PSEG Nuclear LLC
P.O. Box 236, Hancocks Bridge, New Jersey 08038-0236

PSEG
Nuclear LLC

MAR 2 8 2001

LRN-01-0075
LCR S00-06, Sup. 1

United States Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

Gentlemen:

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION IN REGARDS TO REQUEST FOR LICENSE AMENDMENT INCREASED LICENSED POWER LEVEL SALEM GENERATING STATION, UNIT NOS. 1 AND 2 FACILITY OPERATING LICENSE DPR-70 AND DPR-75 DOCKET NOS. 50-272 AND 50-311

On March 8, 2001, the NRC issued a request for additional information (RAI) to support the staff's review of the request for license amendment submitted by PSEG Nuclear LLC on November 10, 2000 requesting an increase in licensed power levels for Salem Generating Station Unit Nos. 1 and 2.

The response to the request for additional information is contained in Attachment 1 including the non-proprietary version of the response to question 7(a). The proprietary version of the response to question 7(a) is contained in Attachment 2 and contains information from WCAP-15553, "Power Calorimetric for the 1.4% Uprating for Public Service Electric and Gas Company Salem Units 1 and 2," that was designated as proprietary information in the November 10, 2000 submittal. Attachment 2 of this submittal should be withheld from public disclosure in accordance with 10CFR2.790 as supported by the affidavit by Westinghouse for withholding proprietary information that is contained as Attachment 8 of the November 10, 2000 submittal.

As stated in question 2 of the NRC's RAI, the staff requests that revised pressure-temperature (P-T) limit curves that do not include the elimination of the flange requirements for Salem Unit Nos. 1 and 2 be submitted. Revised P-T limit curves that do not eliminate the flange requirements and the associated bases change revisions are contained in Attachment 3. Attachment 7 contains Revision 1 of WCAP-15565, and Attachment 8 contains Revision 1 of WCAP-15566 which document the development of the revised pressure-temperature limit curves.

APOI

ATTACHMENTS 2 and 6 OF THIS LETTER CONTAIN PROPRIETARY INFORMATION NOT FOR PUBLIC DISCLOSURE

In Attachment 6 of the November 10, 2000 submittal, PSEG Nuclear requested an exemption form 10 CFR 50 Appendix G to allow the application of WCAP-15315 to eliminate the flange requirements for Salem Unit Nos. 1 and 2, PSEG Nuclear is hereby withdrawing this exemption request.

Based upon the withdrawal of the use of WCAP-15315, the no significant hazards evaluation contained as Attachment 2 of the November 10, 2000 submittal is being revised. The revised no significant hazards evaluation is contained as Attachment 4 of this submittal with the changes marked with revision bars.

Attachment 6 of this submittal contains CE Nuclear Power LLC (CENP) Crossflow uncertainty calculation A-SA2-PS-0001, Revision 0. Attachment 5 is an application and affidavit by CE Nuclear Power LLC for withholding the proprietary document contained in Attachment 6 from public disclosure in accordance with 10CFR2.790. Should you have any questions regarding this request, please contact Mr. Brian Thomas at (856)339-2022.

Sincerely,

D.F. Garchow

D. F. Garchow
Vice President - Operations

Affidavit
Attachments (8)

C **<u>(All without Attachment 2 and 6)</u>**
Mr. H. J. Miller, Administrator - Region I
U. S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

Mr. R. Fretz
Licensing Project Manager - Salem
U. S. Nuclear Regulatory Commission
One White Flint North
Mail Stop O8B2
11555 Rockville Pike
Rockville, MD 20852

USNRC Senior Resident Inspector - Salem (X24)

Mr. K. Tosch, Manager IV
Bureau of Nuclear Engineering
P.O. Box 415
Trenton, NJ 08625

ATTACHMENTS 2 and 6 OF THIS LETTER CONTAIN PROPRIETARY INFORMATION NOT FOR PUBLIC DISCLOSURE

BJT

BC **(All without Attachment 2 and 6)**
Vice President – Technical Support
Director – Performance and Protection (120)
Nuclear Safety and Licensing Manager (N21)
Manager - Business Planning & Co-Owners Affairs (N18)
Salem Operations Manager (S01)
Nuclear Fuel and Reactor Engineering Manager (N20)
Project manager - NRB (N38)
J. Keenan, Esq. (N21)
Records Management (N21)
Microfilm Copy
File Nos. 1.2.1 (Salem)
2.3 (LCR S00-06)

REF: LRN-01-0075
LCR S00-06, Sup. 1

STATE OF NEW JERSEY)
) SS.
COUNTY OF SALEM)

D. F. Garchow, being duly sworn according to law deposes and says:

I am Vice President - Operations of PSEG Nuclear LLC, and as such, I find the matters set forth in the above referenced letter, concerning Salem Generating Station, Units 1 and 2, are true to the best of my knowledge, information and belief.

David F. Garchow

Subscribed and Sworn to before me

this 28th day of March, 2001

Vanita M. Marshall
Notary Public of New Jersey

VANITA M. MARSHALL
NOTARY PUBLIC OF NEW JERSEY
My Commission Expires June 18, 2003

My Commission expires on ____________________

ATTACHMENT 1
SALEM GENERATING STATION
UNIT NOS. 1 AND 2
FACILITY OPERATING LICENSE DPR-70 AND DPR-75
DOCKET NOS. 50-272 AND 50-311
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION
INCREASED LICENSED POWER LEVEL

On March 8, 2001, the NRC issued a request for additional information (RAI) concerning PSEG Nuclear's request for amendment to increase the licensed power level for Salem Unit Nos. 1 and 2. This attachment provides the non-proprietary responses to the RAI questions. The response to question 7(a) which contains Westinghouse proprietary information is contained in attachment 2.

NRC Question:

Section 4.2.3 Steam Generator Blowdown System

1. The submittal contains a statement that the rate of addition of dissolved solids to the secondary system, in addition to being a function of condenser leakage and the quality of secondary makeup water, also depends on the rate of erosion-corrosion within the secondary system. Although the first two sources of dissolved solids do not change with power uprate, generation of particulates by erosion-corrosion may be affected by power uprate due to a change in velocities which may occur in the secondary systems. Please provide justification that the power uprate will not significantly alter generation of particulates by erosion-corrosion.

PSEG Nuclear responseto Q1:

As stated in the November 10, 2000 submittal, Attachment 1 Section 12.5, the Flow Accelerated Corrosion (FAC) Program monitors wall thinning in single and two-phase carbon steel piping systems at Salem as required by GL 89-08. The FAC program monitors the degradation in piping systems based on industry accepted methodology. Sensitivity reviews were performed by varying the power level assumed in the existing CHECWORKS model for several BOP systems. The results of the model runs using the proposed 101.4% power level heat balance parameters indicate that any increase or decrease in particulates generated by flow-accelerated corrosion will not be significant.

NRC Question:

Section 5.2 Reactor Vessel Integrity - Neutron Irradiation

2. Regarding the information in Section 5.2, the Technical Specification changes in Attachment 4, and the exemption request in Attachment 6 to the November 10, 2000, submittal, the NRC staff has not approved the application of WCAP-15315 to remove reactor pressure vessel (RPV) head flange requirements from the Salem licensing basis. The staff was petitioned (as published in Federal Registernotice 65 FR 6044) to undertake rulemaking to modify the requirements of Appendix G to 10 CFR Part 50 as they relate to RPV flange material property issues. The staff is in the process of acting on this petition and will follow the rulemaking process. Therefore, the staff has determined that, since we have not determined the contents of the final rule, it would be inappropriate to grant plant-specific exemptions during the rulemaking process. We request that you submit revised P-T limit curves that do not include the elimination of the flange requirements for Salem, Unit Nos. 1 and 2, to replace those submitted in Attachment 4 to the November 10, 2000, submittal.

PSEG Nuclear response to Q2:

2. PSEG Nuclear is submitting revised P-T curves in Attachment 3 of this submittal incorporating the reactor vessel flange pressure-temperature requirements of Appendix G to 10 CFR Part 50. The revised curves also include the correction of a recently uncovered error in the Westinghouse OPERLIM Code, when using the methodology of the 1995 ASME Code, Section XI, Appendix G, through 1996 Addendum. This correction resulted in a 4% decrease in the pressure limits at the high temperature end of the heatup curves. Associated changes to the Technical Specification bases are also included in Attachment 3.

   With the above change to the P-T curves, PSEG Nuclear is no longer requesting an exemption from Appendix G of 10CFR50 to allow the application of WCAP-15315. Based upon the withdrawal of the use of WCAP-15315, the no significant hazards evaluation contained as Attachment 2 of the November 10, 2000 submittal is being revised. The revised no significant hazards evaluation is contained as Attachment 4 of this submittal with the changes to delete the reference to WCAP-15315 marked with revision bars.

NRC Questions:

3. Regarding the information submitted in Table 4-1 of WCAP-15565, Revision 0, the staff has compared the cited surface (which, based on other information in the WCAP, is apparently at the clad-to-base metal interface) fluence values to the values previously reported by the licensee and contained in the NRC staff's

Reactor Vessel Integrity Database (RVID). The staff noted that while most of the fluence values calculated in WCAP-15565 for post-power uprated conditions did go up, the values cited for all of the longitudinal weld seams (2-042 A, B, and C and 3-042 A, B, and C) decreased slightly.

Please explain how these numbers decreased as a result of the most recent fluence recalculations.

4. Regarding the information submitted in Table 4-1 of WCAP-15566, Revision 0, the staff has compared the cited surface (which, based on other information in the WCAP, is apparently at the clad-to-base metal interface) fluence values to the values previously reported by the licensee and contained in the NRC staff's Reactor Vessel Integrity Database (RVID). The staff noted that while most of the fluence values calculated in WCAP-15566 for post-power uprated conditions did go up, the values cited for intermediate shell longitudinal weld seam 2-442 A and lower shell longitudinal weld seam 3-442 B decreased slightly. Explain how these numbers decreased as a result of the most recent fluence recalculations.

PSEG Nuclear responses to Q3 and Q4:

3 & 4. In performing the fluence calculations for uprate conditions, there were two options available: (1) modify the existing calculations using the same fluence methods to account for increase core thermal power or (2) perform new calculations using different fluence methods. PSEG Nuclear selected the second option listed above for specific reasons noted below.

The pre-uprate Reactor Vessel Integrity Database (RVID) fluence values were generated by PSEG Nuclear using methods developed jointly by PSEG Nuclear and Framatome Technologies. This method consisted of relative-pin power distributions (RPD) being generated by PSEG Nuclear using the PDQ code. These RPDs were then used by PSEG Nuclear to generate an appropriate source term. This source term was then used in the DOT 4.3 code to calculate the flux in the vessel.

Since PSEG Nuclear now uses the Westinghouse core design methods, the uprate fluence values were generated by Westinghouse using methods developed by Westinghouse. The two methods are both valid, but they do provide different, but still valid, fluence results. If option (1) listed above had been selected for the uprate calculations, all of the weld fluences would have been expected to increase in the same relative fraction as the increase in core thermal power. Since option (2) listed above was selected, comparisons of the pre-uprate RVID fluences to the uprate fluences are not legitimate one-to-one comparisons.

The assumptions used in the Westinghouse methods were verified by PSEG Nuclear to be appropriate for use at uprated conditions. The resulting fluences were evaluated as being reasonable and appropriate with respect to the change in fluence methodology. Therefore, even though there are some decreases in weld fluences, these decreases are deemed appropriate.

NRC Question:

5. Explain whether or not a change to the Salem Unit No. 1 or Unit No. 2 low-temperature overpressure protection (LTOP) system (or Pressurizer Overpressure Protection System) pressure setpoint or enable temperature is required as a result of the recalculation of RPV material properties for 32 effective full power years (EFPY) of operation.

PSEG Nuclear response to Q5:

5. No changes to the Pressurizer Overpressure Protection System (POPS) are required as a result of any changes made in the revision of the reactor pressure vessel material properties for 32 EFPY. Please refer to Section 4.1.9 of Attachment 1 of the November 10, 2000 submittal for further information on POPS. The conclusion of Section 4.1.9 is not effected by the revised P-T curves from question 2 above.

NRC Question

Section 5.9 Steam Generators

6. In Section 5.9.5 of the power uprate submittal, PSEG Nuclear stated, without many details, that power uprate will have a negligible impact on the existing and potential tube degradation mechanisms. The NRC staff understands that the Unit No. 2 steam generators are experiencing the following active degradation: primary stress corrosion cracking in hot leg top of tubesheet transition zones, at hot leg dented tube support plate intersections, in low row U-bends, and in tube plugs; outside stress corrosion cracking in the hot leg freespan regions.

   In addition, the following degradation mechanisms have previously occurred in the Unit 2 steam generators: anti-vibration bar wear; thinning at cold leg tube support plate intersections; intergranular attack/stress corrosion cracking at hot leg top of tubesheet (sludge pile); outside diameter stress corrosion cracking at hot leg top of tubesheet and at tube support plate intersections.

   Therefore, in order to verify that General Design Criterion (GDC) No. 14, "Reactor Coolant Pressure Boundary," will continue to be met during future operating cycles at uprated conditions, please address the following:

(a) Confirm whether our understanding is correct, and that the above potential degradation mechanisms are currently active. Provide a brief discussion describing the impact that power uprate will have on each of these degradation mechanisms;

PSEG Nuclear response to Q6a:

6a. Unit 2

For Salem Unit 2 Steam Generators, per the inspection results of outage 2R11, the identified active degradation mechanisms as defined in the EPRI Rev 5 ISI guideline are as follows:

- Primary Water Stress Corrosion Cracking (PWSCC) at the hot leg tubesheet expansion transition zone,
- PWSCC at hot leg dented tube support plate intersections,
- PWSCC in low row U-bends, and
- PWSCC in Alloy 600
- Outside Diameter Stress Corrosion Cracking (ODSCC) in the hot leg sludge pile region
- Anti-Vibration Bar (AVB) wear
- Cold leg thinning

The revised design conditions will have a negligible impact upon the observed degradation mechanisms. The uprate will result in an increase in the design T-hot of 0.5°F. Stress corrosion cracking (SCC) mechanisms are known to be affected by T-hot changes. To assess the sensitivity of the 0.5°F T-hot increase, the Arrhenius Equation was used. To conservatively bound the effect of the increased T-hot, a 1.0°F increase was assumed. The postulated one degree increase in the Salem Unit 2 current T-hot value results in a predicted increase of increased by 2 to 3% (absolute) in ODSCC growth rates, and an increase of 3 to 4% (absolute) in PWSCC growth rates. These changes in growth rates have been incorporated in the Salem Unit 2 Cycle 12 Operational Assessment. Steam pressure fluctuations have a secondary effect upon SCC growth rates. The calculated steam pressure reduction due to uprating is 5 to 6 psi, and has been determined to have an insignificant effect upon operating tube stresses.

AVB wear is not affected by T-hot changes. Secondary system changes (i.e., steam pressure reduction and flow rate increases) can affect AVB wear growth rates, if the associated changes in these parameters are significant. The effects of the 1.4% uprating have been evaluated and found to have a negligible impact upon AVB wear growth rates. AVB wear is monitored during each outage, and site-specific AVB wear growth rates are calculated and are used to develop the operational assessment.

ODSCC and cold leg thinning mechanisms can also be affected by adverse chemistry conditions within the Steam generators. ODSCC mechanisms and cold leg thinning have been addressed through changes in the secondary side chemistry program. Improvements to the secondary side chemistry program are expected to balance any potential impact upon growth rates of ODSCC mechanisms due to uprating. Cold leg thinning is not expected to be affected by T-hot changes due to uprating because cold leg thinning has been associated with localized crevice chemistry conditions. These conditions have been addressed by Chemical Cleaning during 2R10 and by improvements in the secondary side chemistry control program.

Unit 1

The identified on-going degradation mechanisms for Salem Unit 1 Steam generators is AVB wear.

The original Salem Unit 1 Steam generators were replaced with Model F Steam generators in 1998. The Model F SG uses Alloy 600 thermally treated tubing, which has been shown to provide substantial increase in resistance to SCC mechanisms as compared to mill-annealed Alloy 600 tubing. Original and replacement steam generators with Alloy 600 thermally treated tubing have operated since 1980, at higher operating temperatures than Salem Unit 1, with no reported incidence of SCC in domestic units. The 1.4% uprating is expected to have a negligible impact upon SCC initiation in the Salem Unit 1 Steam generators. The only observed degradation mechanism in the Salem Unit 1 steam generators is AVB wear, and this observance is consistent with other domestic Model F Steam generators. AVB wear growth rates are evaluated following each outage in the condition monitoring/operational assessment. The Salem Cycle 15 Operational Assessment will address the effects of uprate on applicable performance criteria.

NRC Question

(b) Also, discuss whether the 40% thoughwall plugging limit for the steam generator tubes in the technical specifications under the power uprate condition satisfies NRC Regulatory Guide 1.121.

PSEG Nuclear response to Q6b:

6b. Steam Generators at Salem are managed in accordance with NEI 97-06, Steam Generator Program, and Technical Specifications. The programmatic requirements of NEI 97-06 for minimum acceptable wall thickness are comparable to the guidance in Regulatory Guide 1.121 in that code safety limits are maintained under normal and faulted conditions (i.e. three times normal operating pressure and 1.4 times main steam line break, respectively). Expected changes in plant parameters that affect degradation growth rates, such as hot leg reactor coolant temperature, change minimally under uprate conditions. The proposed 1.4% uprate will therefore have an insignificant effect on existing and anticipated degradation growth rates, and code safety limits will continue to be programmatically maintained by the station's Steam Generator Program.

Under certain circumstances when flaw growth rates would not support Operational Assessment performance criteria for the planned operational period following an inspection outage, tubes are plugged at less than the current 40% Technical Specification Limit. If the plugging limit were not lowered below the current 40% plugging limit for a specific degradation mechanism and operational assessment performance criteria could not be met for the proposed operating period, the existing Steam Generator Program would require a reduction in the interval between inspections.

These controls in all cases can support the 40% Technical Specification plugging limit by reducing cycle length. If for economic reasons PSEG Nuclear LLC chooses to extend the cycle run time to support the normal 18 month refueling frequency, in some cases an administrative plugging limit below 40% may be implemented.

Based on the above, the 40% through wall plugging limit for the steam generator tubes in the technical specifications under the power uprate conditions satisfies NRC Regulatory Guide 1.121 requirements for minimum acceptable wall thickness.

NRC Question

(c) Will the power uprate impact future tube inspection and inspection frequencies?

PSEG Nuclear response to Q6c:

6c. Steam generator tube inspections and inspection frequencies are driven by the degradation assessment, condition monitoring and operational assessments, and inspection requirements of the EPRI ISI guidelines and Technical Specifications. Inspections are controlled programmatically by evaluating both active and potential degradation mechanisms, industry experience, and plant specific operating experience. The uprate will have minor effects on observed growth rates of existing degradation mechanisms as discussed previously, but will create no new degradation mechanisms. Therefore, no changes to the inspection plan are required.

NRC Question

(d) In Section 5.9.4, U-Bend Fatigue Evaluation, it states that an evaluation found that some steam generator tubes would be susceptible to high cycle fatigue at the uprated conditions with the plant operating at lower steam pressures. Therefore, according to your evaluation for Unit Nos. 1 and 2, which steam generator tubes did you find to be susceptible to U-bend fatigue? Also, where along the tubes are the critical positions? Do these differ between Unit Nos. 1 and 2? If so, why? What are the relevant parameters, with regard to fatigue, at those positions?

Furthermore, in order to independently evaluate the impact that uprated power levels have on certain limiting conditions when comparing current licensed power levels with the proposed uprated levels, please provide the following information described in the table below:

	Unit No. 1		Unit No. 2	
Parameter	**Current Power Level**	**1.4 % Increased**	**Current Power Level**	**1.4 % Increased**
Steam Flow				
Circulation Ratio				
Steam Pressure				
Primary System Temperature				
Amplitude and Direction of the Cyclic Deformation at the Limiting Point along the Tube				
Frequency of Deformation at the Limiting Point along the Tube				
Limiting Number of Cycles				
Expected Number of Cycles to End-of-Service				

PSEG Nuclear Response to Q6d:

6d. At the 1.4% power uprated conditions, tubes recommended for preventative action are tied to the operating full power steam pressure at the steam nozzle outlet. Based upon the analysis, the tubes that would require preventive action for Salem Unit 2 at the given steam pressure are listed in Table 1 below.

