PRESSURE/TEMPERATURE OPERATING CURVES AND ASSESSMENT OF RT_{PTS} CONCERNS FOR KEWAUNEE NUCLEAR PLANT

by

P. K. Nair
E. B. Norris

Final Report

SwRI Project 06-8919

for

Wisconsin Public Service Corporation
P.O. Box 19001
700 N. Adams St.
Green Bay, WI 54307

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ABSTRACT

An assessment of Kewaunee Nuclear Plant Reactor Vessel heatup and cooldown curves and the impact of RT_{PTS} conditions after 11 cycles of operation was conducted.

Revised vessel fluence levels were calculated and projection of vessel ID peak fluence as a function of EFPY was developed. RT_{PTS} values were calculated to satisfy 10 CFR 50.61 requirements. New heatup and cooldown curves were developed for 15 EFPY conditions based on the proposed NRC Regulatory Guide 1.99 Rev. 2.

I. SUMMARY OF RESULTS AND CONCLUSIONS

The analysis of the reactor vessel material conditions was conducted for the Kewaunee Nuclear Power Plant. The following results and conclusions were generated.

1. The reactor vessel after 11 cycles of operation has completed 9.41 effective full power years. The calculated maximum fluence received by the vessel ID is 1.25×10^{19} neutrons/cm^2 (E>1MeV).
2. Projected fluence levels for 15 EFPY and 35 EFPY are 2.0×10^{19} neutrons/cm^2 (E>1MeV) and 4.65×10^{19} neutrons/cm^2 (E>1MeV).
3. For the Linde 1092 material and Cu = 0.24% and Ni = 0.78% there is no pressurized thermal shock concern through 35 EFPY's. This assessment is based on the screening criteria of 10CFR50.61.
4. The capsule lead factors were calculated and are comparable with WCAP 9878 results. SwRI suggests removal of capsule S after 1.5 times peak end of life vessel fluence instead of the current 2. There is no change to the removal schedule for the remaining capsules.
5. Adjusted reference temperatures (ART) for 15 EFPY were calculated for vessel wall inner surface, at 1/4 T and at 3/4 T. The values are 251°F (surface), 226°F (1/4 T) and 183°F (3/4 T).
6. Heatup and cooldown limit curves were developed for 15 EFPY with the above described ART values and instrument errors of -60 psig and +10°F. Regulatory Guide 1.99 Rev. 2 was used in all the analysis.

II. BACKGROUND

The allowable loadings on nuclear pressure vessels are determined by applying the rules in Appendix G, "Fracture Toughness Requirements," of 10CFR50 [1]. In the case of pressure-retaining components made of ferritic materials, the allowable loadings depend on the reference stress intensity factor (K_{IR}) curve indexed to the reference nil ductility temperature (RT_{NDT}) presented in Appendix G, "Protection Against Non-Ductile Failure," of Section III of the ASME Code [2]. The materials in the beltline region of the reactor vessel must be monitored for radiation induced changes in RT_{NDT} per requirements of Appendix H, "Reactor Vessel Material Surveillance Program Requirements," of 10CFR50.

The RT_{NDT} must be established for all materials, including weld metal, heat-affected zone material and base materials that constitute the reactor coolant pressure boundary.

It is well established that ferritic materials undergo an increase in strength and hardness and a decrease in ductility and toughness when exposed to neutron fluences in excess of 10^{17} neutron per cm^2 (E>1MeV) [3]. Also, it has been established that certain elements, particularly copper, and nickel affect the radiation embrittlement response of ferritic materials [4-6]. The relationship between the increase in the transition temperature and the copper and nickel content is defined in NRC Regulatory Guide 1.99 Rev. 2 [7]. This document is currently out "for comment" and is expected to be recommended for all future evaluation of reactor vessel loadings.

Regulatory Guide 1.99, Rev. 2 has modified the Rev. 1 approach to using RT_{NDT} in computations. An adjusted RT_{NDT} is now used and is defined as RT_{ART}. The RT_{ART} incorporates initial RT_{NDT} of the material plus the change in RT_{NDT} due to irradiation and a factor termed as the 'margin'. The 'margin' reflects the two standard deviation of the data base from which the initial RT_{NDT} of the material is assessed.

In July 1985 a new document, 10CFR50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events" [8] was issued. This document was specifically issued to provide guidance and a screening criteria for permissible embrittlement due to irradiation of vessel materials. A new transition temperature, RT_{PTS}, was established. Under the rule, the screening criteria, $RT_{PTS} \leq 270°F$ for plates, forgings and axial welds and $RT_{PTS} \leq 300°F$ for circumferential welds has been established. Materials exceeding the RT_{PTS} limits would require additional evaluation and actions to reduce flux levels and/or provide system modifications to reduce the probability of vessel failure due to postulated pressurized thermal shock events.

In general, the only ferritic pressure boundary materials in a reactor vessel which are expected to receive a fluence to affect both the RT_{ART} and the RT_{PTS} are those materials which are located in the core beltline region of the reactor pressure vessel. Therefore, material surveillance programs include specimens machined from plate or forging materials and weldments which are located in the core beltline region of high flux density. ASTM E 185 [9] describes the recommended practice for monitoring and evaluating the radiation-induced changes occurring in the mechanical properties of presssure vessel beltline materials.

Westinghouse has provided such a surveillance program for the Kewaunee Nuclear Power Plant. A total of six surveillance capsules were installed in the Kewaunee reactor pressure vessel. The six capsules were positioned in the reactor vessel between the thermal shield and the vessel wall. To date two capsules, V and R, have been removed, tested and analyzed. The testing and analysis was performed by Westinghouse and reported in Reports WCAP-8908 [10] and EPRI RP-1021-3 [11]. Capsule R, the most recent capsule to be removed, had experienced five fuel cycles of operation.

In this report, an assessment of the effect of eleven fuel cycles of operation on current and future reactor vessel operating loads is presented. The evaluation includes a new neutron transport calculation, RT_{PTS} considerations and the development of revised heatup and cooldown limit curves based on Regulatory Guide 1.99 Rev. 2.

III. VESSEL FLUENCE DEVELOPMENT

In this section the reactor vessel fluences are calculated based on the current and projected fuel cycle history. The fluences at the ID of the vessel are used to determine the RT_{PTS} concerns for the vessel.

A. Fluence Calculations

Southwest Research Institute performed a two-dimensional discrete ordinates transport calculation with the DOT IV Code, a 47-group Bugle-80 neutron cross section library (based on the Evaluated Nuclear Data File END F/B-IV(12)), a P_3 expansion of the scattering matrix, and an S_8 order of angular quadrature. A one-eighth segment of a plane through the vertical axis at midheight was used to model the core, core barrel, thermal shield, surveillance capsules (and holders), pressure vessel and water regions using R- θ coordinates. All boundary conditions were reflective at the vessel OD. The salient features of the mesh structure used in the analysis are shown in Figure 1. The dimensions used for the mesh were obtained from Reference [13]. In addition to the physical description the DOT IV input included the average core power distribution on a fuel assembly by fuel assembly basis. These values were obtained by averaging the measured values through eleven cycles of operation. Table 1 presents the values used for the analysis.

