single HPSI pump would provide the required flow path.

ATTACHMENT B

JUSTIFICATION, DISCUSSION, AND

SIGNIFICANT HAZARDS CONSIDERATIONS

FOR HEATUP AND COOLDOWN CURVES

The Fort Calhoun Technical Specifications are being amended to update the current heatup and cooldown limit curves for continued safe operation of the reactor vessel and associated primary coolant system beyond 8.5 Equivalent Full Power Years (EFPY). This application requests continued operation through 15 EFPY.

When determining the Limiting Conditions for Operation, the impact of the initial nil-ductility transition reference temperature (RT_{NDT}), the fluence induced RT_{NDT} shift of reactor vessel beltline welds and an appropriate margin must be considered. The fluence induced temperature shift is a function of fluence and the chemical composition of the limiting reactor vessel beltline material. In the past, the absence of specific weld chemical composition data required assumption of upper bound values for beltline weld copper and nickel content. The chemical composition of all Fort Calhoun reactor vessel beltline welds was recently documented through searches of Combustion Engineering (CE) welding records and through analysis of physical weld samples removed from identical welds traced to the reactor vessel head. Additional information on this project is provided in the attached discussion section. With specific weld chemical composition data, it is no longer necessary to assume the upper bound copper and nickel values for these welds. The effect of this change has been evaluated and the lower shell longitudinal weld seam, 3-410 was determined to be the most limiting beltline material with 0.23 w/o copper and 0.95 w/o nickel. The projected fluence at the

critical reactor vessel beltline weld was determined using the following equation:

$$\phi_{I.D.} = [8.8x10^{18} + (EFPY - 5.92)\ (4.8x10^{19})/32]\ (.60)\ \ n/cm^2$$

This equation was developed using results from the analysis of surveillance capsule W-265 which was removed after 5.92 EFPY, the end of Cycle 7. The predicted reactor vessel I.D. fluence was reported to be $8.8x10^{18}$ n/cm^2 by Combustion Engineering in the W-265 analysis report and was projected to reach $4.8x10^{19}$ n/cm^2 by the end-of-life. However, following Cycle 7, reduced fluence core loading patterns were initiated. To take credit for this reduction, an azimuthal flux distribution plot for Cycles 1-9 average, generated using DOT 4.3, was used to determine the azimuthal flux reduction value at the critical weld location (.60) for use in the fluence equation above.

The shift prediction equation developed in the proposed Revision 2 of Regulatory Guide 1.99 was chosen for use in this submittal, however, future shift calculations will re-evaluate its applicability to the Fort Calhoun Station and consider other correlations which may better fit the available data. The corrected attenuation equation developed in the CE Owners Group Response to Proposed Revision 02, Reg. Guide 1.99 was used to predict the fluence into the reactor vessel wall. The root-mean-squares method was used to combine the margin associated with the initial RT_{NDT} (17°F) and the margin associated with the shift prediction equation (28°F). The predicted temperature shift plus a 2 sigma margin were applied to -56°F initial RT_{NDT} baseline curves to generate the proposed 15 EFPY heatup and cooldown limit curves. Likewise, Figure 2-3 has been revised to predict the fluence induced temperature shift, including margin, for the limiting reactor vessel beltline ma-

terial. These curves will ensure that adequate fracture toughness is maintained throughout all conditions of normal operation, including anticipated operational transients and system hydrostatic tests.

The proposed 15 EFPY heatup and cooldown limit curves are required for operation beyond approximately August 25, 1986. Commission approval of the proposed Technical Specification is, therefore, requested prior to August 10, 1986 to ensure adequate curves are available for continued operation throughout the cycle without interruption.

Significant Hazards Considerations

This Technical Specification amendment will not increase the probability of occurrences or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report because the change maintains conservative restrictions on pressure-temperature limits for the reactor vessel based on recently obtained beltline weld chemical composition data and the Cycle 1-9 average azimuthal flux distribution.

The probability of an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report will not be created because this application only revises the heatup and cooldown curves which are bounded by the existing Safety Analysis Report.

The margin of safety as defined in the basis for the Technical Specifications will not be reduced because the shift prediction equation of the proposed Revision 2 of Regulatory Guide 1.99 has been used to determine the value of the RT_{NDT} shift.

ATTACHMENT B

DISCUSSION

The following is a discussion of recently obtained reactor vessel beltline weld chemical composition data for the Fort Calhoun Station and its impact on Fort Calhoun's position with regard to the pressurized thermal shock (PTS) issue.