Table 1

Salem Unit 2 Steam Generator Tubes Requiring Preventative Action

Steam Nozzle Outlet Pressure - PSIA	Tubes Requiring Preventative Action
>800	None
750 to 800	SG 22: R11C16, R11C17, R10C46 SG 23: R8C59
700 to 749	All the above tubes plus: SG 22: R8C61, R10C50
650 to 700	All the above tubes plus: SG 24: R9C35 SG 23: R10C4

All tubes that require plugging for pressures above 700 psia have already been plugged in the Salem Unit 2 steam generators. An item has been entered into PSEG Nuclear's corrective action program (Ref: Notification 20058707) to revise procedures as necessary to ensure SG 24:R9C35 and SG 23:R10C4 are plugged prior to steam generator outlet pressure reaching 700 psia. Lowest recorded pressure at Salem Unit 2 at 100% power was approximately 761 psia (Ref: Salem Calculation S-2-RC-MDC-1827 Rev. 1). This pressure was recorded in the first cycle after chemical cleaning and has since increased.

Note that it is only necessary to address tubes in the Salem Unit 2 steam generators since the prerequisite conditions required to develop high cycle fatigue associated with the 1987 tube rupture at North Anna cannot occur in the replacement Model F steam generators at Salem Unit 1. The tube support plates at Salem Unit 1 are manufactured from 405 stainless steel and have quatrefoil broached tube holes. These tube support plates do not corrode, hence a fixed tube condition (a necessary boundary condition required to develop high cycle fatigue) cannot occur in these steam generators. Therefore it is only necessary to address the Unit 2 steam generator tubes since the tube support plates in this unit are manufactured from carbon steel and as a result are susceptible to conditions that lead to a fixed tube condition.

Table 2 contains a summary of the thermal and hydraulic parameters associated with the Unit 2 steam generators. The table provides the appropriate parameters for both the current and uprated power levels with no steam generator tube plugging and the steam generators at their plugging limit evaluated at either the high T-avg or low T-avg conditions as needed.

Table 2

Salem Unit-2
Predicted Steam Generator Thermal-Hydraulic Characteristics with 1.4% Power Uprate

Power	**Current Rating**			**1.4% Uprate**			
- MWt/SG	**855.75**	**855.75**	**855.75**	**867.75**	**867.75**	**867.75**	**867.75**
Plugging - %	0	25	25	0	20	0	20
Primary Temperatures. - °F							
SG Inlet (T_{hot}) - °F	610.9	612.6	601.3	601.8	601.8	613.1	613.1
SG Outlet (T_{cold}) - °F	544.8	542.9	530.4	530.0	530.0	542.5	542.5
SG $T_{average}$ - °F	577.9	577.8	565.9	565.9	565.9	577.8	577.8
Steam Flow - Million lbs/hour	3.713	3.709	3.700	3.758	3.753	3.769	3.762
Steam Press. - psia	794	762	678	736	689	825	773
Circulation Ratio	5.16	5.17	5.14	5.09	5.06	5.08	5.09

Table 3 contains additional relevant information pertaining to the level of susceptibility of the limiting steam generator tubes. Note that the screening process was performed using stress ratios calculated for the current operating conditions and included all relevant parameters such as: location dependent secondary side fluid velocities, densities and void fractions, and local flow peaking factors. These quantities were used along with low damping, a lower bound fatigue curve, past and future operation conditions, along with the assumption of dented tube conditions to identify which unsupported U- bend tubes would be potentially susceptible to high cycle tube fatigue. It should be noted that the methods used in the analysis did not require the specific calculation of tube displacements, hence these values are not currently available. In addition, the number of allowable cycles for a given tube (used in fatigue usage calculations) is dependent upon the level of alternating stress, which in turn is dependent upon the secondary side operating conditions. Since the level of alternating stress changes with the secondary side conditions, the number of allowable cycles would also change. The values in Table 3 are therefore bounding for predicted uprated operating conditions but actual field values will differ according to actual plant conditions. As stated above, all tubes except SG 24: R9C35 and SG 23: R10C4 have been plugged.

Table 3

Salem Unit 2 Critical Tube Summary For U-bend Tube Fatigue

S/G No.	Tube Row/Col	Current Power Level		1.4 % Uprate Condition	
		Stress Ratio	Fatigue Usage(1)	Stress Ratio (2)	TOTAL FATIGUE USAGE
SG 22	R11C16	0.82	< 0.48	n.a.	~1.0 at Steam Pressure of 800 Psia
SG 22	R11C17	0.81	< 0.48	n.a.	~1.0 at Steam Pressure of 800 Psia
SG 22	R10C46	0.82	< 0.48	n.a.	~1.0 at Steam Pressure of 800 Psia
SG 23	R8C59	0.83	0.48	n.a.	1.0 at Steam Pressure of 800 Psia
SG 22	R8C61	0.68	< 0.48	n.a.	1.0 at Steam Pressure of 749 Psia
SG 22	R10C50	0.66	< 0.48	n.a.	~1.0 at Steam Pressure of 749 Psia
SG 24	R9C35	0.58	< 0.48	n.a.	1.0 at Steam Pressure of 700 Psia
SG 23	R10C4	0.54	< 0.48	n.a.	~1.0 at Steam Pressure of 700 Psia

Notes:

1) Fatigue usage was only calculated for the limiting tube in the original analysis.

2) Stress ratios at uprate power conditions are not meaningful since this quantity does not distinguish between past and future operation at differing power levels. The quantity that is relevant is the total fatigue usage since both past and future operation has been accounted properly.

3) The tubes listed above are not the only active unsupported U-bend tubes remaining in the Salem Unit 2 steam generators. However, it would require that steam pressure fall to below 650 psia for these other unsupported tubes to become potentially susceptible. However, at 650 psia SG outlet pressure, the main turbine throttle valves would have reached the valves-wide-open point, and a significant plant de-rate would have been required for continued operation. Administrative controls will be put in place (Ref: Notification 20058707) to ensure that the effects of secondary side pressure are properly addressed.

NRC Question:

1.4.6 Instrumentation and Controls / Uncertainty Determination

7. In order confirm that licensed power levels will not be exceeded at uprated conditions, the NRC staff needs additional information concerning how instrument uncertainty was calculated. Therefore, the following needs to be addressed:

   (a) Attachment 1, Section 1.4.6, states that CENP has completed the Salem, Unit Nos. 1 and 2, CENP Crossflow uncertainty calculations A-SA1-PS-0001, Revision 0, and A-SA2-PS-0001, Revision 0. Therefore, using a copy of one of these calculations, please provide a further explanation of how the estimated uncertainty of the net heat input from the reactor coolant pump (RCP) to the reactor coolant system (RCS), resulted in the values for total heat input and core power uncertainties, as stated on page 8 of WCAP-15553.

PSEG Nuclear Response to Q7a:

7a. The effect of the reactor coolant pump and total Net Pump Heat Addition (NPHA) are not documented in the reference calculations, rather the explanation is contained on page 8 of WCAP-15553. For the total core power uncertainty calculation, Westinghouse provides allowances for the uncertainty on the determination of primary side heat losses and heat additions. As noted in the WCAP, the allowance for pump heat adder is +15.4 MWt. The uncertainty for this value is estimated to be [ ]$^{+a,c}$. Uncertainties are also estimated for system heat losses (- 2.0 MWt,) and component conduction and convection losses (- 1.4 MWt) as [ ]$^{+a,c}$ respectively. The combination of the losses and heat addition results in a total NPHA input to the RCS of +12 MWt. Westinghouse takes a conservative allowance of [ ]$^{+a,c}$ of the 12 MWt as a percentage of the total core power (3459 MWt) which results in a core power uncertainty of [ ]$^{+a,c}$. As noted in the WCAP the combined uncertainties of the losses and heat additions are less than [ ]$^{+a,c}$, therefore the total allowance of [ ]$^{+a,c}$ is conservative.

Even though the affect of the reactor coolant pump and the total NPHA are not documented in the referenced calculations, a copy of CENP proprietary calculation A-SA2-PS-0001 is being provided in Attachment 6.

NRC Question:

   (b) Section 5.10 of CENPD-397-P-A stated that licensees desiring to lower the total feedwater flow measurement uncertainty can do so by simply improving the accuracy of the feedwater temperature instrumentation. Westinghouse Topical Report WCAP-15553, Table 1, shows a value for the feedwater temperature instrumentation uncertainty. How was this value for the uncertainty determined? Was this value based on actual plant data or was it provided by the instrument supplier?

PSEG Nuclear Response to Q7b:

7b. The uncertainty was determined by a combination of actual plant instrumentation loop testing and vendor specification (RTD Drift) at the temperature of interest. The temperature loops were string (RTD through computer readout) field calibrated by heating the RTDs with a high accuracy dry heat block. The results of the calibration were combined with the RTD vendor's specified drift for a calculated accuracy performed as follows:

Total Accuracy = M&TE+RTD Drift+CPTR Rd+Polynomial Fitting

M&TE = Measurement and Test Equipment Accuracy
RTD Drift = From vendor data
CPTR Rd = Computer Readability
Polynomial Curve Fitting = This error is a non-random error due to curve fitting accuracy

The calculated errors obtained were less than 2°F.

NRC Question:

(c) Westinghouse Topical Report WCAP-15553, Table 1, shows instrumentation uncertainties of [x] pounds per square inch (psi), [y]% flow span, and [z] psi for feedwater pressure (percent span), Steam Generator Blowdown (percent differential pressure (dP) span), and steam pressure (percent span), respectively. Explain how the values for [x], [y], and [z] were calculated.

PSEG Nuclear Response to Q7c:

7c. Westinghouse performed Salem Units 1 and 2 plant specific uncertainty calculations based on the installed instrumentation, plant calibration procedures, as used plant Measurement and Test Equipment (M&TE) and the installed plant computer. To support these calculations, Westinghouse reviewed vendor specific information relative to reference accuracy effects, temperature effects, and static pressure effects, plant calibration procedures for the determination of sensor and rack calibration accuracy and applicable M&TE. After determination of all appropriate allowances, Westinghouse combined these allowances via the Square Root Sum of the Squares (SRSS) method using equation 1 as noted on page 2 of WCAP-15553. The Westinghouse method is conservative with respect to ANSI/ISA S67.04.01-2000 "Setpoints for Nuclear Safety-Related Instrumentation", and Regulatory Guide 1.105, Rev. 3 "Instrument Setpoints for Safety-Related Systems". This approach is the same approach as used for many past Westinghouse submittals for core power uncertainty calculations and previously approved by the NRC via Westinghouse topical report WCAP-8567 "Improved Thermal Design Procedure".

SALEM GENERATING STATION
UNIT NOS. 1 AND 2
FACILITY OPERATING LICENSE DPR-70 AND DPR-75
DOCKET NOS. 50-272 AND 50-311
CHANGE TO FACILITY OPERATING LICENSES

TECHNICAL SPECIFICATION PAGES WITH PROPOSED CHANGES

The following Technical Specifications pages for Facility Operating License No. DPR-70 are to replace the pages contained in Attachment 4 of LR-N00-0387, dated November 10, 2000. Please replace the pages in the November 10, 2000 submittal with the attached pages.

Technical Specification	Page
Figure 3.4-2	3/4 4-26
Figure 3.4-3	3/4 4-27
Bases 3/4.4.9	B 3/4 4-6 B 3/4 4-11

The following Technical Specifications pages for Facility Operating License No. DPR-75 are to replace the pages contained in Attachment 4 of LR-N00-0387, dated November 10, 2000. Please replace the pages in November 10, 2000 submittal with the attached pages.

Technical Specification	Page
Figure 3.4-2	3/4 4-28
Figure 3.4-3	3/4 4-29
Bases 3/4.4.9	B 3/4 4-7 B 3/4 4-12

Insert B1 Here

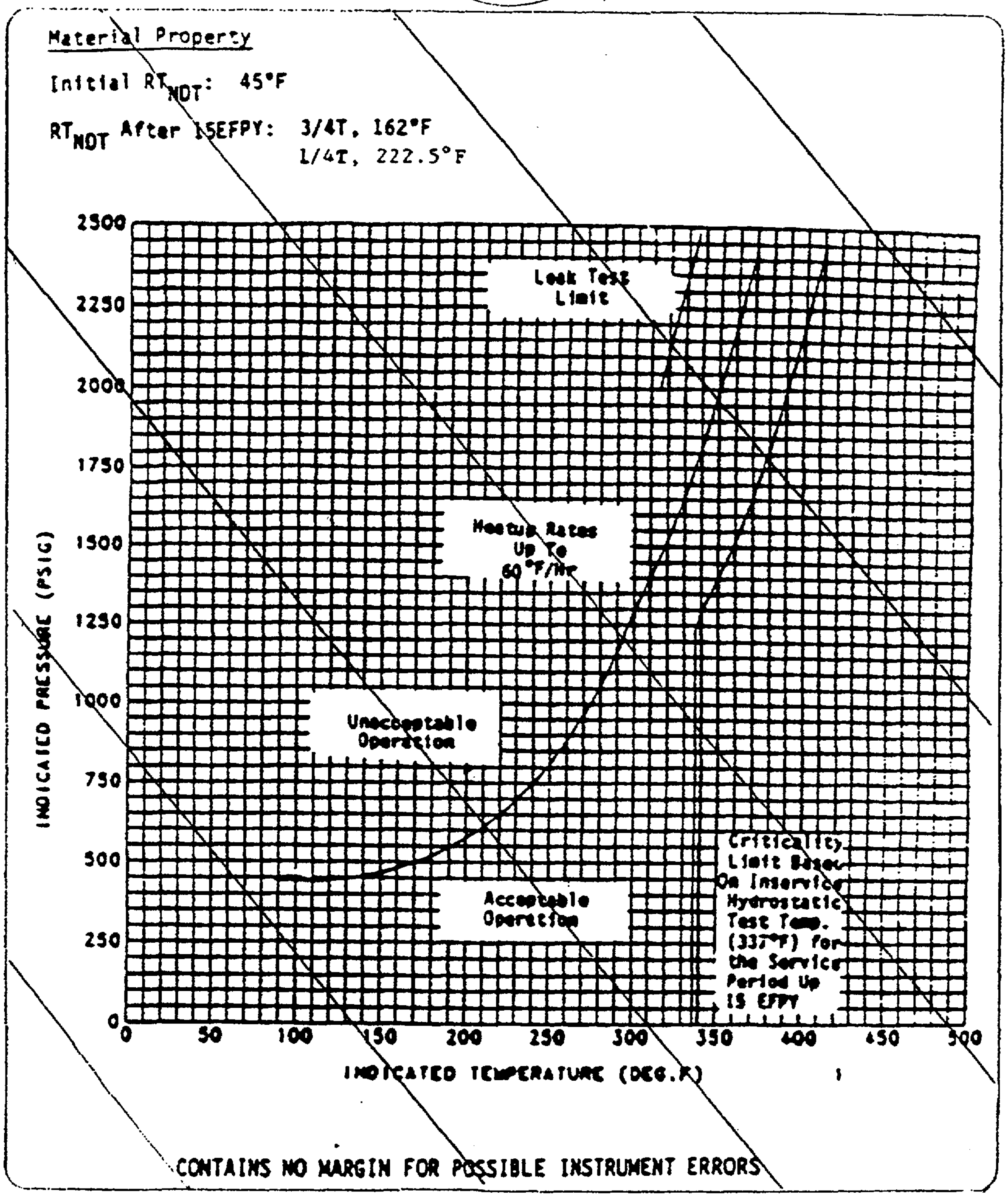
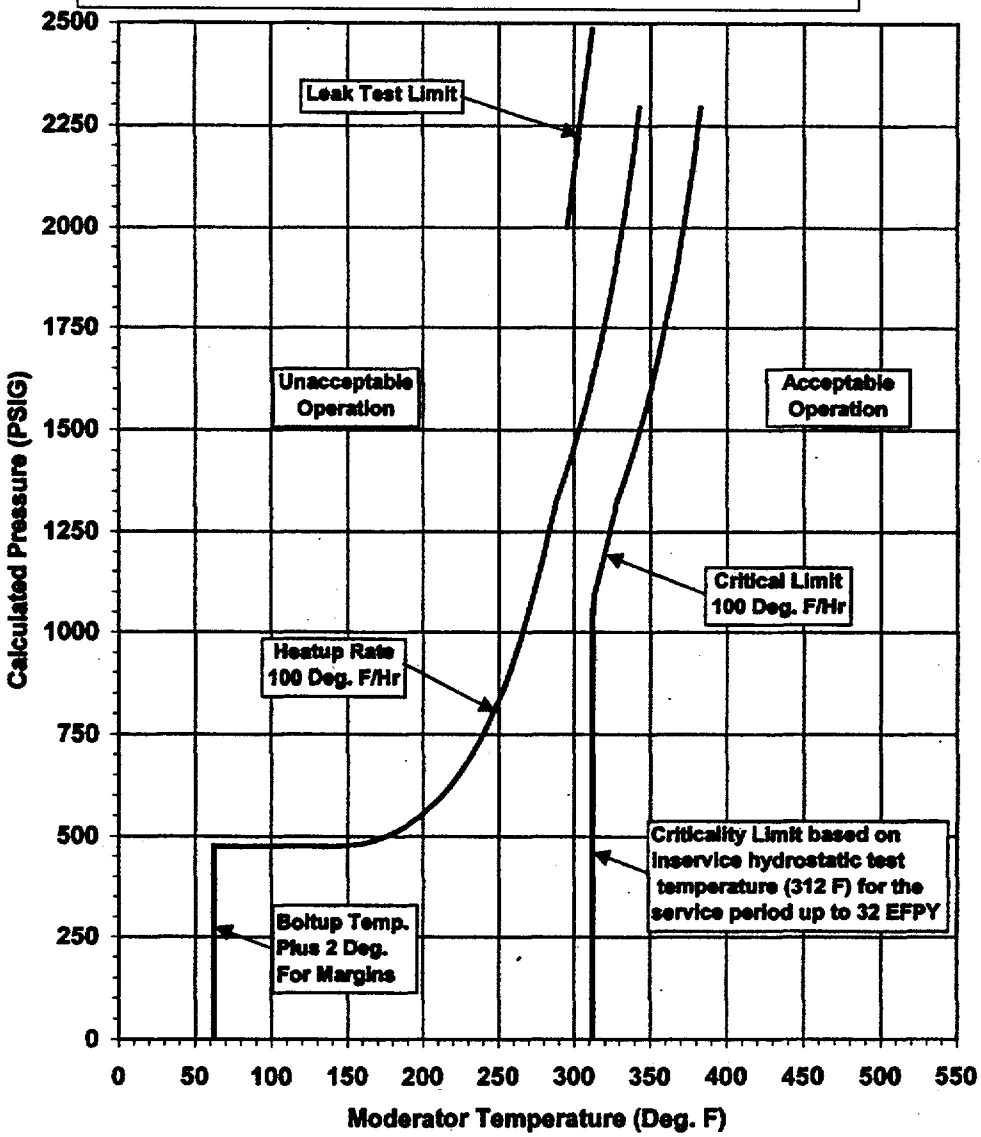


Figure 3.4-2 . Salem Unit 1 Reactor Coolant System Heatup Limitations Applicable for Heatup Rates up to ~~60~~ 100 °F/HR for the Service Period up to ~~15~~ 32 EFPY (with uncertainties for instrumentation errors)

