Docket Nos. 50-272/311

January 29, 1990

Mr. Steven E. Miltenberger
Vice President and Chief Nuclear
Officer
Public Service Electric & Gas Company
Post Office Box 236
Hancocks Bridge, New Jersey 08038

Dear Mr. Miltenberger:

SUBJECT: PRESSURE-TEMPERATURE LIMITS (TAC NOS. 71543/71544/71774/71775)

RE: SALEM GENERATING STATION, UNIT NOS. 1 AND 2

The Commission has issued the enclosed Amendment Nos.108 and 86 to Facility Operating License Nos. DPR-70 and DPR-75 for the Salem Generating Station, Unit Nos. 1 and 2. These amendments consist of changes to the Technical Specifications (TSs) in response to your application dated December 28, 1988 and supplemented on July 31, 1989, October 18, 1989 and December 19, 1989.

These amendments update the pressure-temperature limits to 15 effective full power years for Salem 1 and 10 effective full power years for Salem 2 based on the results of analysis of surveillance capsules. It is noted that the limiting material identified in the enclosed safety evaluation is different from that identified in the current Salem Updated Final Safety Analysis report (UFSAR). This new information (limiting material and chemistry content) should be included in the next revision of the UFSAR, Chapter 5.

A copy of our safety evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

/S/

James C. Stone, Project Manager
Project Directorate I-2
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No.108 to
License No. DPR-70
2. Amendment No. 86 to
License No. DPR-75
3. Safety Evaluation

cc w/enclosures:
See next page

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PDR ADOCK 05000272
P PDC

DISTRIBUTION w/enclosures:

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GHill (8)	Rita Jaques,ARM/LFMB				

[MILT]

PDI-2/LA	PDI-2/PM	OGC	PDI-2/D
MO'Brien	JStone:tr	BMM	WButler
/90	1/19/90	1/16/90	1/25/90

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

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Mr. Steven E. Miltenberger
Public Service Electric & Gas Company

Salem Nuclear Generating Station

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

PUBLIC SERVICE ELECTRIC & GAS COMPANY

PHILADELPHIA ELECTRIC COMPANY

DELMARVA POWER AND LIGHT COMPANY

ATLANTIC CITY ELECTRIC COMPANY

DOCKET NO. 50-272

SALEM GENERATING STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 108
License No. DPR-70

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

   A. The application for amendment filed by the Public Service Electric & Gas Company, Philadelphia Electric Company, Delmarva Power and Light Company and Atlantic City Electric Company (the licensees) dated December 28, 1988 and supplemented on July 31, 1989, October 18, 1989 and December 19, 1989 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 set forth in 10 CFR Chapter I;

   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:

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P PDC

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 1[illegible], 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 its date of issuance and shall be implemented within 60 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Walter R. Butler

Walter R. Butler, Director
Project Directorate I-2
Division of Reactor Projects I/II

Attachment:
Changes to the Technical
Specifications

Date of Issuance: January 29, 1990

ATTACHMENT TO LICENSE AMENDMENT NO. 108

FACILITY OPERATING LICENSE NO. DPR-70

DOCKET NO. 50-272

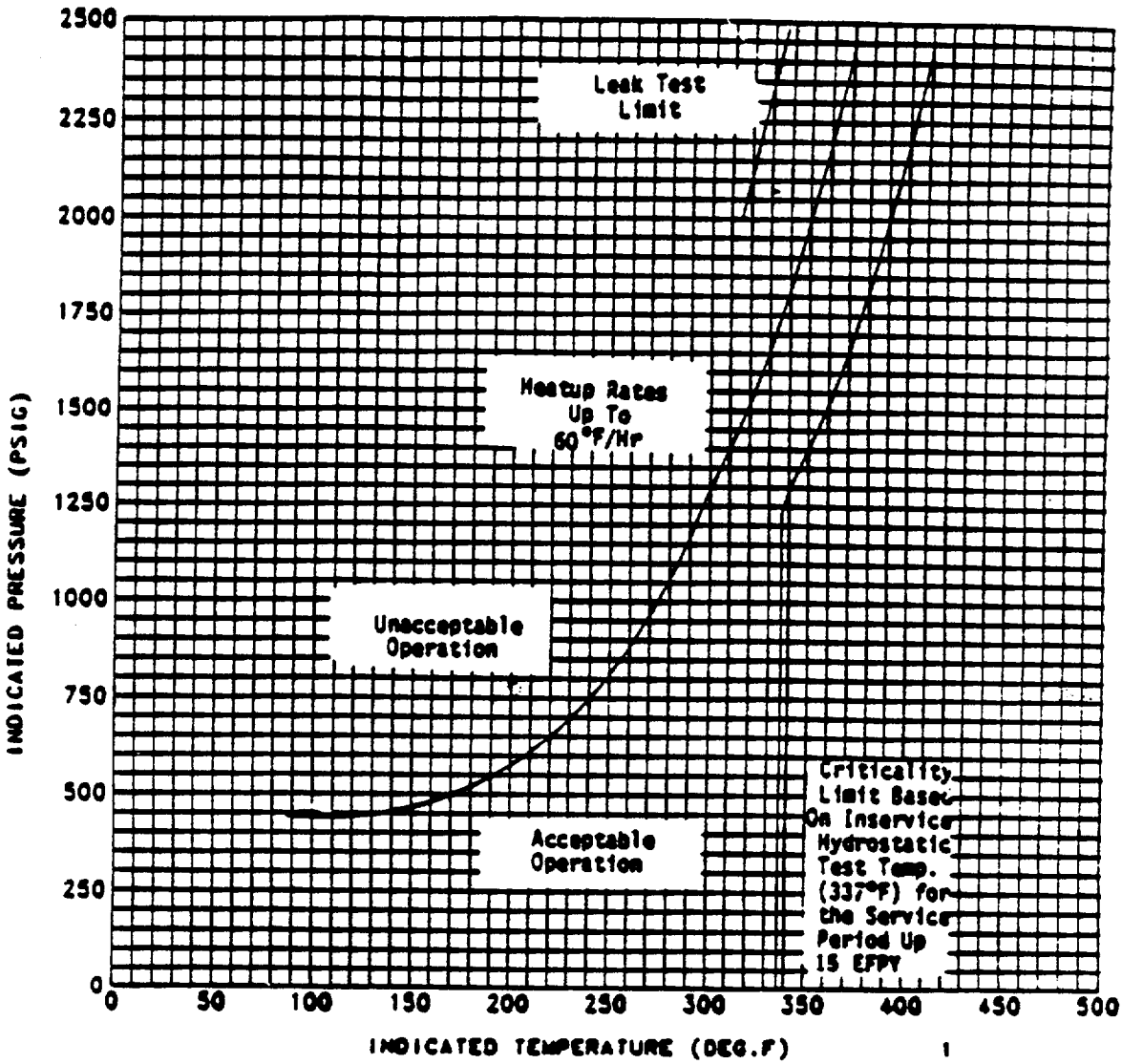
Revise Appendix A as follows:

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 4-11
-	B 3/4 4-12
-	B 3/4 4-13
-	B 3/4 4-14
-	B 3/4 4-15
-	B 3/4 4-16
-	B 3/4 4-17

Material Property

Initial RT_{NDT}: 45°F

RT_{NDT} After 15EFPY: 3/4T, 162°F
1/4T, 222.5°F



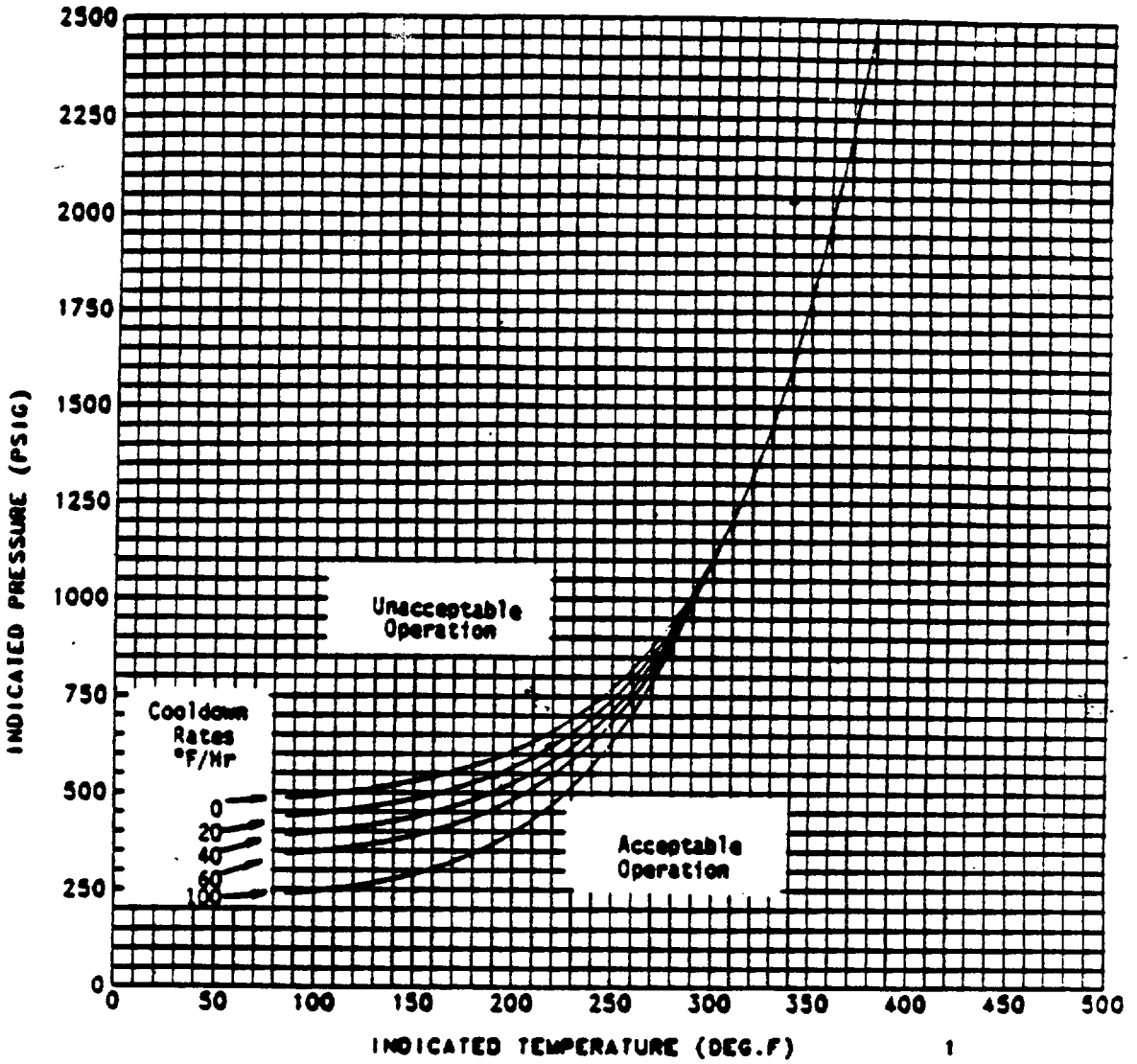
CONTAINS NO MARGIN FOR POSSIBLE INSTRUMENT ERRORS

Figure 3.4-2 . Salem Unit 1 Reactor Coolant System Heatup Limitations Applicable for Heatup Rates up to 60°F/HR for the Service Period up to 15 EFPY

Materia[illegible] roperty Basis

Initial RT_{NDT}: -56°F

RT_{NDT} After 15 EFPY: 1/4T = 222.5°F
3/4T = 162°F



CONTAINS NO MARGIN FOR POSSIBLE INSTRUMENT ERRORS

Figure 3.4-3 Salem Unit 1 Reactor Coolant System Cooldown Limitations Applicable for Cooldown Rates up to 100°F/HR for the Service Period up to 15 EFPY

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

The temperature and pressure changes during heatup and cooldown are limited to be consistent with the requirements given in the ASME Boiler and Pressure Vessel Code, Section III, Appendix G.