In an effort to better address the PTS issue, a search of Combustion Engineering's, (C-E's) weld records was performed in 1984 to determine beltline weld chemistries for the Fort Calhoun Station reactor vessel. Weld chemical composition analyses were obtained for all weld wire heats used in the beltline region except weld wire heat 51989, which was used in the middle shell longitudinal seam welds. This heat was traced to the torus longitudinal seam welds on the Fort Calhoun reactor vessel head and it was determined that weld chip samples could be removed from these welds for analysis. A conference call involving OPPD, C-E and NRC personnel was held on 7/30/85 to review plans for sampling this material during the 1985 refueling outage. As a result of this conversation and others between C-E and NRC personnel, several items of NRC concern were introduced and preliminarily resolved. These NRC concerns are addressed in the following paragraphs.

The first concern was how the location of the weld seams could be accurately distinguished from the surrounding base metal. This was accomplished by polishing the areas to be sampled and then etching them with a nitric acid solution to reveal the weld outline. After the chip samples were removed the sample areas were blended and inspected by magnetic particle testing.

Photographs were taken of the prepared surfaces after etching, after chip removal and after blending at all locations to document that only weld material was removed.

The second concern was the ability to distinguish between uncharted weld repairs and cosmetic welds. C-E has determined that the possibility of an uncharted weld repair was small and that cosmetic welds were not performed on the OD weld surface, rather the submerged arc weld was ground smooth to the surface. To further minimize this concern, duplicate samples were obtained for each weld from different locations.

The results of an optical emission chemical analyses performed on the chip samples, including a check analysis by x-ray fluorescence for copper and nickel content, are shown in Table 1. The report for all elements determined in the optical emission analysis is attached as Table 2. The optical emission values are an average of two analyses. The x-ray fluorescence values represent a single analysis. The chemical analysis results for wire heat 51989 are consistent with the values expected for a weld made with a Mil B-4 wire and a Linde Type 124 flux, as is indicated for these welds by the weld information records. Likewise, the results of the chemical analyses for wire heat 13253 are consistent with a Mil B-4 Modified wire and Linde Type 1092 flux. The D. C. Cook and Salem 2 surveillance welds were also made with heat 13253 and a Linde Type 1092 flux. The nickel content of the Salem 2 (0.72 w/o) and D. C. Cook (0.74 w/o) surveillance welds are almost identical to the Fort Calhoun (0.72 w/o) value, indicating that 13253 was the wire heat used, and further indicating that weld metal was sampled, since the base metal normally contains less than 0.60 w/o nickel. The copper contents between these welds vary significantly (D. C. Cook - 0.27 w/o, Salem 2 - 0.23 w/o, Fort Calhoun -

0.14 w/o). This can be attributed to the variation in copper coating on the coils of the wire making a heat of weld wire. This wide variation in copper has been observed on several other heats of wire for which multiple analyses are available. A portion of the samples from each weld seam was metallographically examined using a Nital solution to reveal the microstructural characteristics. In all cases, the examination showed the fine-grained ferritic structure of weld metal.

The adequacy of a smaller chip sample as opposed to a full boat sample was also questioned. This has been addressed by the fact that the Fort Calhoun closure head has a relatively small allowance for the removal of a sample and anything larger than the proposed chip sample might require a UT inspection and some degree of analysis. Meaningful results have been obtained since it was possible to do metallography and chemical analysis on the same chip specimens.

Using the results of the closure head weld sampling for heats 51989 and 13253, and other available records, copper and nickel contents have been determined for each weld deposit in the Fort Calhoun reactor vessel beltline. These copper and nickel contents are presented in Table 3. The chemistry established for wire heat 27204 resulted from a search in October 1985 of the C-E Metallurgical and Materials Laboratory chemical analysis log books for weld deposit information and a review of data for the Diablo Canyon Unit #1 surveillance weld made with heat 27204. The lower shell longitudinal seam welds were each made using three heats of wire (27204, 12008, and 13253). It is not known whether only one or a combination of two of the wires were used to weld the ID of the seam. It was assumed for conservatism that the weld wire with the highest chemistry factor was used to weld the ID of the seam.

0.14 w/o). This can be attributed to the variation in copper coating on the coils of the wire making a heat of weld wire. This wide variation in copper has been observed on several other heats of wire for which multiple analyses are available. A portion of the samples from each weld seam was metallographically examined using a Nital solution to reveal the microstructural characteristics. In all cases, the examination showed the fine-grained ferritic structure of weld metal.

The adequacy of a smaller chip sample as opposed to a full boat sample was also questioned. This has been addressed by the fact that the Fort Calhoun closure head has a relatively small allowance for the removal of a sample and anything larger than the proposed chip sample might require a UT inspection and some degree of analysis. Meaningful results have been obtained since it was possible to do metallography and chemical analysis on the same chip specimens.

