June 10, 1986

Docket No. 50-272

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Mr. C. A. McNeill, Jr.
Vice President - Nuclear
Public Service Electric and Gas Company
Post Office Box 236
Hancocks Bridge, New Jersey 08038

Dear Mr. McNeill:

The Commission has issued the enclosed Amendment No. 75 to Facility Operating License No. DPR-70 for the Salem Nuclear Generating Station, Unit No. 1. This amendment consists of changes to the Technical Specifications in response to your request dated October 15, 1984 and supplemented October 16, 1985 and January 30, 1986.

The amendment revises the Heatup Limits Curve and the Cooldown Limits Curve for Unit No. 1.

A copy of the Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's next regular bi-weekly Federal Register notice.

Sincerely,

Donald C. Fischer, Senior Project Manager
PWR Project Directorate #3
Division of PWR Licensing-A, NRR

Enclosures:
1. Amendment No. 75 to DPR-70
2. Safety Evaluation

cc: w/enclosures
See next page

| OFC | PAD#3 | PAD#3 | OELD | D/PAD#3 | | | |
|---|---|---|---|---|---|---|---|
| NAME | CVogan | DFischer;ps | J. Moore / M. Karman | SVarga | | | |
| DATE | 5/19/86 | 5/20/86 | 5/27/86 | 6/10/86 | | | |

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

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
PHILADELPHIA ELECTRIC COMPANY
DELMARVA POWER AND LIGHT COMPANY
ATLANTIC CITY ELECTRIC COMPANY

DOCKET NO. 50-272

SALEM NUCLEAR GENERATING STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 75
License No. DPR-70

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

   A. The application for amendment by Public Service Electric and Gas Company, Philadelphia Electric Company, Delmarva Power and Light Company and Atlantic City Electric Company (the licensees) dated October 15, 1984 and supplemented October 16, 1985 and January 30, 1986, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-70 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 75 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga

Steven A. Varga, Director
PWR Project Directorate #3
Division of PWR Licensing-A, NRR

Attachment: 75
Changes to the Technical Specifications

Date of Issuance: June 10, 1986

ATTACHMENT TO LICENSE AMENDMENT NO. 75

FACILITY OPERATING LICENSE NO. DPR-70

DOCKET NO. 50-272

Revise Appendix A as follows:

| Remove Pages | Insert Pages |
|---|---|
| 3/4 4-26 | 3/4 4-26 |
| 3/4 4-27 | 3/4 4-27 |
| ——— | 3/4 4-27a |
| B 3/4 4-6 | B 3/4 4-6 |
| B 3/4 4-7 | B 3/4 4-7 |
| B 3/4 4-8 | B 3/4 4-8 |
| B 3/4 4-9 | B 3/4 4-9 |
| B 3/4 4-10 | B 3/4 4-10 |
| B 3/4 4-11 | ——— |
| B 3/4 4-12 | B 3/4 4-11 |

MATERIAL PROPERTY BASIS

UPPER LIMIT OF REG. GUIDE TREND CURVES (FIGURE B3/4 4-2)

| | | |
|---|---|---|
| COPPER CONTENT | : | 0.35 WT% |
| PHOSPHORUS CONTENT | : | 0.012 WT% |
| $RT_{NDT}$ INITIAL | : | 0°F |
| $RT_{NDT}$ AFTER 10 EFPY | : | 1/4T, 236°F |
| | : | 3/4T, 107°F |