1) The reactor coolant temperature and pressure and system heatup and cooldown rate (with the exception of the pressurizer) shall be limited in accordance with Figures 3.4-2 and 3.4-3 for the service period specified thereon.

   a) Allowable combinations of pressure and temperature for specific temperature change rates are below and to the right of the limit lines shown. Limit lines for cooldown rates between those presented may be obtained by interpolation.

   b) Figures 3.4-2 and 3.4-3 define limits to assure prevention of nondoctile failure only. For normal operation, other inherent plant characteristics, e.g., pump heat addition and pressurizer heater capacity, may limit the heatup and cooldown rates that can be achieved over certain pressure-temperature ranges.

2) These limit lines shall be calculated periodically using methods provided below.

3) The secondary side of the steam generator must not be pressurized above 200 psig if the temperature of the steam generator is below 70°F.

4) The pressurizer heatup and cooldown rates shall not exceed 100°F/hr and 200°F/hr, respectively. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 320°F.

5) System preservice hydrotests and in-service leak and hydrotests shall be performed at pressures in accordance with the requirements of ASME Boiler and Pressure Vessel Code, Section XI.

The fracture toughness properties of the ferritic materials in the reactor vessel are determined in accordance with the NRC Standard Review Plan, ASTM E185-82, and in accordance with additional reactor vessel requirements. These properties are then evaluated in accordance with Appendix G of the 1976 Summer Addenda to Section III of the ASME Boiler and Pressure Vessel Code and the calculation methods described in WCAP-7924-A, "Basis for Heatup and Cooldown Limit Curves, April 1975".

Heatup and cooldown limit curves are calculated using the most limiting value of the nil-ductility reference temperature, RT_{NDT}, at the end of 15 effective full power years of service life. The 15 EFPY service life period is chosen such that the limiting RT_{NDT} at the 1/4T location in the core region is greater than the RT_{NDT} of the limiting unirradiated material. The selection of such a limiting RT_{NDT} assures that all components in the Reactor Coolant System will be operated conservatively in accordance with applicable Code requirements.

The reactor vessel materials have been tested to determine their initial RT_{NDT}; the results of these tests are shown in Table B 3/4.4-1. Reactor operation and resultant fast neutron (E greater than 1 MEV) irradiation can cause an increase in the RT_{NDT}. An adjusted reference temperature, (ART), based upon the fluence and the copper and nickel content of the material in question, can be predicted.

The ART is based upon the largest value of RT_{NDT} computed by the methodology presented in Regulatory Guide 1.99, Revision 2. The ART for each material is given by the following expression:

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

Initial RT_{NDT} is the reference temperature for the unirradiated material. RT_{NDT} is the mean value of the adjustment in reference temperature caused by the irradiation and is calculated as follows:

$$RT_{NDT} = \text{Chemistry Factor x Fluence Factor}$$

The Chemistry Factor, CF (F), is a function of copper and nickel content. It is given in Table B3/4.4-2 for welds and in Table B3/4.4-3 for base metal (plates and forgings). Linear interpolation is permitted.

The predicted neutron fluence as a function of Effective Full Power Years (EFPY) has been calculated and is shown in Figure B3/4.4-1. The fluence factor can be calculated by using Figure B3/4.4-2. Also, the neutron fluence at any depth in the vessel wall is determined as follows:

$$f = (f \text{ surface}) \times (e^{-0.24X})$$

where "f surface" is from Figure B3/4.3-1, and X (in inches) is the depth into the vessel wall.

Finally, the "Margin" is the quantity in °F that is to be added to obtain conservative, upper-bound values of adjusted reference temperature for the calculations required by Appendix G to 10 CFR Part 50.

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

If a measured value of initial RT_{NDT} for the material in question is used, σ_I may be taken as zero. If generic value of initial RT_{NDT} is used, σ_I, should be obtained from the same set of data. The standard deviations, for $_\Delta RT_{NDT}$, σ_Δ, are 28°F for welds and 17°F for base metal, except that σ_Δ need not exceed 0.50 times the mean value of $_\Delta RT_{NDT}$ surface.

The heatup and cooldown limit curves of Figures 3.4-2 and 3.4-3 include predicted adjustments for this shift in RT_{NDT} at the end of 15 EFPY.

Values of ΔRT_{NDT} determined in this manner may be used until the results from the material surveillance program, evaluated according to ASTM E185, are available. Capsules will be removed in accordance with the requirements of ASTM E185-82 and 10 CFR Part 50, Appendix H. The heatup and cooldown curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule exceeds the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure.

Allowable pressure-temperature relationships for various heatup and cooldown rates are calculated using methods derived from Appendix G in Section III of the ASME Boiler and Pressure Vessel Code as required by Appendix G to 10 CFR Part 50 and these methods are discussed in detail in WCAP-7924-A.

The general method for calculating heatup and cooldown limit curves is based upon the principles of the linear elastic fracture mechanics (LEFM) technology. In the calculation procedures a semi-elliptical surface defect with a depth of one-quarter of the wall thickness, T, and a length of 3/2T is assumed to exist at the inside of the vessel wall as well as at the outside of the vessel wall. The dimensions of this postulated crack, referred to in Appendix G of ASME Section III as the reference flaw, amply exceed the current capabilities of inservice inspection techniques. Therefore, the reactor operation limit curves developed for this reference crack are conservative and provide sufficient safety margins for protection against nonductile failure. To assure that the radiation embrittlement effects are accounted for in the calculation of the limit curves, the most limiting value of the nil-ductility reference temperature, RT_{NDT}, is used and this includes the radiation induced shift, ΔRT_{NDT} corresponding to the end of the period for which heatup and cooldown curves are generated.

The ASME approach for calculating the allowable limit curves for various heatup and cooldown rates specifies that the total stress intensity factor, K_I, for the combined therma] and pressure stresses at any time during heatup or cooldown cannot be greater than the reference stress intensity factor, K_{IR}, for the metal temperature at that time. K_{IR} is obtained from the reference fracture toughness curve, defined in Appendix G to the ASME Code. The K_{IR} curve is given by the equation:

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

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

$$C\,K_{IM} + K_{IT} \leq K_{IR} \qquad (2)$$

where K_{IM} is the stress intensity factor caused by membrane (pressure) stress.

K_{IT} is the stress intensity factor caused by the thermal gradients.

K_{IR} is provided by the code as a function of temperature relative to the RT_{NDT} of the material.

C = 2.0 for level A and B service limits, and

C = 1.5 for inservice hydrostatic and leak test operations.

At any time during the heatup or cooldown transient, K_{IR} is determined by the metal temperature at the tip of the postulated flaw, the appropriate value for RT_{NDT}, and the reference fracture toughness curve. The thermal stresses resulting from temperatue gradients through the vessel wall are calculated and then the corresponding (thermal) stress intensity factors, K_{IT}, for the reference flaw are computed. From Equation (2) the pressure stress intensity factors are obtained and from these the allowable pressures are calculated.

COOLDOWN

For the calculation of the allowable pressure versus coolant temperature during cooldown, the Code reference flaw is assumed to exist at the inside of the vessel wall. During cooldown, the controlling location of the flaw is always at the inside of the wall because the thermal gradients produce tensile stresses at the inside, which increase with increasing cooldown rates. Allowable pressure-temperature relations are generated for both steady-state and finite cooldown rate situations. From these relations composite limit curves are constructed for each cooldown rate of interest.

The use of the composite curve in the cooldown analysis is necessary because control of the cooldown procedure is based on measurement of reactor coolant temperature, whereas the limiting pressure is actually dependent on the material temperature at the tip of the assumed flaw. During cooldown, the 1/4T vessel location is at a higher temperature than the fluid adjacent to the vessel ID. This condition, of course, is not true for the steady-state situation. It follows that at any given reactor coolant temperature, the ΔT developed during cooldown results in a higher value of K_{IR} at the 1/4T location for finite cooldown rates than for steady-state operation. Furthermore, if conditions exist such that the increase in K_{IR} exceeds K_{IT}, the calculated allowable pressure during cooldown will be greater than the steady-state value.

The above procedures are needed because there is no direct control on temperature at the 1/4T location, therefore, allowable pressures may unknowingly be violated if the rate of cooling is decreased at various intervals along a cooldown ramp. The use of the composite curve eliminates this problem and assures conservative operation of the system for the entire cooldown period.

HEATUP

Three separate calculations are required to determine the limit curves for finite heatup rates. As is done in the cooldown analysis, allowable pressure-temperature relationships are developed for steady-state conditions as well as finite heatup rate conditions assuming the presence of a 1/4T defect at the inside of the vessel wall. The thermal gradients during heatup produce compressive stress at the inside of the wall that alleviate the tensile stresses produced by internal pressure. The metal temperature at the crack tip lags the coolant temperature therefore, the K_{IR} for the 1/4T crack during heatup is lower than the K_{IR} for the 1/4T crack during steady-state conditions at the same coolant temperature. During heatup, especially at the end of the transient, conditions may exist such that the effects of compressive thermal stresses and different K_{IR}s for steady-state and finite heatup rates do not offset each other and the pressure-temperature curve based on steady-state conditions no longer represents a lower bound of all similar curves for finite heatup rates when the 1/4T flaw is considered. Therefore, both cases have to be analyzed in order to assure that at any coolant temperature the lower value of the allowable pressure calculated for steady-state and finite heatup rates is obtained.

The second portion of the heatup analysis concerns the calculation of pressure-temperature limitations for the case in which a 1/4T deep outside surface flaw is assumed. Unlike the situation at the vessel inside surface, the thermal gradients established at the outside surface during heatup produce stresses which are tensile in nature and thus tend to reinforce any pressure stresses present. These thermal stresses, of course, are dependent on both the rate of heatup and the time (or coolant temperature) along the heatup ramp. Furthermore, since the thermal stresses, at the outside are tensile and increase with increasing heatup rate, a lower bound curve cannot be defined. Rather, each heatup rate of interest must be analyzed on an individual basis.

Following the generation of pressure-temperature curves for both the steady-state and finite heatup rate situations, the final limit curves are produced as follows. A composite curve is constructed based on a point-by-point comparison of the steady-state and finite heatup rate data. At any given temperature, the allowable pressure is taken to be the lesser of the three values taken from the curves under consideration.