The resulting radial and azimuthal distribution of neutrons in the first 18 energy groups were used to calculate the fast (E>1MeV) neutron densities for each mesh point in the matrix. These values at the 13° surveillance capsule location were collapsed with ENDF/B-V cross section values for the following threshold monitor reactions:

$^{54}Fe(n,p)^{54}Mn$

$^{58}Ni(n,p)^{58}Co$

$^{63}Cu(n,\alpha)^{60}Co$

In addition, the three capsule lead factors (13°, 23° and 33° locations) and the azimuthal distribution of fast neutrons at the surveillance capsule and pressure vessel inside radius (I.R.) loci were determined.

The capsule lead factors computed by Southwest Research Institute are compared to the values reported by Westinghouse [11] in Table 2. The two sets of lead factors are in very good agreement considering that the two laboratories utilized different versions of the DOT Code, different sets of cross sections, P_1 and P_3 expansions of the scattering matrices, and individual modeling techniques.

Also included in Table 2 is the capsule removal schedule derived using the SwRI capsule lead factors. The basis for this schedule is ASTM Practice E185-82(9). This schedule, except for the fifth capsule (S), is essentially the same as that proposed in WCAP 9878. The major difference between the Westinghouse and SwRI schedules results from SwRI suggesting that Capsule S be removed after 1.5 times the peak E.O.L. vessel fuence instead of after twice that value as allowed by ASTM E185-82(9). This provides a greater margin for meeting the 1.0 to 2.0 range allowed by the ASTM Practice when making long-term extrapolations.

The azimuthal variations in the fast neutron flux densities calculated for the surveillance capsule and pressure vessel I.D. radial positions are shown in Figure 2. The flux perturbations caused by the presence of the surveillance capsules and holders are clearly evident. Projected vessel I.D. fluences for 19 cycles are presented in Table 3.

A projection of the vessel I.D. peak fluence as a function of Effective Full Power Years (EFPY) of plant operation is given in Figure 3. This was obtained by applying the calculated lead factor for the 13° surveillance capsules to the Capsules R and V exposure values and dosimetry results.

B. RT_{PTS} Assessment

The first step in assessing the effect of irradiation damage to reactor vessel materials is to examine the degree of shift of the material transition temperatures (RT_{PTS}). The screening criteria for this assessment is presented in 10CFR50.61 "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events." The screening criteria is based on the maximum calculated fluence levels at the I.D. of the vessel wall.

The RT_{PTS} is the lower of results obtained by Equations 1 and 2 in 10CFR50.61. The following data is used for the RT_{PTS} evaluation.

1. Equation 1

$$RT_{PTS} = I + M + [-10+470Cu+350CuNi]f^{0.27}$$

where

$I = -56°F$ for Linde 1092 flux

and $M = 59°F$

Equation 2

$$RT_{PTS} = I + M + 283f^{0.194}$$

where

$I = -56°F$

and $M = 34\ °F$

2. Note: Generic values of I and M are being used for the RT_{PTS} evaluation. The reported values for Cu and Ni

Cu = 0.24%

Ni = 0.78%

3. Fluence Levels

At the end of 11 cycles or 9.41 EPFY

$f \times 10^{19} = 1.25 \times 10^{19}$ n/cm^2 (E>1MeV)

At the end of 15 EFPY

$f \times 10^{19} = 2.0 \times 10^{19}$ n/cm^2 (E>1MeV)

At the end of 35 EFPY

$f \times 10^{19} = 4.65 \times 10^{19}$ n/cm^2 (E>1MeV)

The results of the calculations for Equations 1 and 2 are presented in Table 4. It is seen that the lower RT_{PTS} values are represented by Equation 1. The RT_{PTS} after 35 EFPY is below the minimum screening criteria value of 300°F. Hence the vessel will not have a pressurized thermal shock concern according to the requirements in 10CFR50.61.

TABLE 1

AVERAGE THERMAL POWER DISTRIBUTION

CYCLES 1 THROUGH 11

	1	2	3	4	5	6	7
1	1.0171	1.1630	1.1269	1.1970	1.0800	0.9557	0.7043
2		1.1413	1.1831	1.1509	1.1358	1.0849	0.4668
3			1.1014	1.1973	1.0794	0.9309	0.0000
4				1.1086	1.0880	0.5943	0.0000
5					0.6345	0.0000	0.0000
6						0.0000	0.0000
7							0.0000

TABLE 2

CALCULATED LEAD FACTORS FOR SURVEILLANCE CAPSULES

Capsule Code	Position (Degrees)	Lead Factor West.	Lead Factor SwRI	Removal Time	Capsule Fluence (n/cm^2, E>1MEV)
V	77°(13°)	3.37	3.34	1.25 EFPY(a)	5.99×10^{18}
R	257°(13°)	3.37	3.34	4.50 EFPY(a)	2.07×10^{19}
T	67°(23°)	1.94	1.98	11 EFPY(b)	3.3×10^{19}
P	247°(23°)	1.94	1.98	17 EFPY(c)	5.0×10^{19}
S	57°(33°)	1.79	1.96	25 EFPY(d)	7.5×10^{19}
M	237°(33°)	1.79	1.96	(e)	----

(a) Capsule Removed and Fluence Measured (WCAP 9879)

(b) Projected Fluence = 1.0 x 1/4T E.O.L. Vessel Fluence

(c) Projected Fluence = 1.0 x Peak E.O.L. Vessel Fluence

(d) Projected Fluence = 1.5 x Peak E.O.L. Vessel Fluence

(e) Spare Capsule

TABLE 3

PROJECTED VESSEL I.D. FLUENCE

Fuel Cycle	Operating Time (EFPY)	Fast (E >1MeV) Fluence (n/cm^2) (This Cycle)	(Total All Cycles)
1	1.29	1.83×10^{18}	1.83×10^{18}
2	0.68	0.92×10^{18}	2.76×10^{18}
3	1.03	1.40×10^{18}	4.16×10^{18}
4	0.94	1.28×10^{18}	5.44×10^{18}
5	0.69	0.94×10^{18}	6.38×10^{18}
6	0.77	1.05×10^{18}	7.43×10^{18}
7	0.81	1.10×10^{18}	8.53×10^{18}
8	0.79	1.07×10^{18}	9.60×10^{18}
9	0.82	1.12×10^{18}	1.07×10^{19}
10	0.74	1.01×10^{18}	1.17×10^{19}
11	0.84	1.14×10^{18}	1.28×10^{19}
12	0.82	1.12×10^{18}	1.39×10^{19}
13	0.86	1.17×10^{18}	1.51×10^{19}
14	0.87	1.18×10^{18}	1.63×10^{19}
15	0.83	1.13×10^{18}	1.74×10^{19}
16	0.85	1.16×10^{18}	1.86×10^{19}
17	0.85	1.16×10^{18}	1.98×10^{19}
18	0.83	1.13×10^{18}	2.09×10^{19}
19	0.83	1.13×10^{18}	2.20×10^{19}