CURVES APPLICABLE FOR HEATUP RATES UP TO 60°F/HR FOR THE SERVICE PERIOD UP TO 10 EFPY

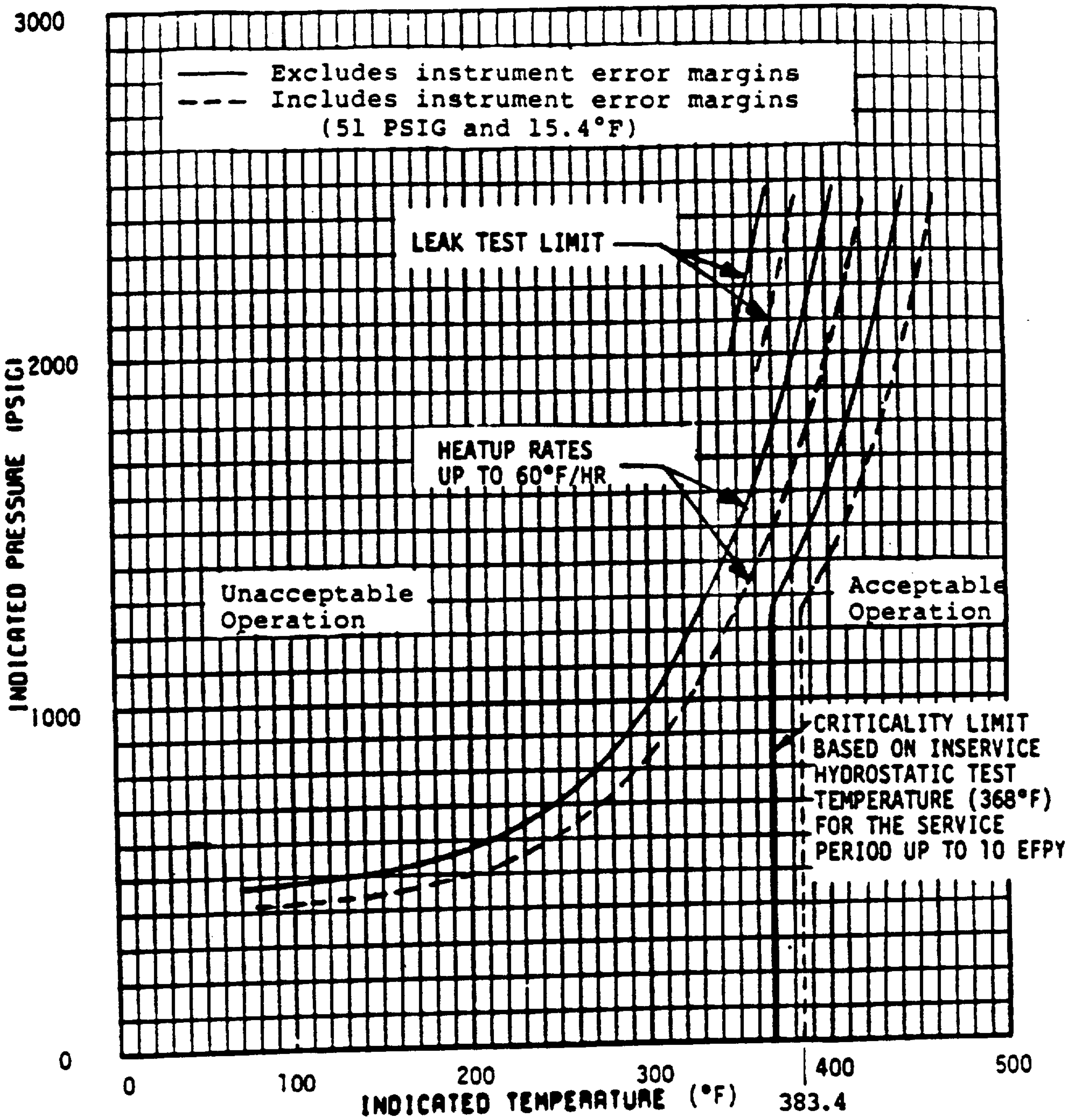


Figure 3.4-2 Salem Unit 1 Reactor Coolant System Heatup Limitations Applicable up to 10 EFPY

MAT'  AL PROPERTY BASIS

UPPER LIMIT OF REG. GUIDE TREND CURVES (FIGURE B 3/4 4-2)
COPPER CONTENT : 0.35 WT%
PHOSPHORUS CONTENT : 0.012 WT%
$RT_{NDT}$ INITIAL : 0°F
$RT_{NDT}$ AFTER 10 EFPY : 1/4T, 236°F
: 3/4T, 107°F