The use of the composite curve is necessary to set conservative heatup limitations because it is possible for conditions to exist such that over the course of the heatup ramp the controlling condition switches from the inside to the outside and the pressure limit must at all times be based on analysis of the most critical criterion.

Finally, the new 10CFR50 rule which addresses the metal temperature of the closure head flange regions is considered. This 10CFR50 rule states that the metal temperature of the closure flange regions must exceed the material RT_{NDT} by at least 120°F for normal operation when the pressure exceeds 20 percent of the preservice hydrostatic test pressure (621 psig for Salem). Table B3/4.4-1 indicates that the limiting RT_{NDT} of 28°F occurs in the closure head flange of Salem Unit 1, and the minimum allowable temperature of this region is 148°F at pressures greater than 621 psig. These limits do not affect Figures 3.4-2 and 3.4-3.

Although the pressurizer operates in temperature ranges above those for which there is reason for concern of non-ductile failure, operating limits are provided to assure compatibility of operation with the fatigue analysis performed in accordance with the ASME Code requirements.

The OPERABILITY of two POPSs or an RCS vent opening of greater than 3.14 square inches ensures that the RCS will be protectd from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of 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 water solid RCS.

TABLE B 3/4.4-1

SALEM UNIT 1 REACTOR VESSEL TOUGHNESS DATA

Component	Plate No. or Weld No.	Material Type	Cu (%)	Ni (%)	T (°F)	50 ft lb 35-Mil Temp (°F)	RT (°F)	Average Upper Shelf Energy: Normal to Principal Working Direction (ft-lb)	Average Upper Shelf Energy: Principal Working Direction (ft-lb)
Cl Hd Dome	B2407-1	A533B, Cl.1	0.20	0.50	-30	99*	39	71.5*	110
Cl Hd Segment	B2406-1	A533B, Cl.1	0.13	0.52	-20	89*	29	97*	125
Cl Hd Segment	B2406-2	A533B, Cl.1	0.16	0.50	-30	85*	25	79*	122
Cl Hd Segment	B2406-3	A533B, Cl.1	0.10	0.53	-50	66*	6	86*	132
Cl Hd Flange	B2811	A508, Cl.2	-	0.72	28*	22*	28	129*	199
Vessel Flange	B2410	A508, Cl.2	-	0.67	60*	0*	60	94*	145
Inlet Nozzle	B2408-1	A508, Cl.2	-	0.68	50*	43*	50	94*	144
Inlet Nozzle	B2408-2	A508, Cl.2	-	0.71	46*	26*	46	102*	157
Inlet Nozzle	B2408-3	A508, Cl.2	-	0.66	47*	37*	47	105*	161
Inlet Nozzle	B2408-4	A508, Cl.2	-	0.65	9*	17*	9	108.5*	167
Outlet Nozzle	B2409-1	A508, Cl.2	-	0.69	60*	95*	60	48*	75
Outlet Nozzle	B2409-2	A508, Cl.2	-	0.69	60*	95*	60	51*	78
Outlet Nozzle	B2409-3	A508, Cl.2	-	0.74	60*	10*	60	79*	121
Outlet Nozzle	B2409-4	A508, Cl.2	-	0.74	60*	13*	60	82*	126
Upper Shell	B2401-1	A533B, Cl.1	0.22	0.48	-30	87*	27	74*	114
Upper Shell	B2401-2	A533B, Cl.1	0.19	0.48	0	80*	20	79*	122
Upper Shell	B2401-3	A533B, Cl.1	0.24	0.51	-10	114*	34	62*	96
Inter Shell	B2402-1	A533B, Cl.1	0.24	0.52	-30	105	45	73	97
Inter Shell	B2402-2	A533B, Cl.1	0.24	0.50	-30	55	-5	91	112
Inter Shell	B2402-3	A533B, Cl.1	0.12	0.50	-40	57	-3	104	127
Lower Shell	B2403-1	A533B, Cl.1	0.19	0.48	-40	70	10	99	143
Lower Shell	B2403-2	A533B, Cl.1	0.19	0.49	-70	86	26	94	128
Lower Shell	B2403-3	A533B, Cl.1	0.19	0.48	-40	90	30	102	131
Bot Hd Segment	B2404-1	A533B, Cl.1	0.10	0.52	10	48*	10	78*	120
Bot Hd Segment	B2404-2	A533B, Cl.1	0.11	0.53	-50	60*	0	86*	132
Bot Hd Segment	B2404-3	A533B, Cl.1	0.12	0.52	10	47*	10	82*	126
Bot Hd Dome	B2405-1	A533B, Cl.1	0.15	0.50	-20	57*	-3	69*	106
Cirum. Weld Bet Nozzle Shell & Int. Shell	8-042	-	0.22	1.02	-	-	-56***	-	-
Cirum. Weld Bet. Int. and Lower Shell	9-042	-	0.25	0.72	-	-	-56***	-	-
Int. Shell Vertical Weld	2-042 [A,B,C]	-	0.18	1.00**	-	-	-56***	-	-
Lower Shell Vertical Weld	3-042 [A,B,C]	-	0.19	1.00**	-	-	-56***	-	-

* Estimated per NRC Standard Review Plan Section 5.3.2.
** Estimated per Regulatory Guide 1.99, Rev. 2
*** Estimated per Pressurized Thermal Shock Rule, 10CFR50.61

TABLE B 3/4.4.2

CHEMISTRY FACTOR FOR WELDS, °F

Copper, Wt-%	Nickel, Wt-% 0	0.20	0.40	0.60	0.80	1.00	1.20
0	20	20	20	20	20	20	20
0.01	20	20	20	20	20	20	20
0.02	21	26	27	27	27	27	27
0.03	22	35	41	41	41	41	41
0.04	24	43	54	54	54	54	54
0.05	26	49	67	68	68	68	68
0.06	29	52	77	82	82	82	82
0.07	32	55	85	95	95	95	95
0.08	36	58	90	106	108	108	108
0.09	40	61	94	115	122	122	122
0.10	44	65	97	122	133	135	135
0.11	49	68	101	130	144	148	148
0.12	52	72	103	135	153	161	161
0.13	58	76	106	139	162	172	176
0.14	61	79	109	142	168	182	188
0.15	66	84	112	146	175	191	200
0.16	70	88	115	149	178	199	211
0.17	75	92	119	151	184	207	221
0.18	79	95	122	154	187	214	230
0.19	83	100	126	157	191	220	238
0.20	88	104	129	160	194	223	245
0.21	92	108	133	164	197	229	252
0.22	97	112	137	167	200	232	257
0.23	101	117	140	169	203	236	263
0.24	105	121	144	173	206	239	268
0.25	110	126	148	176	209	243	272
0.26	113	130	151	180	212	246	276
0.27	119	134	155	184	216	249	280
0.28	122	138	160	187	218	251	284
0.29	128	142	164	191	222	254	287
0.30	131	146	167	194	225	257	290
0.31	136	151	172	198	228	260	293
0.32	140	155	175	202	231	263	296
0.33	144	160	180	205	234	266	299
0.34	149	164	184	209	238	269	302
0.35	153	168	187	212	241	272	305
0.36	158	172	191	216	245	275	308
0.37	162	177	196	220	248	278	311
0.38	166	182	200	223	250	281	314
0.39	171	185	203	227	254	285	317
0.40	175	189	207	231	257	288	320

TABLE B 3/4.4-3

CHEMISTRY FACTOR FOR BASE METAL, °F

Copper, Wt-%	Nickel, Wt-% 0	0.20	0.40	0.60	0.80	1.00	1.20
0	20	20	20	20	20	20	20
0.01	20	20	20	20	20	20	20
0.02	20	20	20	20	20	20	20
0.03	20	20	20	20	20	20	20
0.04	22	26	26	26	26	26	26
0.05	25	31	31	31	31	31	31
0.06	28	37	37	37	37	37	37
0.07	31	43	44	44	44	44	44
0.08	34	48	51	51	51	51	51
0.09	37	53	58	58	58	58	58
0.10	41	58	65	65	67	67	67
0.11	45	62	72	74	77	77	77
0.12	49	67	79	83	86	86	86
0.13	53	71	85	91	96	96	96
0.14	57	75	91	100	106	106	106
0.15	61	80	99	110	115	117	117
0.16	65	84	104	118	123	125	125
0.17	69	88	110	127	132	135	135
0.18	73	92	115	134	141	144	144
0.19	78	97	120	142	150	154	154
0.20	82	102	125	149	159	164	165
0.21	86	107	129	155	167	172	174
0.22	91	112	134	161	176	181	184
0.23	95	117	138	167	184	190	194
0.24	100	121	143	172	191	199	204
0.25	104	126	148	176	199	208	214
0.26	109	130	151	180	205	216	221
0.27	114	134	155	184	211	225	230
0.28	119	138	160	187	216	233	239
0.29	124	142	164	191	221	241	248
0.30	129	146	167	194	225	249	257
0.31	134	151	172	198	228	255	266
0.32	139	155	175	202	231	260	274
0.33	144	160	180	205	234	264	282
0.34	149	164	184	209	238	268	290
0.35	153	168	187	212	241	272	298
0.36	158	173	191	216	245	275	303
0.37	162	177	196	220	248	278	308
0.38	166	182	200	223	250	281	313
0.39	171	185	203	227	254	285	317
0.40	175	189	207	231	257	288	320