Using the results of the closure head weld sampling for heats 51989 and 13253, and other available records, copper and nickel contents have been determined for each weld deposit in the Fort Calhoun reactor vessel beltline. These copper and nickel contents are presented in Table 3. The chemistry established for wire heat 27204 resulted from a search in October 1985 of the C-E Metallurgical and Materials Laboratory chemical analysis log books for weld deposit information and a review of data for the Diablo Canyon Unit #1 surveillance weld made with heat 27204. The lower shell longitudinal seam welds were each made using three heats of wire (27204, 12008, and 13253). It is not known whether only one or a combination of two of the wires were used to weld the ID of the seam. It was assumed for conservatism that the weld wire with the highest chemistry factor was used to weld the ID of the seam. Therefore, the copper and nickel content of wire heat 12008 was used for the evaluation of these weld seams.

Table 1

Wire Type:	Mil B-4	Mil B-4	Mil B-4 Mod	Mil B-4 Mod
Heat No.:	51989	51989	13253	13253
Flux Type:	Linde 124	Linde 124	Linde 1092	Linde 1092
Flux Lot:	3687	3687	3791	3791
Weld Seam:	1-415C	1-415E	2-145/A (near 1-415C)	2-415/A (near 1-415E)
CE Lab No.:	D-41589	D-41591	D-41588	D-41590

	Optical Emission (X-Ray Flour.) W/O	Optical Emission (X-Ray Flour.) W/O	Optical Emission (X-Ray Flour.) W/O	Optical Emission (X-Ray Flour.) W/O
C	0.11	0.096	0.11	0.12
Mn	1.39	1.50	1.10	1.14
P	0.011	0.013	0.010	0.013
S	0.009	0.011	0.008	0.011
Si	0.30	0.36	0.17	0.18
Ni	0.20 (0.18)	0.13 (0.114)	0.72 (0.72)	0.74 (0.72)
Cr	0.08	0.08	0.04	0.04
Mo	0.47	0.52	0.43	0.44
Cu	0.16 (0.17)	0.18 (0.18)	0.14 (0.14)	0.14 (0.14)

COMBUSTION ENGINEERING
METALLURGICAL & MATERIALS LABORATORY

DATE: 11-13-85
P.O. NO.
C-E JOB NO. 99759617
PROJECT NO. 960001

CHEMICAL ANALYSIS
REPORT

C-E Lab No.	D41588	D41589	D41590	D41591
Customer No.				
Description	Area 1 2-415-A	Area 1 1-415-C	Area 2 2-415-A	Area 2 1-415-E
C	.11	.11	.12	.096
Mn	1.10	1.39	1.14	1.50
P	.010	.011	.013	.013
S	.008	.009	.011	.011
Si	.17	.30	.18	.36
Ni	.72	.20	.74	.13
Cr	.04	.08	.04	.08
Mo	.43	.47	.44	.52
V	.003	.004	.003	.004
Cb	< .01	< .01	< .01	< .01
Ti	< .01	< .01	< .01	< .01
Co	.016	.012	.016	.011
Cu	.14	.16	.14	.18
Al	.002	.006	.002	.007
B	< .001	< .001	< .001	< .001
W	< .01	< .01	< .01	< .01
As	.010	.010	.013	.011
Sn	.005	.006	.006	.006
Zr	< .001	< .001	< .001	< .001
N	.009	.015	.009	.009
O				
Fe				

Reactor Vessel Closure Head
Omaha Public Power Dist.
Fort Calhoun Station

Reported by: H. Anthony Stepl

Combustion Engineering, Inc.
911 West Main Street
Chattanooga, Tennessee 37402
(615) 265-4631
Telex 558429

Table 3

Chemical Content of Fort Calhoun
Beltline Welds

Weld Seam	Material (Wire Heat/Flux Lot)	Chemical Content Cu	Chemical Content Ni	Comment/Source
2-410 A/C	51989/3687	0.17	0.17	Fort Calhoun closure head longitudinal weld sample.
3-410 A/C	27204/3774	0.22	1.02	Average of multiple weld deposit records including PG&E Diablo Canyon surveillance weld.
	13253/3774	0.21	0.73	Average of Salem #2 and Cook #1 surveillance welds, and Fort Calhoun closure head torus-to-dome girth seam weld samples.
	12008/3774	0.23	0.95	Average of multiple weld deposit records of tandem arc welds in which second weld wire heat copper content known.
9-410	20291/3833	0.21	0.74	Cooper Station surveillance weld.
8-410	13253/3774	0.21	0.73	(see 3-410)

ATTACHMENT C

Question 1:

Weld metal margin: Why was no margin applied to ΔRT_{NDT}?