INSERT B1

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



Insert C1 Here

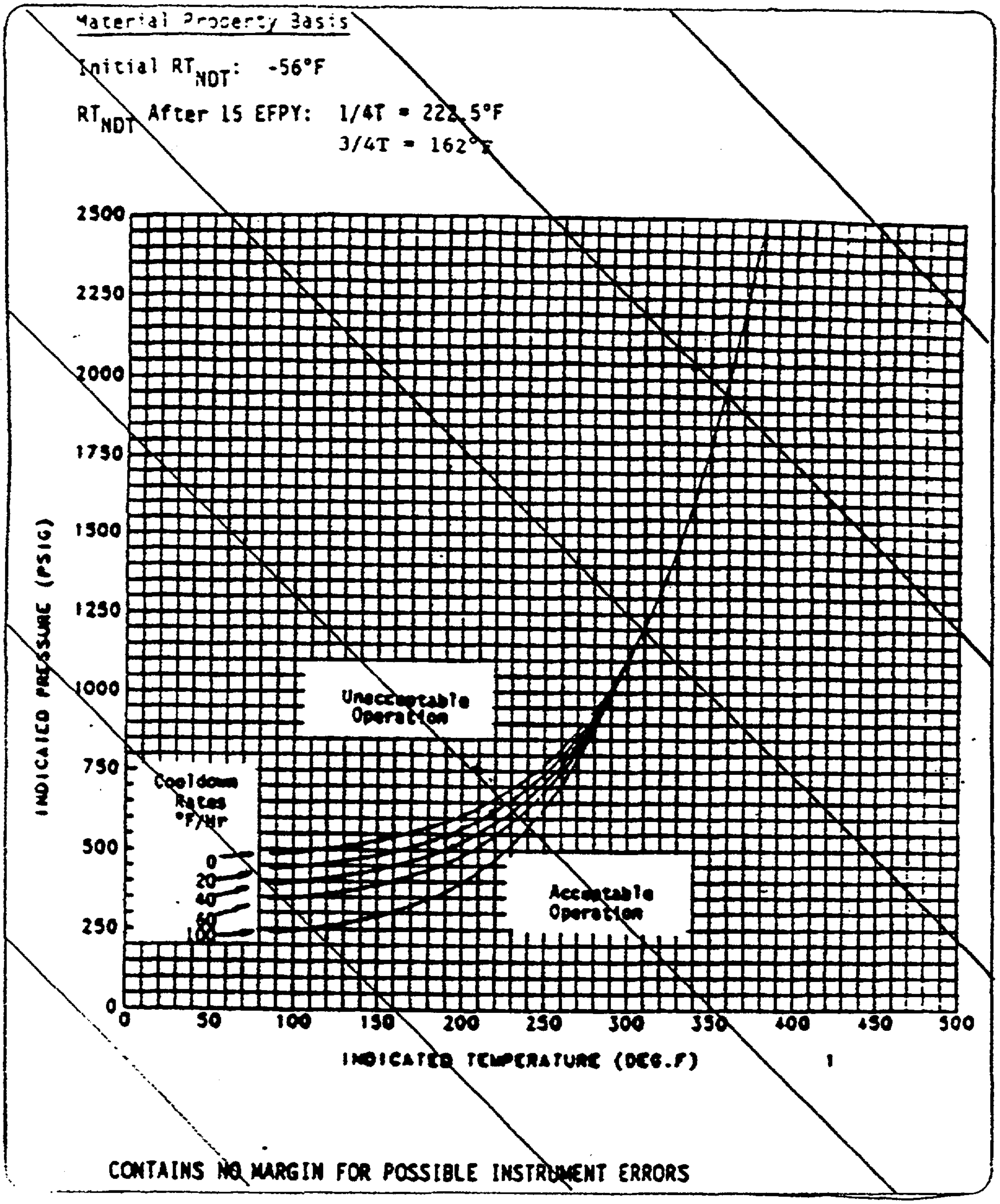


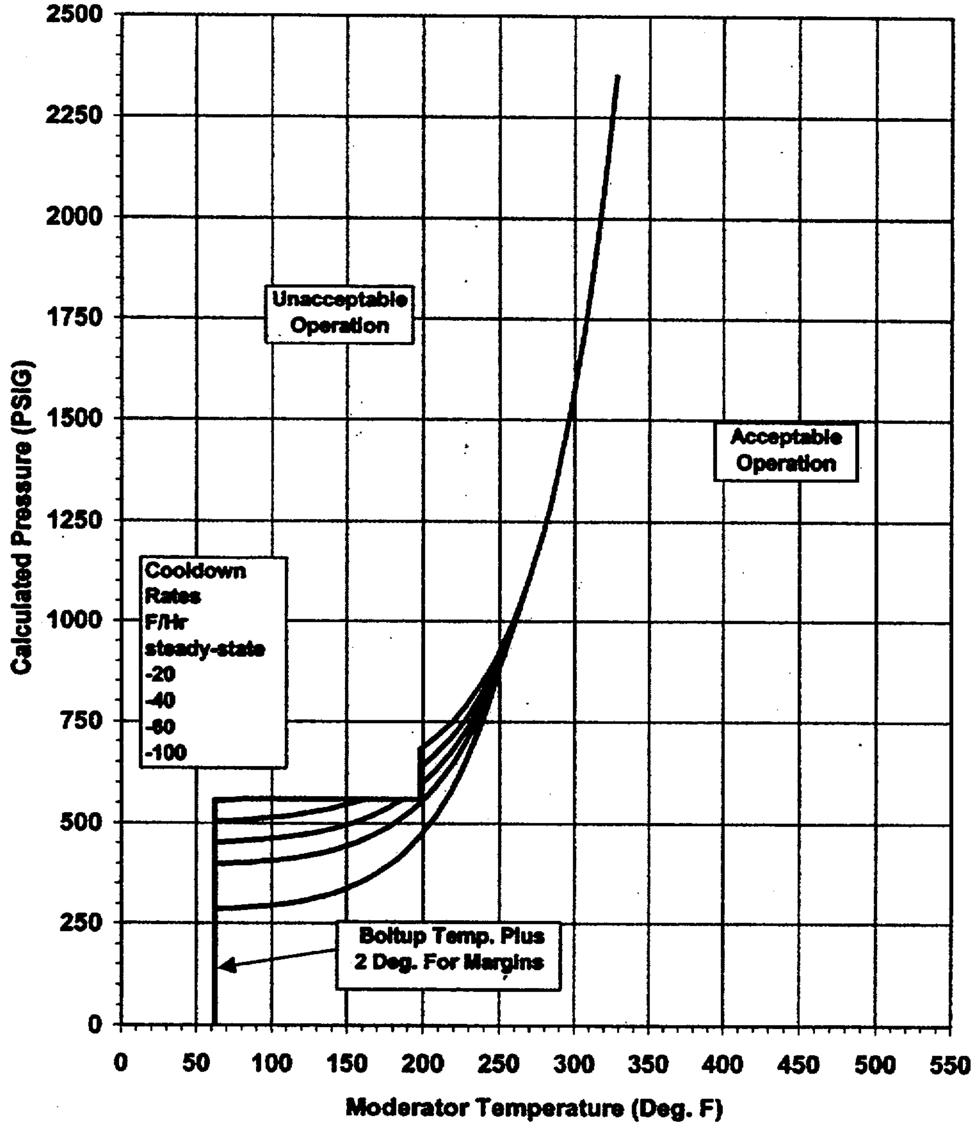
Figure 3.4-3 Salem Unit 1 Reactor Coolant System Cooldown Limitations Applicable for Cooldown Rates up to 100°F/HR for the Service Period up to ~~15~~ EFPY (with uncertainties for instrumentation errors)

32

INSERT C1

Limiting Material Property

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



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. XI

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~~ 1996 Summer Addenda to Section ~~III~~ XI 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"~~. Insert D.1 Here

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~~ 32 effective full power years of service life. The ~~15~~ 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.

INSERT D1

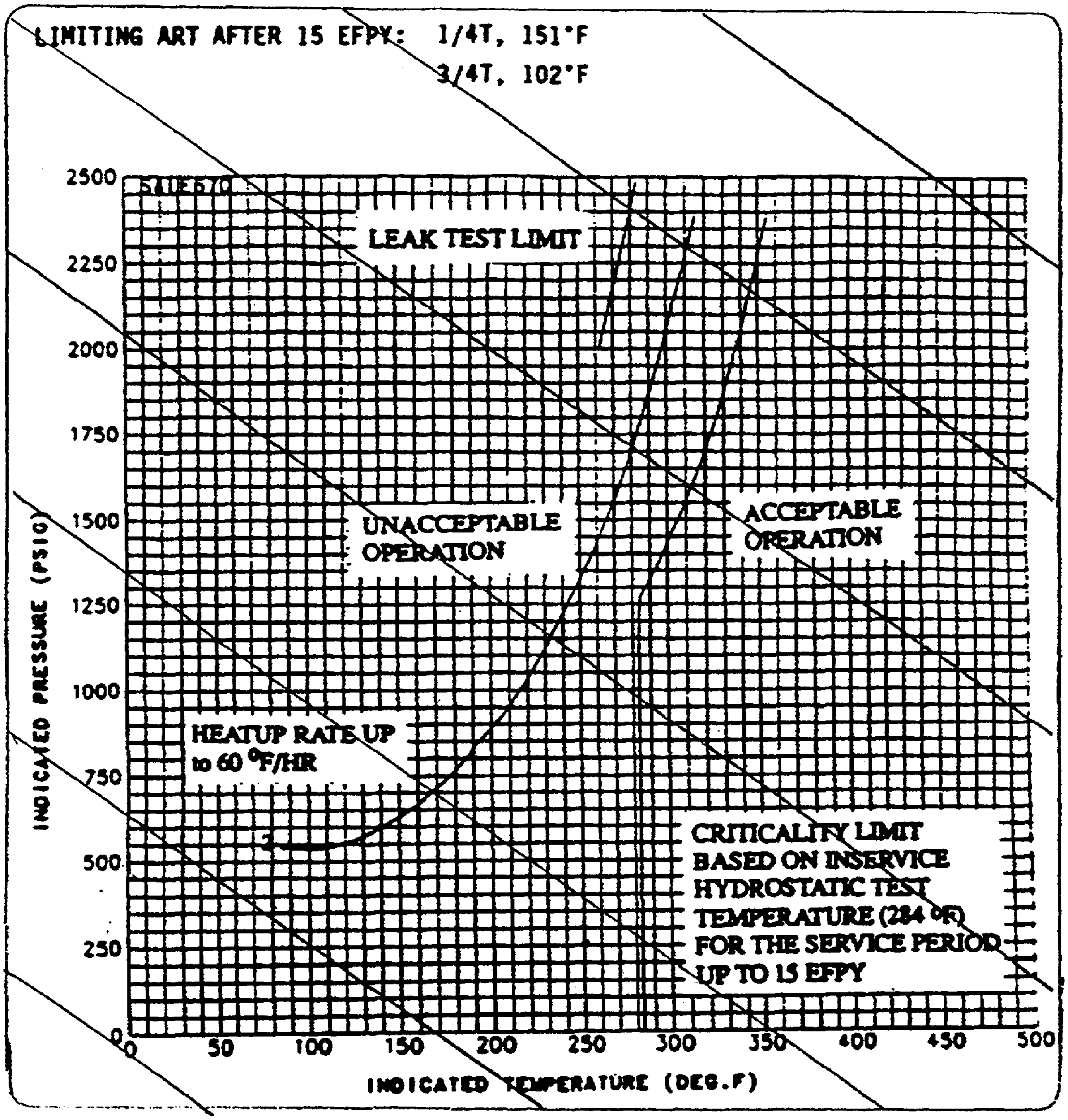
WCAP-14040-NP-A, Rev. 2, “Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves”, January 1996, and ASME Boiler and Pressure Vessel Code Case N-640, “Alternative Reference Fracture Toughness for Development of P-T Limit Curves for Section XI,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

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 ~~28°F~~ 60°F occurs in the ~~closure head~~ vessel flange of Salem Unit 1, and the minimum allowable temperature of this region is ~~140°F~~ 180°F at pressure greater than 621 psig. These limits ~~do not affect~~ 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 injection into a water solid RCS.

Insert B2 Here



SALEM UNIT 2 REACTOR COOLANT SYSTEM HEATUP LIMITATIONS APPLICABLE FOR THE FIRST ~~15~~ EFPY WITH MAXIMUM HEATUP RATE OF ~~60~~ F/HR. CURVE CONTAINS ~~NO~~ MARGIN FOR INSTRUMENT ERRORS.