TABLE 4

RT_{PTS} SCREENING CRITERIA

Fluence n/CM^2	EFPY	RT_{PTS} Equation 1 °F	RT_{PTS} Equation 2 °F
1.25×10^{19}	9.41	182	274
2.0×10^{19}	15.00	206	302
4.65×10^{19}	35.00	258	359

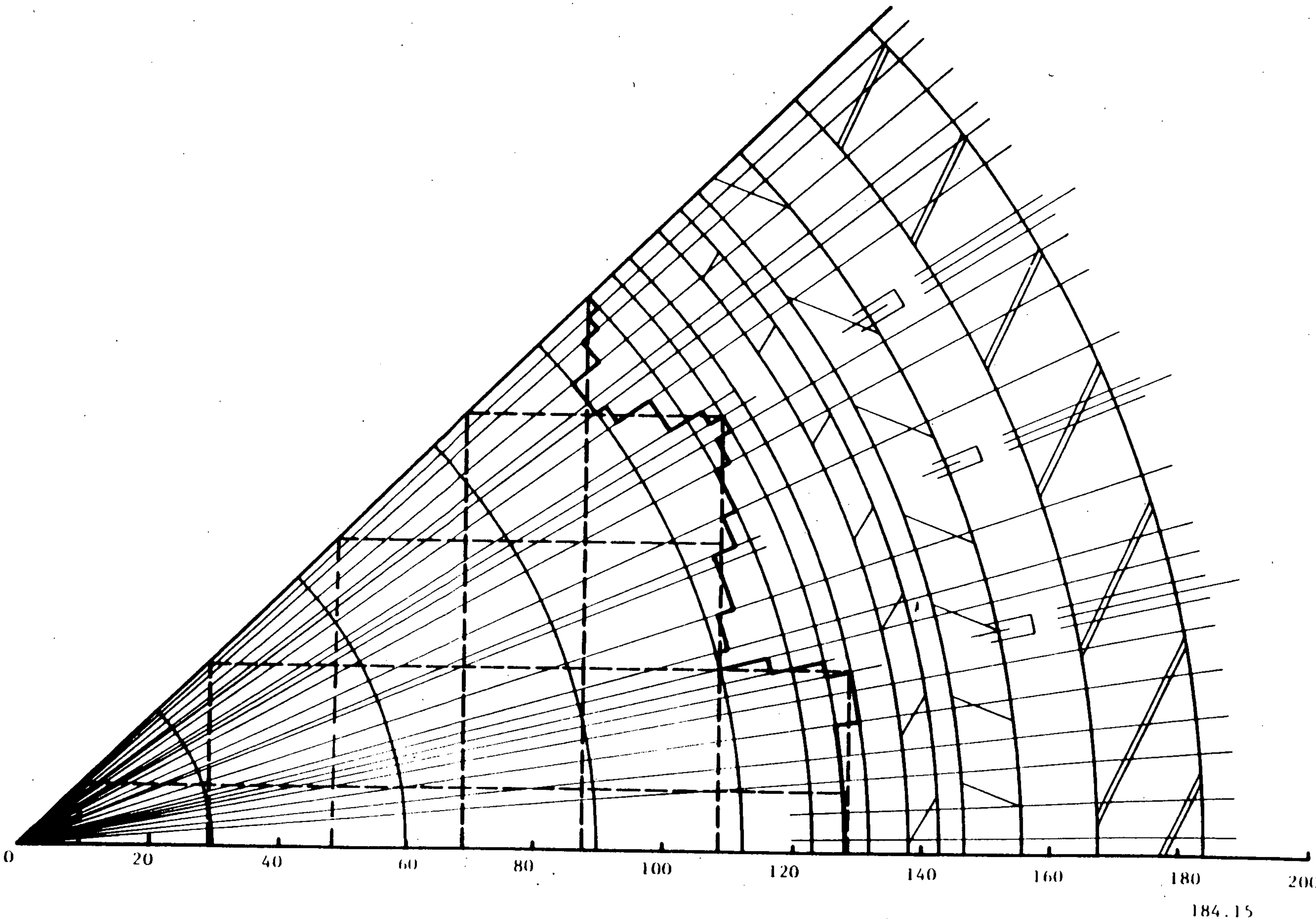

FIGURE 1. MESH FOR THE DISCRETE ORDINATE TRANSPORT CALCULATIONS

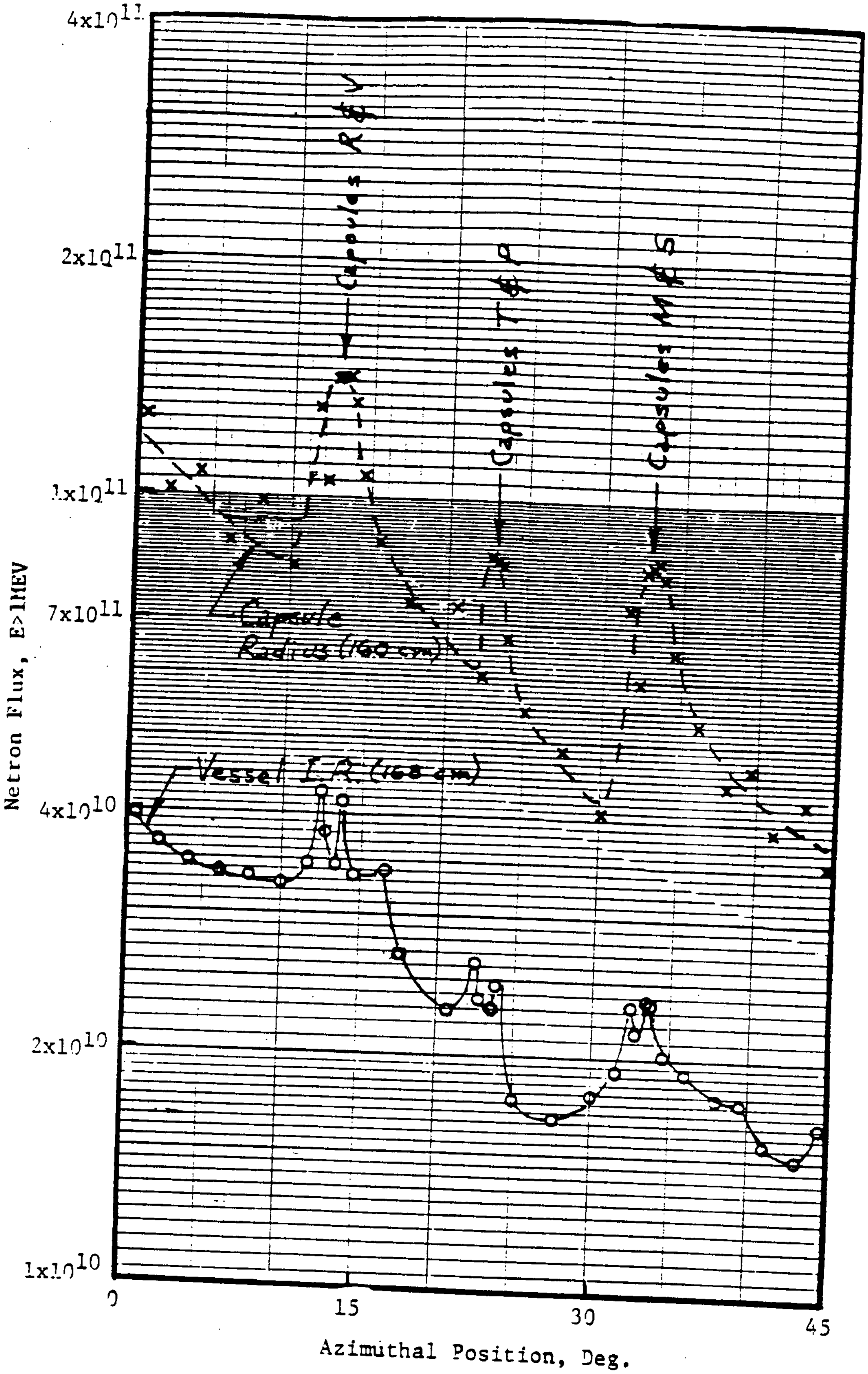

FIGURE 2. RESULTS FROM DOT-IV P3-S8 Rθ TRANSPORT CALCULATION

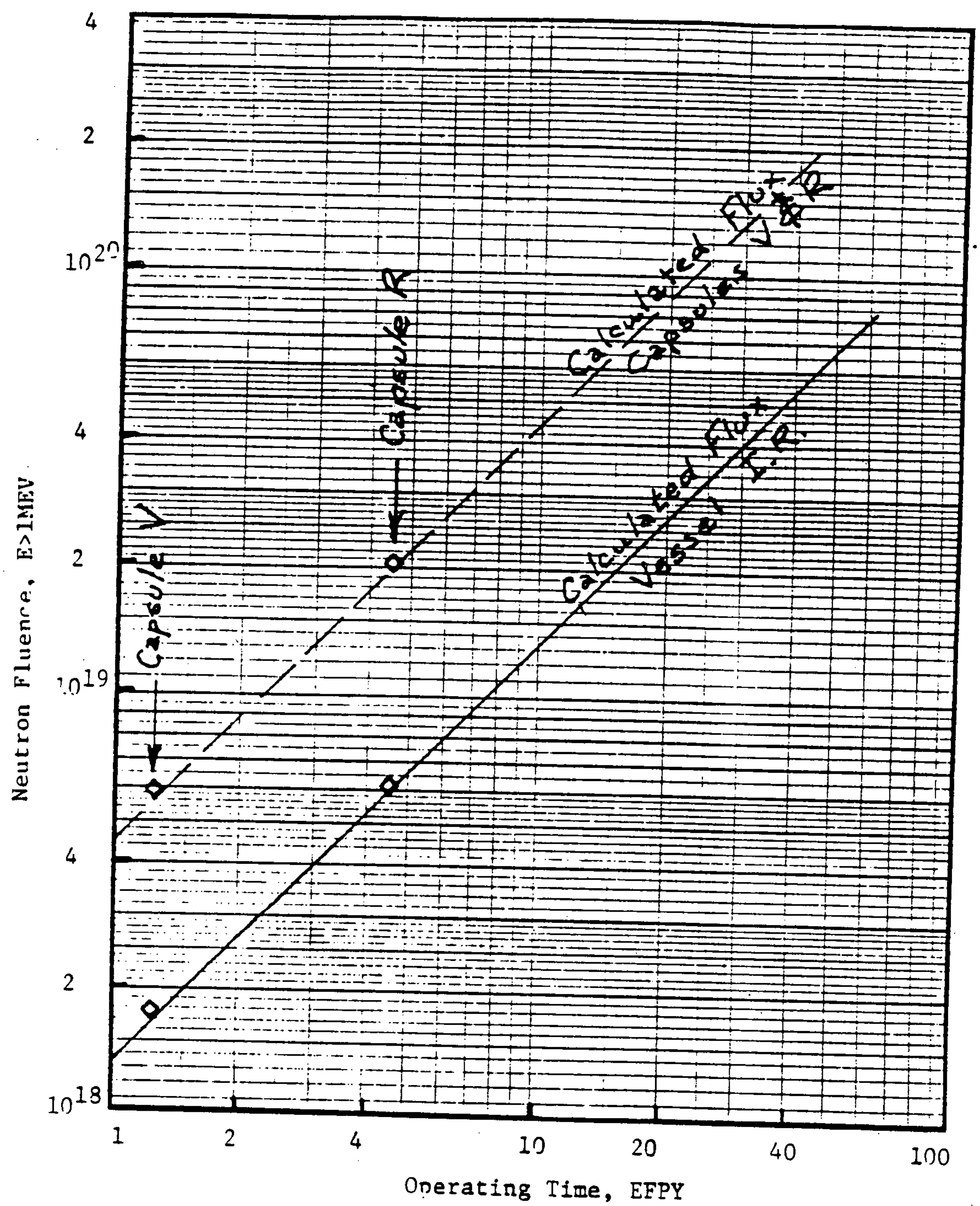

FIGURE 3. CALCULATED FLUENCES AS A FUNCTION OF OPERATING TIME

IV. HEATUP AND COOLDOWN LIMIT CURVES

Heatup and cooldown limit curves are presented in this section for a validity period of 15 effective full power years. These curves were developed based on the Regulatory Guide 1.99 Rev. 2 approach to calculating adjusted reference temperatures for reactor vessel materials and the computation procedure developed by Southwest Research Institute.

The following is a description of the pressure vessel parameters employed as input data in the KNPP analysis.

1. Structural and Thermal Parameters

   Vessel Inner Radius, r_i = 66 inches

   Vessel Outer Radius, r_o = 72.5 inches

   Operating Pressure, P_o = 2235 psig

   Hydrotest Pressure, P_n = 3107 psig

   Initial Temperature, T_o = 60°F

   Final Temperature, T_f = 550°F

   Effective Coolant Flow Rate, Q = 68.2×10^6 lb_m/hr