CURVES APPLICABLE FOR COOLDOWN RATES UP TO 100°F/HR FOR THE SERVICE PERIOD UP TO 10 EFPY

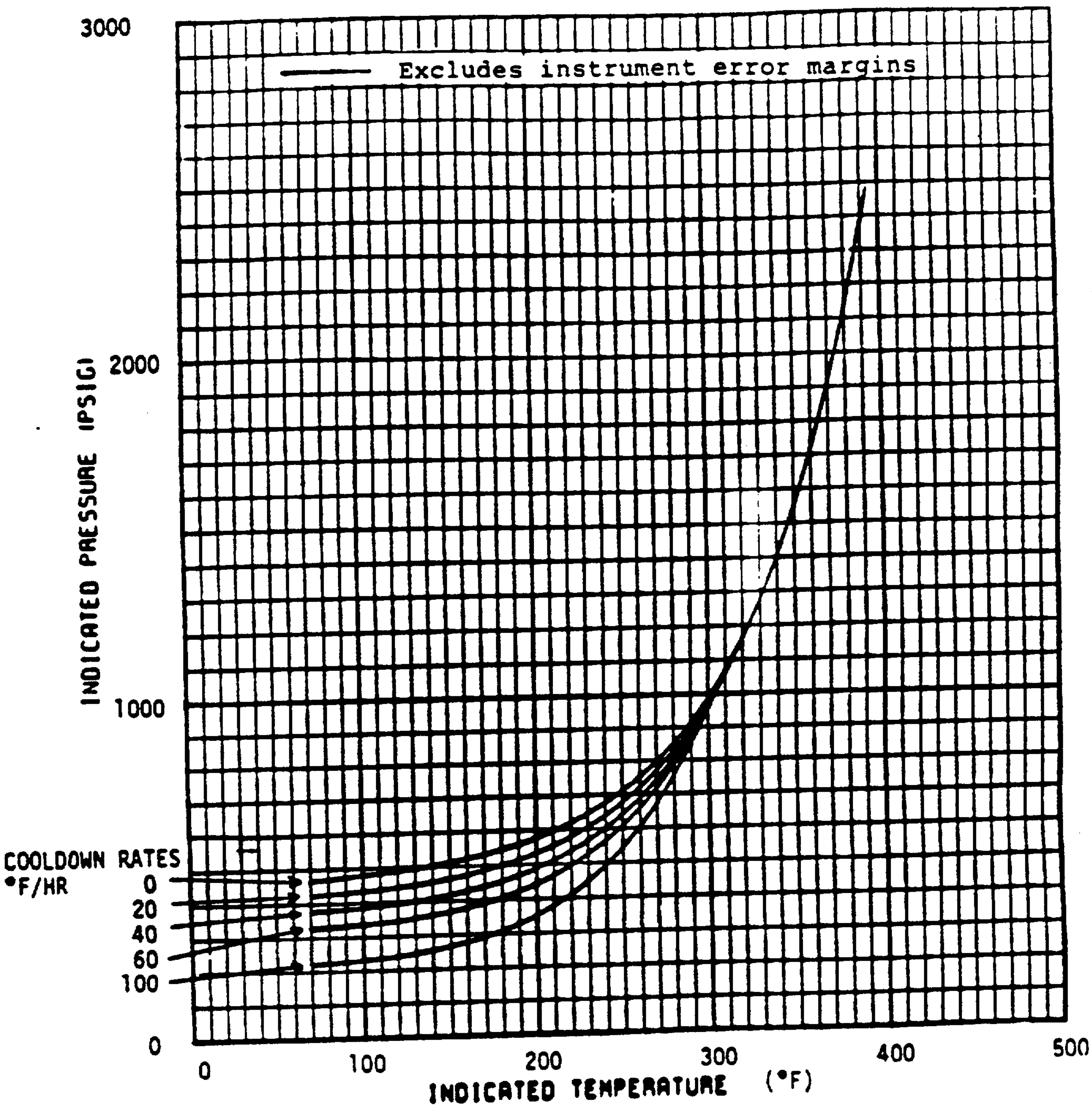


Figure 3.4-3A Salem Unit 1 Reactor Coolant System Cooldown Limitations Applicable up to 10 EFPY (Excluding Instrument Error Margins)

MATERIAL PROPERTY BASIS

UPPER LIMIT OF REG. GUIDE TREND CURVES (FIGURE B 3/4 4-2)

| | | |
|---|---|---|
| COPPER CONTENT | : | 0.35 WT% |
| PHOSPHORUS CONTENT | : | 0.012 WT% |
| $RT_{NDT}$ INITIAL | : | 0°F |
| $RT_{NDT}$ AFTER 10 EFPY | : | 1/4T, 236°F |
| | : | 3/4T, 107°F |

