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Nuclear LLC

NOV 1 0 2000

LR-N00-0387
LCR S00-06

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

Gentlemen:

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

In accordance with 10CFR50.90, PSEG Nuclear LLC hereby requests a change to Facility Operating License Nos. DPR-70 and DPR-75 and to the Technical Specifications (TS) in Appendix A thereto for Salem Generating Station Unit Nos. 1 and 2, respectively. Pursuant to the requirements of 10CFR50.91(b)(1), a copy of this request for amendment has been sent to the State of New Jersey.

The proposed license amendment would increase the licensed core power level for operation to 3459 megawatts, 1.4% greater than the current level. PSEG Nuclear's request is based on reduced uncertainty in core thermal power measurement achieved with the CE Nuclear Power LLC (CENP) Crossflow ultrasonic flow measurement system. CENP topical report CENPD-397-P-A documents the theory, design and operating features of the Crossflow system and its ability to achieve increased accuracy in flow measurement. In a safety evaluation dated March 20, 2000, the NRC approved CENPD-397-P-A for referencing in license applications for power uprate.

An additional change is proposed to remove historical information from the Unit 1 Facility Operating License. The information relates to one-time requirements not applicable to operation at the proposed power level. Editorial changes are also being made to the TS Bases for TS affected by the proposed change.

The proposed change has been evaluated in accordance with 10CFR50.91(a)(1), using the criteria in 10CFR50.92(c), and it has been determined that this request involves no significant hazards considerations.

PSEG Nuclear has reviewed the proposed License Change Request (LCR) against the criteria of 10 CFR 51.22 for environmental considerations. The proposed changes do

ATTACHMENT 7 OF THIS LETTER CONTAINS PROPRIETARY INFORMATION
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not involve a significant hazards consideration, a significant change in the types or a significant increase in the amounts of effluents that may be released offsite, or a significant increase in individual or cumulative occupational radiation exposure. Based on the foregoing, PSEG Nuclear concludes that the proposed change meets the criteria delineated in 10 CFR 51.22(c)(9) for a categorical exclusion from the requirements for an Environmental Impact Statement.

A description of the requested change, the reason for the changes, and the justification for the changes are provided in Attachment 1. The basis for the no significant hazards consideration determination is provided in Attachment 2. The marked up Facility Operating License (FOL) and Technical Specification pages affected by the proposed changes are provided in Attachments 3 and 4.

PSEG Nuclear also requests two exemptions from the requirements of 10 CFR 50.60(a) and 10 CFR 50 Appendix G for use of the following documents as alternatives to requirements described in Appendix G:

- American Society of Mechanical Engineers Code Case N-640, "Alternative Reference Fracture Toughness for Development of P/T Limit Curves for ASME Section XI, Division I," and
- WCAP-15315, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Operating PWR and BWR Plants."

The requests for exemption are provided in Attachments 5 and 6.

Attachment 7 contains an analysis performed by Westinghouse Electric Company LLC of the power calorimetric uncertainty for the 1.4% uprate. Attachment 8 is an application and affidavit by Westinghouse for withholding proprietary information contained in Attachment 7 from public disclosure in accordance with 10 CFR 2.790. Westinghouse is the owner of the information for which withholding is requested. A non-proprietary version of Attachment 7 is provided as Attachment 9 to this letter.

The following regulatory commitments are being made in connection with this proposed change:

1. An impact study including grid stability analysis will be completed before implementation of the proposed change.
2. Operator actions to be taken when the Crossflow system is inoperable will be addressed in procedural guidance as described in section 1.4.2 of Attachment 1 to this request.

PSEG Nuclear requests that approval be provided by May 10, 2001. Upon NRC approval of this proposed change, PSEG Nuclear requests that the amendment be made effective on the date of issuance, but allow an implementation period of sixty days to provide sufficient time for associated administrative activities.

Should you have any questions regarding this request, please contact Mr. Paul Duke at (856) 339-1466.

Sincerely,

D. F. Garchow

D. F. Garchow
Vice President - Technical Support

Affidavit
Attachments (9)

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Records Management (N21)
Microfilm Copy
File Nos. 1.2.1 (Salem)
2.3 (LCR S00-06)

REF: LR-N00-0387
LCR S00-06

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

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

I am Vice President - Technical Support 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.