   Effective Flow Area (A_T) = Flow area between vessel and thermal shield + flow area between core barrel and thermal shield = 16.306 ft^2

   $$\text{Effective Hydraulic Diameter} = \frac{4 \times \text{Effective Flow Area}}{\text{Wetted Perimeter}}$$

   = 12.24 inches

2. Material Parameters

   Basis:
   - Regulatory Guide 1.99 Rev. 2
   - Chemistry Cu = 0.24%
     Ni = 0.78%
   - Initial RT_{NDT} = -56°F
     Standard deviation σ_I = 17°F

     Standard deviation for $\Delta\ RT_{NDT}$
     σ_Δ = 28°F

- Chemistry Factor
  CF = 202.7
- Fluence at Surface = $f \times 10^{19} = 2.0 \times 10^{19}$ n/cm^2

Adjusted Reference Temperature (ART) Calculations

$$ART = \text{Initial } RT_{NDT} + \Delta RT_{NDT} + \text{Margin}$$

$$\Delta RT_{NDT} \text{ surface} = [CF]\ f^{(0.28-0.10 \log f)}$$

$$= 241.035$$

$$\text{Margin} = 2\sqrt{\sigma_I^2 + \sigma_\Delta^2}$$

$$= 65.5$$

ART for 15 EFPY (surface) $\approx$ 251°F

ART in the vessel:

$$\Delta RT_{NDT} = [\Delta RT_{NDT} \text{ surface}]\ e^{-0.067x}$$

ART at 1/4 T = 226°F

ART at 3/4 T = 183°F

The 15 EFPY heatup and cooldown curves were compensated for instrument errors of -60 psig for pressure and of +10°F for temperature. In addition the PT limits have been computed in accordance with the US NRC Standard Review Plan [14] and ASME Section III Appendix G [2]. The heatup curves are presented in Figure 4 and the cooldown curves are shown in Figure 5.