CURVES APPLICABLE FOR COOLDOWN RATES UP TO 100°F/HR FOR THE SERVICE PERIOD UP TO 10 EFPY

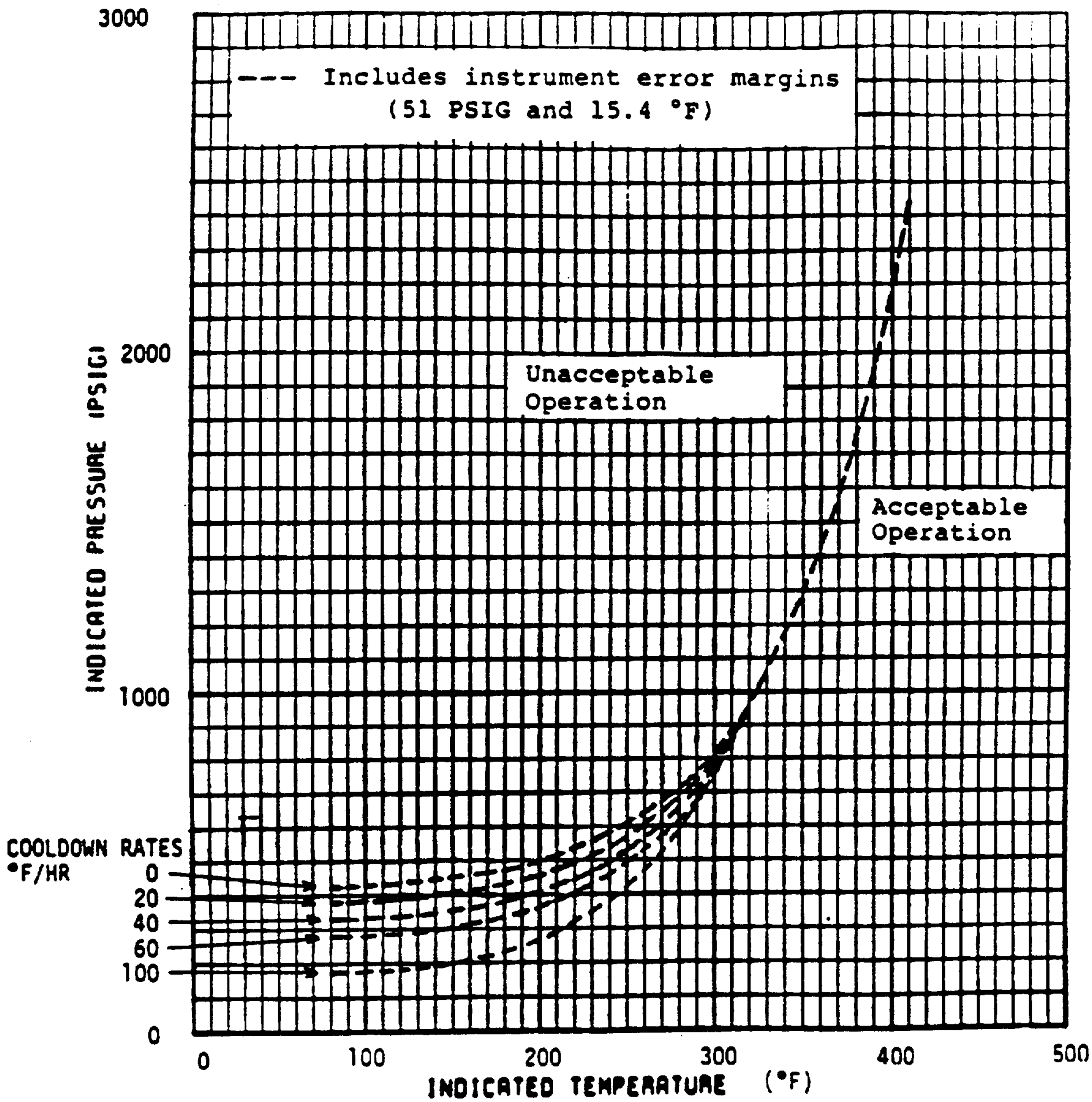


Figure 3.4-3B Salem Unit 1 Reactor Coolant System Cooldown Limitations Applicable up to 10 EFPY (Including Instrument Error Margins)

## 3/4.4.9 PRESSURE/TEMPERATURE LIMITS

All components in the Reactor Coolant System are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in Section 4.1.5 of the FSAR. During startup and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

During heatup, the thermal gradients in the reactor vessel wall produce thermal stresses which vary from compressive at the inner wall to tensile at the outer wall. These thermal induced compressive stresses tend to alleviate the tensile stresses induced by the internal pressure. Therefore, a pressure-temperature curve based on steady state conditions (i.e., no thermal stresses) represents a lower bound of all similar curves for finite heatup rates when the inner wall of the vessel is treated as the governing location.

The heatup analysis also covers the determination of pressure-temperature limitations for the case in which the outer wall of the vessel becomes the controlling location. The thermal gradients established during heatup produce tensile stresses at the outer wall of the vessel. These stresses are additive to the pressure induced tensile stresses which are already present. The thermal induced stresses at the outer wall of the vessel are tensile and are dependent on both the rate of heatup and the time along the heatup ramp; therefore, a lower bound curve similar to that described for the heatup of the inner wall cannot be defined. Consequently, for the cases in which the outer wall of the vessel becomes the stress controlling location, each heatup rate of interest must be analyzed on an individual basis.

The heatup limit curve, Figure 3.4-2, is a composite curve which was prepared by determining the most conservative case, with either the inside or outside wall controlling, for any heatup rate up to 60°F per hour. The cooldown limit curves, Figure 3.4-3, are composite curves which were prepared based upon the same type analysis with the exception that the controlling location is always the inside wall where the cooldown thermal gradients tend to produce tensile stresses while producing compressive stresses at the outside wall. The heatup and cooldown curves were prepared based upon the most limiting value of the predicted adjusted reference temperature at the end of 10 EFPY.

The reactor vessel materials have been tested to determine their initial $RT_{NDT}$; the results of these tests are shown in Table B 3/4.4-1. Reactor operation and resultant fast neutron (E>1 Mev) irradiation will cause an increase in the $RT_{NDT}$. Therefore, an adjusted reference temperature, based upon the fluence and copper content of the material in question, can be predicted using Figures B 3/4.4-1 and B 3/4.4-2. The heatup and cooldown limit curves (Figures 3.4-2 and 3.4-3) include predicted adjustments for this shift in $RT_{NDT}$ at the end of 10 EFPY, as well as adjustments for possible errors in the pressure and temperature sensing instruments.

The actual shift in $RT_{NDT}$ of the vessel material will be established periodically during operation by removing and evaluating, in accordance with ASTM E185-70, reactor vessel material irradiation surveillance specimens installed near the inside wall of the reactor vessel in the core area. Since the neutron spectra at the irradiation samples and vessel inside radius are essentially identical, the measured transition shift for a sample can be applied with confidence to the adjacent section of the reactor vessel. The heatup and cooldown curves must be recalculated when the $\Delta RT_{NDT}$ determined from the surveillance capsule is different from the calculated $\Delta RT_{NDT}$ for the equivalent capsule radiation exposure.

The pressure-temperature limit lines shown on Figure 3.4-2 for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR 50.

The number of reactor vessel irradiation surveillance specimens and the frequencies for removing and testing these specimens are in accordance with the requirements of Appendix H to 10 CFR Part 50.

The limitations imposed on pressurizer heatup and cooldown and spray water temperature differential are provided to assure that the pressurizer is operated within the design criteria assumed for the fatigue analysis performed in accordance with the ASME Code requirements.

