(1) <u>Maximum Power Level</u>

PSEG Nuclear LLC is authorized to operate the facility at a steady state reactor core power level not in excess of 3459 megawatts (one hundred percent of rated core power).

(2) <u>Technical Specifications</u>

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 243 are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

(3) Deleted Per Amendment 22, 11-20-79

(4) <u>Less than Four Loop Operation</u>

PSEG Nuclear LLC shall not operate the reactor at power levels above P-7 (as defined in Table 3.3-1 of Specification 3.3.1.1 of Appendix A to this license) with less than four (4) reactor coolant loops in operation until safety analyses for less than four loop operation have been submitted by the licensees and approval for less than four loop operation at power levels above P-7 has been granted by the Commission by Amendment of this license.

(5) PSEG Nuclear LLC shall implement and maintain in effect all provisions of the approved fire protection program as described in the Updated Final Safety Analysis Report, and as approved in the NRC Safety Evaluation Report dated November 20, 1979, and in its supplements, subject to the following provision:

> PSEG Nuclear LLC may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

PHYSICS TESTS

1.20 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and 1) described in Chapter 14 of the Updated FSAR, 2) authorized under the provisions of 10CFR50.59, or 3) otherwise by the Commission.

PRESSURE BOUNDARY LEAKAGE

1.21 PRESSURE BOUNDARY LEAKAGE shall be leakage (except steam generator tube leakage) through a non-isolable fault in a Reactor Coolant System component body, pipe wall or vessel wall.

PROCESS CONTROL PROGRAM (PCP)

1.22 The PROCESS CONTROL PROGRAM (PCP) shall contain the current formulas, sampling, analyses, test, and determinations to be made to ensure that processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Parts 20, 61, and 71, State regulations, burial ground requirements, and other requirements governing the disposal of radioactive waste.

PURGE - PURGING

1.23 PURGE or PURGING shall be the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration, or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

QUADRANT POWER TILT RATIO

1.24 QUADRANT POWER TILT RATIO shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater. With one excore detector inoperable, the remaining three detectors shall be used for computing the average.

RATED THERMAL POWER

1.25 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3459 MWt.

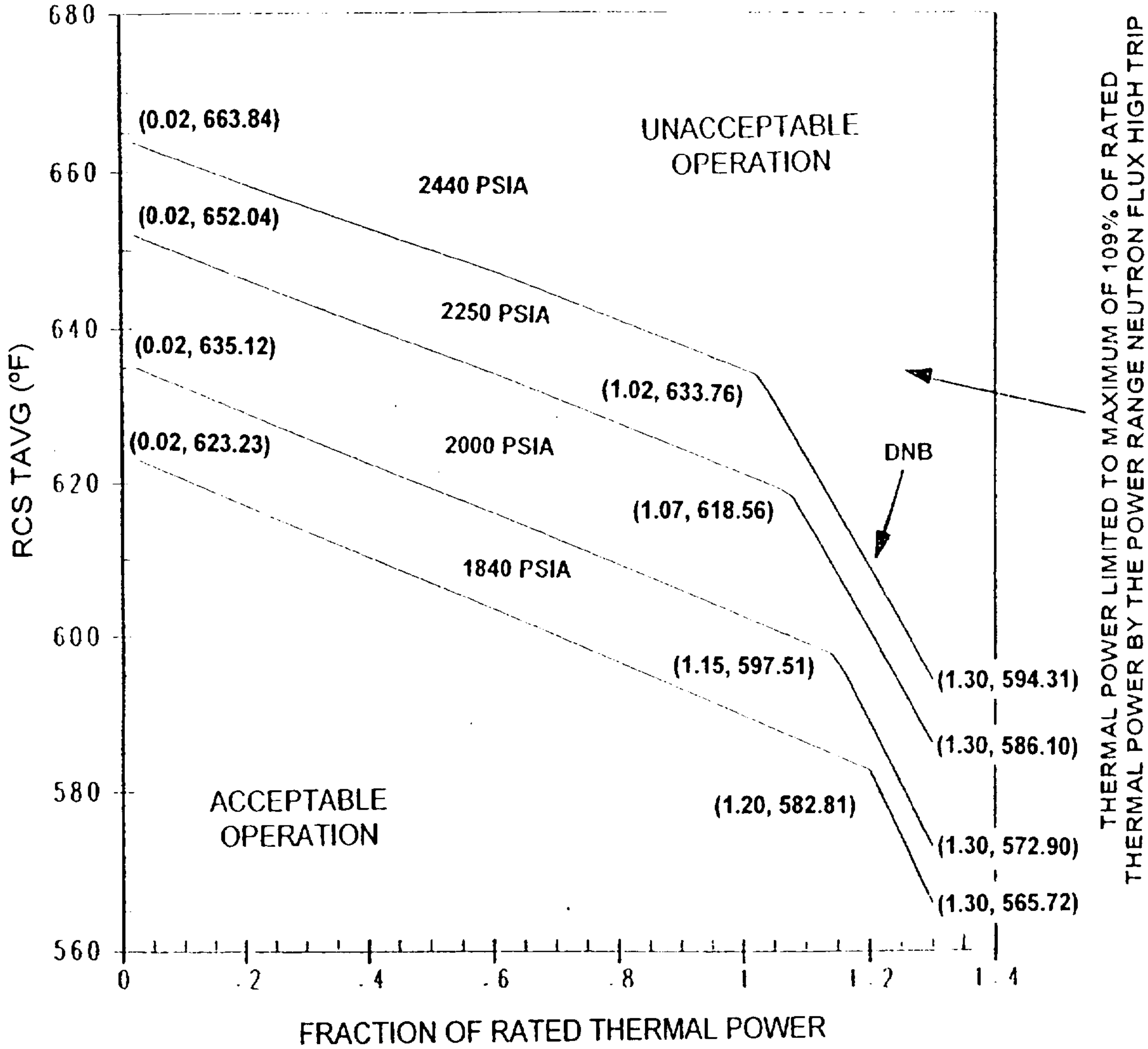

FIGURE 2.1-1
REACTOR CORE SAFETY LIMIT - FOUR LOOPS IN OPERATION

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

Operation with 4 Loops

K1 = 1.22
K2 = 0.02037
K3 = 0.001020

and f_1 (ΔI) is a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

(i) for q_t - q_b between -33 percent and +11 percent, f_1 (ΔI) = 0 (where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and q_t + q_b is total THERMAL POWER in percent of RATED THERMAL POWER).