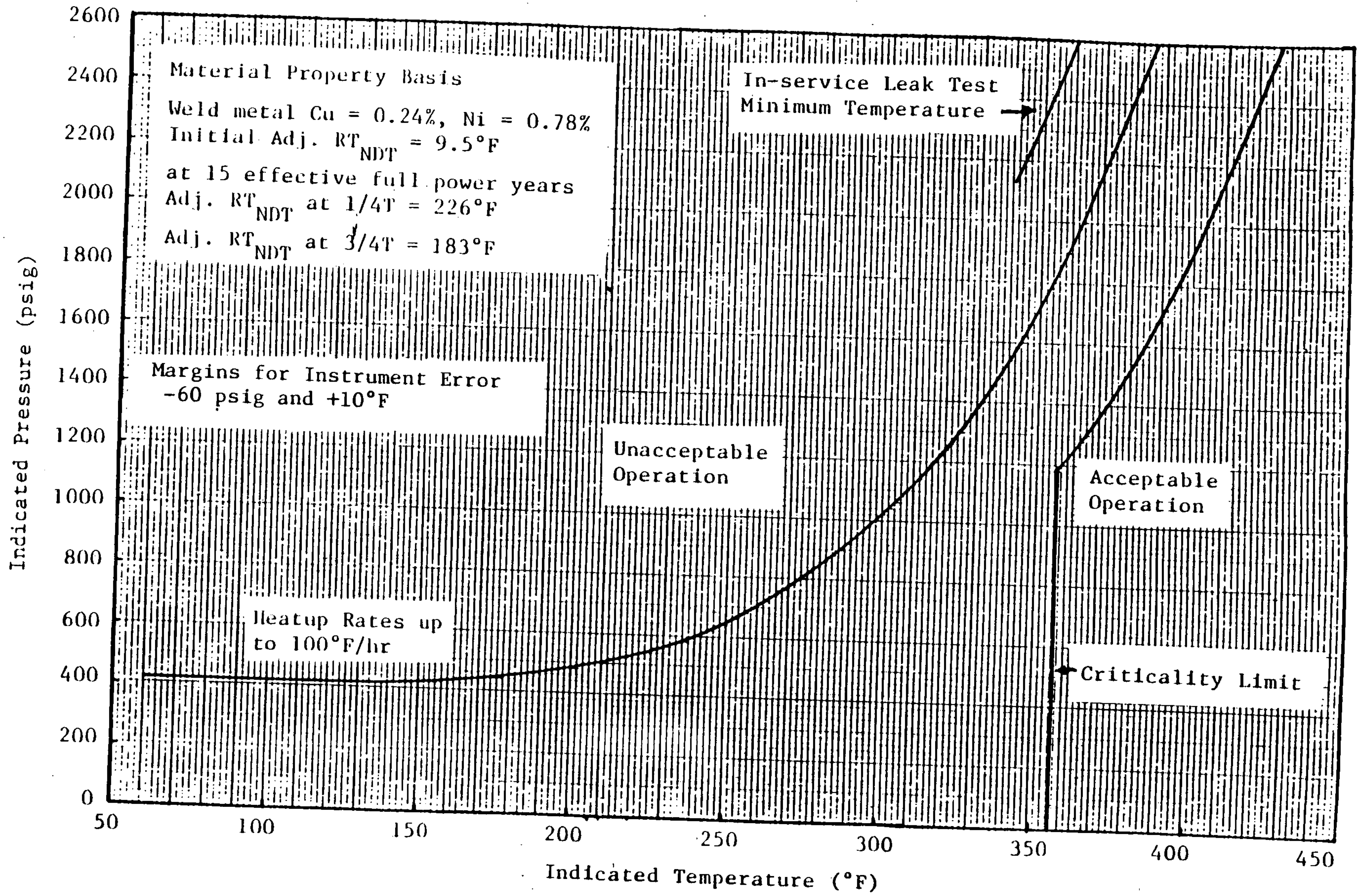

FIGURE 4. KEWAUNEE UNIT NO. 1 COOLANT HEATUP LIMITATION CURVES APPLICABLE FOR PERIODS UP TO 15 EFFECTIVE FULL POWER YEARS