Response: The base curves had some margin included in them. It was $2\sigma_i$ = 34°F. Since some applications of these curves also require the incorporation of σ_Δ, all margin has been removed from the baseline curves. When margin needs to be included, all of the appropriate terms and values can be clearly included.

Question 2:

NRC Staff believes R.G. 1.99, Rev. 2, is more representative than use of Guthrie formula.

Response: The District believes that the Guthrie formula, as defined in 10 CFR 50.61, is a representative formula not only for PTS screening criteria calculation but also for heatup/cooldown curve calculations.

As requested, the District will use proposed Reg. Guide 1.99, Rev. 2, to calculate heatup/cooldown curves for this submittal. The District still believes that the 10 CFR 50.61 equation is valid for this purpose and has not eliminated its use from future submittals. The District further states that 10 CFR 50.61 is the only method it will use to calculate EOL fluences and fluences at screening criteria limits.

Question 3:

Staff calculates 20% of the pre-service system hydro test pressure as 0.2 x 1.25 x 2500 = 625 psia vs the 545 psia in the submittal. Why the discrepancy?

Response: No discrepancy exists. The calculation from the 625 psia to 545 psia is clearly stated in the current Technical Specifications on page 2-7 in section 2.1.2.C.

Question 4:

How did OPPD obtain 162°F for lowest service temperature? Staff calculates 182°F.

Response: The District concurs that 182°F should be used and that the revised submittal will incorporate a lowest service temperature of 182°F.

Question 5:

Submittal has omitted a page of references.

Response: The District has included the references in the text.

Question 6:

How was the fluence attenuated through the vessel wall?

Response: The attenuation was based on information contained in W-265, a surveillance capsule examination report prepared by Combustion Engineering. Page 36 of the report showed ID, 1/4t, and 3/4t fluences. These values were ratioed. The ratios were used for future attenuation calculations. The ratios are used only when attenuation is not clearly defined.

In the current submittal using proposed R.G. 1.99, Rev. 2, the District will use the following equation for attenuation:

$$\Delta RT_{NDT}/\Delta RT_{NDT_{surface}} = e^{-(0.067x + 0.0025x^2)_f} \; 0.0208x$$

Question 7:

The fluence in the May 7, 1986 PTS submittal and the fluence in this submittal are not consistent. Why the discrepancy?

Response: The fluences for each submittal were calculated with different assumptions. The PTS submittal was calculated with no flux reduction credit assumed for Cycles 1 through 7. Credit was taken for fuel loading pattern changes (low radial leakage) made for Cycle 8 and beyond. The equation stated below mathematically reflects these assumptions.

$$\phi_{ID} = 8.8\text{x}10^{18} + 0.7(\text{EFPY} - 5.92)(4.8\text{x}10^{19})/32 \text{ n/cm}^2$$

The Technical Specification submittal utilized DOT 4.3 calculations to determine a Cycle 1-9 average azimuthal flux reduction. This was incorporated into the fluence extrapolation equation as shown below.

Longitudinal Welds

$$\phi_{ID} = [8.8\text{x}10^{18} + (\text{EFPY} - 5.92)(4.8\text{x}10^{19})/32] \; (.6) \quad \text{n/cm}^2$$

Circumferential Welds

$$\phi_{ID} = [8.8\text{x}10^{18} + (\text{EFPY} - 5.92)(4.8\text{x}10^{19})/32] \; (.92) \quad \text{n/cm}^2$$

Question 8:

The 100°/hour heatup curves for the 3/4t should be included.

Response: The District's submittal will include 3/4t curves when they are limiting.

Question 9:

There are no pressure temperature limits for heatup or cooldown for the reactor critical.

Response: The District removed these curves via a previous amendment. They are not restrictive since present Technical Specifications require all 4

reactor coolant pumps to be in operation prior to criticality. Additionally, the inlet temperature must be at least 515°F prior to criticality. The only exception is for physics testing when the power level shall be 10^{-1}% or less. Further, the reactor cannot be made or maintained critical unless the inlet temperature is NDTT + 120°F. These restrictions are delineated in Technical Specification sections; 2.1.1(1), 1.3 Table 1.1 No. 2, 2.10.1(1), and 2.10.1(2) which are located on pages 2-1, 1-10a and 2-48, respectively. The District feels that incorporating additional if not less restrictive guidance could be confusing and lead to a violation of the more restrictive requirements.