(ii) for each percent that the magnitude of (q_t - q_b) exceeds -33 percent, the ΔT trip setpoint shall be automatically reduced by 2.34 percent of its value at RATED THERMAL POWER.

(iii) for each percent that the magnitude of (q_t - q_b) exceeds +11 percent, the ΔT trip setpoint shall be automatically reduced by 2.37 percent of its value at RATED THERMAL POWER.

Limiting Material Property		
	@1/4T	@3/4T
	Weld 3-042C	Plate B2402-1
Initial RT_{NDT}	-56°F	45°F
RT_{NDT} after 32 EFPY	232°F	171°F

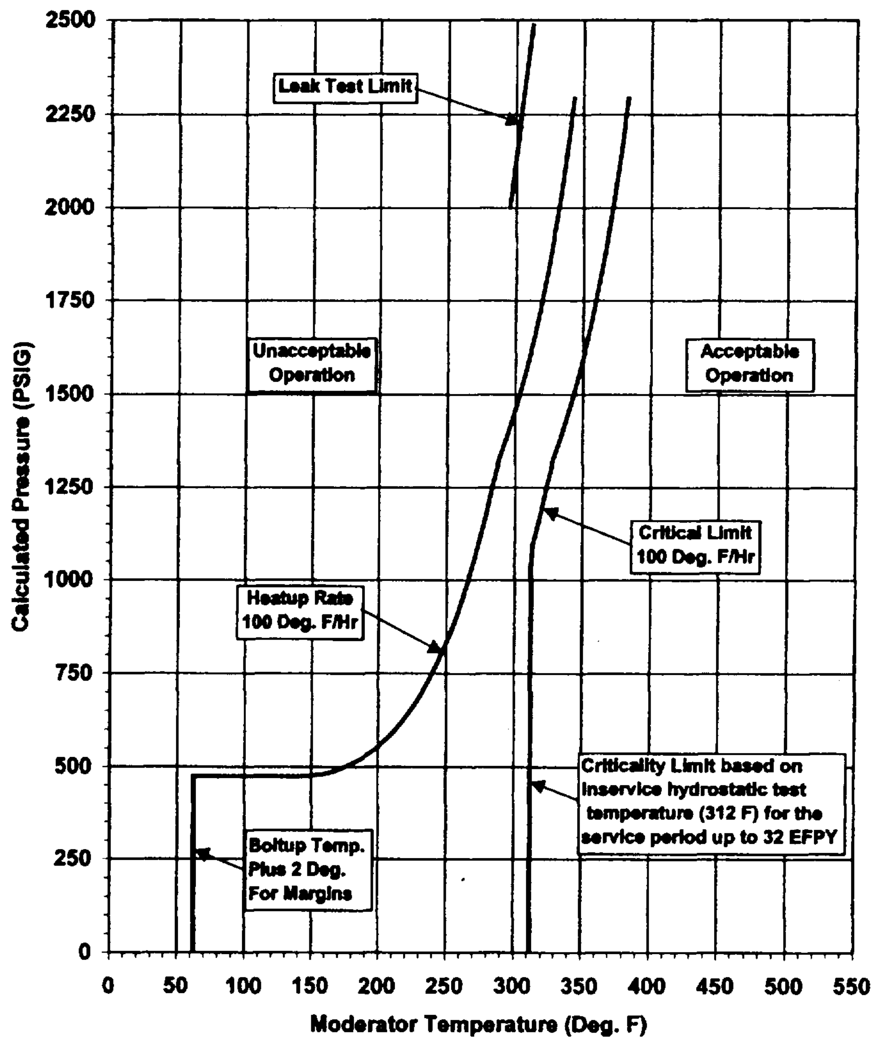

Figure 3.4-2 Salem Unit 1 Reactor Coolant System Heatup Limitations Applicable for Heatup Rates up to 100°F/HR for the Service Period up to 32 EFPY (with uncertainties for instrumentation errors).

Limiting Material Property

	@1/4T	@3/4T
	Weld 3-042C	Plate B2402-1
Initial RT_{NDT}	-56°F	45°F
RT_{NDT} after 32 EFPY	232°F	171°F