Subscribed and Sworn to before me
this 10 day of November, 2000

Notary Public of New Jersey

My Commission expires on 10/13/2002

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
CHANGE TO TECHNICAL SPECIFICATIONS
INCREASED LICENSED POWER LEVEL

DESCRIPTION OF THE PROPOSED CHANGE:

The proposed license amendment would revise the Salem Generating Station Unit Nos. 1 and 2 Facility Operating Licenses and Technical Specification to increase licensed power level for operation to 3459 MWt, 1.4% greater than the current level. The proposed changes are indicated on the marked up pages in Attachments 3 and 4 and 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 provided previously 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.

Along with the proposal to increase licensed power level to 3459 MWt, PSEG Nuclear also proposes continued use of the topical reports identified in Technical Specification section 6.9.1.9.b. These reports describe the NRC approved methods which support the Salem safety analyses. In many of these topical reports, reference is made to the use of a 2% uncertainty for reactor power, consistent with 10 CFR 50 Appendix K. PSEG Nuclear proposes that these topical reports be approved for use consistent with this amendment request.

REASON FOR THE PROPOSED CHANGE:

Salem Units 1 and 2 are currently licensed to operate at a maximum power level of 3411 MWt. The current licensed power level includes a 2% margin in the ECCS evaluation model to allow for uncertainties in core thermal power measurement. The 2% margin was required by 10 CFR 50, Appendix K. The NRC recently revised

Appendix K to permit licensees to use an assumed power level less than 1.02 times the licensed power level, provided the new power level is demonstrated to account for uncertainties due to power level instrumentation error. The revised requirements were issued June 1, 2000 (65 FR 34913) with an effective date of July 31, 2000.

PSEG Nuclear will install Crossflow ultrasonic flow measurement (UFM) systems for feedwater flow measurement in Salem Units 1 and 2 before implementation of the proposed uprate. Use of the Crossflow UFM system will reduce core power measurement uncertainty to less than 0.6 percent. Based on this, PSEG Nuclear proposes to reduce the power measurement uncertainty required by 10 CFR 50 Appendix K to permit an increase of 1.4 percent in the licensed power level. The reduction in power measurement uncertainty does not constitute a significant change to the emergency core cooling system (ECCS) evaluation model as defined in 10 CFR 50.46(a)(3)(i).

Uncertainty in feedwater flow measurement is the most significant contributor to core power measurement uncertainty. Use of the Crossflow UFM system provides a more accurate measurement of feedwater flow than the instrumentation currently installed in Salem. CENP topical report CENPD-397-P-A documents the theory, design and operating features of the Crossflow system and its ability to achieve increased accuracy of flow measurement. In a safety evaluation dated March 20, 2000, the NRC approved CENPD-397-P-A for referencing in license applications for power uprate.

JUSTIFICATION OF REQUESTED CHANGES:

1 INTRODUCTION

1.1 BACKGROUND

The Salem Generating Stations Units 1 and 2 are presently licensed for a full core power rating of 3411 MWt. Through the use of more accurate feedwater flow measurement equipment, approval is being requested to increase this core power by 1.4 percent to 3459 MWt. This corresponds to an uprated NSSS power of 3471 MWt. PSEG Nuclear evaluated the impact of a 1.4 percent core power uprate on plant systems, components, and safety analyses. Results of the evaluation are summarized in the following sections.

1.2 APPROACH

The evaluation of the proposed increase in licensed power level has been completed consistent with the methodology established in WCAP-10263, "A Review Plan for Uprating the Licensed Power of a PWR Power Plant," issued in 1983. Since its submittal to the Nuclear Regulatory Commission (NRC), the methodology has been successfully used as the basis for power uprate projects on over 20 pressurized water reactor (PWR) units, including Diablo Canyon Units 1 and 2, Turkey Point Units 3 and 4, and Comanche Peak Unit 2 (SER dated September 30, 1999).

The methodology in WCAP-10263 establishes the general approach and criteria for uprate projects including the broad categories that must be addressed, such as NSSS performance parameters, design transients, systems, components, accidents, and nuclear fuel as well as interfaces between the NSSS and balance-of-plant (BOP) systems. Inherent in this methodology are key points that include the use of well-defined analysis input assumptions and parameter values, use of currently approved analytical techniques, and use of currently applicable licensing criteria and standards.