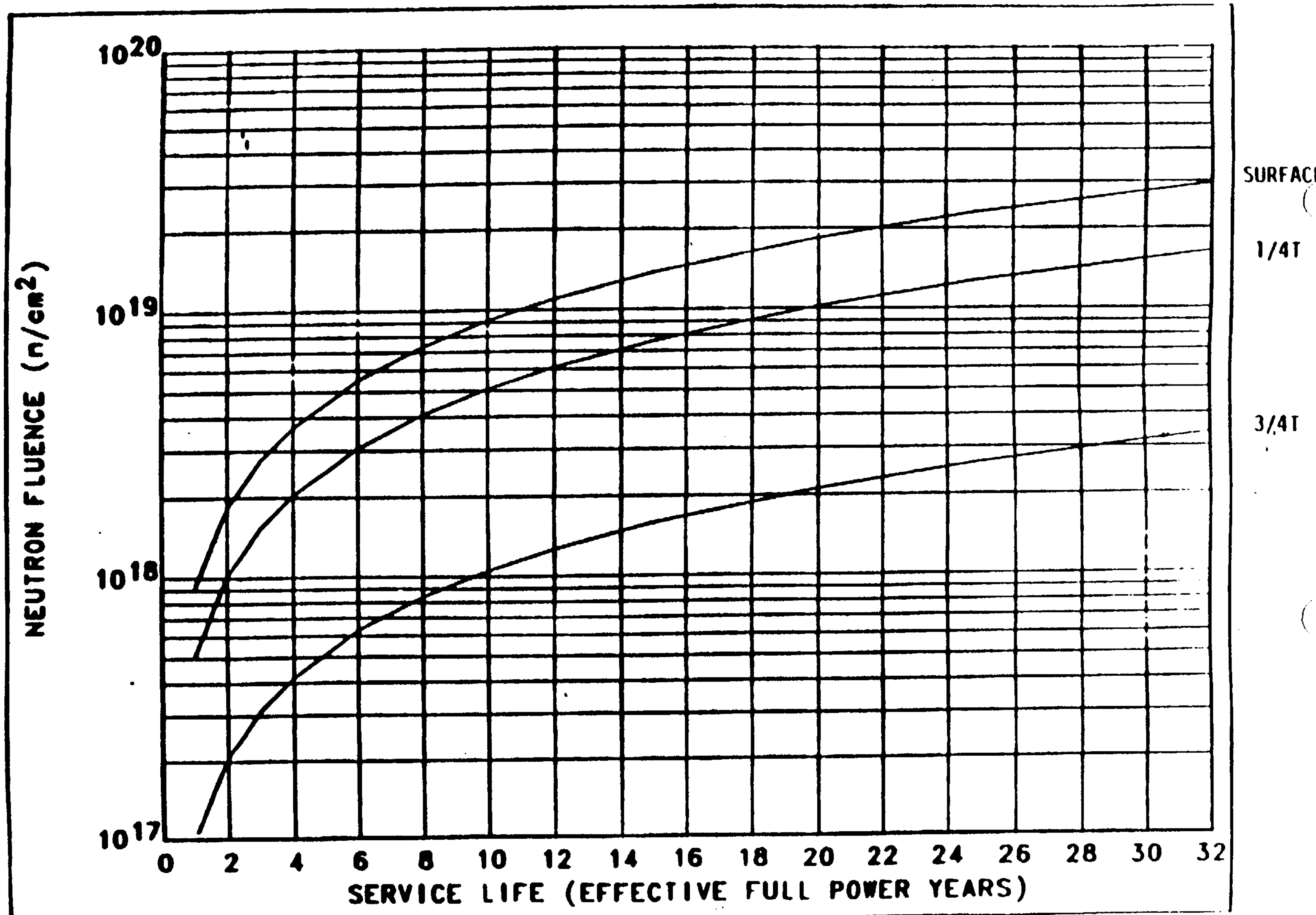


Figure B3/4.4-1 Fast Neutron Fluence (E>1 MeV) as a Function of Full Power Service Life (EFPY)