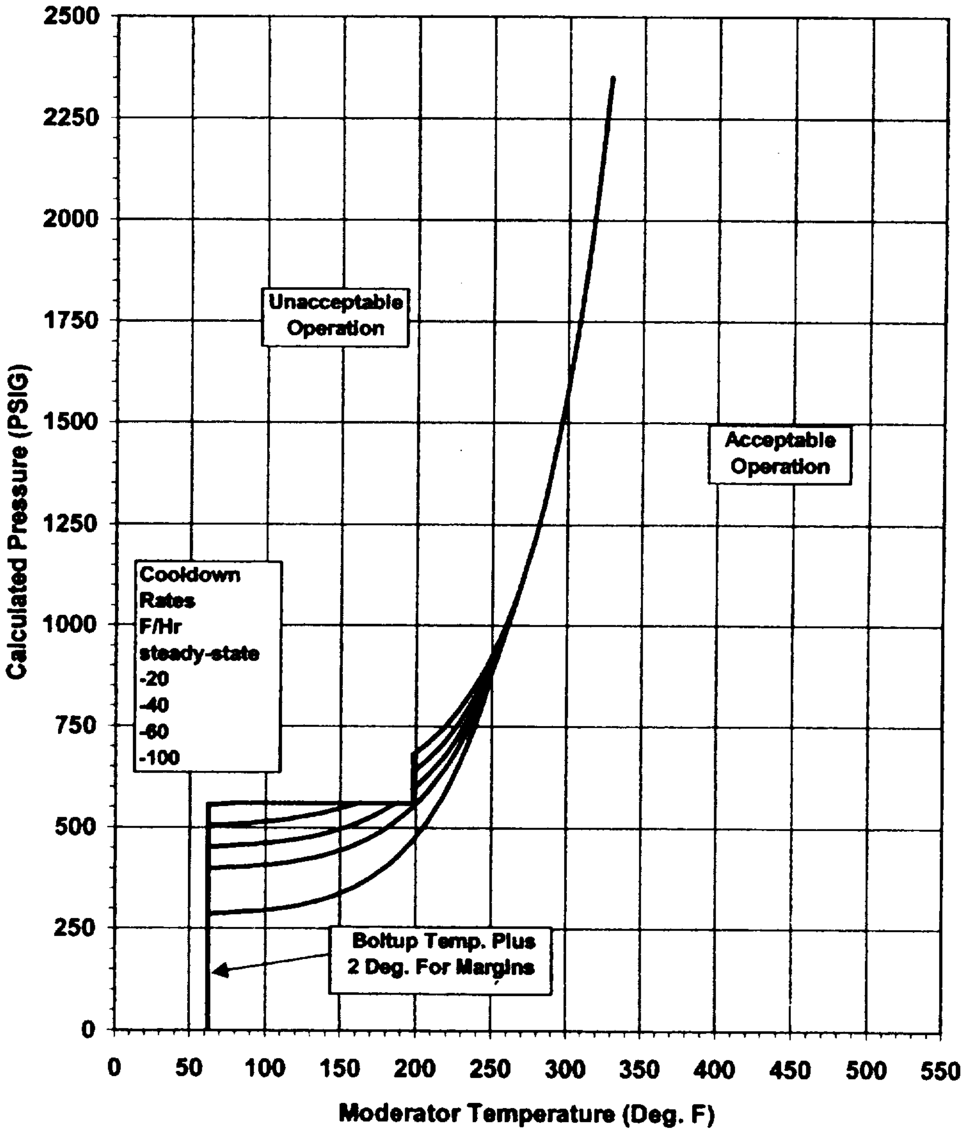

Figure 3.4-3 Salem Unit 1 Reactor Coolant System Cooldown Limitations Applicable for Cooldown Rates up to 100°F/HR for the Service Period up to 32 EFPY (with uncertainties for instrumentation errors)

TABLE 3.7-1

MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT
WITH INOPERABLE STEAM LINE SAFETY VALVES DURING 4 LOOP OPERATION

Maximum Number of Inoperable Safety Valves on Any Operating Steam Generator	Maximum Allowable Power Range Neutron Flux High Setpoint (Percent of RATED THERMAL POWER)
1	85
2	63
3	41

TABLE 3.7-2

MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT
WITH INOPERABLE STEAM LINE SAFETY VALVES DURING 3 LOOP OPERATION

Maximum Number of Inoperable Safety Valves on Any Operating Steam Generator*	Maximum Allowable Power Range Neutron Flux High Setpoint (Percent of RATED THERMAL POWER)
1	59
2	43
3	28

* At least two safety valves shall be OPERABLE on the non-operating steam generator.

2. WCAP-8385, Power Distribution Control and Load Following Procedures - Topical Report, September 1974 (W Proprietary) Methodology for Specification 3/4.2.1 Axial Flux Difference. Approved by Safety Evaluation dated January 31, 1978.

3. WCAP-10054-P-A, Rev. 1, Westinghouse Small Break ECCS Evaluation Model Using NOTRUMP Code, August 1985 (W Proprietary), Methodology for Specification 3/4.2.2 Heat Flux Hot Channel Factor. Approved for Salem by NRC letter dated August 25, 1993.

4. WCAP-10266-P-A, Rev. 2, The 1981 Version of Westinghouse Evaluation Model Using BASH Code, Rev. 2. March 1987 (W Proprietary) Methodology for Specification 3/4.2.2 Heat Flux Hot Channel Factor. Approved by Safety Evaluation dated November 13, 1986.

5. WCAP-12472-P-A, BEACON - Core Monitoring and Operations Support System, Revision 0, (W Proprietary). Approved February 1994.

6. CENPD-397-P-A, Rev. 1, Improved Flow Measurement Accuracy Using Crossflow Ultrasonic Flow Measurement Technology, May 2000.

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any mid-cycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the U.S. Nuclear Regulatory Commission, Document Control Desk, Washington, D.C. 20555, with a copy to the Administrator, USNRC Region I within the time period specified for each report.

6.9.3 Violations of the requirements of the fire protection program described in the Updated Final Safety Analysis Report which would have adversely affected the ability to achieve and maintain safe shutdown in the event of a fire shall be submitted to the U. S. Nuclear Regulatory Commission, Document Control Desk, Washington, DC 20555, with a copy to the Regional Administrator of the Regional Office of the NRC via the Licensee Event Report System within 30 days.

6.9.4 When a report is required by ACTION 8 or 9 of Table 3.3-11 "Accident Monitoring Instrumentation", a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring for inadequate core cooling, the cause of the inoperability, and the plans and schedule for restoring the instrument channels to OPERABLE status.

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 XI, 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 1996 Summer Addenda to Section XI of the ASME Boiler and Pressure Vessel Code and the calculation methods described in WCAP-14040-NP-A, Rev. 2, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves", January 1996, and ASME Boiler and Pressure Vessel Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves for Section XI, Division 1", approved March 1999.

Heatup and cooldown limit curves are calculated using the most limiting value of the nil-ductility reference temperature, RT_{NDT}, at the end of 32 effective full power years of service life. The 32 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:

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

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 B3/4.4-1, 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 ΔRT_{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 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 32 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 XI 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-14040-NP-A, Rev. 2, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves", January 1996, and ASME Boiler and Pressure Vessel Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves for Section XI, Division 1", approved March 1999.