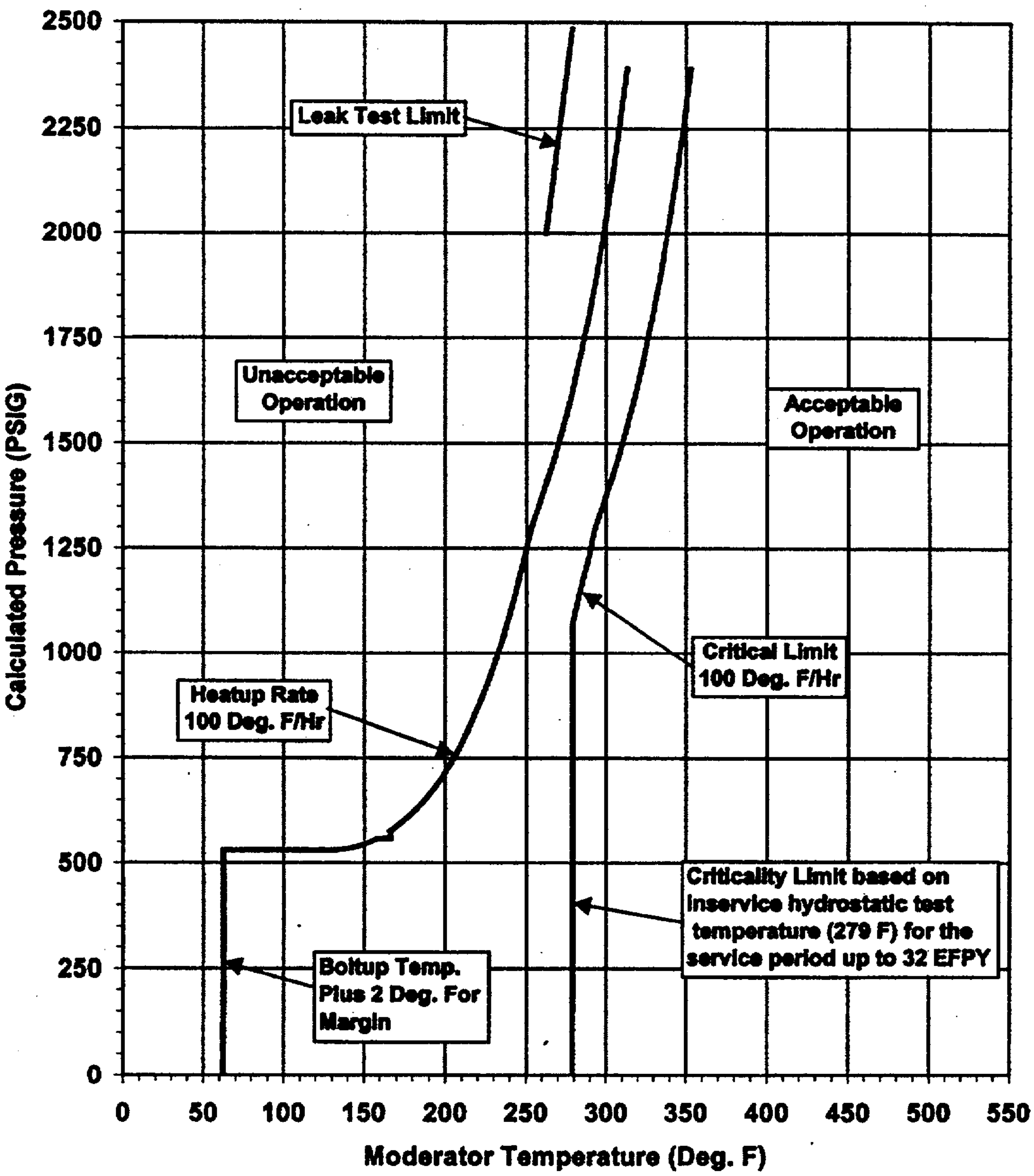
32

100

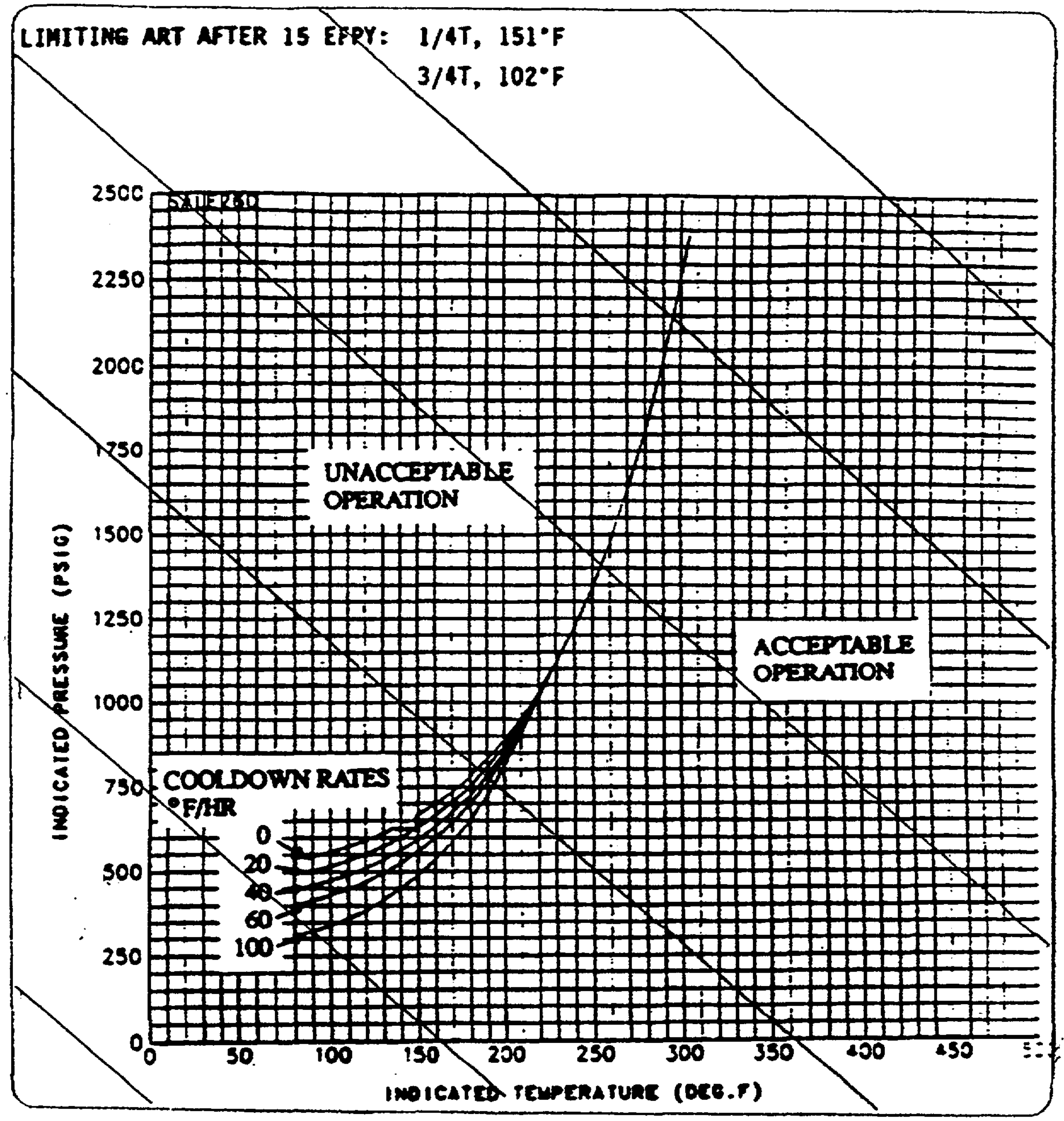
Figure 3.4-2

INSERT B2

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



Insert
C2
Here



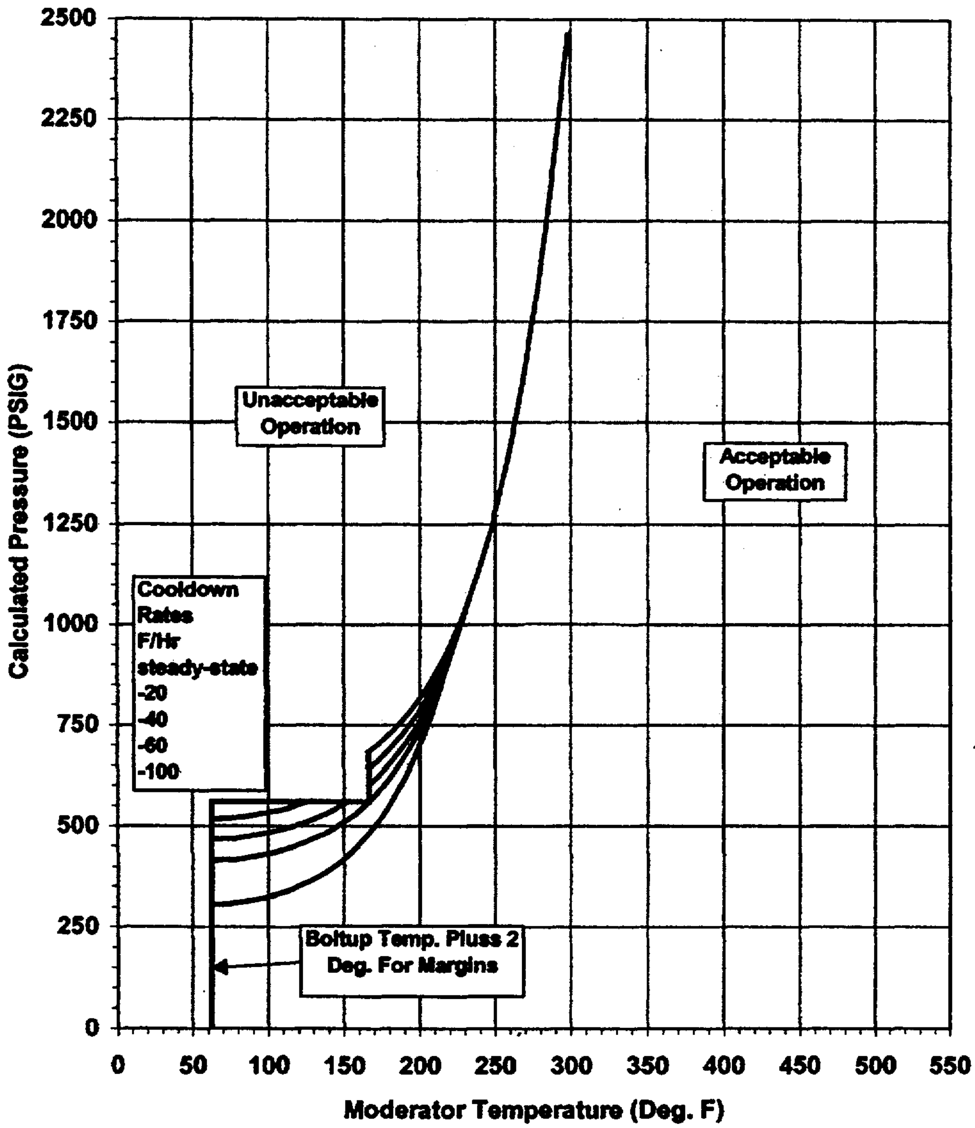
SALEM UNIT 2 REACTOR COOLANT SYSTEM COOLDOWN LIMITATIONS APPLICABLE FOR THE FIRST 15 EFPY WITH MAXIMUM COOLDOWN RATE OF 100 F/HR. CURVE CONTAINS NO MARGIN FOR INSTRUMENT ERRORS. (32)

Figure 3.4-3

INSERT C2

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



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~~ 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 ~~1976~~ 1996 Summer Addenda to Section ~~III~~ XI 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"~~ Insert D2 Here.

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~~ 32 effective full power years of service life. The ~~15~~ 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.

INSERT D2

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.

2

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 RTNDT 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 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 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.

are incorporated into

**SALEM GENERATING STATION
UNIT NOS. 1 AND 2
FACILITY OPERATING LICENSE DPR-70 AND DPR-75
DOCKET NOS. 50-272 AND 50-311
CHANGE TO TECHNICAL SPECIFICATIONS**

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATION

PSEG Nuclear LLC has determined that operation of Salem Generating Station Unit Nos. 1 and 2, in accordance with the proposed changes does not involve a significant hazards consideration. In support of this determination, an evaluation of each of the three standards of 10CFR50.92 is provided below.

REQUESTED CHANGE

The proposed license amendment increases the licensed power level for operation to 3459 MWt, 1.4% greater than the current level. Changes to the Facility Operating License and associated Technical Specifications are described below:

A. Increase in Licensed Core Power Level

1. Paragraph 2.C.(1) in Facility Operating Licenses DPR-70 and DPR-75 is revised to authorize operation at a steady state reactor core power level not in excess of 3459 megawatts (one hundred percent of rated core power).

2. The definition of RATED THERMAL POWER in Technical Specification 1.25 is revised to reflect the increase from 3411 MWt to 3459 MWt.

3. Technical Specification Table 3.7-1, Maximum Allowable Thermal Power With Inoperable Steam Line Safety Valves, and its associated Bases are revised to reflect the increase in core power.

4. Technical Specification 6.9.1.9, Core Operating Limits Report, is revised to add a reference to Topical Report CENPD-397-P-A, Revision 01, "Improved Flow Measurement Accuracy Using Crossflow Ultrasonic Flow Measurement Technology," May 2000.

B. **Reactor Core Safety Limits and Reactor Trip Setpoints**

1. Technical Specification Figure 2.1-1, Reactor Core Safety Limit, is revised to reflect the new safety limits required to prevent core exit boiling at the new core power of 3459 MWt.

2. The Overtemperature ΔT (OTΔT) f(ΔI) penalties in Technical Specification Table 2.2-1, Reactor Trip System Instrumentation Trip Setpoints, are revised to support the increase in core power.

C. **New Heatup and Cooldown Curves**

1. Technical Specification Figures 3.4-2 and 3.4-3, Reactor Coolant System Heatup and Cooldown Curves, and their associated Bases are revised to support the increase in core power based on uprated fluence projections. The revised curves are applicable for the service period up to 32 effective full power years (EFPY). The maximum heatup rate for Figure 3.4-2, Reactor Coolant System Heatup Limitations, is being changed from 60°F/hr to 100°F/hr. The revised curves are being adjusted to account for pressure and temperature instrument uncertainties and the curves are being extended to show minimum boltup temperature. The values in Bases Table B 3/4.4-1, Reactor Vessel Toughness Data, for Unit 1 and 2 are being updated to reflect information related to reactor pressure vessel integrity previously provided to the NRC in response to Generic Letter 92-01 and its supplement.

D. **Editorial Changes**

1. In TS Bases 3/4.4.9, references to ASME Boiler and Pressure Vessel Code, Section III, Appendix G are being changed to Section XI, Appendix G which is the correct reference for requirements related to reactor vessel pressure-temperature limits.

2. In TS Bases 3/4.4.9, corrections are being made to the symbol "ΔRT_{NDT}" in cases where the symbol is represented incorrectly.

3. In TS Bases 3/4.4.9, a reference to Figure B3/4.3-1 is being revised to the correct number, Figure B3/4.4-1.

E. **Removal of Historical Information from Unit 1 Facility Operating License**

1. Paragraph 2.C.(1) of the Unit 1 Facility Operating License is revised to delete reference to Attachment 1 which identified incomplete preoperational tests, startup tests and other items which were required to be completed before proceedings to certain specified Operational Modes during the initial startup of Unit 1. The NRC authorized full power operation for Unit 1 by letter dated April 6, 1977. The Unit 2 Facility Operating License does not contain a similar requirement.

BASIS

1. *The proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.*

A. Increase in Licensed Core Power Level

The comprehensive analytical efforts performed to support the proposed uprate conditions included a review and evaluation of all components and systems (including interface systems and control systems) that could be affected by this change. Evaluation of accident analyses including steam generator tube rupture (SGTR) dose-related events confirmed the effects of the proposed uprate are bounded by the current dose analyses. All systems will function as designed, and all performance requirements for these systems have been evaluated and found acceptable. Changes to the maximum allowable thermal power with inoperable steam line safety valves ensure that all current analyses supporting the allowable power levels remain bounding for uprated conditions. Addition of Topical Report CENPD-397-P-A, Revision 1, to the list of documents describing methods for determination of core operating limits ensures use of a previously approved method for determination of feedwater flow measurement uncertainty. The proposed changes do not affect any accident initiators and do not affect the ability of any systems, structures or components to mitigate the consequences of accidents. Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

B. Reactor Core Safety Limits and Reactor Trip Setpoints

Neither the core limits curve nor the OTΔT Delta I penalties initiate any accident. Therefore, the probability of an accident has not been increased. Dose consequences have been analyzed or evaluated with respect to these parameters, and the 10 CFR 100 acceptance criteria continue to be met. Therefore, the proposed changes to the reactor core safety limits and to the reactor trip setpoints do not involve a significant increase in the probability or consequences of an accident previously evaluated.

C. New Heatup and Cooldown Curves

The revised curves support the increase in core power based on uprated fluence projections and are applicable for the service period up to 32 effective full power years (EFPY). There are no changes being made to the reactor coolant system (RCS) pressure boundary or to RCS material, design or construction standards. The proposed heatup and cooldown curves define limits that continue to ensure the prevention of nonductile failure of the RCS pressure boundary. The design-basis events that were protected have not changed. The modification of the heatup and cooldown curves does not alter any assumptions previously made in the radiological consequence evaluations since the integrity of the RCS pressure boundary is unaffected. Therefore, the proposed changes will not significantly increase the probability or consequences of an accident previously evaluated.

D. Editorial Changes
The proposed editorial changes involve typographical errors. These changes do not affect any accident initiators and do not affect the ability of any systems, structures or components to mitigate the consequences of accidents. Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

E. Removal of Historical Information from Unit 1 Facility Operating License
The reference to Attachment 1 in Paragraph 2.C.(1) of the Unit 1 Facility Operating License is being deleted because it refers to one-time requirements that are not applicable to operation at the proposed power level. The change does not affect any accident initiators and does not affect the ability of any systems, structures or components to mitigate the consequences of accidents. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. *The proposed changes do not create the possibility of a new or different kind of accident from any accident previously analyzed.*

A. Increase in Licensed Core Power Level
No new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of the proposed change. Systems, structures and components previously required for mitigation of design basis events remain capable of performing their design function. The proposed change has no adverse effects on any safety-related system and does not challenge the performance or integrity of any safety-related system. Therefore, the possibility of a new or different kind of accident is not created.

B. Reactor Core Safety Limits and Reactor Trip Setpoints
The proposed changes to the reactor core limits figure and to the OTΔT F Delta I penalties do not introduce any new accident scenarios, failure mechanisms, or limiting single failures. The proposed changes have no adverse effects on any safety-related system and do not challenge the performance or integrity of any safety-related system. No new or different type of equipment will be installed. The OTΔT and OPΔT reactor trip system (RTS) functions continue to ensure all accident analyses criteria are met. Therefore, the possibility of a new or different kind of accident is not created.

C. New Heatup and Cooldown Curves
Revisions to the heatup and cooldown curves do not involve any new components or plant procedures. The proposed changes do not create any new single failure or cause any systems, structures or components to be operated beyond their design bases. Therefore, the proposed license amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

D. Editorial Changes
These proposed changes do not involve any potential initiating events that would create the possibility of a new or different kind of accident. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

E. Removal of Historical Information from Unit 1 Facility Operating License
The reference to Attachment 1 in Paragraph 2.C.(1) of the Unit 1 Facility Operating License is being deleted because it refers to one-time requirements that are not applicable to operation at the proposed power level. The change does not involve any potential initiating events that would create the possibility of a new or different kind of accident. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. *The proposed changes do not involve a significant reduction in a margin of safety.*

A. Increase in Licensed Core Power Level
The proposed change does not involve a significant reduction in a margin of safety. All analyses supporting the proposed uprate conditions reflect the rated thermal power value. All acceptance criteria continue to be met. Therefore, the proposed change does not involve a significant reduction in the margin of safety.

B. Reactor Core Safety Limits and Reactor Trip Setpoints
The core safety limits curve represents the locus of conditions where limits would be exceeded. The particular limits are the core exit boiling limits and departure from nucleate boiling ratio (DNBR) limits. The OTΔT setpoints are defined to protect against violating these limits. A re-analysis has been performed verifying that the revised core safety limits curves are protected by the OTΔT setpoints provided. The calculations are based on PSEG Nuclear, LLC instrumentation and calibration/functional test methods and include allowances for the uprated conditions. All analyses and evaluations supporting the proposed uprated conditions are acceptable. All acceptance criteria continue to be met. As such, the proposed changes do not involve a significant reduction in the margin of safety.

C. New Heatup and Cooldown Curves
The proposed figures define the limits for ensuring prevention of nonductile failure for the reactor coolant system based on the methods described in ASME Code Case N-640. The effect of the change is to permit plant operation within different pressure-temperature limits, but still with adequate margin to assure the integrity of the reactor coolant system pressure boundary. Therefore, the proposed change does not involve a significant reduction in the margin of safety.

D. Editorial Changes

These changes are editorial in nature. The proposed changes will make the information in the TS consistent with that already approved by the NRC. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

E. Removal of Historical Information from Unit 1 Facility Operating License

The reference to Attachment 1 in Paragraph 2.C.(1) of the Unit 1 Facility Operating License is being deleted because it refers to one-time requirements that are not applicable to operation at the proposed power level. The change does not affect the ability of any system, structure or component to perform its specified function. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

CONCLUSION

Based on the preceding discussion, PSEG Nuclear has concluded that the proposed changes to the Technical Specifications do not involve a significant hazards consideration insofar as the changes: (i) do not involve a significant increase in the probability or consequences of an accident previously evaluated, (ii) do not create the possibility of a new or different kind of accident from any accident previously evaluated, and (iii) do not involve a significant reduction in a margin of safety.

WCAP-15565, Revision 1, “Salem Unit 1 Heatup and Cooldown Curves for Normal Operation”

WCAP-15565, Revision 1

Salem Unit 1 Heatup and Cooldown Curves for Normal Operation

Ed Terek

February 2001

Approved: AJ Kuempel for C. H. Boyd 2/23/01
C. H. Boyd, Manager
Equipment & Materials Technology

Approved:
D. M. Trombola, Manager
Mechanical Systems Integration

Westinghouse Electric Company LLC
Energy Systems
P.O. Box 355
Pittsburgh, PA 15230-0355

PREFACE

This report has been technically reviewed and verified by:

Reviewer: Tom Laubham

Revision 1:

Revision 0 of WCAP-15565 documented heatup and cooldown limit curves that were generated without the vessel flange requirements of 10 CFR 50, Appendix G. Revision 1 of WCAP-15565 documents the heatup and cooldown limit curves with the flange requirement included.

TABLE OF CONTENTS

LIST OF TABLES

LIST OF TABLES -(Continued)

LIST OF FIGURES

EXECUTIVE SUMMARY

This report documents the development pressure-temperature limit curves for the PSEG Nuclear LLC Salem Unit 1 electric generating plant for normal operation at 32 and 48 EFPY. These pressure-temperature limit curves include the 1.4% uprating fluence values and utilize the methodology from the 1995 ASME Boiler and Pressure Vessel Code, Section XI, Appendix G, through 1996 addendum[3]. Regulatory Guide 1.99, Revision 2[1] is used for the calculation of Adjusted Reference Temperature (ART) values at the ¼T and ¾T locations. The ¼T and ¾T values are summarized in Table 4-14. The pressure-temperature limit curves were generated with margins for instrumentation errors for heatup rates of 60 and 100°F/hr and cooldown rates of 0, 20, 40, 60 and 100°F/hr. These curves can be found in Figures 5-1 through 5-4. In addition, these heatup and cooldown pressure-temperature limit curves include ASME Code Case N-640[10], which allows the use of the K_{Ic} methodology. Revision 0 of this report provides justification for the removal of the reactor vessel flange temperature-pressure requirements of Appendix G to 10 CFR Part 50[2] and documents the development of curves that do not include the vessel flange requirements. Revision 1 of this report contains curves that include the reactor vessel flange temperature-pressure requirements of Appendix G to 10 CFR Part 50[2].

1 INTRODUCTION

Heatup and cooldown limit curves are calculated using the adjusted RT_{NDT} (reference nil-ductility temperature) corresponding to the limiting beltline region material of the reactor vessel. The adjusted RT_{NDT} of the limiting material in the core region of the reactor vessel is determined by using the unirradiated reactor vessel material fracture toughness properties, estimating the radiation-induced ΔRT_{NDT}, and adding a margin. The unirradiated RT_{NDT} is designated as the higher of either the drop weight nil-ductility transition temperature (NDTT) or the temperature at which the material exhibits at least 50 ft-lb of impact energy and 35-mil lateral expansion (normal to the major working direction) minus 60°F.

RT_{NDT} increases as the material is exposed to fast-neutron radiation. Therefore, to find the most limiting RT_{NDT} at any time period in the reactor's life, ΔRT_{NDT} due to the radiation exposure associated with that time period must be added to the unirradiated RT_{NDT} (IRT_{NDT}). The extent of the shift in RT_{NDT} is enhanced by certain chemical elements (such as copper and nickel) present in reactor vessel steels. The Nuclear Regulatory Commission (NRC) has published a method for predicting radiation embrittlement in Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials"[1]. Regulatory Guide 1.99, Revision 2, is used for the calculation of Adjusted Reference Temperature (ART) values (IRT_{NDT} + ΔRT_{NDT} + margins for uncertainties) at the ¼T and ¾T locations, where T is the thickness of the vessel at the beltline region measured from the clad/base metal interface. The most limiting ART values are used in the generation of heatup and cooldown pressure-temperature limit curves for normal operation. Calculated capsule and vessel fluence projections[4,7] were used in determination of the most limiting ART values. The fluence evaluations used the ENDF/B-VI scattering cross-section data set. This is consistent with the methods presented in WCAP-14040-NP-A, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves"[8].

The heatup and cooldown curves documented in this report were generated using the most limiting ART values and the NRC approved methodology documented in WCAP-14040-NP-A, Revision 2[8], "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves" with exception of the following: 1) The K_{Ic} critical stress intensities are used in place of the K_{Ia} critical stress intensities. This methodology is taken from approved ASME Code Case N-640[10], and 2) The 1995 Version of Appendix G to Section XI[3] through the 1996 addendum was utilized rather than the 1989 version.

2 PURPOSE

PSEG Nuclear LLC contracted Westinghouse to generate new heatup and cooldown curves for Salem Unit 1 at 32 and 48 EFPY based upon the 1.4% uprating projected fluence values using the latest Code Methodologies and the elimination of the flange requirement. The heatup and cooldown curves were generated with margins for instrumentation errors: 18°F for temperature uncertainty and 61 psig for pressure uncertainty and 2°F temperature uncertainty for boltup. The curves include a hydrostatic leak test limit curve from 2485 to 2000 psig.

The purpose of this report is to present the calculations and the development of the PSEG Nuclear LLC Salem Unit 1 heatup and cooldown curves for 32 and 48 EFPY. This report documents the calculated adjusted reference temperature (ART) values following the methods of Regulatory Guide 1.99, Revision 2[1], for all the beltline materials and the development of the heatup and cooldown pressure-temperature limit curves for normal operation.

Revision 0 of this report documented the development of pressure-temperature limit curves for normal operation with the flange requirements of Appendix G to 10 CFR Part 50 eliminated.

The purpose of Revision 1 of this report is to document the development of pressure-temperature limit curves for normal operation with the flange requirements of Appendix G to 10 CFR Part 50 included. All other assumptions and calculations remain the same as Revision 0.