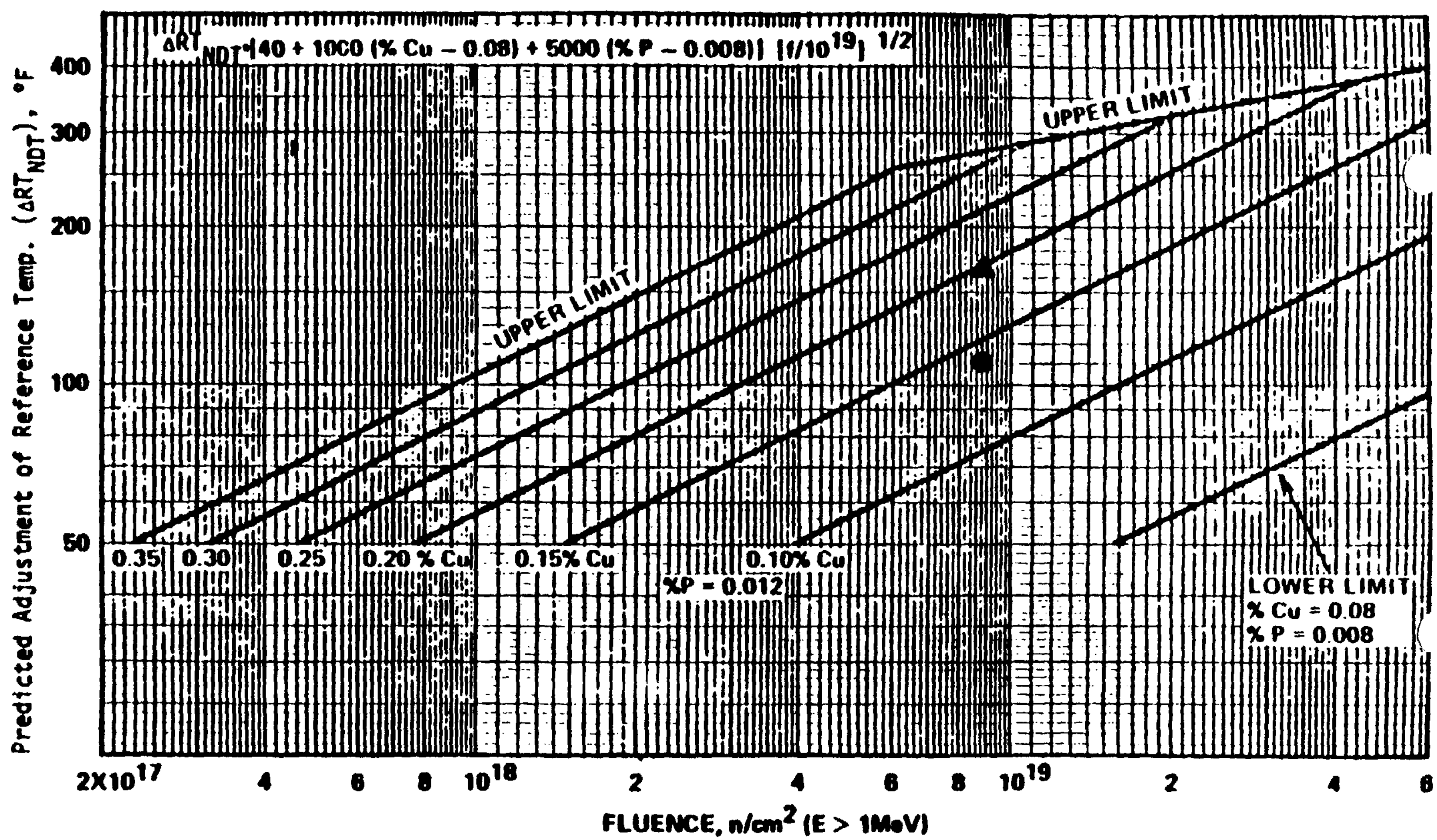


Figure B3/4.4-2 Effect of Fluence and Copper and Phosphorus Contents on $\Delta RT_{NDT}$ for Reactor Vessel Steels

▲ Weld Metal (Based on Capsule Y results)

● Shell Plate B2402-3 (Based on Capsule Y results)