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 XI 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_{IC}, for the metal temperature at that time. K_{IC} is obtained from the reference fracture toughness curve, defined in ASME Code Case N-640. The K_{IC} curve is given by the equation:

$$K_{IC} = 33.2 + 20.734 \exp [0.02(T-RT_{NDT})] \qquad (1)$$

where K_{IC} 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:

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

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_{IC} 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_{IC} 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 temperature 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_{IC} at the 1/4T location for finite cooldown rates than for steady-state operation. Furthermore, if conditions exist such that the increase in K_{IC} 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_{IC} for the 1/4T crack during heatup is lower than the K_{IC} 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_{IC}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 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 60°F occurs in the vessel flange of Salem Unit 1, and the minimum allowable temperature of this region is 180°F at pressures greater than 621 psig. These limits are incorporated into 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 POPS 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 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 an intermediate head safety injection pump and its injection into a water solid RCS, or the start of a high head safety injection pump in conjunction with a running positive displacement 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	0.53	-30	105	45	91	97
Inter Shell	B2402-2	A533B, Cl.1	0.24	0.53	-30	55	-5	98	112
Inter Shell	B2402-3	A533B, Cl.1	0.22	0.51	-40	57	-3	104	127
Lower Shell	B2403-1	A533B, Cl.1	0.19	0.48	-40	70	4	93	143
Lower Shell	B2403-2	A533B, Cl.1	0.19	0.49	-70	86	18	83	128
Lower Shell	B2403-3	A533B, Cl.1	0.19	0.48	-40	90	6	85	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
Circum Weld Bet Nozzle Shell & Int. Shell	8-042	-	0.22	1.02	-	-	-56***	-	-
Circum Weld Bet. Int. and Lower Shell	9-042	-	0.22	0.73	-	-	-56***	112	-
Int. Shell Vertical Weld	2-042 [A,B,C]	-	0.18	1.04	-	-	-56***	96.2	-
Lower Shell Vertical Weld	3-042 [A,B,C]	-	0.19	1.04	-	-	-56***	112	-

* Estimated per NRC Standard Review Plan Section 5.3.2.
*** Estimated per Pressurized Thermal Shock Rule, 10 CFR 50.61

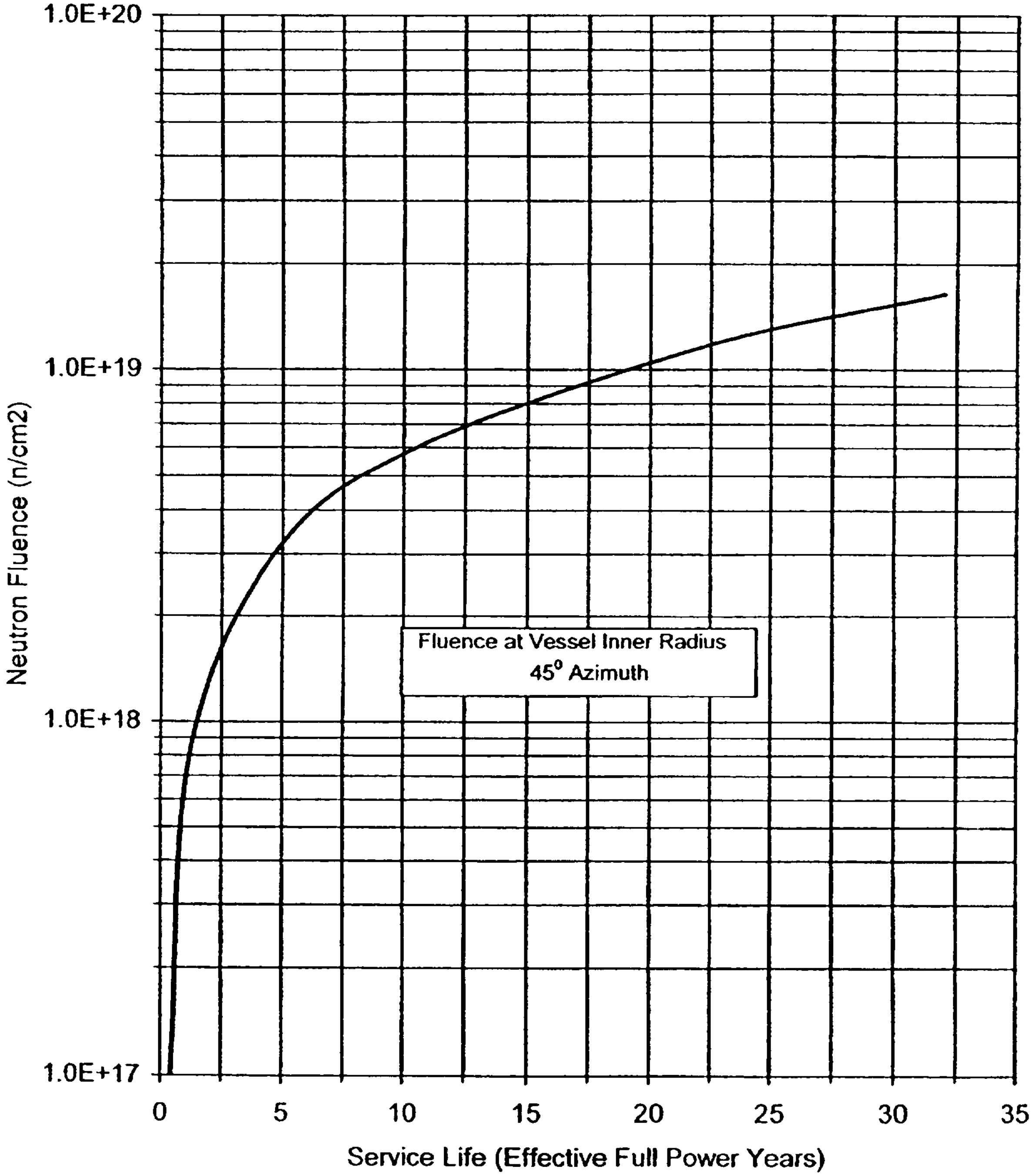

Figure B 3/4.4-1 Fast neutron fluence (E>1MeV) as a function of full power service life (EFPY)

3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES

The OPERABILITY of the main steam line code safety valves ensures that the secondary system pressure will be limited to within 110% of its design pressure of 1085 psig during the most severe anticipated system operational transient. The maximum relieving capacity is associated with a turbine trip from 100% RATED THERMAL POWER coincident with an assumed loss of condenser heat sink (i.e., no steam bypass to the condenser).

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Code, 1971 Edition. The total relieving capacity for all valves on all of the steam lines is 16.66×10^6 lbs/hr which is 110.3 percent of the maximum calculated steam flow of 15.10×10^6 lbs/hr at 100% RATED THERMAL POWER. A minimum of 2 OPERABLE safety valves per OPERABLE steam generator ensures that sufficient relieving capacity is available for the allowable THERMAL POWER restriction in Table 3.7-2.