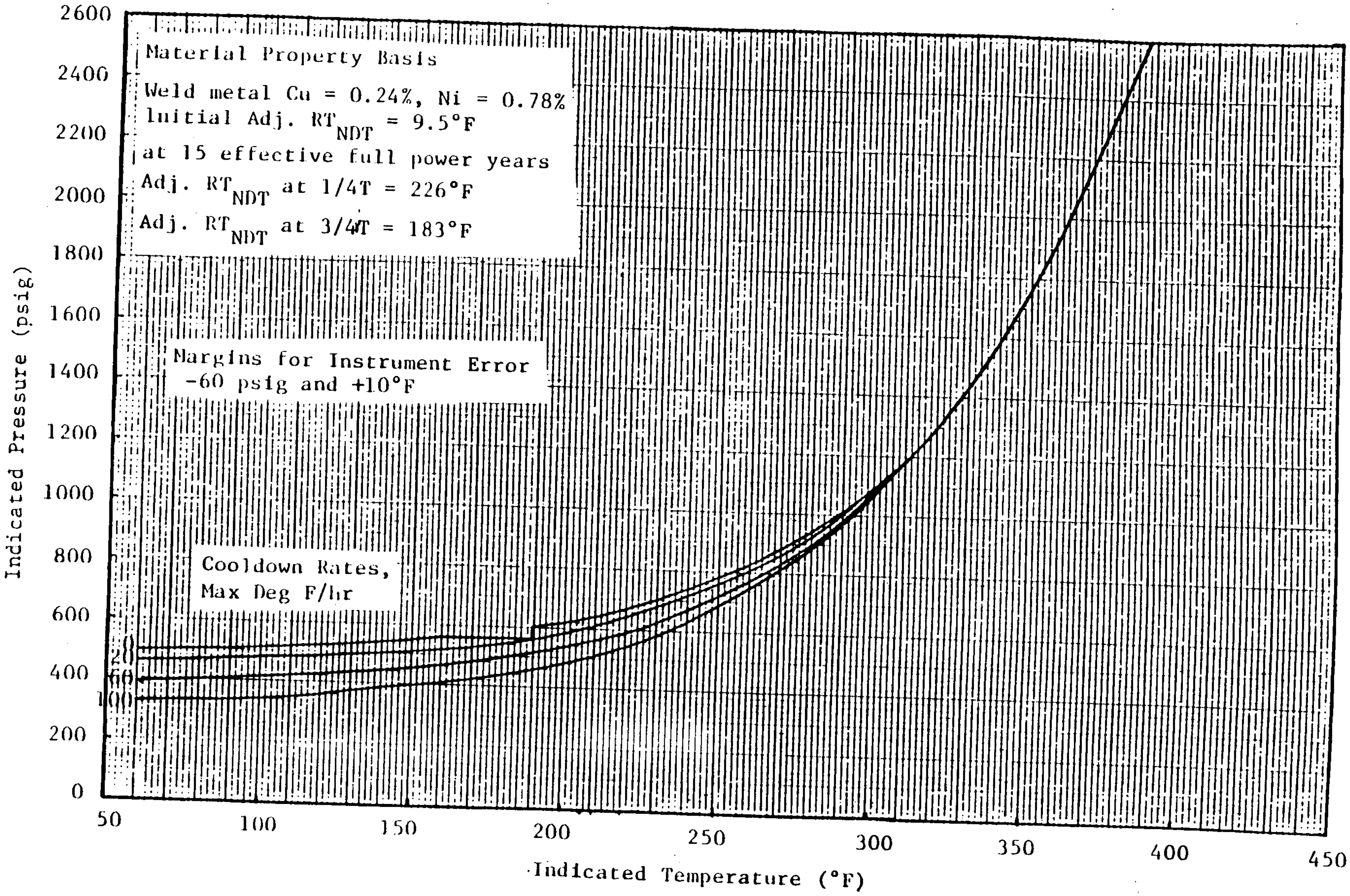

FIGURE 5. KEWAUNEE UNIT NO. 1 COOLANT COOLDOWN LIMITATIONS APPLICABLE FOR PERIODS UP TO 15 EFFECTIVE FULL POWER YEARS

V. REFERENCES

1. Title 10, Code of Federal Regulations, Part 50, "Licensing of Production and Utilization Facilities," 1983 Revision.

2. ASME Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components," 1983 Edition through the Summer 1984 Addenda.

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6. Perrin, J.F., Wullaert, R.A., Odette, G.R., and Lombrozo, P.M., "Physically Based Regression Correlations of Embrittlement Data from Reactor Pressure Vessel Surveillance Programs," EPRI NP-3319, January 1984.

7. Regulatory Guide 1.99, Revision 2, Office of Standards Development, U.S. Nuclear Regulatory Commission, Feb. 1986 (for comment).

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9. "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels, E706(IF)," 1985 ASTM Annual Book of Standards, Vol. 12.02.

10. Yanichko, S.E., Anderson, S.L., and Scott, K.V., "Analysis of Capsule V from the Wisconsin Public Service Corporation Kewaunee Nuclear Plant Reactor Vessel Radiation Surveillance Program," WCAP 8908, January 1977.

11. Yanichko, S.E., Anderson, S.L., Shogan, R.P., and Lott, R.G., "Analysis of Capsule R from Wisconsin Public Service Corporation Kewaunee Nuclear Plant Reactor Vessel Radiation Surveillance Program," EPRI RP1021-3, WCAP-9878, March 1981.

12. Magurno, B.A., "ENDF/B-IV Dosimetry File," BNL-NCS-50446 (ENDF-212), Brookhaven National Lab, April 1975.

13. Letter transmittal K.A. Hoops to P.K. Nair, "Information Necessary for Preparation of P/T Curves and Assessment of RT_{PTS} Concerns," dated November 6, 1985.

14. US NRC Standard Review Plan, NUREG-0800, Section 5.3.2, Pressure-Temperature Limits, Revision 1, July 1981.

Proposed Amendment No. 71

Attachment C

To

Letter from C. W. Giesler (WPSC) to H. R. Denton (NRC)

Dated

April 29, 1986

<u>Description, Safety Evaluation, and Significant Hazards Determination</u>

Proposed Amendment No. 71 to the KNPP Technical Sepcifications

The specific changes in this proposed amendment, along with their safety evaluations and significant hazards determinations, are identified below.

List of Figures

Page TS viii

Description of Change

Figures TS 3.1-3 and TS 3.1-4 have been deleted and the list of figures for the technical specifications has been revised to reflect the deletion. Figure TS 3.1-3 provided the effect of fluence and copper content on the shift of RT_{NDT} for the reactor vessel. Figure TS 3.1-4 provided the relationship between fluence and EFPY. Technical Specification Amendment No. 17 removed reference to these figures in 1978. Both of the figures provide information that is not necessary to be included in Technical Specifications. In addition, with the recent research in these areas, both figures are obsolete.

Safety Evaluation

Both of the curves have not been referenced by any Technical Specification or its basis since 1978. Removing these curves does not reduce any requirements in the Technical Specifications or change the intent of any Technical Specification. Thus, the deletion of these curves is not a safety concern.

Significant Hazards Determination

The proposed change does not involve a significant hazards consideration because operation of the KNPP in accordance with this change would not:

1) Significantly increase the probability or consequences of an accident previously evaluated. This change would have no effect on any previous evaluation. It removes superfluous information from the Technical Specification and, therefore, is an administrative change.

2) Create the possibility of a new or different type of accident from any previously analyzed. This change does not change the meaning or intent of any Technical Specification or reduce any requirements. Thus, this change does not create the possibility of a new or different type of accident.

3) Significantly reduce the margin of safety as defined in the basis for any Technical Specification. The removal of the superfluous information does not affect any margin of safety.

This change deletes unnecessary information from the Technical Specifications. This change does not present a significant hazard concern.

Section 3.0, Limiting Conditions for Operation

TS 3.1, Page TS 3.1-3

Description of Change

The expiration date for the heatup and cooldown limit curves has been changed from 10 effective full power years (EFPY) to 15 EFPY. This makes TS 3.1.b.1 consistent with the expiration date noted on the revised heatup and cooldown limit curves.

Safety Evaluation

This change is administrative in nature since it implements another Technical Specification change that has a determination of no significant hazard. The safety analysis and significant hazards determination for the other change (revision to the heatup and cooldown limit curves) are provided under TS 3.1, Figures TS 3.1-1 and TS 3.1-2. In conclusion, this change is administrative in nature and involves no safety concern.

Significant Hazards Determination

The proposed change does not involve a significant hazards consideration because operation of the KNPP in accordance with this change would not:

1) Significantly increase the probability or consequences of an accident previously evaluated. This change achieves consistency with another portion of the Technical Specifications that has a determination of no significant hazard. Therefore, this change does not increase the probability or consequences of an accident.

2) Create the possibility of a new or different type of accident from any previously analyzed. Achieving consistency with another portion of the Technical Specifications that has a determination of no significant hazard does not create the possibility of a new or different kind of accident.

3) Significantly reduce the margin of safety as defined in the basis for any Technical Specification. This change ensures consistency with another portion of the Technical Specifications that has a determination of no significant hazard. Therefore, the margin of safety is acceptable as discussed under the significant hazards determination for TS 3.1, Figures TS 3.1-1 and TS 3.1-2.