## TABLE B 3/4.4-1
## SALEM UNIT 1
## REACTOR VESSEL TOUGHNESS DATA (UNIRRADIATED)

| Component | Heat No. | Code No. | Material Type | Cu (%) | P (%) | $T_{NDT}$ (°F) | 50 ft lb 35-Mil Temp (°F) | $RT_{NDT}$ (°F) | Average Shelf Energy NMWD (ft lb) | Average Shelf Energy MWD (ft lb) |
|---|---|---|---|---|---|---|---|---|---|---|
| Cl Hd Dome | A0610 | B2407-1 | A533B, Cl.1 | 0.20 | 0.011 | -30 | 99* | 39 | - | 110 |
| Cl Hd Segment | C1544 | B2406-1 | A533B, Cl.1 | 0.13 | 0.010 | -20 | 89* | 29 | - | 129 |
| Cl Hd Segment | C1544 | B2406-2 | A533B, Cl.1 | 0.16 | 0.012 | -30 | 85* | 25 | - | 122 |
| Cl Hd Segment | B5852 | B2406-3 | A533B, Cl.1 | 0.10 | 0.009 | -50 | 66* | 6 | - | 132 |
| Cl Hd Flange | 123P409 | B2811 | A508, Cl.2 | - | 0.010 | 28* | 22* | 28 | - | 199 |
| Vessel Flange | 5P1191 / 4P1019 | B2410 | A508, Cl.2 | - | 0.009 | 60* | 0* | 50 | - | 145 |
| Inlet Nozzle | 123P403 | B2408-1 | A508, Cl.2 | - | 0.010 | 50* | 43* | 50 | - | 144 |
| Inlet Nozzle | 125P544 | B2408-2 | A508, Cl.2 | - | 0.011 | 46* | 26* | 46 | - | 157 |
| Inlet Nozzle | 123P403 | B2408-3 | A508, Cl.2 | - | 0.010 | 47* | 37* | 47 | - | 161 |
| Inlet Nozzle | 125P544 | B2408-4 | A508, Cl.2 | - | 0.010 | 9* | 17* | 9 | - | 167 |
| Outlet Nozzle | ZT2550 | B2409-1 | A508, Cl.2 | - | 0.010 | 60* | 95* | 60 | - | 75 |
| Outlet Nozzle | ZT2550 | B2409-2 | A508, Cl.2 | - | 0.011 | 60* | 95* | 60 | - | 78 |
| Outlet Nozzle | ZT2585 | B2409-3 | A508, Cl.2 | - | 0.013 | 60* | 10* | 60 | - | 121 |
| Outlet Nozzle | ZT2585 | B2409-4 | A508, Cl.2 | - | 0.012 | 60* | 13* | 60 | - | 126 |
| Upper Shell | A0497 | B2401-1 | A533B, Cl.1 | 0.22 | 0.012 | -30 | 87* | 27 | - | 114 |
| Upper Shell | A0495 | B2401-2 | A533B, Cl.1 | 0.19 | 0.011 | 0 | 80* | 20 | - | 122 |
| Upper Shell | A0512 | B2401-3 | A533B, Cl.1 | 0.24 | 0.011 | -10 | 114* | 34 | - | 96 |
| Inter Shell | C1354 | B2402-1 | A533B, Cl.1 | 0.24 | 0.010 | -30 | 105 | 45 | 73.0 | 97 |
| Inter Shell | C1354 | B2402-2 | A533B, Cl.1 | 0.24 | 0.010 | -30 | 55 | -5 | 91.5 | 117 |
| Inter Shell | C1397 | B2402-3 | A533B, Cl.1 | 0.22 | 0.011 | -40 | 57 | -3 | 104.0 | 127 |
| Lower Shell | C1356 | B2403-1 | A533B, Cl.1 | 0.19 | 0.011 | -40 | 70 | 10 | 99.0 | 143 |
| Lower Shell | C1356 | B2403-2 | A533B, Cl.1 | 0.19 | 0.012 | -70 | 86 | 26 | 94.0 | 128 |
| Lower Shell | C1356 | B2403-3 | A533B, Cl.1 | 0.19 | 0.010 | -40 | 90 | 30 | 102.0 | 131 |
| Bot Hd Segment | A0705 | B2404-1 | A533B, Cl.1 | 0.10 | 0.009 | 10 | 48* | 10 | - | 120 |
| Bot Hd Segment | A0705 | B2404-2 | A533B, Cl.1 | 0.11 | 0.010 | -50 | 60* | 0 | - | 132 |
| Bot Hd Segment | A0705 | B2404-3 | A533B, Cl.1 | 0.12 | 0.008 | 10 | 47* | 10 | - | 126 |
| Bot Hd Dome | A0705 | B2405-1 | A533B, Cl.1 | 0.15 | 0.010 | -20 | 57* | -3 | - | 106 |
| Surveillance Weld | | | | 0.16 | 0.019 | 0* | -38** | 0 | | 104** |

NMWD - Normal to Major Working Direction
MWD - Major Working Direction
* - Estimated per NRC Standard Review Plan Branch Technical Position MTEB 5-2
** - Actual transverse data obtained from surveillance program (from minimum data points).

The OPERABILITY of two POPSs or an RCS vent opening of greater than 3.14 square inches ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more the RCS cold legs are less than or equal to 312°F. Either POPS has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either (1) the start of an idle RCP with the secondary water temperature of the steam generator less than or equal to 50°F above the RCS cold leg temperatures, or (2) the start of a safety injection pump and its injection into a wate solid RCS.

## 3/4.4.10 STRUCTURAL INTEGRITY

The inspection programs for ASME Code Class 1, 2 and 3 components ensure that the structural integrity of these components will be maintain at an acceptable level throughout the life of the plant. To the extent applicable, the inspection program for these components is in compliance with Section XI of the ASME Boiler and Pressure Vessel Code.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 75 TO FACILITY OPERATING LICENSE NO. DPR-70

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
PHILADELPHIA ELECTRIC COMPANY
DELMARVA POWER AND LIGHT COMPANY, AND
ATLANTIC CITY ELECTRIC COMPANY

SALEM NUCLEAR GENERATION STATION, UNIT NO. 1

DOCKET NO. 50-272

## INTRODUCTION

In a letter from E. A. Liden to S. A. Varga dated October 15, 1984, Public Service Electric & Gas Company (the licensee) proposed an amendment to their Facility Operating License DPR-70 for Salem Generating Station, Unit 1 (Salem-1). The amendment proposed to revise the reactor coolant system pressure/temperature limits, which are contained in Section 3.4.9.1 of the Technical Specifications and was based on the analysis of Capsule T data. During the staff review of this information, the results of the analysis of Capsule Y data were made available for use as input for the staff evaluation. These data were provided in letters dated October 16, 1985 and January 30, 1986 from C. A. McNeill to S. A. Varga. The revised curves are to be applicable for 10 effective full power years (EFPY). The bases for these changes were the test results from the Salem-1 surveillance program, which are contained in Report WCAP-10694, "Analysis of Capsule Y From The Public Service Electric and Gas Company Salem Unit 1 Reactor Vessel

Radiation Surveillance Program" and Report WCAP-9678, "Analysis of Capsule T From The Public Service Electric and Gas Company Salem Unit 1 Reactor Vessel Radiation Surveillance Program". The information provided in the October 16, 1985 and the January 30, 1986 submittals served to enhance the accuracy of the revised pressure/temperature limits.