STARTUP and/or POWER OPERATION is allowable with safety valves inoperable within the limitations of the ACTION requirements on the basis of the reduction in secondary system steam flow and THERMAL POWER required by the reduced reactor trip settings of the Power Range Neutron Flux channels. The reactor trip setpoint reductions are derived on the following bases:

For 4 loop operation

$$SP = \frac{(X) - (Y)(V)}{X} \times (109)$$

For 3 loop operation

$$SP = \frac{(X) - (Y)(U)}{X} \times (76)$$

Where:

SP = reduced reactor trip setpoint in percent of RATED THERMAL POWER

V = maximum number of inoperable safety valves per steam line

(2) PSEG Nuclear LLC, pursuant to Section 104b of the Act and 10 CFR part 50, "Domestic Licensing of Production and Utilization Facilities," to possess, use and operate the facility at the designated location in Salem County, New Jersey, in accordance with the limitations set forth in this license;

(3) PSEG Nuclear LLC, pursuant to the Act and 10 CFR Part 70, to receive, possess and use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Final Safety Analysis Report, as supplemented and amended;

(4) PSEG Nuclear LLC, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess and use at any time any byproduct, source or special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration and as fission detectors in amounts as required;

(5) PSEG Nuclear LLC, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(6) PSEG Nuclear LLC, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

PSEG Nuclear LLC is authorized to operate the facility at steady state reactor core power levels not in excess of 3459 megawatts (thermal).

PHYSICS TESTS

1.20 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and 1) described in Chapter 14 of the Updated FSAR, 2) authorized under the provisions of 10CFR50.59, or 3) otherwise by the Commission.

PRESSURE BOUNDARY LEAKAGE

1.21 PRESSURE BOUNDARY LEAKAGE shall be leakage (except steam generator tube leakage) through a non-isolable fault in a Reactor Coolant System component body, pipe wall or vessel wall.

PROCESS CONTROL PROGRAM (PCP)

1.22 The PROCESS CONTROL PROGRAM (PCP) shall contain the current formulas, sampling, analyses, test, and determinations to be made to ensure that processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Parts 20, 61, and 71, State regulations, burial ground requirements, and other requirements governing the disposal of radioactive waste.

PURGE - PURGING

1.23 PURGE or PURGING shall be the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration, or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

QUADRANT POWER TILT RATIO

1.24 QUADRANT POWER TILT RATIO shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater. With one excore detector inoperable, the remaining three detectors shall be used for computing the average.

RATED THERMAL POWER

1.25 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3459 MWt.

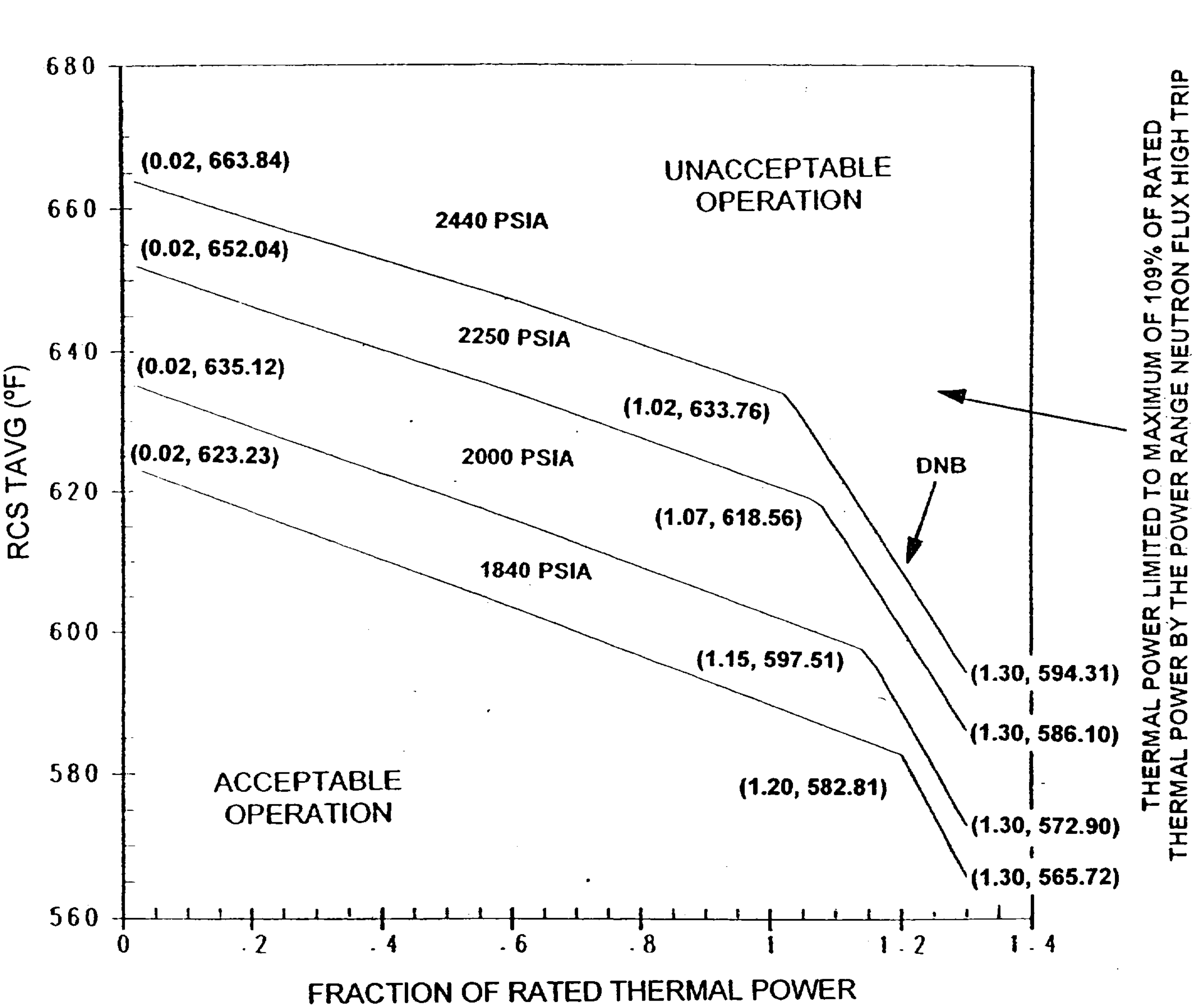

FIGURE 2.1-1 REACTOR CORE SAFETY LIMIT - FOUR LOOPS IN OPERATION

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

Operation with 4 Loops

K_1 = 1.22
K_2 = 0.02037
K_3 = 0.001020

and f_1 (ΔI) is a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

(i) for q_t - q_b between -33 percent and +11 percent, f_1 (ΔI) = 0 where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and q_t + q_b is total THERMAL POWER in percent of RATED THERMAL POWER).