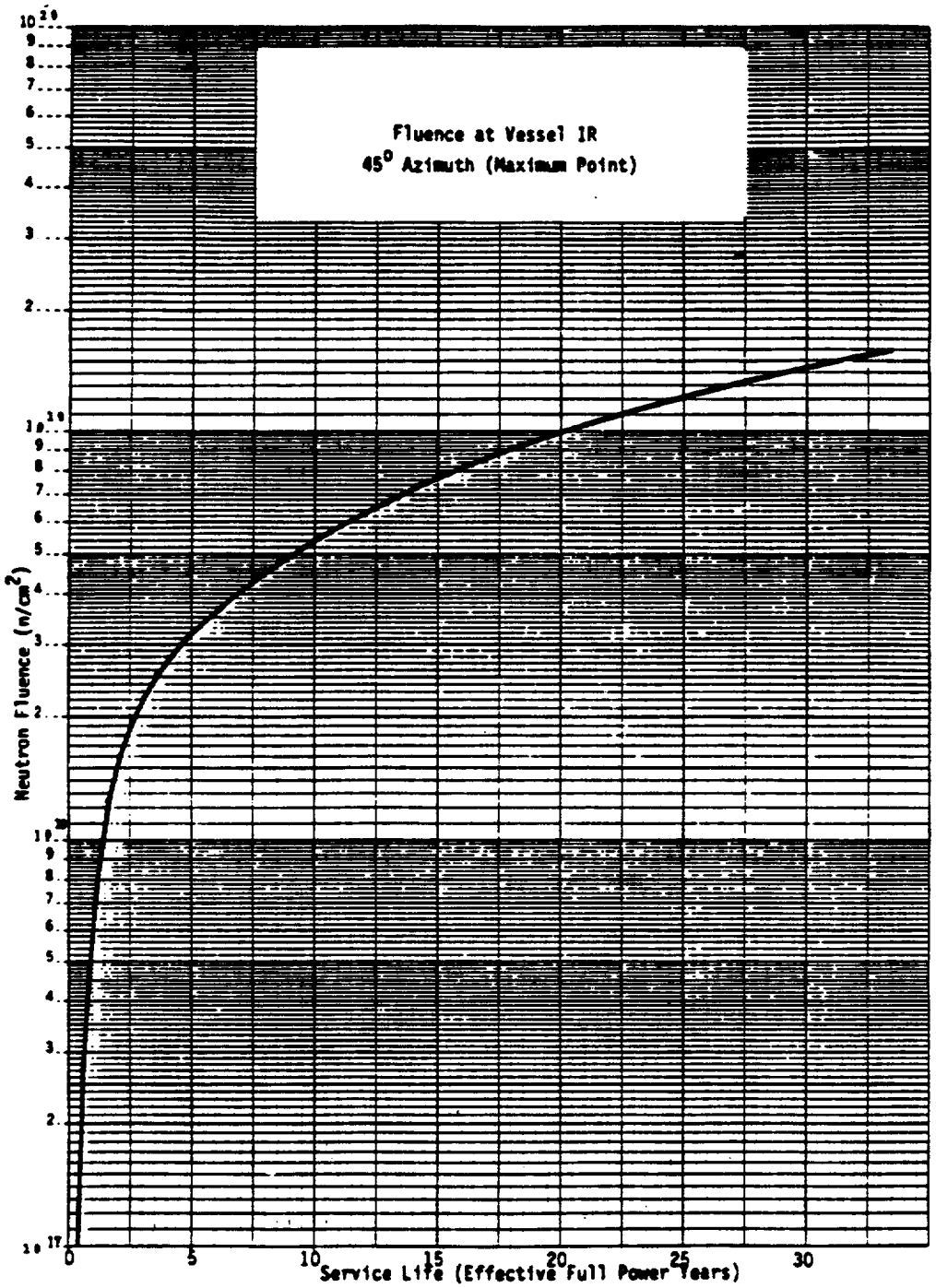
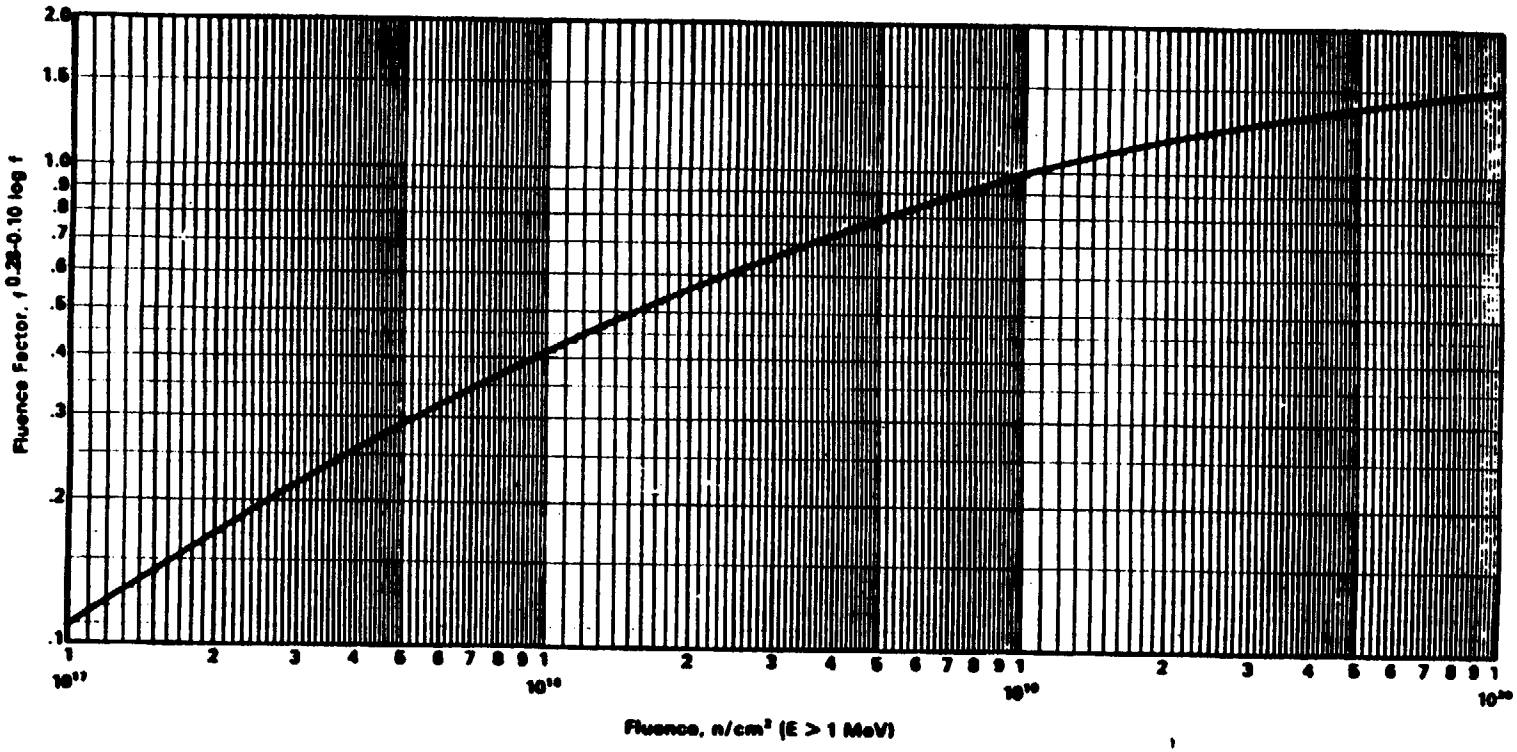


Figure B 3/4.4-1 Fast neutron fluence (E > 1MeV) as a function of of full power service life (EFPY)



Fluence factor for use in the expression for ΔRT_{NDT}

FIGURE B 3/4 4-2

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 maintained 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.

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

PUBLIC SERVICE ELECTRIC & GAS COMPANY

PHILADELPHIA ELECTRIC COMPANY

DELMARVA POWER AND LIGHT COMPANY

ATLANTIC CITY ELECTRIC COMPANY

DOCKET NO. 50-311

SALEM GENERATING STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 86
License No. DPR-75

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

   A. The application for amendment filed by the Public Service Electric & Gas Company, Philadelphia Electric Company, Delmarva Power and Light Company and Atlantic City Electric Company (the licensees) dated December 28, 1988 and supplemented on July 31, 1989, October 18, 1989 and December 19, 1989 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 set forth in 10 CFR Chapter I;

   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-75 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 86 , 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 its date of issuance and shall be implemented within 60 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Walter R. Butler

Walter R. Butler, Director
Project Directorate I-2
Division of Reactor Projects I/II

Attachment:
Changes to the Technical
Specifications

Date of Issuance: January 29, 1990

ATTACHMENT TO LICENSE AMENDMENT NO. 86

FACILITY OPERATING LICENSE NO. DPR-75

DOCKET NO. 50-311

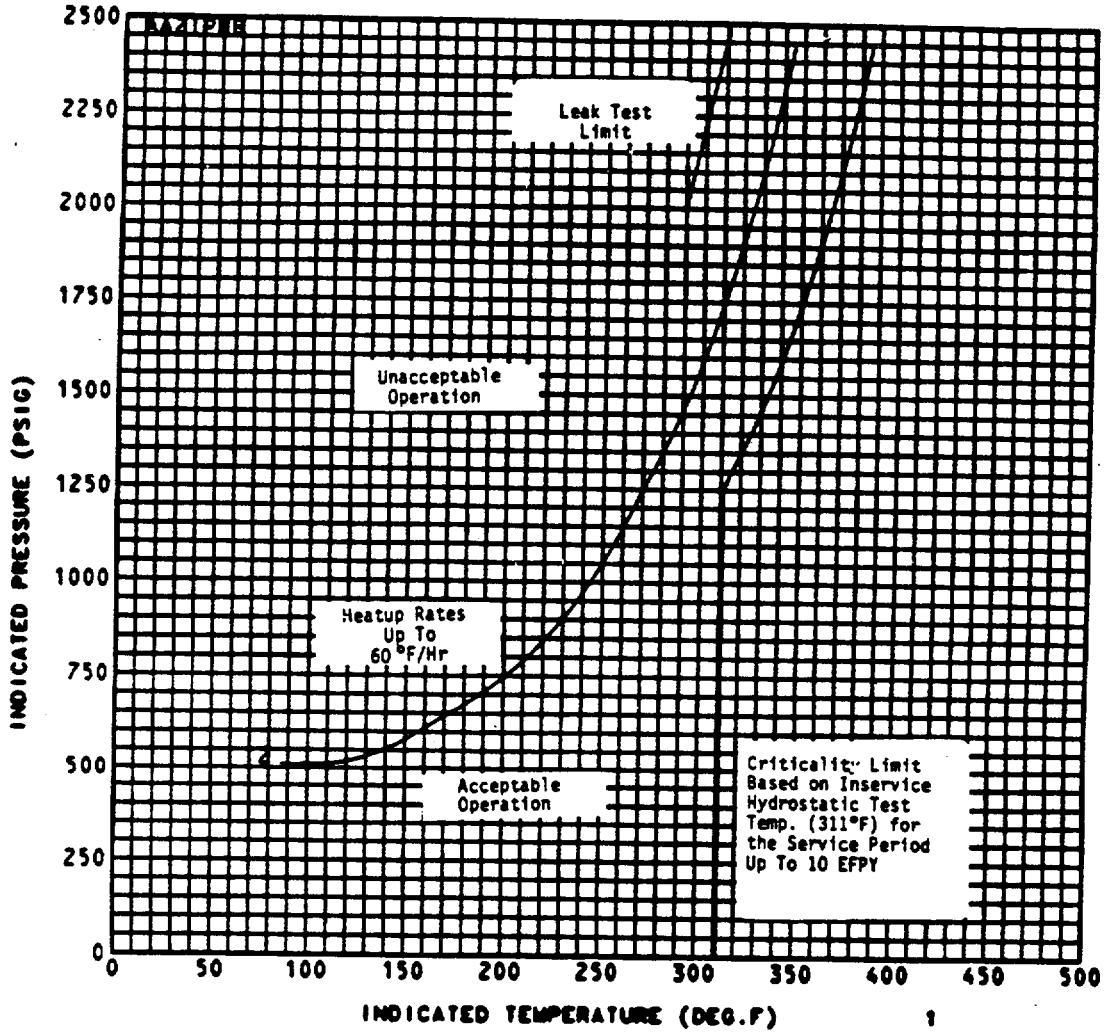
Revise Appendix A as follows:

Remove Pages	Insert Pages
3/4 4-28	3/4 4-28
3/4 4-29	3/4 4-29
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 4-11
B 3/4 4-12	B 3/4 4-12
B 3/4 4-13	B 3/4 4-13
B 3/4 4-14	B 3/4 4-14
B 3/4 4-15	B 3/4 4-15
B 3/4 4-16	B 3/4 4-16
B 3/4 4-17	B 3/4 4-17
-	B 3/4 4-18

MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL:	LONGITUDINAL WELD
COPPER CONTENT:	0.35 WT%
NICKEL CONTENT:	1.00 WT%
INITIAL RT_{NDT}:	-56°F
RT_{NDT} AFTER 10 EFPY:	1/4 T, 178.6°F 3/4 T, 116.1°F

CURVES APPLICABLE FOR HEATUP RATES UP TO 60°F/HR FOR THE SERVICE PERIOD UP TO 10 EFPY AND CONTAINS NO MARGIN FOR POSSIBLE INSTRUMENT ERRORS



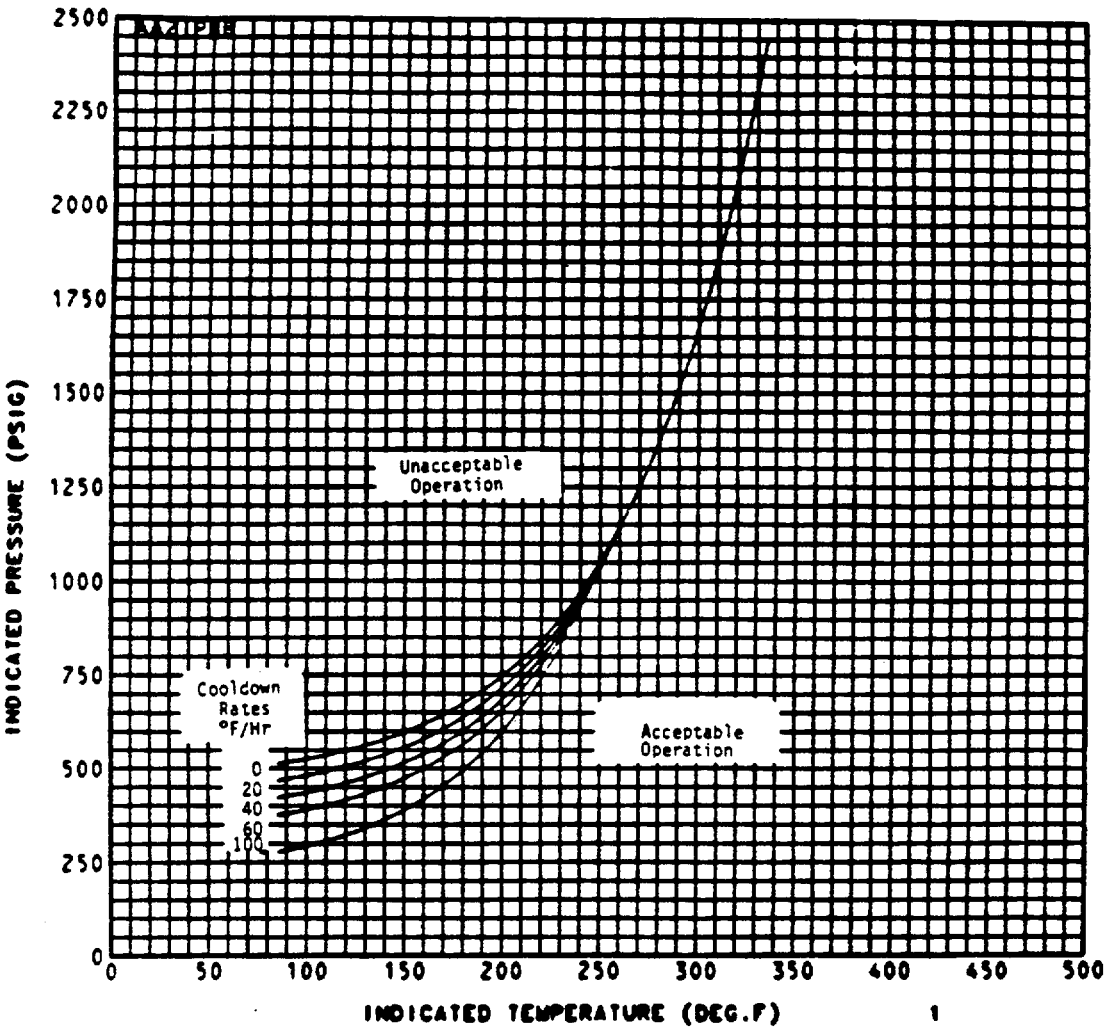
Salem Unit 2 reactor coolant system heatup limitations applicable for the first 10 EFPY with maximum heatup rate of 60°F/hr

FIGURE 3.4-2

MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL: LONGITUDINAL WELD
COPPER CONTENT: 0.35 WT%
NICKEL CONTENT: 1.00 WT%
INITIAL RT_{NDT}: -56°F
RT_{NDT} AFTER 10 EFPY: 1/4T, 178.6°F
3/4T, 116.1°F

CURVES APPLICABLE FOR COOLDOWN RATES UP TO 100°F/HR FOR THE SERVICE PERIOD UP TO 10 EFPY AND CONTAINS NO MARGIN FOR POSSIBLE INSTRUMENT ERRORS



Salem Unit 2 reactor coolant system cooldown limitations applicable for the first 10 EFPY

FIGURE 3.4-3

3/4.4.10 PRESSURE/TEMPERATURE LIMITS

The temperature and pressure changes during heatup and cooldown are limited to be consistent with the requirements given in the ASME Boiler and Pressure Vessel Code, Section III, Appendix G.

1) The reactor coolant temperature and pressure and system heatup and cooldown rate (with the exception of the pressurizer) shall be limited in accordance with Figures 3.4-2 and 3.4-3 for the service period specified thereon.

   a) Allowable combinations of pressure and temperature for specific temperature change rates are below and to the right of the limit lines shown. Limit lines for cooldown rates between those presented may be obtained by interpolation.

   b) Figures 3.4-2 and 3.4-3 define limits to assure prevention of nonductile failure only. For normal operation, other inherent plant characteristics, e.g., pump heat addition and pressurizer heater capacity, may limit the heatup and cooldown rates that can be achieved over certain pressure-temperature ranges.

2) These limit lines shall be calculated periodically using methods provided below.

3) The secondary side of the steam generator must not be pressurized above 200 psig if the temperature of the steam generator is below 70°F.

4) The pressurizer heatup and cooldown rates shall not exceed 100°F/hr and 200°F/hr, respectively. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 320°F.

5) System preservice hydrotests and in-service leak and hydrotests shall be performed at pressures in accordance with the requirements of ASME Boiler and Pressure Vessel Code, Section XI.

The fracture toughness properties of the ferritic materials in the reactor vessel are determined in accordance with the NRC Standard Review Plan, ASTM E185-82, and in accordance with additional reactor vessel requirements. These properties are then evaluated in accordance with Appendix G of the 1976 Summer Addenda to Section III of the ASME Boiler and Pressure Vessel Code and the calculation methods described in WCAP-7924-A, "Basis for Heatup and Cooldown Limit Curves, April 1975".

Heatup and cooldown limit curves are calculated using the most limiting value of the nil-ductility reference temperature, RT_{NDT}, at the end of 15 effective full power years of service life. The 15 EFPY service life period is chosen such that the limiting RT_{NDT} at the 1/4T location in the core region is greater than the RT_{NDT} of the limiting unirradiated material. The selection of such a limiting RT_{NDT} assures that all components in the Reactor Coolant System will be operated conservatively in accordance with applicable Code requirements.

The reactor vessel materials have been tested to determine their initial RT_{NDT}; the results of these tests are shown in Table B 3/4.4-1. Reactor operation and resultant fast neutron (E greater than 1 MEV) irradiation can cause an increase in the RT_{NDT}. An adjusted reference temperature, (ART), based upon the fluence and the copper and nickel content of the material in question, can be predicted.

The ART is based upon the largest value of RT_{NDT} computed by the methodology presented in Regulatory Guide 1.99, Revision 2. The ART for each material is given by the following expression:

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

Initial RT_{NDT} is the reference temperature for the unirradiated material. ΔRT_{NDT} is the mean value of the adjustment in reference temperature caused by the irradiation and is calculated as follows:

$$\Delta RT_{NDT} = \text{Chemistry Factor} \times \text{Fluence Factor}$$

The Chemistry Factor, CF (F), is a function of copper and nickel content. It is given in Table B3/4.4-2 for welds and in Table B3/4.4-3 for base metal (plates and forgings). Linear interpolation is permitted.

The predicted neutron fluence as a function of Effective Full Power Years (EFPY) has been calculated and is shown in Figure B3/4.4-1. The fluence factor can be calculated by using Figure B3/4.4-2. Also, the neutron fluence at any depth in the vessel wall is determined as follows:

$$f = (f \text{ surface}) \times (e^{-0.24X})$$

where "f surface" is from Figure B3/4.3-1, and X (in inches) is the depth into the vessel wall.

Finally, the "Margin" is the quantity in ^{o}F that is to be added to obtain conservative, upper-bound values of adjusted reference temperature for the calculations required by Appendix G to 10 CFR Part 50.

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

If a measured value of initial RT_{NDT} for the material in question is used, σ_I may be taken as zero. If generic value of initial RT_{NDT} is used, σ_I should be obtained from the same set of data. The standard deviations, for ΔRT_{NDT}, σ_Δ, are $28^{o}F$ for welds and $17^{o}F$ for base metal, except that σ_Δ need not exceed 0.50 times the mean value of ΔRT_{NDT} surface.

The heatup and cooldown limit curves of Figures 3.4-2 and 3.4-3 include predicted adjustments for this shift in RT_{NDT} at the end of 15 EFPY.

Values of ΔRT_{NDT} determined in this manner may be used until the results from the material surveillance program, evaluated according to ASTM E185, are available. Capsules will be removed in accordance with the requirements of ASTM E185-82 and 10 CFR Part 50, Appendix H. The heatup and cooldown curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule exceeds the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure.

Allowable pressure-temperature relationships for various heatup and cooldown rates are calculated using methods derived from Appendix G in Section III of the ASME Boiler and Pressure Vessel Code as required by Appendix G to 10 CFR Part 50 and these methods are discussed in detail in WCAP-7924-A.

The general method for calculating heatup and cooldown limit curves is based upon the principles of the linear elastic fracture mechanics (LEFM) technology. In the calculation procedures a semi-elliptical surface defect with a depth of one-quarter of the wall thickness, T, and a length of 3/2T is assumed to exist at the inside of the vessel wall as well as at the outside of the vessel wall. The dimensions of this postulated crack, referred to in Appendix G of ASME Section III as the reference flaw, amply exceed the current capabilities of inservice inspection techniques. Therefore, the reactor operation limit curves developed for this reference crack are conservative and provide sufficient safety margins for protection against nonductile failure. To assure that the radiation embrittlement effects are accounted for in the calculation of the limit curves, the most limiting value of the nil-ductility reference temperature, RT_{NDT}, is used and this includes the radiation induced shift, ΔRT_{NDT} corresponding to the end of the period for which heatup and cooldown curves are generated.