3 CRITERIA FOR ALLOWABLE PRESSURE-TEMPERATURE RELATIONSHIPS

3.1 Overall Approach

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 Code Case N-640, "Alternative Reference Fracture Toughness for Development of PT Limit Curves for Section XI"[10] of the ASME Appendix G to Section XI[3]. The K_{Ic} curve is given by the following equation:

$$K_{Ic} = 33.2 + 20.734 * e^{[0.02(T - RT_{NDT})]} \quad (1)$$

where,

K_{Ic} = reference stress intensity factor as a function of the metal temperature T and the metal reference nil-ductility temperature RT_{NDT}

This K_{Ic} curve is based on the lower bound of static critical K_I values measured as a function of temperature on specimens of SA-533 Grade B Class1, SA-508-1, SA-508-2, SA-508-3 steel.

3.2 Methodology for Pressure-Temperature Limit Curve Development

The governing equation for the heatup-cooldown analysis is defined in Appendix G of the ASME Code as follows:

$$C^* K_{Im} + K_{It} < K_{Ic} \quad (2)$$

where,

K_{Im} = stress intensity factor caused by membrane (pressure) stress

K_{It} = stress intensity factor caused by the thermal gradients

K_{Ic} = function of temperature relative to the RT_{NDT} of the material

C = 2.0 for Level A and Level B service limits

C = 1.5 for hydrostatic and leak test conditions during which the reactor core is not critical

For membrane tension, the corresponding K_I for the postulated defect is:

$$K_{Im} = M_m \times (pR_i / t) \tag{3}$$

where, M_m for an inside surface flaw is given by:

M_m = 1.85 for $\sqrt{t} < 2$,
M_m = $0.926\sqrt{t}$ for $2 \leq \sqrt{t} \leq 3.464$,
M_m = 3.21 for $\sqrt{t} > 3.464$

Similarly, M_m for an outside surface flaw is given by:

M_m = 1.77 for $\sqrt{t} < 2$,
M_m = $0.893\sqrt{t}$ for $2 \leq \sqrt{t} \leq 3.464$,
M_m = 3.09 for $\sqrt{t} > 3.464$

Where: p = internal pressure, Ri = vessel inner radius, and t = vessel wall thickness.

For bending stress, the corresponding K_I for the postulated defect is:

K_{Ib} = M_b * Maximum Stress, where M_b is two-thirds of M_m

The maximum K_I produced by radial thermal gradient for the postulated inside surface defect of G-2120 is $K_{It} = 0.953 \times 10^{-3}$ x CR x $t^{2.5}$, where CR is the cooldown rate in °F/hr., or for a postulated outside surface defect, $K_{It} = 0.753 \times 10^{-3}$ x HU x $t^{2.5}$, where HU is the heatup rate in °F/hr.

The through-wall temperature difference associated with the maximum thermal K_I can be determined from Fig. G-2214-1. The temperature at any radial distance from the vessel surface can be determined from Fig. G-2214-2 for the maximum thermal K_I.

(a) The maximum thermal K_I relationship and the temperature relationship in Fig. G-2214-1 are applicable only for the conditions given in G-2214.3(a)(1) and (2).

(b) Alternatively, the K_I for radial thermal gradient can be calculated for any thermal stress distribution and at any specified time during cooldown for a ¼-thickness inside surface defect using the relationship:

$$K_{It} = (1.0359C_0 + 0.6322C_1 + 0.4753C_2 + 0.3855C_3) * \sqrt{\pi a} \tag{4}$$

or similarly, K_{IT} during heatup for a ¼-thickness outside surface defect using the relationship:

$$K_{It} = (1.043C_0 + 0.630C_1 + 0.481C_2 + 0.401C_3) * \sqrt{\pi a} \quad (5)$$

where the coefficients C_0, C_1, C_2 and C_3 are determined from the thermal stress distribution at any specified time during the heatup or cooldown using the form:

$$\sigma(x) = C_0 + C_1(x/a) + C_2(x/a)^2 + C_3(x/a)^3 \quad (6)$$

and x is a variable that represents the radial distance from the appropriate (i.e., inside or outside) surface to any point on the crack front and a is the maximum crack depth.

Note, that equations 1 through 6 were implemented in the OPERLIM computer code, which is the program used to generate the pressure-temperature (P-T) limit curves. No other changes were made to the OPERLIM computer code with regard to P-T calculation methodology. Therefore, the P-T curve methodology is unchanged from that described in WCAP-14040, "Methodology used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves"[8] with the exceptions just described above.

At any time during the heatup or cooldown transient, K_{Ic} is determined by the metal temperature at the tip of a postulated flaw at the ¼T and ¾T location, the appropriate value for RT_{NDT}, and the reference fracture toughness curve. The thermal stresses resulting from the 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.

For the calculation of the allowable pressure versus coolant temperature during cooldown, the reference flaw of Appendix G to the ASME Code 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 the 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 ¼T vessel location is at a higher temperature than the fluid adjacent to the vessel inner diameter. This condition, of course, is not true for the steady-state situation. It follows that, at any given reactor coolant temperature, the ΔT (temperature) developed during cooldown results in a higher value of K_{Ic} at the ¼T location for finite cooldown rates than for steady-state operation. Furthermore, if conditions exist so 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 ¼T location and, 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 ensures conservative operation of the system for the entire cooldown period.

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 ¼T defect at the inside of the wall. The heatup results in compressive stresses at the inside surface 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 ¼T crack during heatup is lower than the K_{Ic} for the ¼T crack during steady-state conditions at the same coolant temperature. During heatup, especially at the end of the transient, conditions may exist so that the effects of compressive thermal stresses and lower K_{Ic} values 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 ¼T flaw is considered. Therefore, both cases have to be analyzed in order to ensure 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 the pressure-temperature limitations for the case in which a ¼T flaw located at the ¼T location from the outside surface 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 therefore tend to reinforce any pressure stresses present. These thermal stresses are dependent on both the rate of heatup and the time (or coolant temperature) along the heatup ramp. Since the thermal stresses at the outside are tensile and increase with increasing heatup rates, each heatup rate 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 by constructing a composite curve 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 wherein, 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.

3.3 Closure Head/Vessel Flange Requirements

10 CFR Part 50, Appendix G addresses the metal temperature of the closure head flange and vessel flange regions. This rule states that the metal temperature of the closure flange regions must exceed the material unirradiated RT_{NDT} by at least 120°F for normal operation when the pressure exceeds 20 percent of the pre-service hydrostatic test pressure (3106 psig).

The limiting unirradiated RT_{NDT} of 60°F occurs in the vessel flange of the Salem Unit 1 reactor vessel, so the minimum allowable temperature of this region is 180°F at pressure greater than 621 psig with uncertainties of 18°F and 61 psig. This limit is reflected in the heatup and cooldown curves shown in Figures 5-1 through 5-4.

3.4 Minimum Boltup Temperature

The minimum boltup temperature is equal to the material RT_{NDT} of the stressed region. The RT_{NDT} is calculated in accordance with the methods described in Branch Technical Position MTEB 5-2. The Westinghouse position is that the boltup temperature be no lower than 60°F. Thus, the minimum boltup temperature should be 60°F or the material RT_{NDT}, whichever is higher. This limit (including a 2°F uncertainty) is reflected in the heatup and cooldown curves shown in Figures 5-1 through 5-4.

4 CALCULATION OF ADJUSTED REFERENCE TEMPERATURE

From Regulatory Guide 1.99, Revision 2, the adjusted reference temperature (ART) for each material in the beltline region is given by the following expression:

$$ART = Initial\ RT_{NDT} + \Delta RT_{NDT} + Margin \quad (7)$$

Initial RT_{NDT} is the reference temperature for the unirradiated material as defined in paragraph NB-2331 of Section III of the ASME Boiler and Pressure Vessel Code[6]. If measured values of initial RT_{NDT} for the material in question are not available, generic mean values for that class of material may be used if there are sufficient test results to establish a mean and standard deviation for the class.

ΔRT_{NDT} is the mean value of the adjustment in reference temperature caused by irradiation and is calculated as follows:

$$\Delta RT_{NDT} = CF * f^{(0.28-0.10\log f)} \quad (8)$$

To calculate ΔRT_{NDT} at any depth (e.g., at ¼T or ¾T), the following formula must first be used to attenuate the fluence at the specific depth.

$$f_{(depth x)} = f_{surface} * e^{(-0.24x)} \quad (9)$$

where x inches (vessel beltline thickness is 8.625 inches[5]) is the depth into the vessel wall measured from the vessel clad/base metal interface. The resultant fluence is then placed in Equation 8 to calculate the ΔRT_{NDT} at the specific depth.

The Westinghouse Radiation Engineering and Analysis group evaluated the vessel fluence projections[4,7] The evaluation used the ENDF/B-VI scattering cross-section data set. This is consistent with the methods presented in WCAP-14040-NP-A, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves"[8]. Tables 4-1 and 4-2, herein, contain the calculated vessel surface fluence values along with the Regulatory Guide 1.99, Revision 2, ¼T and ¾T calculated fluences used to calculate the ART values for all beltline materials in the Salem Unit 1 reactor vessel.

TABLE 4-1
Summary of the Calculated Peak Clad/Base Metal Interface Pressure Vessel Neutron Fluence Values at 32 EFPY used for the Calculation of ART Values (n/cm^2, E > 1.0 MeV)

Material	Surface*	¼ T	¾ T
Intermediate Shell B2402-1	1.64×10^{19}	0.977×10^{19}	0.347×10^{19}
Intermediate Shell B2402-2	1.64×10^{19}	0.977×10^{19}	0.347×10^{19}
Intermediate Shell B2402-3	1.64×10^{19}	0.977×10^{19}	0.347×10^{19}
Lower Shell B2403-1	1.64×10^{19}	0.977×10^{19}	0.347×10^{19}
Lower Shell B2403-2	1.64×10^{19}	0.977×10^{19}	0.347×10^{19}
Lower Shell B2403-3	1.64×10^{19}	0.977×10^{19}	0.347×10^{19}
Intermediate to Lower Shell Circumferential Weld Seam 9-042 (Heat # 13253)	1.64×10^{19}	0.977×10^{19}	0.347×10^{19}
Longitudinal Weld Seams 2-042 A&B (Heat # 39B196/34B009 +NI200)	1.18×10^{19}	0.703×10^{19}	0.250×10^{19}
Longitudinal Weld Seam 2-042 C (Heat # 39B196/34B009 +NI200)	0.685×10^{19}	0.408×10^{19}	0.145×10^{19}
Longitudinal Weld Seams 3-042 A&B (Heat # 34B009+NI200, 13253)	1.08×10^{19}	0.664×10^{19}	0.229×10^{19}
Longitudinal Weld Seam 3-042 C (Heat # 34B009+NI200, 13253)	1.64×10^{19}	0.977×10^{19}	0.347×10^{19}

* Surface fluence values are the calculated clad/base metal interface values.

TABLE 4-2

Summary of the Calculated Peak Clad/Base Metal Interface Pressure Vessel Neutron Fluence Values at 48 EFPY used for the Calculation of ART Values (n/cm^2, E > 1.0 MeV)

Material	Surface*	¼ T	¾ T
Intermediate Shell B2402-1	2.42×10^{19}	1.44×10^{19}	0.512×10^{19}
Intermediate Shell B2402-2	2.42×10^{19}	1.44×10^{19}	0.512×10^{19}
Intermediate Shell B2402-3	2.42×10^{19}	1.44×10^{19}	0.512×10^{19}
Lower Shell B2403-1	2.42×10^{19}	1.44×10^{19}	0.512×10^{19}
Lower Shell B2403-2	2.42×10^{19}	1.44×10^{19}	0.512×10^{19}
Lower Shell B2403-3	2.42×10^{19}	1.44×10^{19}	0.512×10^{19}
Intermediate to Lower Shell Circumferential Weld Seam 9-042 (Heat # 13253)	2.42×10^{19}	1.44×10^{19}	0.512×10^{19}
Longitudinal Weld Seams 2-042 A&B (Heat # 39B196/34B009 +NI200)	1.75×10^{19}	1.04×10^{19}	0.371×10^{19}
Longitudinal Weld Seam 2-042 C (Heat # 39B196/34B009 +NI200)	1.03×10^{19}	0.614×10^{19}	0.218×10^{19}
Longitudinal Weld Seams 3-042 A&B (Heat # 34B009+NI200, 13253)	1.61×10^{19}	0.960×10^{19}	0.341×10^{19}
Longitudinal Weld Seam 3-042 C (Heat # 34B009+NI200, 13253)	2.42×10^{19}	1.44×10^{19}	0.512×10^{19}

* Surface fluence values are the calculated clad/base metal interface values.

The calculated integrated neutron exposure of the Salem Unit 1 surveillance capsules tested to date is given in Table 4-3.

TABLE 4-3
Calculated Integrated Neutron Exposure of the Salem Unit 1
Surveillance Capsules Tested to Date

Capsule	Fluence
T	2.73×10^{18} n/cm^2, (E > 1.0 MeV)
Y	9.13×10^{18} n/cm^2, (E > 1.0 MeV)
Z	1.33×10^{19} n/cm^2, (E > 1.0 MeV)
S	2.12×10^{19} n/cm^2, (E > 1.0 MeV)

Margin is calculated as, $M = 2\sqrt{\sigma_I^2 + \sigma_\Delta^2}$. The standard deviation for the initial RT_{NDT} margin term, σ_i, is 0°F when the initial RT_{NDT} is a measured value, and 17°F when a generic value is used. The standard deviation for the ΔRT_{NDT} margin term, σ_Δ, is 17°F for plates when surveillance capsule data is not used and 8.5°F for plates when surveillance capsule data is used. For welds, σ_Δ is 28°F when surveillance capsule data is not used and 14°F when surveillance capsule data is used. In addition, σ_Δ need not exceed one-half the mean value of ΔRT_{NDT}.

Contained in Table 4-4 is a summary of the Measured 30 ft-lb transition temperature shifts of the beltline materials[4]. These measured shift values were obtained using CVGRAPH, Version 4.1, which is a hyperbolic tangent curve-fitting program.

TABLE 4-4
Measured 30 ft-lb Transition Temperature Shifts of the Beltline Materials Contained in the Surveillance Program

Material	**Capsule**	**Measured 30 ft-lb Transition Temperature Shift[a]**
Intermediate Shell Plate B2402-1 (Longitudinal Orientation)	T	105.89°F
	Z	175.36°F
	S	172.61°F
Intermediate Shell Plate B2402-2 (Longitudinal Orientation)	T	87.17°F
	Z	153.82°F
Intermediate Shell Plate B2402-3 (Longitudinal Orientation)	T	66.07°F
	Y	114.75°F
	Z	105.77°F
Surveillance Program Weld Metal	Y	186.95°F
	S	230.65°F

Notes:
(a) Calculated using measured Charpy data plotted using CVGRAPH, Version 4.1.

Table 4-5 contains a summary of the weight percent of copper, the weight percent of nickel and the initial RT_{NDT} of the beltline materials. The weight percent values of Cu and Ni given in Table 4-5 were used to generate the calculated chemistry factor (CF) values based on Tables 1 and 2 of Regulatory Guide 1.99, Revision 2, and presented in Table 4-7. Table 4-6 provides the calculation of the CF values based on surveillance capsule data, Regulatory Guide 1.99, Revision 2, Position 2.1, which are also summarized in Table 4-7.

TABLE 4-5
Reactor Vessel Beltline Material Unirradiated Toughness Properties

Material Description	**Cu (%)**	**Ni(%)**	**Initial RT_{NDT}**
Closure Head Flange B2811	- -	- -	28°F
Vessel Flange B2410	- -	- -	60°F
Intermediate Shell B2402-1[(a)]	0.24	0.53	45°F
Intermediate Shell B2402-2[(a)]	0.24	0.53	-5°F
Intermediate Shell B2402-3[(a)]	0.22	0.51	-3°F
Lower Shell B2403-1[(a)]	0.19	0.48	4°F
Lower Shell B2403-2[(a)]	0.19	0.49	18°F
Lower Shell B2403-3[(a)]	0.19	0.48	6°F
Intermediate to Lower Shell Circumferential Weld Seam 9-042 (Heat # 13253)	0.22	0.73	-56°F [(c)]
Longitudinal Weld Seams 2-042 A, B & C (Heat # 39B196/34B009 +NI200)	0.18	1.04	-56°F [(c)]
Longitudinal Weld Seams 3-042 A , B & C (Heat # 34B009+NI200, 13253)	0.19	1.04	-56°F [(c)]

Notes:
(a) Taken from WCAP-14702[5]
(b) The surveillance program weld metal was fabricated with 3/16" diameter high manganese moly wire, heat #39B196, Linde 1092 flux, lot #3617. This weld metal is only representative of the beltline welds, not identical. Hence, the weld metal surveillance data was not used in any ART calculations.
(c) Generic mean values per 10 CFR 50.61.

TABLE 4-6
Calculation of Chemistry Factors using Salem Unit 1 Surveillance Capsule Data

Material	Capsule	Capsule f[a]	FF[b]	ΔRT_{NDT}[c]	FF*ΔRT_{NDT}	FF^2
Intermediate Shell B2402-1[d] (Longitudinal)	T	0.273	0.646	105.89°F	68.40°F	0.417
	Y	0.913	0.974	--	--	--
	Z	1.33	1.079	175.36°F	189.21°F	1.166
	S	2.12	1.204	172.61°F	207.82°F	1.440
				Sum =	465.43°F	3.023
	$CF_{B2402\text{-}1} = \Sigma(FF \times \Delta RT_{NDT}) \div \Sigma(FF^2) = (465.43°F \div 3.023) = 153.6°F$					
Intermediate Shell B2402-2[d] (Longitudinal)	T	0.273	0.646	87.17°F	56.31°F	0.417
	Y	0.913	0.974	--	--	--
	Z	1.33	1.079	153.82°F	165.97°F	1.166
	S	2.12	1.204	--	--	--
				Sum =	222.28°F	1.583
	$CF_{B2402\text{-}2} = \Sigma(FF \times \Delta RT_{NDT}) \div \Sigma(FF^2) = (222.28°F \div 1.583) = 140.4°F$					
Intermediate Shell B2402-3[d] (Longitudinal)	T	0.273	0.646	66.07°F	42.68°F	0.417
	Y	0.913	0.974	114.75°F	111.77°F	0.949
	Z	1.33	1.079	105.77°F	114.13°F	1.166
	S	2.12	1.204	--	--	--
				Sum =	268.58°F	2.532
	$CF_{B2402\text{-}3} = \Sigma(FF \times \Delta RT_{NDT}) \div \Sigma(FF^2) = (268.58°F \div 2.532) = 106.1°F$					
Surveillance Program Weld Metal[d]	T	0.273	0.646	--	--	--
	Y	0.913	0.974	186.95°F	182.09°F	0.949
	Z	1.33	1.079	--	--	--
	S	2.12	1.204	230.65°F	277.70°F	1.440
				Sum =	459.79°F	2.389
	$CF_{Sur.\ Weld} = \Sigma(FF \times \Delta RT_{NDT}) \div \Sigma(FF^2) = (459.79°F \div 2.389) = 192.5°F$					

Notes:

(a) f = calculated fluence values.[4,7] (x 10^{19} n/cm^2, E > 1.0 MeV).

(b) FF = fluence factor = $f^{(0.28 - 0.1*\log f)}$.

(c) ΔRT_{NDT} values are the measured 30 ft-lb shift values.[4]

(d) Per Reference 12, all surveillance data is credible. However, the surveillance weld metal is only representative of the beltline welds, not identical. Hence, the weld metal surveillance data was not used in any ART calculations. It is only presented here for completeness.