(ii) for each percent that the magnitude of (q_t - q_b) exceeds -33 percent, the ΔT trip setpoint shall be automatically reduced by 2.34 percent of its value at RATED THERMAL POWER.

(iii) for each percent that the magnitude of (q_t - q_b) exceeds +11 percent, the ΔT trip setpoint shall be automatically reduced by 2.37 percent of its value at RATED THERMAL POWER.

Limiting Material Property Weld 3-442 A&C Initial RT_{NDT} -56°C RT_{NDT} after 32 EFPY: 1/4T 199°F 3/4T 140°F

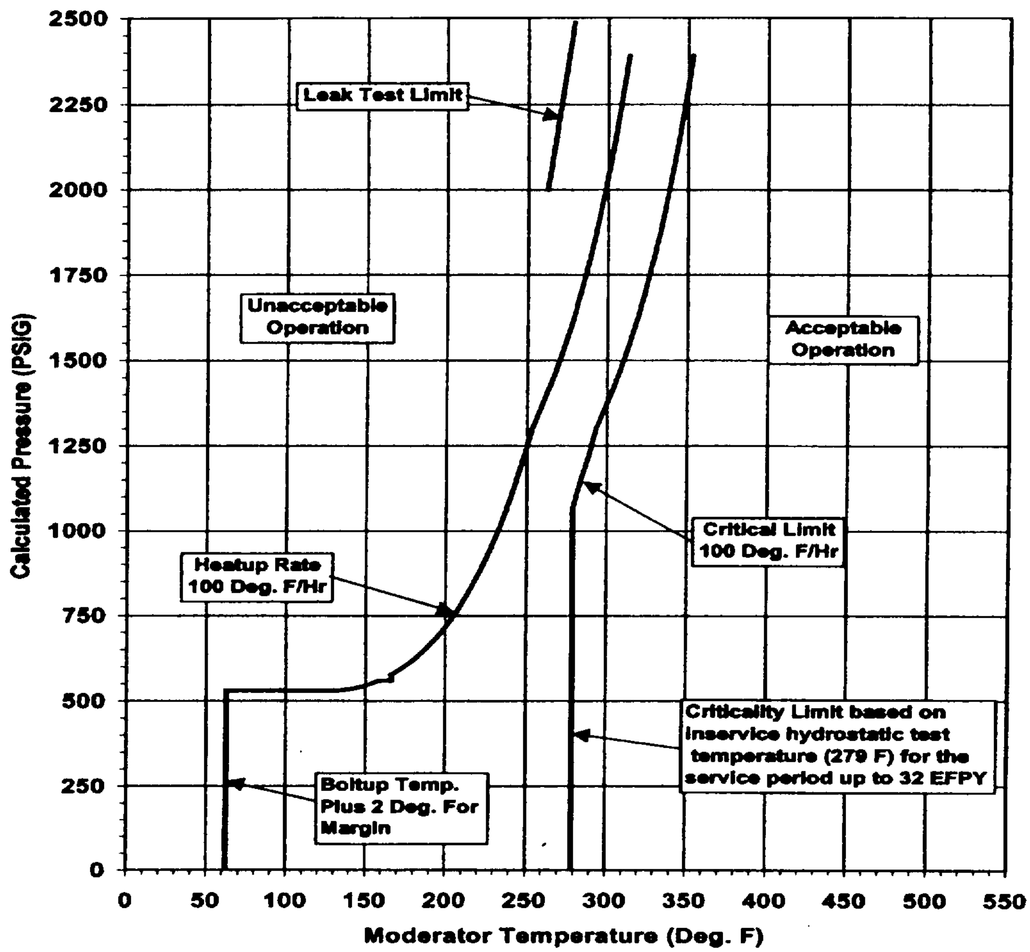

Figure 3.4-2

SALEM UNIT 2 REACTOR COOLANT SYSTEM HEATUP LIMITATIONS APPLICABLE FOR THE FIRST 32 EFPY WITH MAXIMUM HEATUP RATE OF 100 F/HR. CURVE CONTAINS MARGIN FOR INSTRUMENT ERRORS