This change is required to achieve consistency with the revised heatup and cooldown limit curves and does not represent a significant hazards concern. The significant hazards determination for the revision to the curves is discussed under TS 3.1, Figures TS 3.1-1 and TS 3.1-2.

TS 3.1, Page TS 3.1-4

Description of Change

The basis for TS 3.1.b has been revised to make it consistent with the method used to prepare the revised heatup and cooldown limit curves. As allowed by the ASME Boiler and Pressure Vessel Code, the revised curves use a safety factor of 1.0 on the stress intensity factor induced by the thermal gradient. Therefore, any reference to a 1.25 safety factor for thermal stresses has been removed.

Safety Evaluation

This change to the basis for TS 3.1.b is necessary to ensure the basis accurately reflects the methods used to prepare the revised heatup and cooldown limit curves. The safety evaluation of the methodology is provided in the discussion of the changes to Technical Specifications Figures TS 3.1-1 and TS 3.1-2. In addition, a detailed description of the methodology is provided in Attachment B to this letter. Therefore, this change to page TS 3.1-4 is administrative in nature as it ensures consistency with a portion of the Technical Specifications that has a determination of no significant hazard.

Significant Hazards Determination

The proposed change does not involve a significant hazards consideration because operation of the KNPP in accordance with this change would not:

1) Significantly increase the probability or consequences of an accident previously evaluated. This change ensures consistency with a portion of the Technical Specifications that has a determination of no significant hazard. Therefore, this change does not increase the probability or consequences of an accident.

2) Create the possibility of a new or different type of accident from any previously analyzed. Achieving consistency with a portion of the Technical Specifications that has a determination of no significant hazard does not create the possibility of a new or different kind of accident.

3) Significantly reduce the margin of safety as defined in the basis for any technical specification. This change does reduce a safety factor in a technical specification basis. However, this is done to ensure the basis accurately reflects the method used to prepare the revised heatup and cooldown limit curves. The basis for using the smaller safety factor is discussed under TS 3.1, Figures TS 3.1-1 and TS 3.1-2.

This change achieves consistency with the method used to prepare the revised heatup and cooldown limit curves and does not represent a significant hazards concern. The significant hazards determination for the revision to the curves is discussed under TS 3.1, Figures TS 3.1-1 and TS 3.1-2.

TS 3.1, Page TS 3.1-5

Description of Change

The changes to this page correct two grammatical errors. The word effect has been replaced by affect and K_{IM} has been replaced by K_{Im}. These changes are administrative in nature.

Safety Evaluation

These changes to the basis for TS 3.1.b correct two grammatical errors. No change has been made to the meaning or intent by these changes. This is an administrative change to Technical Specifications and it involves no safety concern.

Significant Hazards Determination

The proposed changes do not involve a significant hazards consideration because operation of the KNPP in accordance with these changes would not:

1) Significantly increase the probability or consequences of an accident previously evaluated. These changes are merely the correction of grammatical errors and cannot increase the probability or consequences of an accident.

2) Create the possibility of a new or different type of accident from any previously analyzed. The correction of these grammatical errors does not create the possibility of a new or different kind of accident.

3) Significantly reduce the margin of safety as defined in the basis for any Technical Specification. An administrative change does not reduce any margin of safety.

These changes have not affected the meaning or intent of the basis for TS 3.1.b and they do not represent a significant hazards concern.

TS 3.1, Page TS 3.1-6

Description of Change

Two changes were made to page TS 3.1-6. Both changes correct errors contained in the basis for TS 3.1.b. A miswording in the third paragraph has been corrected to read "Standard Review Plan...and Reference (1)" rather than "Standard Review Plan...in Reference (1)." Reference (1) is Section III, Appendix G of the ASME code.

The other correction was made to the fourth paragraph. Recent review of WCAP 9878 indicated that capsule R received above the 10 effective full power years stated by the basis. Even when accurate, the data presented by this statement does not provide any information necessary to be included in the basis for the heatup and cooldown curves. Therefore, the sentence was deleted and the reference number for WCAP 9878 was added to the previous sentence.

Safety Evaluation

The deletion of information on the approximate full power years of operation the most recently analyzed capsule had received does not remove any necessary information from the basis. This knowledge would not assist in the use of the heatup and cooldown limit curves.

This type of information is not appropriate for inclusion in the Technical Specifications, and if it is required for a study or analysis, it can be obtained from data in WCAP 9878. Also, the correction of the miswording ("and" versus "in") improves the accuracy of the basis and does not affect the intent of the Technical Specifications. Therefore, these changes to the basis do not present a safety concern.

Significant Hazards Determination

The proposed changes do not involve a significant hazards consideration because operation of the KNPP in accordance with these changes would not:

1) Significantly increase the probability or consequences of an accident previously evaluated. The removal of this incorrect statement does not increase the probability or consequence of an accident previously

evaluated. The removal of this information from Technical Specifications does not affect the intent of TS 3.1.b and does not imply a change in the manner in which the expiration date of the heatup and cooldown limit curves is calculated or interpreted. The correction of the miswording also does not increase the probability or consequence of an accident previously evaluated. This type of administrative correction does not affect the intent of the Technical Specifications.

2) Create the possibility of a new or different type of accident from any previously analyzed. As stated above, the removal of the incorrect statement from the basis does not affect the manner in which TS 3.1.b is interpreted or used. This is also true for the correction of the miswording. Therefore, no new or different type of accident has been created.

3) Significantly reduce the margin of safety as defined in the basis for any Technical Specification. Neither of these changes affect the manner in which TS 3.1.b is interpreted or used. Therefore, these changes have not affected any margin of safety.

Based on the above considerations, this change does not represent a significant hazards concern.

TS 3.1, Page TS 3.1-7

Description of Change

The expiration date for the heatup and cooldown curves has been changed from 10 EFPY to 15 EFPY. This change makes the basis consistent with the expiration date noted on the revised heatup and cooldown limit curves. Reference one has been updated to reflect the version of the ASME code applicable to the revised limit curves. In addition, reference three has been changed to the actual basis used for the revised curves, replacing the basis used for the past curves.

Safety Evaluation

All of these changes are administrative in nature. The change in the expiration date is required to make the basis consistent with the revised curves which have a determination of no significant hazard. The change to reference one is required to ensure the correct version of the ASME code is referenced by the basis. The change to reference three is required to ensure the correct reference is provided for discussion of the basis for the heatup and cooldown limit curves. None of these changes affect the meaning or intent of TS 3.1.b. All ensure consistency between portions of the Technical Specifications.

Significant Hazards Determination

The proposed changes do not involve a significant hazards consideration because operation of the KNPP in accordance with these changes would not:

1) Significantly increase the probability or consequences of an accident previously evaluated. These changes achieve consistency with a portion of the Technical Specifications that has a determination of no significant hazard. Therefore, these changes do not increase the probability or consequences of an accident.