TABLE 4-7
Summary of the Salem Unit 1 Reactor Vessel Beltline Material Chemistry Factors
Based on Regulatory Guide 1.99, Revision 2, Position 1.1 and Position 2.1

Material	Chemistry Factor	
	Position 1.1	Position 2.1
Intermediate Shell B2402-1*	161.9°F	153.6°F
Intermediate Shell B2402-2*	161.9°F	140.4°F
Intermediate Shell B2402-3*	148.9°F	106.1°F
Lower Shell B2403-1	128.8°F	--
Lower Shell B2403-2	129.9°F	--
Lower Shell B2403-3	128.8°F	--
Intermediate to Lower Shell Circumferential Weld Seam 9-042 (Heat # 13253)	188.5°F	--
Intermediate Shell Longitudinal Weld Seam 2-042 A,B &C (Heat # 39B196/34B009 +NI200)	217.2°F	--
Lower Shell Longitudinal Weld Seam 3-042 A,B & C (Heat # 34B009+NI200, 13253)	223.6°F	--

*Surveillance Material

Contained in Tables 4-8 and 4-9 are summaries of the fluence factors (FF) used in the calculation of adjusted reference temperatures for the Salem Unit 1 reactor vessel beltline materials for 32 and 48EFPY.

TABLE 4-8
Summary of the Calculated Fluence Factors used for the Generation of the
32 EFPY Heatup and Cooldown Curves

Material	¼ T f (n/cm^2, E > 1.0 MeV)	¼ T FF[(a)]	¾ T f (n/cm^2, E >1.0 MeV)	¾ T FF[(b)]
Intermediate Shell B2402-1	9.77 x 10^{18}	0.993	3.47 x 10^{18}	0.708
Intermediate Shell B2402-2	9.77 x 10^{18}	0.993	3.47 x 10^{18}	0.708
Intermediate Shell B2402-3	9.77 x 10^{18}	0.993	3.47 x 10^{18}	0.708
Lower Shell B2403-1	9.77 x 10^{18}	0.993	3.47 x 10^{18}	0.708
Lower Shell B2403-2	9.77 x 10^{18}	0.993	3.47 x 10^{18}	0.708
Lower Shell B2403-3	9.77 x 10^{18}	0.993	3.47 x 10^{18}	0.708
Intermediate to Lower Shell Circumferential Weld Seam 9-042 (Heat # 13253)	9.77 x 10^{18}	0.993	3.47 x 10^{18}	0.708
Intermediate Shell Longitudinal Weld Seams 2-042 A & B (Heat # 39B196/34B009 +NI200)	7.03 x 10^{18}	0.901	2.50 x 10^{18}	0.620
Intermediate Shell Longitudinal Weld Seam 2-042 C (Heat # 39B196/34B009 +NI200)	4.08 x 10^{18}	0.750	1.45 x 10^{18}	0.495
Lower Shell Longitudinal Weld Seams 3-042 A&B (Heat # 34B009+NI200, 13253)	6.44 x 10^{18}	0.880	2.29 x 10^{18}	0.602
Lower Shell Longitudinal Weld Seam 3-042 C (Heat # 34B009+NI200, 13253)	9.77 x 10^{18}	0.993	3.47 x 10^{18}	0.708

Notes:
(a) Fluence Factor at the ¼T vessel thickness location.
(b) Fluence Factor at the ¾T vessel thickness location.

TABLE 4-9
Summary of the Calculated Fluence Factors used for the Generation of the
48 EFPY Heatup and Cooldown Curves

Material	¼ T f (n/cm², E > 1.0 MeV)	¼ T FF[(a)]	¾ T f (n/cm², E > 1.0 MeV)	¾ T FF[(b)]
Intermediate Shell B2402-1	1.44×10^{19}	1.101	5.12×10^{18}	0.813
Intermediate Shell B2402-2	1.44×10^{19}	1.101	5.12×10^{18}	0.813
Intermediate Shell B2402-3	1.44×10^{19}	1.101	5.12×10^{18}	0.813
Lower Shell B2403-1	1.44×10^{19}	1.101	5.12×10^{18}	0.813
Lower Shell B2403-2	1.44×10^{19}	1.101	5.12×10^{18}	0.813
Lower Shell B2403-3	1.44×10^{19}	1.101	5.12×10^{18}	0.813
Intermediate to Lower Shell Circumferential Weld Seam 9-042 (Heat #13253)	1.44×10^{19}	1.101	5.12×10^{18}	0.813
Intermediate Shell Longitudinal Weld Seams 2-042 A&B (Heat # 39B196/34B009 +NI200)	1.04×10^{19}	1.010	3.71×10^{18}	0.726
Intermediate Shell Longitudinal Weld Seam 2-042 C (Heat # 39B196/34B009 +NI200)	6.14×10^{18}	0.860	2.18×10^{18}	0.590
Lower Shell Longitudinal Weld Seams 3-042 A&B (Heat # 34B009+NI200, 13253)	9.60×10^{18}	1.00	3.41×10^{18}	0.700
Lower Shell Longitudinal Weld Seam 3-042 C (Heat # 34B009+NI200, 13253)	1.44×10^{19}	1.101	5.12×10^{18}	0.813

Notes:
(a) Fluence Factor at the ¼T vessel thickness location.
(b) Fluence Factor at the ¾T vessel thickness location.

Contained in Tables 4-10 through 4-13 are the calculations of the ART values used for the generation of the 32 EFPY and 48 EFPY heatup and cooldown curves.

TABLE 4-10

Calculation of the ART Values for the ¼T Location @ 32 EFPY

Material	CF (°F)	FF	IRT_{NDT}[a] (°F)	ΔRT_{NDT}[b] (°F)	Margin (°F)	ART[c] (°F)
Intermediate Shell B2402-1	161.9	0.993	45	160.8	34	240
→ Using Surveillance Data	153.6	0.993	45	152.5	17	215
Intermediate Shell B2402-2	161.9	0.993	-5	160.8	34	190
→ Using Surveillance Data	140.4	0.993	-5	139.4	17	151
Intermediate Shell B2402-3	148.9	0.993	-3	147.9	34	179
→ Using Surveillance Data	106.1	0.993	-3	105.4	17	119
Lower Shell B2403-1	128.8	0.993	4	127.9	34	166
Lower Shell B2403-2	129.9	0.993	18	129.0	34	181
Lower Shell B2403-3	128.8	0.993	6	127.9	34	168
Intermediate to Lower Shell Circumferential Weld Seam 9-042 (Heats:13253)	188.5	0.993	-56	187.2	65.5	197
Intermediate Shell Longitudinal Weld Seams 2-042 A&B (Heat # 39B196/34B009 +NI200)	217.2	0.901	-56	195.7	65.5	205
Intermediate Shell Longitudinal Weld Seam 2-042 C (Heat # 39B196/34B009 +NI200)	217.2	0.750	-56	162.8	65.5	172
Lower Shell Longitudinal Weld Seams 3-042 A&B (Heat # 34B009+NI200, 13253)	223.6	0.880	-56	196.8	65.5	206
Lower Shell Longitudinal Weld Seam 3-042 C (Heat # 34B009+NI200, 13253)	223.6	0.993	-56	222.0	65.5	**232**

NOTES:

(a) Initial RT_{NDT} values are measured values for the plate material and generic mean values for the weld metal.

(b) $\Delta RT_{NDT} = CF * FF$

(c) $ART = I + \Delta RT_{NDT} + M$ (This value was rounded per ASTM E29, using the "Rounding Method".)

TABLE 4-11

Calculation of the ART Values for the ¾T Location @ 32 EFPY

Material	CF (°F)	FF	IRT_{NDT}[(a)] (°F)	ΔRT_{NDT}[(b)] (°F)	Margin (°F)	ART[(c)] (°F)
Intermediate Shell B2402-1	161.9	0.708	45	114.6	34	194
→ Using Surveillance Data	153.6	0.708	45	108.7	17	171
Intermediate Shell B2402-2	161.9	0.708	-5	114.6	34	144
→ Using Surveillance Data	140.4	0.708	-5	99.4	17	113
Intermediate Shell B2402-3	148.9	0.708	-3	105.4	34	136
→ Using Surveillance Data	106.1	0.708	-3	75.1	17	89
Lower Shell B2403-1	128.8	0.708	4	91.2	34	129
Lower Shell B2403-2	129.9	0.708	18	92.0	34	144
Lower Shell B2403-3	128.8	0.708	6	91.2	34	131
Intermediate to Lower Shell Circumferential Weld Seam 9-042 (Heat # 13253)	188.5	0.708	-56	133.5	65.5	143
Intermediate Shell Longitudinal Weld Seams 2-042 A&B (Heat # 39B196/34B009 +NI200)	217.2	0.620	-56	134.7	65.5	144
Intermediate Shell Longitudinal Weld Seam 2-042 C (Heat # 39B196/34B009 +NI200)	217.2	0.500	-56	108.6	65.5	118
Lower Shell Longitudinal Weld Seams 3-042 A&B (Heat # 34B009+NI200, 13253)	223.6	0.600	-56	134.2	65.5	144
Lower Shell Longitudinal Weld Seam 3-042 C (Heat # 34B009+NI200, 13253)	223.6	0.708	-56	158.3	65.5	168

NOTES:

(a) Initial RT_{NDT} values are measured values for the plate material and generic mean values for the weld metal.

(d) ΔRT_{NDT} = CF * FF

(c) ART = I + ΔRT_{NDT} + M (This value was rounded per ASTM E29, using the "Rounding Method".)

TABLE 4-12

Calculation of the ART Values for the ¼T Location @ 48 EFPY

Material	CF (°F)	FF	IRT_{NDT}[(a)] (°F)	ΔRT_{NDT}[(b)] (°F)	Margin (°F)	ART[(c)] (°F)
Intermediate Shell B2402-1	161.9	1.101	45	178.3	34	257
→ Using Surveillance Data	153.6	1.101	45	169.1	17	231
Intermediate Shell B2402-2	161.9	1.101	-5	178.3	34	207
→ Using Surveillance Data	140.4	1.101	-5	154.6	17	167
Intermediate Shell B2402-3	148.9	1.101	-3	163.9	34	195
→ Using Surveillance Data	106.1	1.101	-3	116.8	17	131
Lower Shell B2403-1	128.8	1.101	4	141.8	34	180
Lower Shell B2403-2	129.9	1.101	18	143.0	34	195
Lower Shell B2403-3	128.8	1.101	6	141.8	34	182
Intermediate to Lower Shell Circumferential Weld Seam 9-042 (Heats # 13253)	188.5	1.101	-56	207.5	65.5	217
Intermediate Shell Longitudinal Weld Seams 2-042 A&B (Heat # 39B196/34B009 +NI200)	217.2	1.010	-56	219.4	65.5	229
Intermediate Shell Longitudinal Weld Seam 2-042 C (Heat # 39B196/34B009 +NI200)	217.2	0.860	-56	186.8	65.5	196
Lower Shell Longitudinal Weld Seam 3-042 A&B (Heat # 34B009+NI200, 13253)	223.6	1.000	-56	223.6	65.5	233
Lower Shell Longitudinal Weld Seam 3-042 C (Heat # 34B009+NI200, 13253)	223.6	1.101	-56	246.2	65.5	**256**

Notes:

(a) Initial RT_{NDT} values are measured values for the plate material and generic mean values for the weld metal.

(b) $\Delta RT_{NDT} = CF * FF$

(c) ART = Initial RT_{NDT} + ΔRT_{NDT} + Margin (°F) ; (Rounded per ASTM E29, using the "Rounding Method")

TABLE 4-13

Calculation of the ART Values for the ¾T Location @ 48 EFPY

Material	CF (°F)	FF	IRT_{NDT}[(a)] (°F)	ΔRT_{NDT}[(b)] (°F)	Margin (°F)	ART[(c)] (°F)
Intermediate Shell B2402-1	161.9	0.813	45	131.6	34	211
→ Using Surveillance Data	153.6	0.813	45	124.9	17	187
Intermediate Shell B2402-2	161.9	0.813	-5	131.6	34	161
→ Using Surveillance Data	140.4	0.813	-5	114.1	17	126
Intermediate Shell B2402-3	148.9	0.813	-3	121.1	34	152
→ Using Surveillance Data	106.1	0.813	-3	86.3	17	100
Lower Shell B2403-1	128.8	0.813	4	104.7	34	143
Lower Shell B2403-2	129.9	0.813	18	105.6	34	158
Lower Shell B2403-3	128.8	0.813	6	104.7	34	145
Intermediate to Lower Shell Circumferential Weld Seam 9-042 (Heat # 13253)	188.5	0.813	-56	153.3	65.5	163
Intermediate Shell Longitudinal Weld Seams 2-042 A&B (Heat # 39B196/34B009 +NI200)	217.2	0.726	-56	157.7	65.5	167
Intermediate Shell Longitudinal Weld Seam 2-042 C (Heat # 39B196/34B009 +NI200)	217.2	0.590	-56	128.1	65.5	138
Lower Shell Longitudinal Weld Seams 3-042 A&B (Heat # 34B009+NI200, 13253)	223.6	0.700	-56	156.5	65.5	166
Lower Shell Longitudinal Weld Seam 3-042 C (Heat # 34B009+NI200, 13253)	223.6	0.813	-56	181.8	65.5	**191**

Notes:

(a) Initial RT_{NDT} values are measured values for the plate material and generic mean values for the weld metal.

(b) $\Delta RT_{NDT} = CF * FF$

(c) ART = Initial RT_{NDT} + ΔRT_{NDT} + Margin (°F) ; (Rounded per ASTM E29, using the "Rounding Method")

The longitudinal weld seam 3-042 C is the limiting beltline material for the ¼T case at 32 EFPY and the ¼T and ¾T cases at 48 EFPY. However, the intermediate shell plate B2402-1 using credible surveillance capsule data is limiting for the ¾T 32 EFPY case. Contained in Table 4-14 is a summary of the limiting ARTs to be used in the generation of the Salem Unit 1 reactor vessel heatup and cooldown curves.

TABLE 4-14
Summary of the Limiting ART Values Used in the
Generation of the Salem Unit 1 Heatup/Cooldown Curves

EFPY	1/4T Limiting ART	3/4T Limiting ART
32	232	171
48	256	191

5 HEATUP AND COOLDOWN PRESSURE-TEMPERATURE LIMIT CURVES

Pressure-temperature limit curves for normal heatup and cooldown of the primary reactor coolant system have been calculated for the pressure and temperature in the reactor vessel beltline region using the methods discussed in Sections 3 and 4 of this report. This approved methodology is also presented in WCAP-14040-NP-A[8], dated January 1996.

Figures 5-1 and 5-3 present the heatup curves with margins of 18°F and 61psig for instrumentation errors for heatup rates of 60 and 100°F/hr. These curves are applicable for 32 EFPY and 48 EFPY, respectively, for the Salem Unit 1 reactor vessel. Additionally, Figures 5-2 and 5-4 present the cooldown curves with margins of 18°F and 61 psig for instrumentation errors for cooldown rates of 0, 20, 40, 60, and 100°F/hr. These curves are also applicable for 32 EFPY and 48 EFPY, respectively, for the Salem Unit 1 reactor vessel. Figures 5-1 through 5-4 include the boltup temperature of 60°F with a margin of 2°F for measurement uncertainty. Allowable combinations of temperature and pressure for specific temperature change rates are below and to the right of the limit lines shown in Figures 5-1 through 5-4. This is in addition to other criteria which must be met before the reactor is made critical, as discussed in the following paragraphs.

The reactor must not be made critical until pressure-temperature combinations are to the right of the criticality limit line shown in Figures 5-1 and 5-3 (for the specific heatup rate being utilized). The straight-line portion of the criticality limit is at the minimum permissible temperature for the 2485 psig inservice hydrostatic test as required by Appendix G to 10 CFR Part 50. The governing equation for the hydrostatic test is defined in Appendix G to Section XI of the ASME Code[3] as follows:

$$1.5K_{Im} < K_{Ic} \qquad (10)$$

where,

K_{Im} is the stress intensity factor covered by membrane (pressure) stress,

$K_{Ic} = 33.2 + 20.734e^{[0.02(T - RT_{NDT})]}$,

T is the minimum permissible metal temperature, and

RT_{NDT} is the metal reference nil-ductility temperature

The criticality limit curve specifies pressure-temperature limits for core operation to provide additional margin during actual power production as specified in Reference 2. The pressure-temperature limits for core operation (except for low power physics tests) are that the reactor vessel must be at a temperature equal to or higher than the minimum temperature required for the inservice hydrostatic test, and at least 40°F higher than the minimum permissible temperature in the corresponding pressure-temperature curve for heatup and cooldown calculated as described in Section 3 of this report. The vertical line drawn from these points on the pressure-temperature curve, intersecting a curve 40°F higher than the pressure-temperature limit curve, constitutes the limit for core operation for the reactor vessel.

Figures 5-1 through 5-4 define all of the above limits for ensuring prevention of nonductile failure for the Salem Unit 1 reactor vessel. The data points for these heatup and cooldown pressure-temperature limit curves are presented in Tables 5-1 through 5-4, respectively.

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: Lower Shell Axial Weld Seam 3-042C (¼T)
Intermediate Shell Plate B2402-1 (¾T)

LIMITING ART VALUES AT 32 EFPY: ¼T ART = 232°F
¾T ART = 171°F

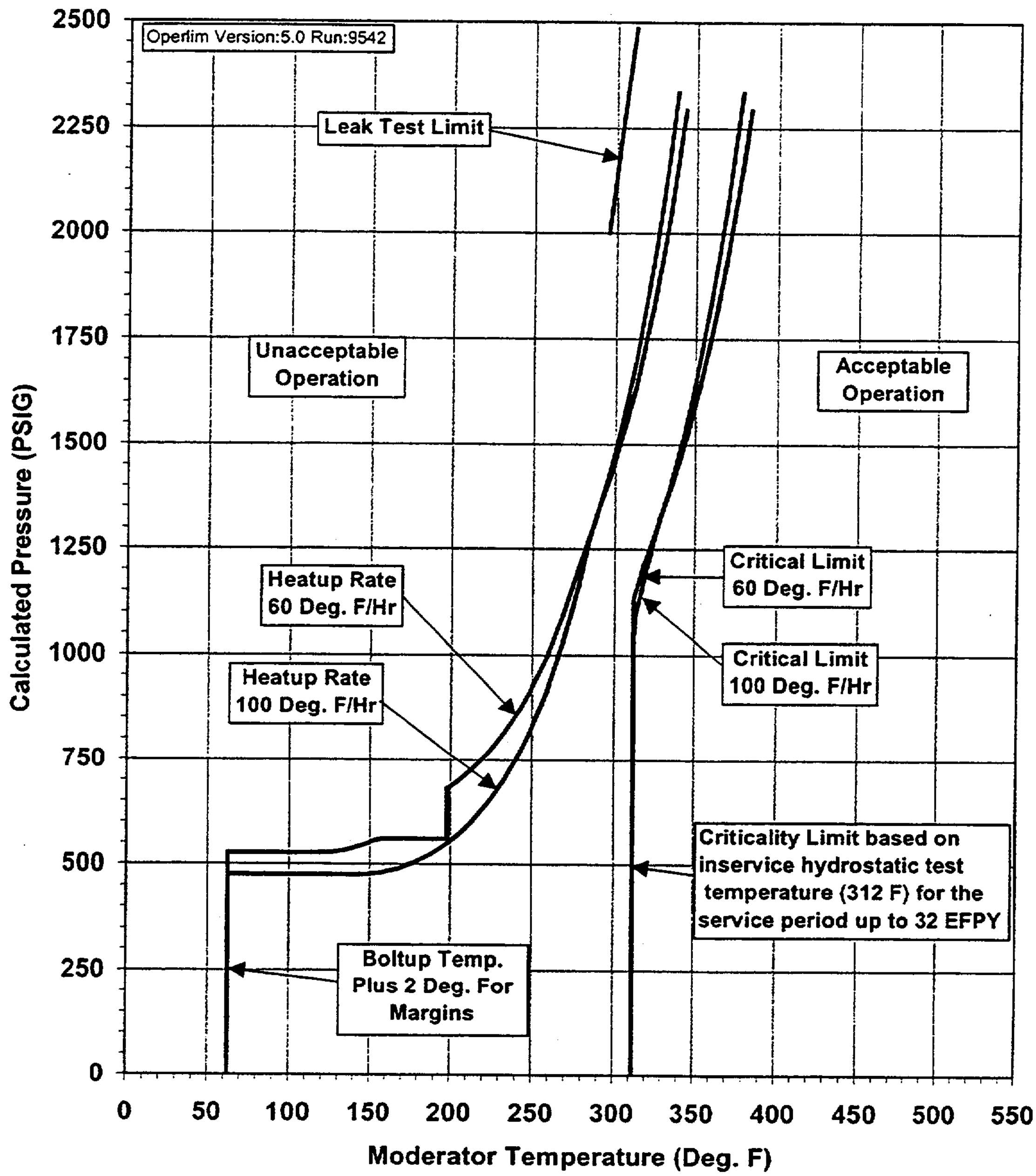


Figure 5-1. Salem Unit 1 Reactor Coolant System Heatup Limitations (Heatup Rates of 60 and 100°F/hr) Applicable to 32 EFPY (with Margins for Instrumentation Errors)