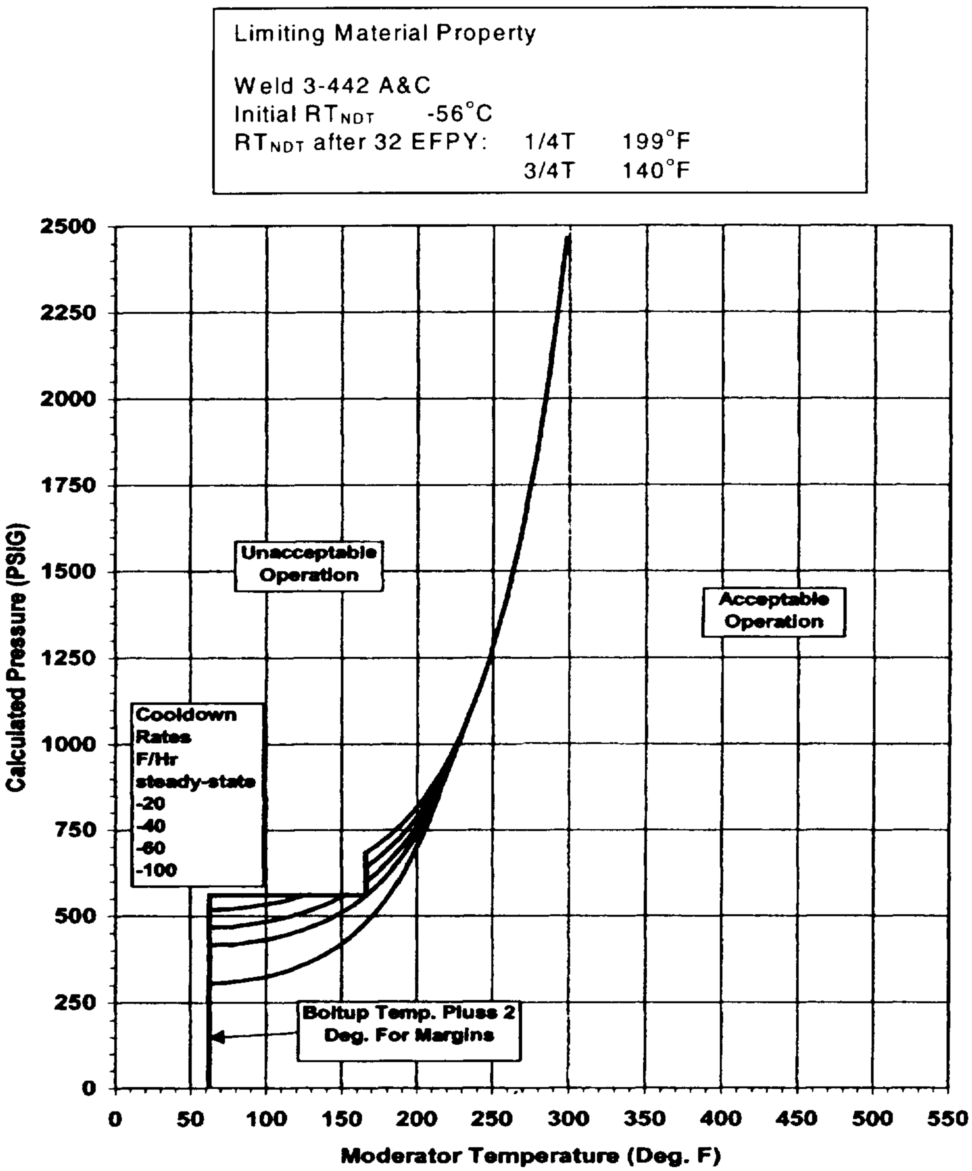

Figure 3.4-3

SALEM UNIT 2 REACTOR COOLANT SYSTEM COOLDOWN LIMITATIONS APPLICABLE FOR THE FIRST 32 EFPY WITH MAXIMUM COOLDOWN RATE OF 100 F/HR. CURVE CONTAINS MARGIN FOR INSTRUMENT ERRORS.

TABLE 3.7-1

MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES DURING 4 LOOP OPERATION

Maximum Number of Inoperable Safety Valves on Any Operating Steam Generator	Maximum Allowable Power Range Neutron Flux High Setpoint (Percent of RATED THERMAL POWER)
1	85
2	63
3	41

2. WCAP-8385, Power Distribution Control and Load Following Procedures - Topical Report, September 1974 (W Proprietary) Methodology for Specification 3/4.2.1 Axial Flux Difference Approved by Safety Evaluation dated January 31, 1978.

3. WCAP-10054-P-A, Rev. 1, Westinghouse Small Break ECCS Evaluation Model Using NOTRUMP Code, August 1985 (W Proprietary), Methodology for Specification 3/4.2.2 Heat Flux Hot Channel Factor. Approved for Salem by NRC letter dated August 25, 1993.

4. WCAP-10266-P-A, Rev. 2, The 1981 Version of Westinghouse Evaluation Model Using BASH Code, Rev. 2. March 1987 (W Proprietary) Methodology for Specification 3/4.2.2 Heat Flux Hot Channel Factor. Approved by Safety Evaluation dated November 13, 1986.

5. WCAP-12472-P-A, BEACON - Core Monitoring and Operations Support System, Revision 0, (W Proprietary). Approved February 1994.

6. CENPD-397-P-A, Rev. 1, Improved Flow Measurement Accuracy Using Crossflow Ultrasonic Flow Measurement Technology, May 2000

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any mid-cycle revisions or supplements shall be provided upon issuance for each reload cycle to the NRC.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the U.S. Nuclear Regulatory Commission, Document Control Desk, Washington, D.C. 20555, with a copy to the Administrator, USNRC Region I within the time period specified for each report.

6.9.3 Violations of the requirements of the fire protection program described in the Updated Final Safety Analysis Report which would have adversely affected the ability to achieve and maintain safe shutdown in the event of a fire shall be submitted to the U. S. Nuclear Regulatory Commission, Document Control Desk, Washington, DC 20555, with a copy to the Regional Administrator of the Regional Office of the NRC via the Licensee Event Report System within 30 days.

6.9.4 When a report is required by ACTION 8 OR 9 of Table 3.3-11 "Accident Monitoring Instrumentation", a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring for inadequate core cooling, the cause of the inoperability, and the plans and schedule for restoring the instrument channels to OPERABLE status.

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 XI, 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 1996 Summer Addenda to Section XI of the ASME Boiler and Pressure Vessel Code and the calculation methods described in WCAP-14040-NP-A, Rev. 2, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves", January 1996, and ASME Boiler and Pressure Vessel Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves for Section XI, Division 1", approved March 1999.

Heatup and cooldown limit curves are calculated using the most limiting value of the nil-ductility reference temperature, RT_{NDT}, at the end of 32 effective full power years of service life. The 32 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 B3/4.4-1, 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 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°F for welds and 17°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 32 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 XI 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-14040-NP-A, Rev. 2, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves", January 1996, and ASME Boiler and Pressure Vessel Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves for Section XI, Division 1", approved March 1999.