The ASME approach for calculating the allowable limit curves for various heatup and cooldown rates specifies that the total stress intensity factor, K_I, for the combined thermal and pressure stresses at any time during heatup or cooldown cannot be greater than the reference stress intensity factor, K_{IR}, for the metal temperature at that time. K_{IR} is obtained from the reference fracture toughness curve, defined in Appendix G to the ASME Code. The K_{IR} curve is given by the equation:

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

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

$$C\, K_{IM} + K_{IT} \leq K_{IR} \qquad (2)$$

where K_{IM} is the stress intensity factor caused by membrane (pressure) stress.

K_{IT} is the stress intensity factor caused by the thermal gradients.

K_{IR} is provided by the code as a function of temperature relative to the RT_{NDT} of the material.

C = 2.0 for level A and B service limits, and

C = 1.5 for inservice hydrostatic and leak test operations.

At any time during the heatup or cooldown transient, K_{IR} is determined by the metal temperature at the tip of the postulated flaw, the appropriate value for RT_{NDT}, and the reference fracture toughness curve. The thermal stresses resulting from temperatue gradients through the vessel wall are calculated and then the corresponding (thermal) stress intensity factors, K_{IT}, for the reference flaw are computed. From Equation (2) the pressure stress intensity factors are obtained and from these the allowable pressures are calculated.

COOLDOWN

For the calculation of the allowable pressure versus coolant temperature during cooldown, the Code reference flaw is assumed to exist at the inside of the vessel wall. During cooldown, the controlling location of the flaw is always at the inside of the wall because the thermal gradients produce tensile stresses at the inside, which increase with increasing cooldown rates. Allowable pressure-temperature relations are generated for both steady-state and finite cooldown rate situations. From these relations composite limit curves are constructed for each cooldown rate of interest.

The use of the composite curve in the cooldown analysis is necessary because control of the cooldown procedure is based on measurement of reactor coolant temperature, whereas the limiting pressure is actually dependent on the material temperature at the tip of the assumed flaw. During cooldown, the 1/4T vessel location is at a higher temperature than the fluid adjacent to the vessel ID. This condition, of course, is not true for the steady-state situation. It follows that at any given reactor coolant temperature, the ΔT developed during cooldown results in a higher value of K_{IR} at the 1/4T location for finite cooldown rates than for steady-state operation. Furthermore, if conditions exist such that the increase in K_{IR} exceeds K_{IT}, the calculated allowable pressure during cooldown will be greater than the steady-state value.

The above procedures are needed because there is no direct control on temperature at the 1/4T location, therefore, allowable pressures may unknowingly be violated if the rate of cooling is decreased at various intervals along a cooldown ramp. The use of the composite curve eliminates this problem and assures conservative operation of the system for the entire cooldown period.

HEATUP

Three separate calculations are required to determine the limit curves for finite heatup rates. As is done in the cooldown analysis, allowable pressure-temperature relationships are developed for steady-state conditions as well as finite heatup rate conditions assuming the presence of a 1/4T defect at the inside of the vessel wall. The thermal gradients during heatup produce compressive stress at the inside of the wall that alleviate the tensile stresses produced by internal pressure. The metal temperature at the crack tip lags the coolant temperature. Therefore the K_{IR} for the 1/4T crack during heatup is lower than the K_{IR} for the 1/4T crack during steady-state conditions at the same coolant temperature. During heatup, especially at the end of the transient, conditions may exist such that the effects of compressive thermal stresses and different K_{IR}s for steady-state and finite heatup rates do not offset each other and the pressure-temperature curve based on steady-state conditions no longer represents a lower bound of all similar curves for finite heatup rates when the 1/4T flaw is considered. Therefore, both cases have to be analyzed in order to assure that at any coolant temperature the lower value of the allowable pressure calculated for steady-state and finite heatup rates is obtained.

The second portion of the heatup analysis concerns the calculation of pressure-temperature limitations for the case in which a 1/4T deep outside surface flaw is assumed. Unlike the situation at the vessel inside surface, the thermal gradients established at the outside surface during heatup produce stresses which are tensile in nature and thus tend to reinforce any pressure stresses present. These thermal stresses, of course, are dependent on both the rate of heatup and the time (or coolant temperature) along the heatup ramp. Furthermore, since the thermal stresses, at the outside are tensile and increase with increasing heatup rate, a lower bound curve cannot be defined. Rather, each heatup rate of interest must be analyzed on an individual basis.

Following the generation of pressure-temperature curves for both the steady-state and finite heatup rate situations, the final limit curves are produced as follows. A composite curve is constructed based on a point-by-point comparison of the steady-state and finite heatup rate data. At any given temperature, the allowable pressure is taken to be the lesser of the three values taken from the curves under consideration.

The use of the composite curve is necessary to set conservative heatup limitations because it is possible for conditions to exist such that over the course of the heatup ramp the controlling condition switches from the inside to the outside and the pressure limit must at all times be based on analysis of the most critical criterion.

Finally, the new 10CFR50 rule which addresses the metal temperature of the closure head flange regions is considered. This 10CFR50 rule states that the metal temperature of the closure flange regions must exceed the material RT_{NDT} by at least 120°F for normal operation when the pressure exceeds 20 percent of the preservice hydrostatic test pressure (621 psig for Salem). Table B3/4.4-1 indicates that the limiting RT_{NDT} of 28°F occurs in the closure head flange of Salem Unit 2, and the minimum allowable temperature of this region is 148°F at pressures greater than 621 psig. These limits do not affect Figures 3.4-2 and 3.4-3.

Although the pressurizer operates in temperature ranges above those for which there is reason for concern of non-ductile failure, operating limits are provided to assure compatibility of operation with the fatigue analysis performed in accordance with the ASME Code requirements.

The OPERABILITY of two POPSs or an RCS vent opening of greater than 3.14 square inches ensures that the RCS will be protectd from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of 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 water solid RCS.

TABLE B 3/4.4-1

SALEM UNIT 2 REACTOR VESSEL TOUGHNESS DATA

Component	Plate No. or Weld No.	Material Type	Cu (%)	Ni (%)	T (°F)	50 ft lb 35-Mil Temp (°F)	RT (°F)	Average Upper Shelf Energy: Normal to Principal Working Direction (ft-lb)	Average Upper Shelf Energy: Principal Working Direction (ft-lb)
Closure Hd Dome	B4708	A533BCL1	0.11	0.70	-40	45*	-15*	82.5	127
Closure Hd Peel	B5007-3	A533BCL1	0.12	0.57	-20	15*	-20*	97*	149
Closure Hd Peel	B4707-1	A533BCL1	0.10	0.55	0	51*	0*	84*	129
Closure Hd Peel	B4707-3	A533BCL1	0.13	0.63	0	66*	6*	84*	129.5
Closure Hd Flng	B4702-1	A508CL2	-	0.68	28*	39*	28*	104*	160
Vessel Flange	B5001	A508CL2	-	0.70	12*	4*	12*	107*	164
Inlet Nozzle	B4703-1	A508CL2	-	0.69	60*	62*	60*	>72*	>111**
Inlet Nozzle	B4703-2	A508CL2	-	0.69	60*	25*	60*	>61*	> 94**
Inlet Nozzle	B4703-3	A508CL2	-	0.68	60*	32*	60*	>71*	>109**
Inlet Nozzle	B4703-4	A508CL2	-	0.81	60*	40*	60*	80*	123.5
Outlet Nozzle	B4704-1	A508CL2	-	0.84	60*	8*	60*	82*	126
Outlet Nozzle	B4704-2	A508CL2	-	0.77	60*	20*	60*	75*	116
Outlet Nozzle	B4704-3	A508CL2	-	0.69	28*	8*	28*	82*	126
Outlet Nozzle	B4704-4	A508CL2	-	0.71	60*	40*	60*	77*	119
Upper Shell	B4711-1	A533BCL1	0.11	0.55	0*	50*	0*	87*	134
Upper Shell	B4711-2	A533BCL1	0.14	0.56	-10	60*	0*	79*	122
Upper Shell	B4711-3	A533BCL1	0.12	0.58	-10	88*	28*	69*	107
Inter. Shell	B4712-1	A533BCL1	0.13	0.56	0	<60	0	105	138
Inter. Shell	B4712-2	A533BCL1	0.14	0.60	-20	72	12	97	127.5
Inter. Shell	B4712-3	A533BCL1	0.11	0.57	-50	70	10	107	116
Lower Shell	B4713-1	A533BCL1	0.12	0.60	-10	68	8	98	127
Lower Shell	B4713-2	A533BCL1	0.12	0.57	-20	68	8	103	135.5
Lower Shell	B4713-3	A533BCL1	0.12	0.58	-10	70	10	122	135.5
Bottom Hd Peel	B4709-1	A533BCL1	0.12	0.60	-30	54*	-6*	90*	139
Bottom Hd Peel	B4709-2	A533BCL1	0.12	0.58	-20	42*	-18*	89*	137.5
Bottom Hd Peel	B4709-3	A533BCL1	0.11	0.56	-20	71*	11*	93*	143
Bottom Head	B4710	A533BCL1	0.12	0.60	-30	60*	0*	77*	118
Cirum. Weld Bet Nozzle Shell & Int. Shell	8-442	-	0.28	0.74	-	-	-56***	-	-
Cirum. Weld Bet Int. Shell & Lower Shell	9-442	-	0.175	0.20****	-	-	-56***	-	-
Int. Shell Vertical Weld	2-442 [A,B,C]	-	0.23	0.73	-	-	-56***	-	-
Lower Shell Vertical Weld	3-442 [A,B,C]	-	0.20	0.86	-	-	-56***	-	-

* Estimated per NRC Standard Review Plan Section 5.3.2.
** 100% Shear not reached
*** Estimated per Pressurized Thermal Shock Rule, 10CFR50.61
****Estimated value for type MIL B-4 wire heats.