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: Lower Shell Axial Weld Seam 3-042C (¼T)
Intermediate Shell Plate B2402-1 (¾T)

LIMITING ART VALUES AT 32 EFPY: ¼T ART = 232°F
¾T ART = 171°F

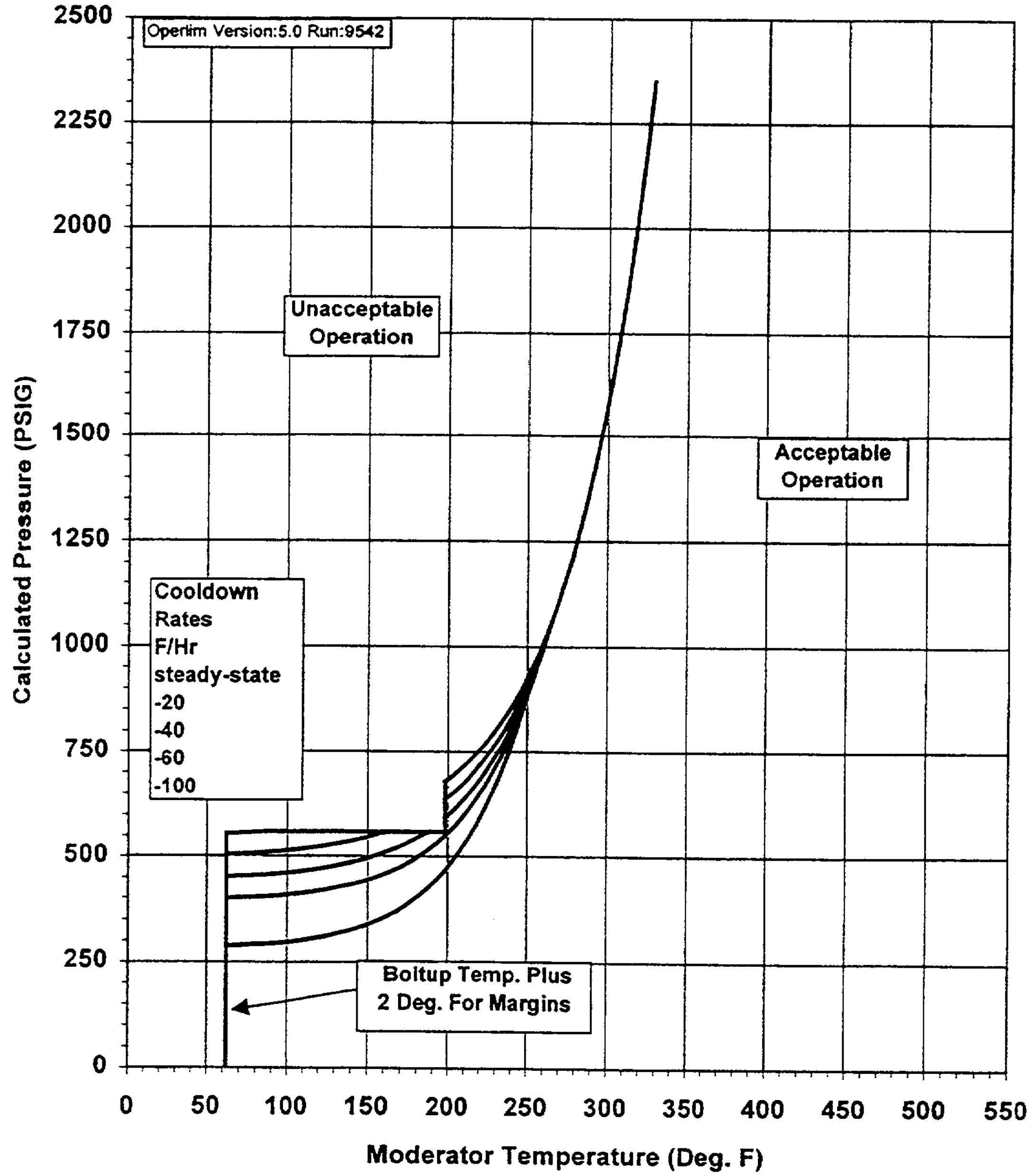


Figure 5-2. Salem Unit 1 Reactor System Cooldown Limitations (Cooldown Rates of 0, 20, 40, 60 and 100°F/hr) Applicable to 32 EFPY (with Margins for Instrumentation Errors)

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: Lower Shell Axial Weld Seam 3-042C

LIMITING ART VALUES AT 48 EFPY: ¼T ART = 256°F
¾T ART = 191°F

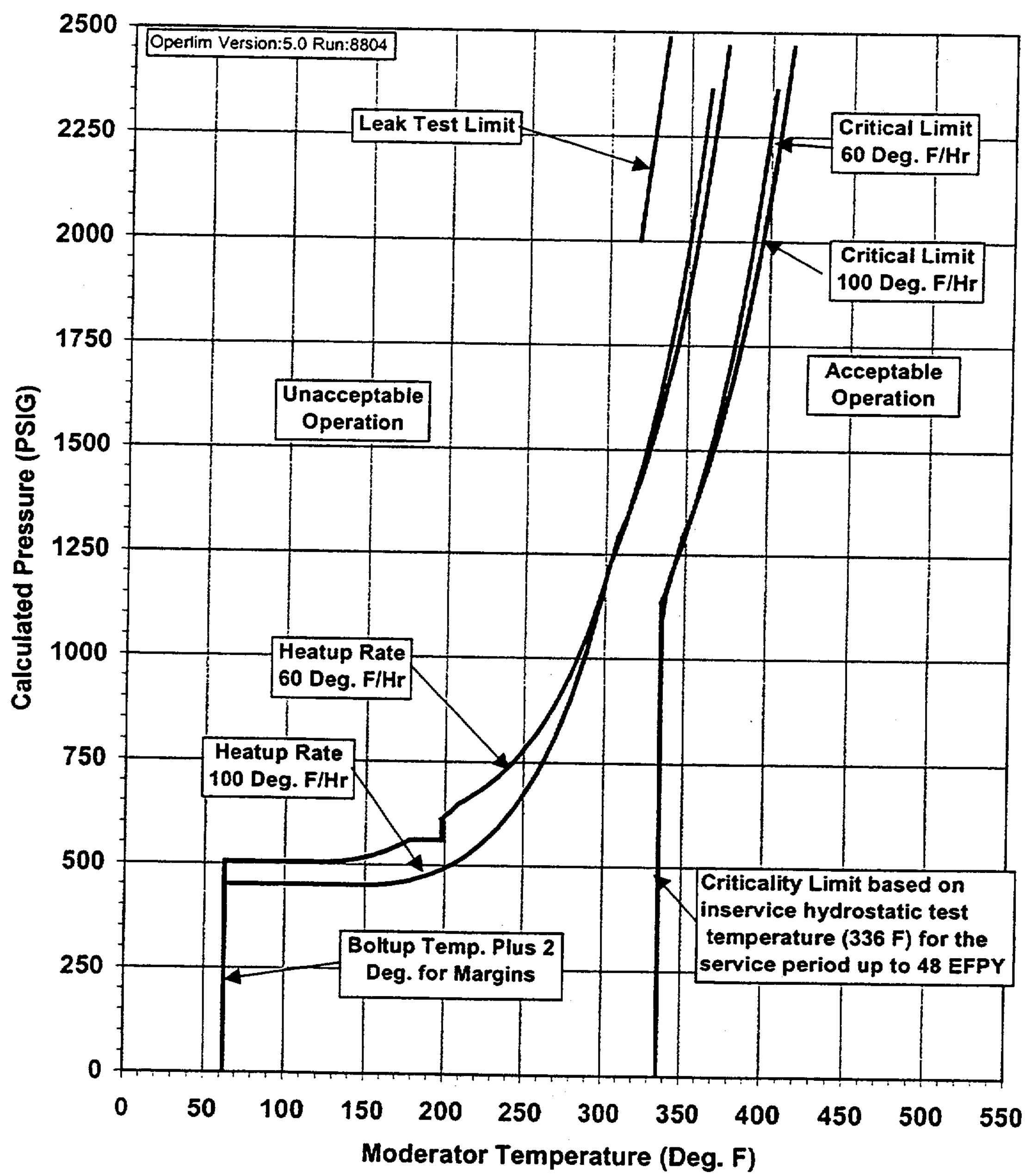


Figure 5- 3. Salem Unit 1 Reactor Coolant System Heatup Limitations (Heatup Rates of 60 and 100°F/hr) Applicable to 48 EFPY (with Margins for Instrumentation Errors)

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: Lower Shell Axial Weld Seam 3-042C

LIMITING ART VALUES AT 48 EFPY: ¼T ART = 256°F
¾T ART = 191°F

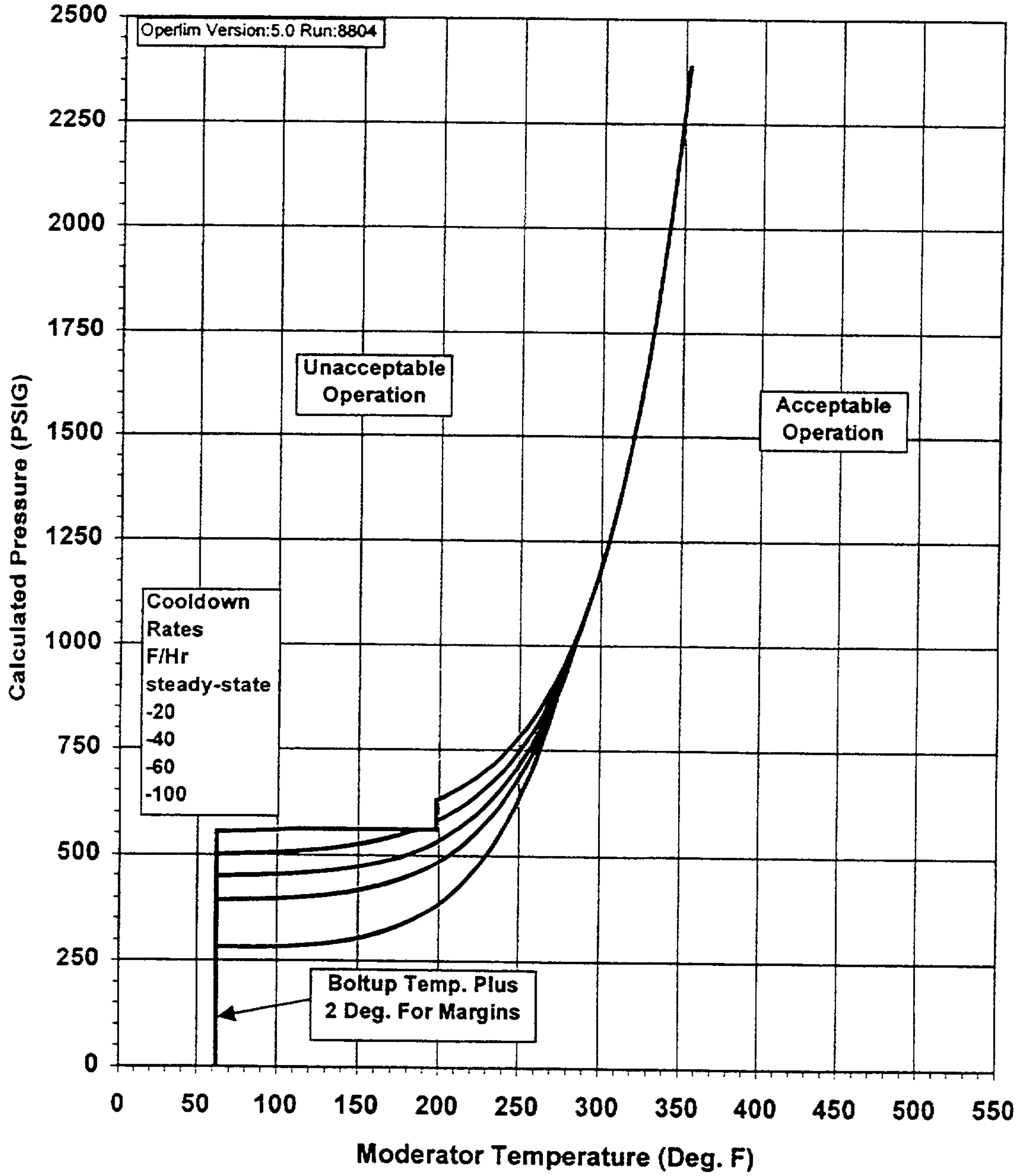


Figure 5- 4. Salem Unit 1 Reactor Coolant System Cooldown Limitations (Cooldown Rates of 0, 20, 40, 60, 100°F/hr) Applicable to 48 EFPY (with Margins for Instrumentation Errors)

TABLE 5-1
Salem Unit 1 Heatup Curve Data Points Applicable to 32 EFPY Using
1996 Appendix G and Code Case N-640
(w/Margins for instrumentation Errors and w/Flange Requirements)

60°F/hr. Heatup		60°F/hr. Crit. Limit		100°F/hr. Heatup		100°F/hr. Crit. Limit		Inservice Leak Test	
Temp.	Press.	Temp.	Press.	Temp.	Press.	Temp.	Press.	Temp.	Press.
(°F)	(psig)	(°F)	(psig)	(°F)	(psig)	(°F)	(psig)	(°F)	(psig)
62	0	312	0	62	0	312	0	295	2000
62	526	312	526	62	476	312	476	312	2485
78	526	312	526	78	476	312	476		
83	526	312	527	83	476	312	476		
88	526	312	528	88	476	312	478		
93	526	312	529	93	476	312	478		
98	526	312	532	98	476	312	481		
103	526	312	534	103	476	312	481		
108	526	312	536	108	476	312	485		
113	526	312	540	113	476	312	486		
118	526	312	541	118	476	312	490		
123	526	312	548	123	476	312	491		
128	528	312	549	128	476	312	496		
133	532	312	556	133	476	312	499		
138	536	312	560	138	476	312	504		
143	541	312	560	143	476	312	507		
148	548	312	560	148	476	312	513		
153	556	312	560	153	478	312	518		
158	560	312	560	158	481	312	523		
163	560	312	560	163	485	312	531		
168	560	312	560	168	490	312	535		
173	560	312	560	173	496	312	545		
178	560	312	560	178	504	312	549		
183	560	312	560	183	513	312	560		

TABLE 5-1 (continued)
Salem Unit 1 Heatup Curve Data Points Applicable to 32 EFPY Using
1996 Appendix G and Code Case N-640
(w/Margins for instrumentation Errors and w/Flange Requirements)

60°F/hr. Heatup		60°F/hr. Crit. Limit		100°F/hr. Heatup		100°F/hr. Crit. Limit		Inservice Leak Test	
Temp.	Press.	Temp.	Press.	Temp.	Press.	Temp.	Press.	Temp.	Press.
(°F)	(psig)	(°F)	(psig)	(°F)	(psig)	(°F)	(psig)	(°F)	(psig)
188	560	312	560	188	523	312	560		
193	560	312	695	193	535	312	560		
198	560	312	721	198	549	312	564		
198	560	312	749	203	564	312	582		
198	680	312	780	208	582	312	601		
203	694	312	815	213	601	312	623		
208	710	312	853	218	623	312	647		
213	727	312	895	223	647	312	674		
218	746	312	942	228	674	312	704		
223	767	312	976	233	704	312	737		
228	790	312	1000	238	737	312	774		
233	816	312	1027	243	774	312	815		
238	844	312	1057	248	815	312	860		
243	876	312	1090	253	860	312	910		
248	910	312	1126	258	910	312	965		
253	949	312	1126	263	965	312	1025		
258	991	313	1147	268	1025	313	1092		
263	1038	318	1210	273	1092	318	1166		
268	1089	323	1263	278	1166	323	1248		
273	1147	328	1320	283	1248	328	1325		
278	1210	333	1383	288	1325	333	1377		
283	1263	338	1452	293	1377	338	1434		
288	1320	343	1528	298	1434	343	1497		
293	1383	348	1612	303	1497	348	1566		

TABLE 5-1 (continued)
Salem Unit 1 Heatup Curve Data Points Applicable to 32 EFPY Using
1996 Appendix G and Code Case N-640
(w/Margins for instrumentation Errors and w/Flange Requirements)

60°F/hr. Heatup		60°F/hr. Crit. Limit		100°F/hr. Heatup		100°F/hr. Crit. Limit		Inservice Leak Test	
Temp.	Press.	Temp.	Press.	Temp.	Press.	Temp.	Press.	Temp.	Press.
(°F)	(psig)	(°F)	(psig)	(°F)	(psig)	(°F)	(psig)	(°F)	(psig)
298	1452	353	1705	308	1566	353	1642		
303	1528	358	1807	313	1642	358	1726		
308	1612	363	1920	318	1726	363	1818		
313	1705	368	2044	323	1818	368	1920		
318	1807	373	2182	328	1920	373	2032		
323	1920	378	2333	333	2032	378	2165		
328	2044			338	2156	383	2292		
333	2182			343	2292				
338	2333								

TABLE 5-2
Salem Unit 1 Cooldown Curve Data Points Applicable to 32 EFPY Using
1996 Appendix G and Code Case N-640
(w/Margins for instrumentation Errors and w/Flange Requirements)

Steady State		20°F/hr. Cooldown		40°F/hr. Cooldown		60°F/hr. Cooldown		100°F/hr. Cooldown	
Temp.	Press.	Temp.	Press.	Temp.	Press.	Temp.	Press.	Temp.	Press.
(°F)	(psig)	(°F)	(psig)	(°F)	(psig)	(°F)	(psig)	(°F)	(psig)
62	0	62	0	62	0	62	0	62	0
62	555	62	504	62	452	62	399	62	287
68	556	68	505	68	453	68	400	68	288
73	557	73	506	73	454	73	401	73	289
78	558	78	507	78	455	78	402	78	290
83	559	83	508	83	456	83	403	83	291
88	560	88	510	88	457	88	404	88	292
93	560	93	511	93	459	93	405	93	294
98	560	98	513	98	460	98	407	98	295
103	560	103	515	103	462	103	409	103	297
108	560	108	517	108	464	108	411	108	299
113	560	113	519	113	467	113	413	113	302
118	560	118	522	118	469	118	416	118	305
123	560	123	525	123	472	123	419	123	309
128	560	128	528	128	476	128	423	128	313
133	560	133	532	133	480	133	427	133	317
138	560	138	536	138	484	138	431	138	323
143	560	143	540	143	489	143	436	143	328
148	560	148	545	148	494	148	442	148	335
153	560	153	551	153	500	153	448	153	343
158	560	158	557	158	506	158	455	158	351
163	560	163	560	163	514	163	463	163	361
168	560	168	560	168	522	168	472	168	371
173	560	173	560	173	531	173	482	173	383

TABLE 5-2 (continued)
Salem Unit 1 Cooldown Curve Data Points Applicable to 32 EFPY Using
1996 Appendix G and Code Case N-640
(w/Margins for instrumentation Errors and w/Flange Requirements)