## EVALUATION AND SUMMARY

Pressure-temperature limits must be calculated in accordance with the requirements of Appendix G, 10 CFR 50, which became effective on July 16, 1983. Pressure-temperature limits that are calculated in accordance with the requirements of Appendix G, 10 CFR 50 are dependent upon the initial $RT_{NDT}$ for the limiting materials in the beltline, and closure flange regions of the reactor vessel and the increase in $RT_{NDT}$ resulting from neutron irradiation damage to the limiting beltline material. The Salem-1 reactor vessel was procured to ASME Code requirements, which did not specify fracture toughness testing to determine the initial $RT_{NDT}$ for each vessel material. The licensee indicates that the initial $RT_{NDT}$ for the limiting materials in the closure flange and beltline regions of the Salem vessel were estimated using the method recommended by the staff in Branch Technical Position MTEB 5-2, "Fracture Toughness Requirements." This method result in an initial $RT_{NDT}$ for the limiting beltline base metal and weld metal of 45°F and 0°F, respectively and an initial $RT_{NDT}$ for the limiting closure flange material of 50°F.

The increase in $RT_{NDT}$ resulting from neutron irradiation damage was estimated by the licensee using the empirical relationship documented in Regulatory Guide 1.99, Rev. 1, April 1977, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials." This method of predicting neutron irradiation damage is dependent upon the predicted amount of neutron fluence and the amounts of residual elements (copper and phosphorus) in the beltline material. The neutron fluence predictions were verified by measurements from passive neutron flux monitors and by analysis, which was made with the DOT two-dimensional discrete ordinates code. Inputs into the analysis included 47 neutron energy groups, P3 expansion of the scattering cross section, and power distributions representative of time-averaged conditions derived from statistical studies of long-term operation of Westinghouse 4-loop plants. The cross sections used in the analysis were obtained from the SAILOR cross section library. Using this method of analysis, the measured saturated activity and neutron fluences (E>1MeV) for five foil reactions, which were calculated from neutron dosimetry in Capsules T and Y, were less than that predicted from the design basis calculated neutron fluences. The authors of WCAP 10694 recommended that projections of vessel toughness into the future be based on the design calculated fluence levels, since the calculated fluence levels were based on conservative representations of core power distributions derived for long-term operation while the Capsule data are representative only of past operation. The staff agrees with this recommendation.

The predicted amounts of neutron irradiation damage are based on design basis calculated neutron fluences and the increase in reference temperature ($\Delta RT_{NDT}$) using the curves in Regulatory Guide 1.99, Rev. 1. The prediction curves in Regulatory Guide 1.99, Rev. 1 are dependent upon the amounts of residual elements in the beltline material. The licensee in a Report entitled, "Fracture Toughness Analysis For Salem Unit 1 and 2 Reactor Pressure Vessels to Protect Against Pressurized Thermal Shock Events 10 CFR 50.61" has identified the residual elements in each weld and plate in the Salem-1 beltline. This report was contained in a letter from C. A. McNeill, Jr., to S. A. Varga, dated January 20, 1986. Based on the chemical composition of the beltline materials that were reported in this report, the limiting beltline material would be Plate No. B2402-1. In Table 1 we have compared the increase in $\Delta RT_{NDT}$ from the surveillance material to the increase in $\Delta RT_{NDT}$ predicted by Regulatory Guide 1.99, Rev. 1. This comparison indicates that the predicted increase in $\Delta RT_{NDT}$ for the plate material is greater than the increase in $\Delta RT_{NDT}$ measured from the surveillance material. Thus, the prediction method in Regulatory Guide 1.99, Rev. 1, should conservatively predict the increase in $\Delta RT_{NDT}$ for the Salem-1 beltline plate material.

The staff has used the method of calculating pressure-temperature limits in USNRC Standard Review Plan 5.3.2, NUREG-0800, Rev. 1, July 1981 to evaluate the proposed pressure-temperature limits. The amount of neutron irradiation damage was calculated using design basis calculated neutron fluences and the Regulatory Guide 1.99, Rev. 1, prediction curves. Our conclusion is that the proposed pressure-temperature limits meet the safety margins of Appendix G, 10 CFR 50 for 10 EFPY and may be incorporated into the plant's technical specifications.

## ENVIRONMENTAL CONSIDERATION

This amendment involves a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR Sec 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

## CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: June 10, 1986

PRINCIPAL CONTRIBUTOR:

B. Elliot

Table 1: Increase in Reference Temperature for Capsules T and Y Surveillance Material

| Surveillance Material | Neutron Fluence $(E>1MeV)\times10^{18}\eta/cm^2$ | Increase in Reference Temperature (°F) From Surveillance Capsule | Increase in Reference Temperature (°F) Predicted by Reg. Guide 1.99, Rev. 1 |
|---|---|---|---|
| Base Metal | | | |
| Plate B2402-1 | 2.40 | 100 | 100 |
| Plate B2402-2 | 2.40 | 100 | 113 |
| Plate B2402-3 | 2.40 | 75 | 98 |
| Plate B2402-3 | 8.91 | 110 | 189 |
| Weld Metal | | | |
| Weld 9-042 | 8.91 | 165 | 156 |
| Correlation Monitor | | | |
| HSST Plate 02 | 2.40 | 60 | 59 |
| HSST Plate 02 | 8.91 | 125 | 113 |