TABLE B 3/4.4.2

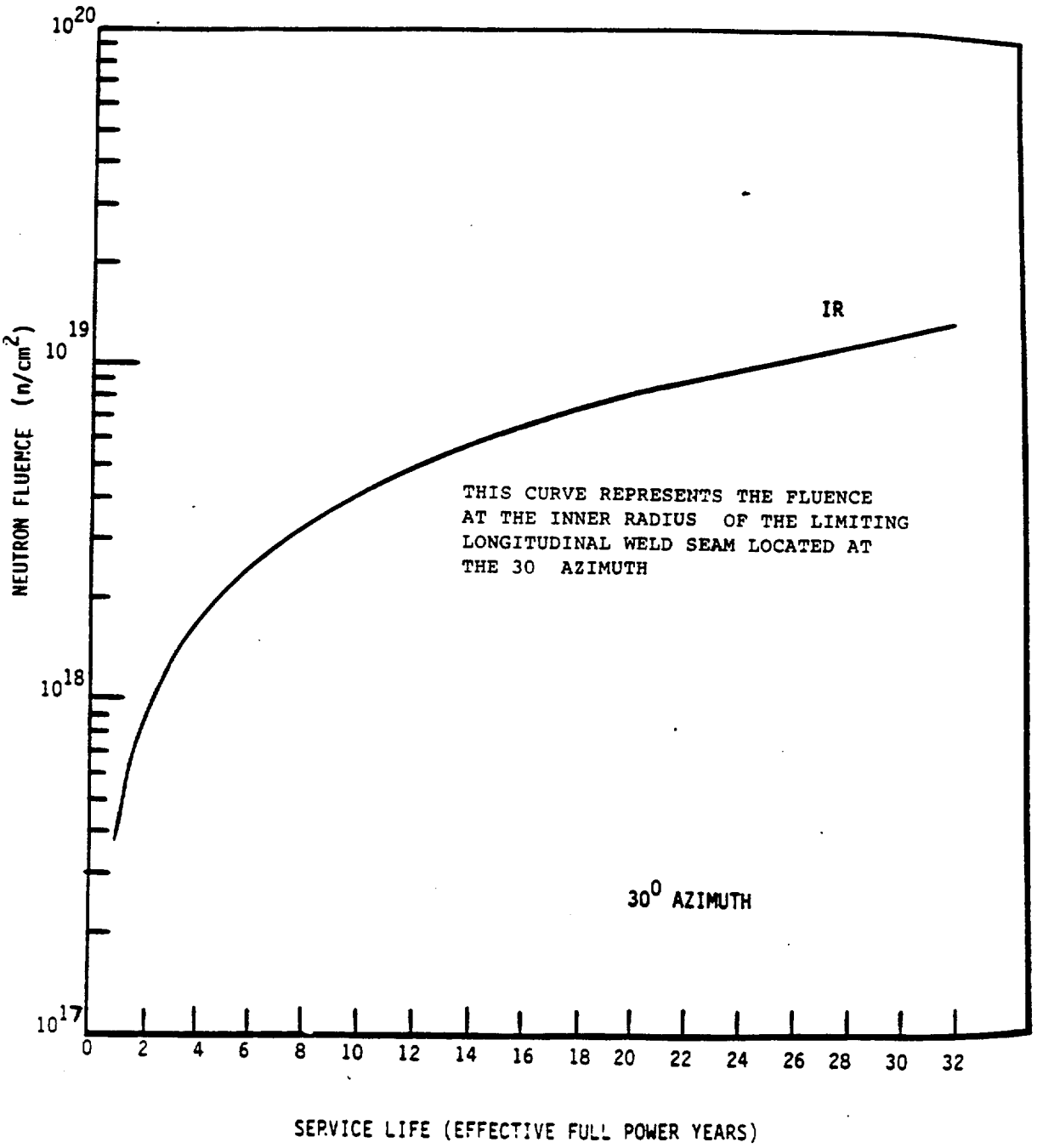
CHEMISTRY FACTOR FOR WELDS, °F

Copper, Wt-%	Nickel, Wt-% 0	0.20	0.40	0.60	0.80	1.00	1.20
0	20	20	20	20	20	20	20
0.01	20	20	20	20	20	20	20
0.02	21	26	27	27	27	27	27
0.03	22	35	41	41	41	41	41
0.04	24	43	54	54	54	54	54
0.05	26	49	67	68	68	68	68
0.06	29	52	77	82	82	82	82
0.07	32	55	85	95	95	95	95
0.08	36	58	90	106	108	108	108
0.09	40	61	94	115	122	122	122
0.10	44	65	97	122	133	135	135
0.11	49	68	101	130	144	148	148
0.12	52	72	103	135	153	161	161
0.13	58	76	106	139	162	172	176
0.14	61	79	109	142	168	182	188
0.15	66	84	112	146	175	191	200
0.16	70	88	115	149	178	199	211
0.17	75	92	119	151	184	207	221
0.18	79	95	122	154	187	214	230
0.19	83	100	126	157	191	220	238
0.20	88	104	129	160	194	223	245
0.21	92	108	133	164	197	229	252
0.22	97	112	137	167	200	232	257
0.23	101	117	140	169	203	236	263
0.24	105	121	144	173	206	239	268
0.25	110	126	148	176	209	243	272
0.26	113	130	151	180	212	246	276
0.27	119	134	155	184	216	249	280
0.28	122	138	160	187	218	251	284
0.29	128	142	164	191	222	254	287
0.30	131	146	167	194	225	257	290
0.31	136	151	172	198	228	260	293
0.32	140	155	175	202	231	263	296
0.33	144	160	180	205	234	266	299
0.34	149	164	184	209	238	269	302
0.35	153	168	187	212	241	272	305
0.36	158	172	191	216	245	275	308
0.37	162	177	196	220	248	278	311
0.38	166	182	200	223	250	281	314
0.39	171	185	203	227	254	285	317
0.40	175	189	207	231	257	288	320

TABLE B 3/4.4-3

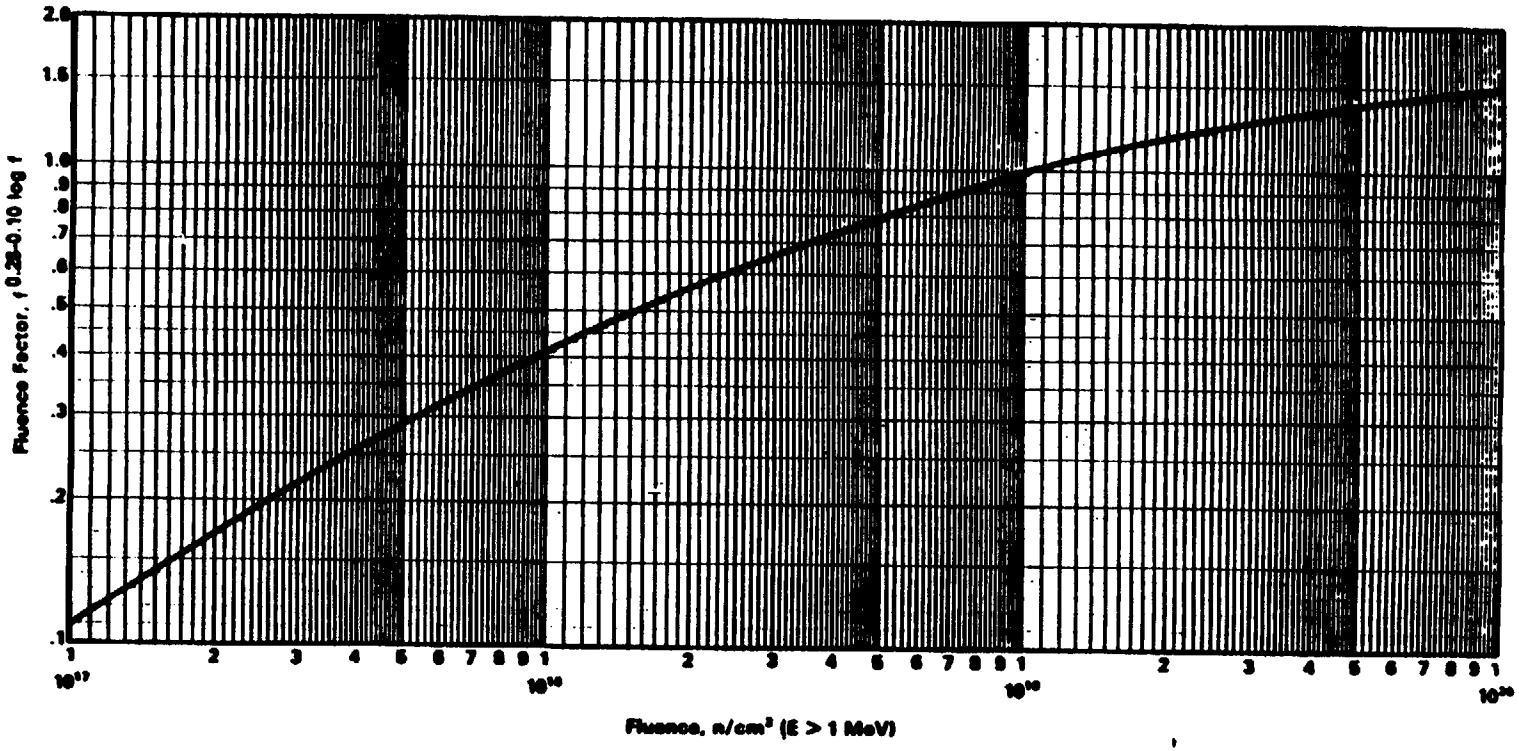
CHEMISTRY FACTOR FOR BASE METAL, °F

Copper, Wt-%	Nickel, Wt-% 0	0.20	0.40	0.60	0.80	1.00	1.20
0	20	20	20	20	20	20	20
0.01	20	20	20	20	20	20	20
0.02	20	20	20	20	20	20	20
0.03	20	20	20	20	20	20	20
0.04	22	26	26	26	26	26	26
0.05	25	31	31	31	31	31	31
0.06	28	37	37	37	37	37	37
0.07	31	43	44	44	44	44	44
0.08	34	48	51	51	51	51	51
0.09	37	53	58	58	58	58	58
0.10	41	58	65	65	67	67	67
0.11	45	62	72	74	77	77	77
0.12	49	67	79	83	86	86	86
0.13	53	71	85	91	96	96	96
0.14	57	75	91	100	105	106	106
0.15	61	80	99	110	115	117	117
0.16	65	84	104	118	123	125	125
0.17	69	88	110	127	132	135	135
0.18	73	92	115	134	141	144	144
0.19	78	97	120	142	150	154	154
0.20	82	102	125	149	159	164	165
0.21	86	107	129	155	167	172	174
0.22	91	112	134	161	176	181	184
0.23	95	117	138	167	184	190	194
0.24	100	121	143	172	191	199	204
0.25	104	126	148	176	199	208	214
0.26	109	130	151	180	205	216	221
0.27	114	134	155	184	211	225	230
0.28	119	138	160	187	216	233	239
0.29	124	142	164	191	221	241	248
0.30	129	146	167	194	225	249	257
0.31	134	151	172	198	228	255	266
0.32	139	155	175	202	231	260	274
0.33	144	160	180	205	234	264	282
0.34	149	164	184	209	238	268	290
0.35	153	168	187	212	241	272	298
0.36	158	173	191	216	245	275	303
0.37	162	177	196	220	248	278	308
0.38	166	182	200	223	250	281	313
0.39	171	185	203	227	254	285	317
0.40	175	189	207	231	257	288	320



Fast neutron fluence (E > 1 MeV) as a function of full power service life (EFPY)

FIGURE B 3/4.4-1



Fluence Factor for use in the expression for ΔRT_{NDT}

FIGURE B 3/4.4-2

3/4.4.11 STRUCTURAL INTEGRITY

The inservice inspection and testing programs for ASME Code Class 1, 2 and 3 components ensure that the structural integrity and operational readiness of these components will be maintained at an acceptable level through the life of the plant. These programs are in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR Part 50.55a(g) except where specific written relief has been granted by the Commission pursuant to 10 CFR Part 50.55a(g)(6)(i).