2) Create the possibility of a new or different type of accident from any previously analyzed. Achieving consistency with a portion of the Technical Specifications that has a determination of no significant hazard does not create the possibility of a new or different kind of accident.

3) Significantly reduce the margin of safety as defined in the basis for any Technical Specification. These changes ensure consistency with another portion of the Technical Specifications that has a determination of no significant hazard. Therefore, the margin of safety is acceptable as discussed under the significant hazards determination for TS 3.1, Figures TS 3.1-1 and TS 3.1-2.

These changes ensure that the basis concurs with the limit curves and the method used to prepare the curves. Therefore, as stated above, these changes are administrative in nature and do not represent a significant hazards concern. The significant hazards determination for the revision to the curves is discussed below under TS 3.1, Figures TS 3.1-1 and TS 3.1-2.

TS 3.1, Pages TS 3.1-8 through TS 3.1-11a

Description of Change

Blank page TS 3.1-8 has been deleted and pages TS 3.1-9 through TS 3.1-11a have been renumbered as pages TS 3.1-8 through TS 3.1-11.

Safety Evaluation

The deletion of a blank page and the subsequent renumbering of the following pages are administrative changes to the Technical Specifications. No change was made to the content of any of the pages. Therefore, these changes do not represent a safety concern.

Significant Hazards Determination

The proposed changes do not involve a significant hazards consideration because operation of the KNPP in accordance with these changes would not:

1) Significantly increase the probability or consequences of an accident previously evaluated. These changes are administrative in nature with no effect on the content of the Technical Specifications. Therefore, these changes do not increase the probability or consequences of an accident.

2) Create the possibility of a new or different type of accident from any previously analyzed. These changes are administrative in nature and, therefore, do not create the possibility of a new or different kind of accident.

3) Significantly reduce the margin of safety as defined in the basis for any Technical Specification. These changes have no effect on the content of the Technical Specifications. Therefore, no margin of safety has been changed.

These changes are administrative in nature and, as shown above, do not represent a significant hazards concern.

TS 3.1, Figures TS 3.1-1 and TS 3.1-2

Description of Change

The previous heatup and cooldown limit curves have been replaced with revised limit curves which are valid to 15 EFPY. These curves have been prepared in a manner consistent with NRC Regulatory Standard Review Plan Section 5.3.2, "Pressure-Temperature Limits," Draft Regulatory Guide 1.99, Revision 2, and ASME Boiler and Pressure Vessel Code Section III, Appendix G, 1983 edition through Summer, 1984 addenda. The major differences in the preparation of the revised limit curves compared to the previous curves are:

1) The assumption of higher copper and nickel contents for the limiting reactor vessel weld. The basis for the new assumptions is described in Attachment A to this letter.

2) The use of Draft Revision 2 of Regulatory Guide 1.99 instead of Revision 1. Revision 2 is a technical improvement over the methodology of Revision 1.

Safety Evaluation

The method used in preparing the heatup and cooldown limit curves is presented in Attachment B to this letter. The method used is consistent with ASME Boiler and Pressure Vessel Code Section III, Appendix G, NRC Regulatory Standard Review Plan Section 5.3.2, and Draft Regulatory Guide 1.99, Revision 2. The safety factors and margins applied in the preparation of the limit curves meet the criteria set forth by these documents. Therefore, while a smaller safety factor was applied to the thermal stress intensity factor than previously used, the current safety factor meets the code requirements. In the same manner, even though different material assumptions were used, these assumptions followed the guidance of Draft Regulatory Guide 1.99, Revision 2.

Therefore, the preparation of the heatup and cooldown limit curves meet the applicable safety criteria and do not represent a safety concern.

Significant Hazards Determination

The proposed change does not involve a significant hazards consideration because operation of the KNPP in accordance with this change would not:

1) Significantly increase the probability or consequences of an accident previously evaluated. The revised heatup and cooldown limit curves meet the applicable requirements that ensure the conservatism of the curves. The revised safety factor applied to the thermal stress intensity factor meets the requirements of the ASME Boiler and Pressure Vessel Code and the criteria in the NRC Regulatory Standard Review Plan Section 5.3.2. The use of updated material information that had been

revised in the conservative direction is also not a concern because this data agrees with the guidance set forth in Draft Regulatory Guide 1.99, Revision 2. With the preparation of the limit curves in accordance with the latest criteria and guidance, there is no significant hazards concern for any postulated change in the probability or consequences of an accident previously evaluated.

2) Create the possibility of a new or different type of accident from any previously analyzed. The revised heatup and cooldown curves do not create the possibility of a new or different type of accident. The curves were prepared in accordance with regulatory requirements and require plant operation within more limiting requirements to allow operation to 15 EFPY instead of 10 EFPY. Thus, no new or different type of accident has been created.

3) Significantly reduce the margin of safety as defined in the basis for any Technical Specification. The safety factors and margins used in the preparation of the limit curves meet the requirements of the ASME Code and the guidance of Draft Regulatory Guide 1.99, Revision 2. The safety factor which was applied to the thermal stress intensity factor meets the requirements of the ASME code and the criteria in the Standard Review Plan. The use of updated material information does not present a safety concern as this data agrees with the guidance set forth in Draft Regulatory Guide 1.99, Revision 2. In addition, the margins added to the material reference temperature were consistent with the draft regulatory guide.

Based on the safety evaluation and the above considerations, WPSC has determined that this change does not involve a significant hazards concern.

TS 3.1, Figures TS 3.1-1 and TS 3.1-2

Description of Change

Figures TS 3.1-3 and TS 3.1-4 have been deleted and blank pages have been inserted in their place. Figure TS 3.1-3 provided the effect of fluence and copper content on the shift of RT_{NDT} for the reactor vessel. Figure TS 3.1-4 provided the relationship between fluence and EFPY. Technical Specification Amendment No. 17 removed reference to these figures in 1978. Both of the figures provide information that is not necessary to be included in Technical Specifications. In addition, with the recent research in these areas, both figures are obsolete.

Safety Evaluation

Both of the curves have not been referenced by any Technical Specification or its basis since 1978. Removing these curves does not reduce any requirements in the Technical Specifications or change the intent of any Technical Specification. Thus, the deletion of these curves is not a safety concern.

Significant Hazards Determination

The proposed change does not involve a significant hazards consideration because operation of the KNPP in accordance with this change would not:

1) Significantly increase the probability or consequences of an accident previously evaluated. This change would have no effect on any previous evaluation. It removes superfluous information from the Technical Specifications and, therefore, is an administrative change.

2) Create the possibility of a new or different type of accident from any previously analyzed. This change does not change the meaning or intent of any Technical Specification or reduce any requirements. Thus, this change does not create the possibility of ā new or different type of accident.

3) Significantly reduce the margin of safety as defined in the basis for any Technical Specification. The removal of the superfluous information does not affect any margin of safety.

This change deletes unnecessary information from the Technical Specifications. This change does not present a significant hazard concern.

Proposed Amendment No. 71

Attachment D

To

Letter from C. W. Giesler (WPSC) to H. R. Denton (NRC)

Dated

April 29, 1986

Pages Affected by Proposed Amendment No. 71