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 XI 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_{IC}, for the metal temperature at that time. K_{IC} is obtained from the reference fracture toughness curve, defined in ASME Code Case N-640. The K_{IC} curve is given by the equation:

$$K_{IC} = 33.2 + 20.734 \exp\left[0.02(T-RT_{NDT})\right] \qquad (1)$$

where K_{IC} 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:

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

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_{IC} 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_{IC} 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 temperature 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_{IC} at the 1/4T location for finite cooldown rates than for steady-state operation. Furthermore, if conditions exist such that the increase in K_{IC} 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_{IC} for the 1/4T crack during heatup is lower than the K_{IC} 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_{IC}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 are incorporated into 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 protected 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 an Intermediate Head Safety Injection pump and its injection into a water solid RCS, or the start of a High Head Safety Injection pump in conjunction with a running Positive Displacement 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 Shell Energy Normal to Principal Working Direction (ft-lb)	Average Upper Shell 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	106	138
Inter. Shell	B4712-2	A533BCL1	0.12	0.62	-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	121	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
Circum. Weld Bet Nozzle Shell & Int. Shell	8-442	-	0.28	0.74	-	-	-56***	-	-
Circum. Weld Bet Int. Shell & Lower Shell	9-442	-	0.197	0.060	-	-	-56***	99.7	-
Int. Shell Vertical Weld	2-442 [A,B,C]	-	0.219	0.735	-	-	-56***	96.2	-
Lower Shell Vertical Weld	3-442 [A,B,C]	-	0.213	0.867	-	-	-56***	114	-

* Estimated per NRC Standard Review Plan Section 5.8.2.
** 100% Shear not reached
*** Estimate per Pressurized Thermal Shock Rule, 10 CFR 50.61

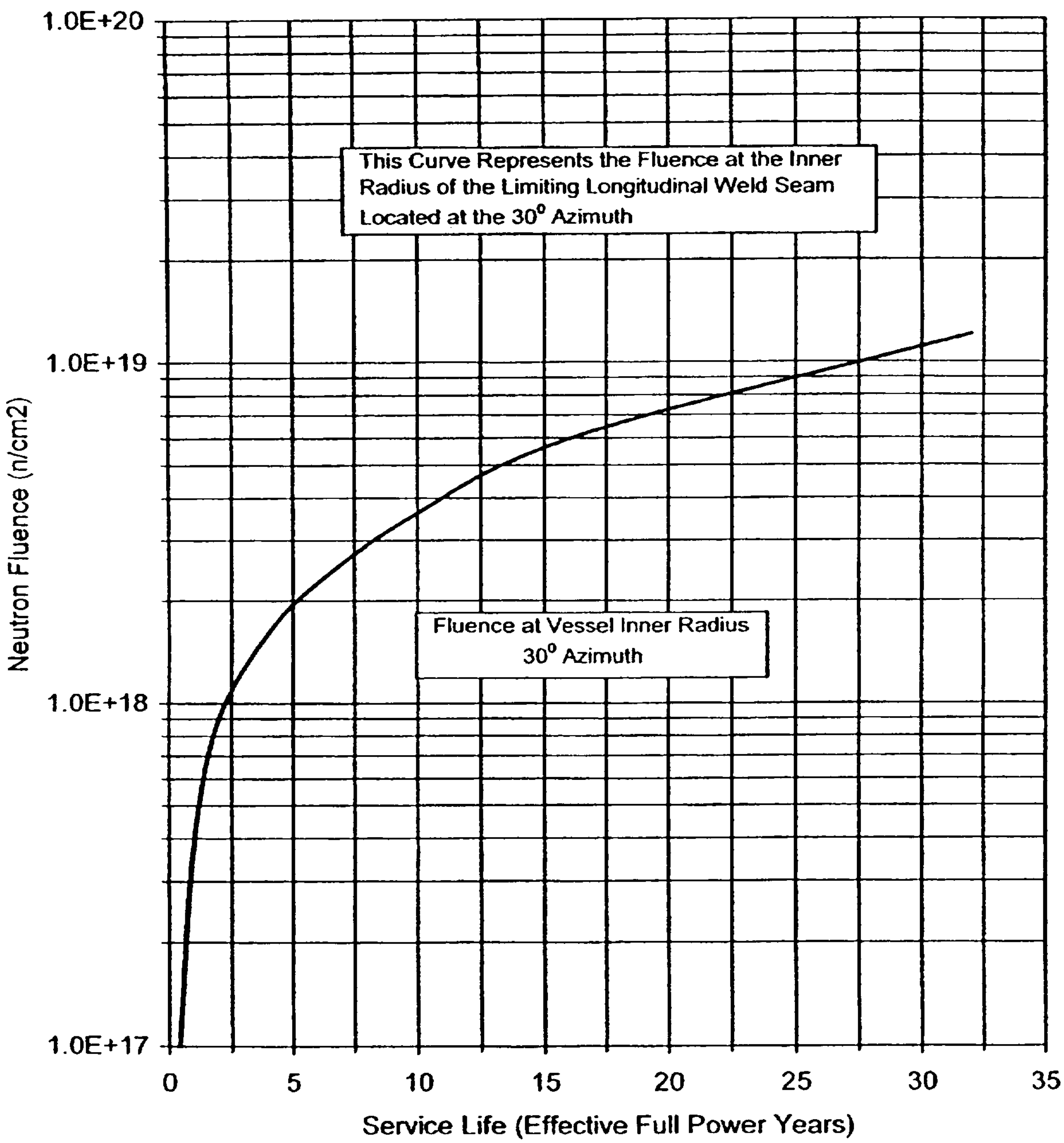

Figure B 3/4.4-1 Fast neutron fluence (E > 1 MeV) as a function of full power service life (EFPY)

3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES

The OPERABILITY of the main steam line code safety valves ensures that the secondary system pressure will be limited to within 110% of its design pressure of 1085 psig during the most severe anticipated system operational transient. The maximum relieving capacity is associated with a turbine trip from 100% RATED THERMAL POWER coincident with an assumed loss of condenser heat sink (i.e., no steam bypass to the condenser).

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Code, 1971 Edition. The total relieving capacity for all valves on all of the steam lines is 16.66 x 10^6 lbs/hr which is 110.4% of the maximum calculated steam flow of 15.08 x 10^6 lbs/hr at 100% RATED THERMAL POWER. A minimum of 2 OPERABLE safety valves per OPERABLE steam generator ensures that sufficient relieving capacity is available for the allowable THERMAL POWER restriction in Table 3.7-2.

STARTUP and/or POWER OPERATION is allowable with safety valves inoperable within the limitations of the ACTION requirements on the basis of the reduction in secondary system steam flow and THERMAL POWER required by the reduced reactor trip settings of the Power Range Neutron Flux channels. The reactor trip setpoint reductions are derived on the following bases:

For 4 loop operation

$$SP = \frac{(X) - (Y)(V)}{X} \times (109)$$

For 3 loop operation

$$SP = \frac{(X) - (Y)(U)}{X} \times (76)$$

Where:

SP = reduced reactor trip setpoint in percent of RATED THERMAL POWER

V = maximum number of inoperable safety valves per steam line