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NOS. 108 AND 86 TO FACILITY OPERATING

LICENSE NOS. DPR-70 AND DPR-75

PUBLIC SERVICE ELECTRIC & GAS COMPANY

PHILADELPHIA ELECTRIC COMPANY

DELMARVA POWER AND LIGHT COMPANY

ATLANTIC CITY ELECTRIC COMPANY

SALEM GENERATING STATION, UNIT NOS. 1 AND 2

DOCKET NOS. 50-272 AND 50-311

1.0 INTRODUCTION

In response to Generic Letter 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Effect on Plant Operations," the Public Service Electric and Gas Company requested permission to revise the pressure/temperature (P/T) limits in the Salem Nuclear Generating Plant Technical Specifications. The request was documented in a letter from the licensee dated December 28, 1988 and supplemented in letters dated July 31, 1989, October 18, 1989 and December 19, 1989. The supplemental letters dated July 31, 1989 and October 18, 1989, provided additional technical information and revised technical specification pages based on the results of the technical review. The supplemental letter dated December 19, 1989 provided corrected technical specification pages only and did not change the scope of the amendment request or affect the staff's original no significant hazards determination. The purpose of the revision is to change the effectiveness of the P/T limits to 15 effective full power years (EFPY) for Salem Unit 1 and 10 EFPY for Unit 2. The proposed P/T limits were developed based on Regulatory Guide (RG) 1.99, Revision 2. The proposed revision provides up-to-date P/T limits for the operation of the reactor coolant system during heatup, cooldown, criticality, and hydrotest.

To evaluate the P/T limits, the staff uses the following NRC regulations and guidance: Appendices G and H of 10 CFR Part 50; the ASTM Standards and the ASME Code, which are referenced in Appendices G and H; 10 CFR 50.36(c)(2); RG 1.99, Rev. 2; Standard Review Plant (SRP) Section 5.3.2; and Generic Letter 88-11.

Each licensee authorized to operate a nuclear power reactor is required by 10 CFR 50.36 to provide Technical Specifications for the operation of the plant. In particular, 10 CFR 50.36(c)(2) requires that limiting conditions of operation be included in the Technical Specifications. The P/T limits are among the limiting conditions of operation in the Technical Specifications for all commercial nuclear plants in the U.S. Appendices G and H of 10 CFR Part 50 describe specific

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requirements for fracture toughness and reactor vessel material surveillance that must be considered in setting P/T limits. An acceptable method for constructing the P/T limits is described in SRP Section 5.3.2.

Appendix G of 10 CFR Part 50 specifies fracture toughness and testing requirements for reactor vessel materials in accordance with the ASME Code and, in particular, that the beltline materials in the surveillance capsules be tested in accordance with Appendix H of 10 CFR Part 50. Appendix H, in turn, refers to ASTM Standards. These tests define the extent of vessel embrittlement at the time of capsule withdrawal in terms of the increase in reference temperature. Appendix G also requires the licensee to predict the effects of neutron irradiation on vessel embrittlement by calculating the adjusted reference temperature (ART) and Charpy upper shelf energy (USE). Generic Letter 88-11 requested that licensees and permittees use the methods in Regulatory Guide 1.99, Revision 2, to predict the effect of neutron irradiation on reactor vessel materials. This guide defines the ART as the sum of unirradiated reference temperature, the increase in reference temperature resulting from neutron irradiation, and a margin to account for uncertainties in the prediction method.

Appendix H of 10 CFR Part 50 requires the licensee to establish a surveillance program to periodically withdraw surveillance capsules from the reactor vessel. Appendix H refers to the ASTM Standards which, in turn, require that the capsules be installed in the vessel before startup and that they contain test specimens made from plate, weld, and heat-affected-zone (HAZ) materials of the reactor beltline.

2.0 EVALUATION

A. Salem Unit 1

The staff evaluated the effect of neutron irradiation embrittlement on each beltline material in the Salem Unit 1 reactor vessel in accordance with RG 1.99, Rev. 2. The staff has determined that the limiting material at 1/4T (T = reactor beltline thickness) for 15 EFPY is the lower shell longitudinal seam weld, 3-042C. This weld uses Linde flux 1092 with heat number 34B009 and has 0.19% copper, 1.00% nickel, and an unirradiated RT_{NDT} of - 56°F (Ref. 1). However, a conservative 0.35% copper was used in the licensee's calculation of the ART. The staff also used the 0.35% copper in the calculation. At the 3/4T location, the limiting material is the intermediate shell plate, B2402-1. The plate has 0.22% copper, 0.53% nickel, and an unirradiated RT_{NDT} of 45°F.

The licensee removed three surveillance capsules T, Y, and Z from Unit 1 and the results of capsule analysis were reported in References 2, 3, and 4. The limiting material, the longitudinal weld, was not included in any of the three capsules but the plate, B2402-1, was included in the T and Z capsules. Therefore, the ART of the weld was calculated based on Section 1.1 of RG 1.99 and the ART of the plate was calculated based on Section 2.1. The staff as well as the licensee calculated

an ART of 222.5°F for the longitudinal weld at 1/4T for the proposed 15 EFPY P/T limits. For plate B-2402-1, the licensee calculated an ART of 162°F and the staff, 145°F, at the 3/4T location. The difference between the two ARTs is that the licensee used a conservative margin.

Substituting the ART of 222.5°F and 162°F into equations in SRP 5.3.2, the staff verified that the proposed Unit 1 P/T limits for heatup, cooldown, and hydrotest meet the beltline material requirements in Appendix G of 10 CFR Part 50.

Because the ART of 222.5°F exceeds the Appendix G allowable of 200°F, Section IV.B of Appendix G states that the reactor vessel must be designed to permit a thermal annealing treatment at a sufficiently high temperature to recover material toughness properties of ferritic materials of the reactor vessel beltline. The Salem Units 1 and 2 reactor vessels are designed to permit a thermal annealing treatment as stated in Section 5.4.3.1 of the Salem Updated Final Safety Analysis Report; therefore, the licensee satisfies Section IV.B of Appendix B.

In addition to beltline materials, Appendix G of 10 CFR Part 50 also imposes P/T limits based on the reference temperature for the reactor vessel closure flange materials. Section IV.2 of Appendix G states that when the pressure exceeds 20% of the preservice system hydrostatic test pressure, the temperature of the closure flange regions highly stressed by the bolt preload must exceed the reference temperature of the material in those regions by at least 120°F for normal operation and by 90°F for hydrostatic pressure tests and leak tests. Based on the flange reference temperature of 28°F, the staff has determined that the proposed P/T limits satisfy Section IV.2 of Appendix G.

Section IV.B of Appendix G requires that the predicted Charpy USE at end of life be above 50 ft-lb. The Plate B2402-1 has the lowest unirradiated USE of 73 ft-lb. The percentage of USE reduction due to irradiation was calculated based on surveillance capsule results as allowed by RG 1.99. The staff used the irradiated USE data of the B2404-1 plate to calculated an end of life USE of 57 ft-lb. This value is greater than 50 ft-lb and, therefore, satisfies the Appendix G requirement.

B. Salem Unit 2

For Salem Unit 2, the staff has confirmed the licensee's calculation that the limiting material at 1/4T and 3/4T for 10 EFPY is the same longitudinal weld as in Salem Unit 1. The ART for the weld at 1/4T is 178.6°F and at 3/4T, 116.1°F, with the neutron fluence at vessel inside diameter of 4.16E18 n/cm². The licensee has removed capsules T & U from the Unit 2 reactor (Reference 5&6); however, the surveillance data of the plate and weld specimens were not used because they were not limiting. Substituting both ARTs into equations in SRP 5.3.2, the staff verified that the proposed Unit 2 P/T limits for heat up, cooldown, and hydrotest meet the beltline material requirements in Appendix G of 10 CFR Part 50.

The Unit 2 closure flange has the same RT_{NDT} of 28°F as that of Unit 1. Based on the RT_{NDT}, the staff determined that the proposed P/T limits satisfy Section IV.2 of Appendix G. The staff also determined that the USEs in beltline materials satisfy the 50 ft-lb requirement. This was based on the lowest initial USE of 69 ft-lb of the plate B4711-3 and the end of life fluence of 1.7E19 n/cm².

C. CONCLUSION

The staff concludes that the proposed P/T limits for the reactor coolant system for heatup, cooldown, leak test, and criticality are valid through 15 EFPY for Unit 1 and 10 EFPY for Unit 2 because the limits conform to the requirements of Appendices G and H of 10 CFR Part 50. The licensee's submittal also satisfies Generic Letter 88-11, because the licensee used the method in RG 1.99, Rev. 2, to calculate the ART. Hence, the proposed P/T limits may be incorporated into the Salem Units 1 and 2 Technical Specifications.

The limiting material identified in this evaluation is different from the one in the current Salem Updated Final Safety Analysis Report; therefore, the licensee should revise the limiting material and chemistry contents of the material in Chapter 5 of the UFSAR.

Administrative changes (e.g. page numbers, paragraph numbers, typographical errors) were made, with the concurrence of the licensee, to the revised technical specification pages. These changes brought the revised technical specification pages into agreement with the marked up technical specification pages that were included with the amendment request.

D. REFERENCES

1. December 28, 1988, Letter from S. E. Miltenberger (PSE&G) to NRC Document Control Desk, Subject: Request for Amendment Salem Units 1 and 2.

2. November 7, 1988, Letter from S. LaBruna (PSE&G) to NRC Document Control Desk, Subject: Surveillance Specimen Capsule Test Report, Salem Units 1.

3. Yanichko S. E., et al, "Analysis of Capsule T from the Public Service Electric and Gas Company Salem Unit 1 Reactor Vessel Radiation Surveillance Program," Westinghouse, WCAP-9678, February 1980.

4. Boggs, R. S., et al, "Analysis of Capsule Y from the Public Service Electric and Gas Company Salem Unit 1 Reactor Vessel Radiation Surveillance Program," Westinghouse, WCAP-10694, December 1984.

5. Boggs, R. S., et al, "Analysis of Capsule T from the Public Service Electric and Gas Company Salem Unit 2 Reactor Vessel Radiation Surveillance Program," Westinghouse, WCAP-10492, March 1984.

6. Yanichko, S. E., et al, "Analysis of Capsule U from the Public Service Electric and Gas Company Salem Unit 2 Reactor Vessel Radiation Surveillance Program," Westinghouse, WCAP-11554, September 1987.

3.0 ENVIRONMENTAL CONSIDERATION

These amendments involve a change to a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes to the surveillance requirements. The staff has determined that the amendments involve 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 the amendments involve no significant hazards consideration and there has been no public comment on such finding. Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 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 the amendments.

4.0 CONCLUSION

The Commission made a proposed determination that the amendments involve no significant hazards consideration which was published in the Federal Register (54 FR 51258) on December 13, 1989. No public comments were received. The State of New Jersey notified the NRC that they did not have any comments by letter dated May 1, 1989.

The staff has 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 the amendments will not be inimical to the common defense and security nor to the health and safety of the public.

Principal Contributor: J. Tsao

Dated: January 29, 1990