Steady State		20°F/hr. Cooldown		40°F/hr. Cooldown		60°F/hr. Cooldown		100°F/hr. Cooldown	
Temp.	Press.	Temp.	Press.	Temp.	Press.	Temp.	Press.	Temp.	Press.
(°F)	(psig)	(°F)	(psig)	(°F)	(psig)	(°F)	(psig)	(°F)	(psig)
178	560	178	560	178	541	178	493	178	397
183	560	183	560	183	552	183	505	183	412
188	560	188	560	188	560	188	519	188	429
193	560	193	560	193	560	193	534	193	447
198	560	198	560	198	560	198	551	198	468
198	560	198	560	198	560	203	569	203	491
198	680	198	637	198	593	208	590	208	517
203	694	203	652	203	610	213	613	213	545
208	710	208	669	208	629	218	639	218	577
213	727	213	688	213	650	223	667	223	613
218	746	218	709	218	673	228	699	228	652
223	767	223	732	223	699	233	734	233	696
228	790	228	758	228	727	238	772	238	744
233	816	233	786	233	758	243	815	243	798
238	844	238	817	238	793	248	863	248	858
243	876	243	852	243	832	253	915	253	924
248	910	248	890	248	874	258	974	258	991
253	949	253	933	253	922	263	1038	263	1038
258	991	258	980	258	974	268	1089	268	1089
263	1038	263	1032	263	1032	273	1147	273	1147
268	1089	268	1089	268	1089	278	1210	278	1210
273	1147	273	1147	273	1147	283	1280	283	1280
278	1210	278	1210	278	1210	288	1357	288	1357
283	1280	283	1280	283	1280	293	1442	293	1442

TABLE 5-2 (continued)
Salem Unit 1 Cooldown Curve Data Points Applicable to 32 EFPY Using
1996 Appendix G and Code Case N-640
(w/Margins for instrumentation Errors and w/Flange Requirements)

Steady State		20°F/hr. Cooldown		40°F/hr. Cooldown		60°F/hr. Cooldown		100°F/hr. Cooldown	
Temp.	Press.	Temp.	Press.	Temp.	Press.	Temp.	Press.	Temp.	Press.
(°F)	(psig)	(°F)	(psig)	(°F)	(psig)	(°F)	(psig)	(°F)	(psig)
288	1357	288	1357	288	1357	298	1536	298	1536
293	1442	293	1442	293	1442	303	1640	303	1640
298	1536	298	1536	298	1536	308	1755	308	1755
303	1640	303	1640	303	1640	313	1883	313	1883
308	1755	308	1755	308	1755	318	2023	318	2023
313	1883	313	1883	313	1883	323	2179	323	2179
318	2023	318	2023	318	2023	328	2350	328	2350
323	2179	323	2179	323	2179				
328	2350	328	2350	328	2350				

TABLE 5-3
Salem Unit 1 Heatup Curve Data Points Applicable to 48 EFPY Using
1996 Appendix G and Code Case N-640
(w/Margins for instrumentation Errors and w/Flange Requirements)

60°F/hr. Heatup		60°F/hr. Crit. Limit		100°F/hr. Heatup		100°F/hr. Crit. Limit		Inservice Leak Test	
Temp.	Press.	Temp.	Press.	Temp.	Press.	Temp.	Press.	Temp.	Press.
(°F)	(psig)	(°F)	(psig)	(°F)	(psig)	(°F)	(psig)	(°F)	(psig)
62	0	336	0	62	0	336	0	319	2000
62	506	336	506	62	450	336	450	336	2485
78	506	336	506	78	450	336	451		
83	506	336	507	83	450	336	451		
88	506	336	507	88	450	336	452		
93	506	336	509	93	450	336	452		
98	506	336	509	98	450	336	455		
103	506	336	512	103	450	336	455		
108	506	336	513	108	450	336	458		
113	506	336	516	113	450	336	458		
118	506	336	518	118	450	336	462		
123	506	336	520	123	450	336	462		
128	506	336	525	128	450	336	467		
133	507	336	526	133	450	336	468		
138	509	336	533	138	450	336	473		
143	512	336	535	143	450	336	475		
148	516	336	540	148	450	336	480		
153	520	336	548	153	450	336	483		
158	526	336	549	158	451	336	489		
163	533	336	559	163	452	336	493		
168	540	336	560	168	455	336	499		
173	549	336	560	173	458	336	504		
178	559	336	560	178	462	336	510		
183	560	336	560	183	467	336	517		

TABLE 5-3 (continued)
Salem Unit 1 Heatup Curve Data Points Applicable to 48 EFPY Using
1996 Appendix G and Code Case N-640
(w/Margins for instrumentation Errors and w/Flange Requirements)

60°F/hr. Heatup		60°F/hr. Crit. Limit		100°F/hr. Heatup		100°F/hr. Crit. Limit		Inservice Leak Test	
Temp.	Press.	Temp.	Press.	Temp.	Press.	Temp.	Press.	Temp.	Press.
(°F)	(psig)	(°F)	(psig)	(°F)	(psig)	(°F)	(psig)	(°F)	(psig)
188	560	336	560	188	473	336	522		
193	560	336	611	193	480	336	532		
198	560	336	627	198	489	336	536		
198	560	336	646	203	499	336	549		
198	611	336	667	208	510	336	552		
203	627	336	690	213	522	336	560		
208	646	336	715	218	536	336	560		
213	658	336	743	223	552	336	569		
218	670	336	775	228	569	336	589		
223	683	336	809	233	589	336	611		
228	697	336	847	238	611	336	635		
233	713	336	889	243	635	336	662		
238	731	336	935	248	662	336	692		
243	750	336	960	253	692	336	725		
248	771	336	983	258	725	336	761		
253	795	336	1008	263	761	336	801		
258	821	336	1035	268	801	336	846		
263	850	336	1066	273	846	336	895		
268	882	336	1099	278	895	336	950		
273	918	336	1136	283	950	336	1010		
278	957	336	1176	288	1010	336	1076		
283	1000	338	1221	293	1076	338	1149		
288	1048	343	1270	298	1149	343	1230		
293	1100	348	1324	303	1223	348	1319		

TABLE 5-3 (continued)
Salem Unit 1 Heatup Curve Data Points Applicable to 48 EFPY Using
1996 Appendix G and Code Case N-640
(w/Margins for instrumentation Errors and w/Flange Requirements)

60°F/hr. Heatup		60°F/hr. Crit. Limit		100°F/hr. Heatup		100°F/hr. Crit. Limit		Inservice Leak Test	
Temp.	Press.	Temp.	Press.	Temp.	Press.	Temp.	Press.	Temp.	Press.
(°F)	(psig)	(°F)	(psig)	(°F)	(psig)	(°F)	(psig)	(°F)	(psig)
298	1159	353	1333	308	1290	353	1337		
303	1223	358	1396	313	1337	358	1390		
308	1275	363	1466	318	1390	363	1447		
313	1333	368	1544	323	1447	368	1511		
318	1396	373	1629	328	1511	373	1580		
323	1466	378	1723	333	1580	378	1657		
328	1544	383	1827	338	1657	383	1742		
333	1629	388	1941	343	1742	388	1836		
338	1723	393	2068	348	1836	393	1938		
343	1827	398	2207	353	1938	398	2052		
348	1941	403	2361	358	2052	403	2177		
353	2068			363	2177	408	2315		
358	2207			368	2315	413	2465		
363	2361			373	2465				

TABLE 5-4
Salem Unit 1 Cooldown Curve Data Points Applicable to 48 EFPY Using
1996 Appendix G and Code Case N-640
(w/Margins for instrumentation Errors and w/Flange Requirements)

Steady State		20°F/hr. Cooldown		40°F/hr. Cooldown		60°F/hr. Cooldown		100°F/hr. Cooldown	
Temp.	Press.	Temp.	Press.	Temp.	Press.	Temp.	Press.	Temp.	Press.
(°F)	(psig)	(°F)	(psig)	(°F)	(psig)	(°F)	(psig)	(°F)	(psig)
62	0	62	0	62	0	62	0	62	0
62	553	62	500	62	448	62	394	62	283
68	553	68	501	68	448	68	394	68	283
73	554	73	501	73	449	73	395	73	283
78	554	78	502	78	449	78	395	78	283
83	554	83	503	83	450	83	396	83	283
88	555	88	504	88	451	88	396	88	283
93	556	93	504	93	451	93	397	93	283
98	557	98	505	98	452	98	397	98	284
103	558	103	506	103	453	103	398	103	285
108	560	108	508	108	454	108	399	108	285
113	560	113	509	113	455	113	401	113	287
118	560	118	510	118	457	118	402	118	288
123	560	123	512	123	459	123	404	123	290
128	560	128	514	128	460	128	406	128	292
133	560	133	516	133	463	133	408	133	294
138	560	138	519	138	465	138	410	138	297
143	560	143	521	143	468	143	413	143	300
148	560	148	524	148	471	148	416	148	304
153	560	153	528	153	474	153	420	153	308
158	560	158	531	158	478	158	424	158	313
163	560	163	535	163	483	163	429	163	318
168	560	168	540	168	487	168	434	168	324
173	560	173	545	173	493	173	440	173	331

TABLE 5-4 (continued)
Salem Unit 1 Cooldown Curve Data Points Applicable to 48 EFPY Using
1996 Appendix G and Code Case N-640
(w/Margins for instrumentation Errors and w/Flange Requirements)

Steady State		20°F/hr. Cooldown		40°F/hr. Cooldown		60°F/hr. Cooldown		100°F/hr. Cooldown	
Temp.	Press.	Temp.	Press.	Temp.	Press.	Temp.	Press.	Temp.	Press.
(°F)	(psig)	(°F)	(psig)	(°F)	(psig)	(°F)	(psig)	(°F)	(psig)
178	560	178	551	178	499	178	446	178	339
183	560	183	557	183	506	183	454	183	348
188	560	188	560	188	513	188	462	188	358
193	560	193	560	193	521	193	471	193	369
198	560	198	560	198	531	198	481	198	381
198	560	198	560	203	541	203	492	203	395
198	629	198	580	208	552	208	505	208	410
203	638	203	589	213	565	213	519	213	428
208	647	208	600	218	579	218	534	218	447
213	658	213	612	223	595	223	552	223	468
218	670	218	624	228	612	228	571	228	492
223	683	223	639	233	631	233	592	233	519
228	697	228	654	238	653	238	616	238	548
233	713	233	672	243	676	243	642	243	581
238	731	238	691	248	703	248	671	248	617
243	750	243	712	253	732	253	703	253	658
248	771	248	736	258	764	258	739	258	703
253	795	253	762	263	799	263	779	263	753
258	821	258	791	268	839	268	823	268	808
263	850	263	823	273	883	273	872	273	870
268	882	268	859	278	931	278	926	278	938
273	918	273	898	283	984	283	986	283	1000
278	957	278	941	288	1044	288	1048	288	1048
283	1000	283	989	293	1100	293	1100	293	1100

TABLE 5-4 (continued)
Salem Unit 1 Cooldown Curve Data Points Applicable to 48 EFPY Using
1996 Appendix G and Code Case N-640
(w/Margins for instrumentation Errors and w/Flange Requirements)

Steady State		20°F/hr. Cooldown		40°F/hr. Cooldown		60°F/hr. Cooldown		100°F/hr. Cooldown	
Temp.	Press.	Temp.	Press.	Temp.	Press.	Temp.	Press.	Temp.	Press.
(°F)	(psig)	(°F)	(psig)	(°F)	(psig)	(°F)	(psig)	(°F)	(psig)
288	1048	288	1043	298	1159	298	1159	298	1159
293	1100	293	1100	303	1223	303	1223	303	1223
298	1159	298	1159	308	1294	308	1294	308	1294
303	1223	303	1223	313	1373	313	1373	313	1373
308	1294	308	1294	318	1460	318	1460	318	1460
313	1373	313	1373	323	1556	323	1556	323	1556
318	1460	318	1460	328	1662	328	1662	328	1662
323	1556	323	1556	333	1780	333	1780	333	1780
328	1662	328	1662	338	1910	338	1910	338	1910
333	1780	333	1780	343	2053	343	2053	343	2053
338	1910	338	1910	348	2212	348	2212	348	2212
343	2053	343	2053	353	2387	353	2387	353	2387
348	2212	348	2212						
353	2387	353	2387						

6 REFERENCES

1. Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials", U.S. Nuclear Regulatory Commission, May, 1988.

2. 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements", Federal Register, Volume 60, No. 243, dated December 19, 1995.

3. Section XI of the ASME Boiler and Pressure Vessel Code, Appendix G, "Fracture Toughness Criteria for Protection Against Failure.", Dated December 1995 through 1996 addendum.

4. WCAP-14635, "Analysis of Capsule S from the Public Service Electric and Gas Company Salem Unit 1 Reactor Vessel Radiation Program", P. A. Grendys, et. al., June 1996.

5. WCAP-14702, "Salem Unit 1 Heatup and Cooldown Limit Curves for Normal Operation", T. J. Laubham, April 1997.

6. 1989 Section III, Division 1 of the ASME Boiler and Pressure Vessel Code, Paragraph NB-2331, "Material for Vessels".

7. LTR-REA-00-621, "Reactor Vessel Fluences for the Salem 1.4% Uprate Project", S.L. Anderson, June 27, 2000.

8. WCAP-14040-NP-A, Revision 2, "Methodology used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves", J. D. Andrachek, et al., January 1996.

9. CE NPSD-1039, Rev. 2, "Best Estimate Copper and Nickel Values in CE Fabricated Reactor Vessel Welds, Appendix A, CE Reactor Vessel Weld Properties Database, Volume 1," CEOG Task 902, June 1997.

10. ASME Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves for Section XI, Division 1", February 26, 1999.

11. WCAP-15315, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation For Operating PWR and BWR Plants", W. Bamford, et.al., October 1999.

12. WCAP-15567, "Evaluation of Pressurized Thermal Shock for Salem Unit 1", E. Terek, et. al., November 2000.

APPENDIX A

PROJECTED UPPER SHELF ENERGY VALUES

FOR SALEM UNIT 1

APPENDIX A - PREDICTED EOL USE VALUES

Per Regulatory Guide 1.99, Revision 2, the Charpy upper-shelf energy is assumed to decrease as a function of fluence and copper content as indicated in Figure 2 of the guide when surveillance data is not used. Linear interpolation is permitted. In addition, if surveillance data is used, the decrease in upper-shelf energy may be obtained by plotting the reduced plant surveillance data on Figure 2 of the guide and fitting the data with a line drawn parallel to the existing lines as the upper bound of all the data. This line should be used in preference to the existing graph.

The EOL (32 EFPY) and license renewal (48 EFPY) USE values can be predicted using the ¼T fluence projections at 32 and 48 EFPY, the copper content of the beltline materials and/or the results of the capsules tested to date using Figure 2 in Regulatory Guide 1.99, Revision 2. The peak vessel clad/base metal interface fluence value was used to determine the EOL (32 EFPY) and license renewal (48 EFPY) USE values of all the beltline materials.

The Salem Unit 1 reactor vessel beltline region minimum thickness is 8.625 inches.

The calculation of the ¼T vessel fluence values at 32 EFPY for the beltline materials is contained in Table A-1.

The calculation of the EOL USE values at 32 EFPY for the beltline materials is contained in Table A-2.

The calculation of the ¼T vessel fluence value at 48 EFPY for the beltline materials is contained in Table A-3.

The calculation of the EOL USE values at 48 EFPY for the beltline materials is contained in Table A-4.

TABLE A-1

EOL (32 EFPY) ¼T Fluence Values for all the Salem Unit 1 Beltline Materials.

Material	f @ 32 EFPY[a]	¼T f @ 32 EFPY[b]
Intermediate Shell B2402-1	1.64	0.98
Intermediate Shell B2402-2	1.64	0.98
Intermediate Shell B2402-3	1.64	0.98
Lower Shell B2403-1	1.64	0.98
Lower Shell B2403-2	1.64	0.98
Lower Shell B2403-3	1.64	0.98
Intermediate to Lower Shell Circumferential Weld Seam 9-042 (Heat # 13253)	1.64	0.98
Longitudinal Weld Seams 2-042 A, B & C (Heat # 39B196/34B009 +NI200)	1.64	0.98
Longitudinal Weld Seams 3-042 A, B & C (Heat # 34B009+NI200, 13253)	1.64	0.98

Notes:

(a) f @ 32 EFPY is the 32 EFPY fluence at the clad/base metal interface ($x\ 10^{19}$ $n.cm^2$, E > 1.0 MeV).

(b) ¼T f @ 32 EFPY = f @ 32 EFPY * $e^{(-0.24*X)}$, ($x\ 10^{19}$ $n.cm^2$, E > 1.0 MeV) where X is the depth into the vessel wall (X = 0.25 * 8.625 inches = 2.15625 inches)

TABLE A-2

Salem 1 Predicted End-of-License (32 EFPY) USE Calculations for all the Beltline Region Materials

Material	**Weight % of Cu**	**¼T EOL Fluence (10^{19} n/cm^2, E>1.0MeV)**	**Unirradiated USE (ft-lb)**	**Projected USE Decrease (%)**	**Projected EOL USE (ft-lb)**
Intermediate Shell B2402-1	0.24	0.98	91	19	74
Intermediate Shell B2402-2	0.24	0.98	98	15	83
Intermediate Shell B2402-3	0.22	0.98	104	16	87
Lower Shell B2403-1	0.19	0.98	93	29	66
Lower Shell B2403-2	0.19	0.98	83	29	59
Lower Shell B2403-3	0.19	0.98	85	29	60
Intermediate to Lower Shell Circumferential Weld Seam 9-042 (Heat # 13253)	0.22	0.98	112	36	72
Longitudinal Weld Seams 2-042 A, B & C (Heat # 39B196/34B009 + NI200)	0.18	0.98	96.2	32	65
Longitudinal Weld Seams 3-042 A, B & C (Heat # 34B009+NI200, 13253)	0.19	0.98	112	32	76

TABLE A-3

EOL (48 EFPY) ¼T Fluence Values for all the Salem Unit 1 Beltline

Material	f @ 48 EFPY[a]	¼T f @ 48 EFPY[b]
Intermediate Shell B2402-1	2.42	1.44
Intermediate Shell B2402-2	2.42	1.44
Intermediate Shell B2402-3	2.42	1.44
Lower Shell B2403-1	2.42	1.44
Lower Shell B2403-2	2.42	1.44
Lower Shell B2403-3	2.42	1.44
Intermediate to Lower Shell Circumferential Weld Seam 9-042 (Heat # 13253)	2.42	1.44
Intermediate Shell Longitudinal Weld Seams 2-042 A, B & C (Heat # 39B196/34B009 +NI200)	2.42	1.44
Lower Shell Longitudinal Weld Seams 3-042 A, B & C (Heat # 34B009+NI200, 13253)	2.42	1.44

Notes:

(b) f @ 48 EFPY is the 48 EFPY fluence at the clad/base metal interface ($x\ 10^{19}$ n.cm^2, E > 1.0 MeV).

(b) ¼T f @ 48 EFPY = f @ 48 EFPY * $e^{(-0.24*X)}$, ($x\ 10^{19}$ n.cm^2, E > 1.0 MeV) where X is the depth into the vessel wall (X = 0.25 * 8.625 inches = 2.15625 inches)

TABLE A-4

Salem 1 Predicted Life Extension (48 EFPY) USE Calculations for all the Beltline Region Materials

Material	Weight % Cu	¼T Life Extension Fluence (x 10^{19} n/cm^2)	Unirradiated USE (ft-lb)	Projected USE Decrease (%)	Projected Life Extension USE (ft-lb)
Intermediate Shell B2402-1	0.24	1.44	91	21	72
Intermediate Shell B2402-2	0.24	1.44	98	16	82
Intermediate Shell B2402-3	0.22	1.44	104	17	86
Lower Shell B2403-1	0.19	1.44	93	32	63
Lower Shell B2403-2	0.19	1.44	83	32	56
Lower Shell B2403-3	0.19	1.44	85	32	58
Intermediate to Lower Shell Circumferential Weld Seam 9-042 (Heats:13253)	0.22	1.44	112	40	67
Longitudinal Weld Seam 2-042 (Heat: 39B196/34B009 +NI200)	0.18	1.44	96.2	35	63
Longitudinal Weld Seam 3-042 (Heat: 34B009+NI200, 13253)	0.19	1.44	112	35	73