The results of PSEG Nuclear's evaluation are summarized in the following sections of this attachment. Section 2 of this attachment discusses the revised NSSS design thermal and hydraulic parameters that were modified as a result of the 1.4 percent uprate and that serve as the basis for all of the NSSS analyses and evaluations. Section 3 concludes that no design transient modifications are required to accommodate the revised NSSS design conditions. Sections 4 and 5 present the NSSS system and component evaluations completed for the revised design conditions. Section 6 provides the results of the accident analyses and evaluations performed for the steam generator tube rupture, mass and energy release, loss-of-coolant-accident (LOCA), and non-LOCA areas. Section 7 contains the results of fuel-related analyses. Sections 8 and 9 summarize the effects of the uprate on plant electrical and balance of plant (BOP) systems. Section 10 provides a summary of the radiological evaluation. Section 11 discusses the effect of the uprate on plant operations, and Section 12 describes evaluations of other licensing requirements. The results of the analyses and evaluations performed demonstrate that all acceptance criteria continue to be met.

1.3 GENERAL LICENSING APPROACH FOR PLANT ANALYSES USING PLANT POWER LEVEL

The reactor core power and NSSS thermal power are used as inputs to most plant safety, component, and system analyses. These analyses generally model the core and/or NSSS thermal power in one of four ways.

First, some analyses apply a 2 percent increase to the initial power level to account solely for the power measurement uncertainty. These analyses have not been re-performed for the 1.4 percent uprate conditions because the sum of increased core power level (1.4 percent) and the decreased power measurement uncertainty (less than 0.6 percent) fall within the previously analyzed conditions.

The power calorimetric uncertainty calculation described in Section 6.7 indicates that with the Crossflow device installed, the power measurement uncertainty (based on a 95 percent probability at a 95 percent confidence interval) is less than 0.6 percent. Thus, these analyses only need to reflect a 0.6 percent power measurement uncertainty. Accordingly, the existing 2 percent uncertainty can be allocated such that 1.4 percent is applied to provide sufficient margin to address the uprate to 3459 MWt, and 0.6 percent is retained in the analysis to still account for the power measurement uncertainty. In addition, for these types of analyses, it is shown that they still employ other conservative assumptions not affected by the 1.4 percent uprated power. Taken together, the use of the calculated 95/95 power measurement uncertainty and retention of conservative assumptions indicate that the margin of safety for these analyses would not be reduced.

Second, some analyses employ a nominal power level. These analyses have either been evaluated or re-performed for the 1.4 percent increased power level. The results demonstrate that the applicable analysis acceptance criteria continue to be met at the 1.4 percent conditions.

Third, some of the analyses already employ a core power level in excess of the proposed 3459 MWt. These analyses were previously performed at a higher power level as part of prior plant programs. For these analyses, some of this available margin has been used to offset the 1.4 percent uprate. Consequently, the analyses have been evaluated to confirm that sufficient analysis margin exists to envelope the 1.4 percent uprate.

Fourth, some of the analyses are performed at zero percent power conditions or do not actually model the core power level. Consequently, these analyses have not been re-performed since they are unaffected by the core power level.

1.4 CROSSFLOW ULTRASONIC FLOW MEASUREMENT

The Crossflow system uses a cross correlation technique to determine the velocity of the fluid by measuring the time a unique pattern of eddies takes to pass between two sets of ultrasonic transducers, each transducer set at a known distance apart, injecting ultrasonic signals perpendicular to the pipe axis.

This flow measurement method yields highly accurate flow readings and has been approved by the NRC for power uprate applications as documented in CENPD-397-P-A, Rev. 01.

1.4.1 Use Of Crossflow To Determine Calorimetric Power

The Crossflow system receives feedwater pressure, feedwater temperature and feedwater flow inputs that can be manually inputted to the Crossflow computer or transmitted via datalink from the Plant Computer. The Crossflow computer then determines fluid velocity in the common header and converts the fluid velocity to a mass flow by using the feedwater temperature and pressure as calculation inputs. The Crossflow feedwater mass flow is periodically compared to the feedwater venturi mass flow to determine the correction factor that must be applied to the venturi mass flow to obtain the corrected mass flow signal. This corrected mass flow is then used to determine power. This power determination will be used directly to calibrate the nuclear instruments in accordance with Technical Specification Surveillance Requirements.

1.4.2 Crossflow Failure

Crossflow system failures are detected and transmitted to the plant computer which causes an overhead annunciator point to alarm for Crossflow abnormal conditions so that the operators are aware of Crossflow status. The Crossflow system does not perform any safety function and is not used to directly control any plant systems. Therefore, system inoperability has no immediate effect on thermal power measurement uncertainty or plant operation.

If the Crossflow system becomes unavailable, plant operation at a core thermal power level of 3459 MWt may continue for 24 hours after the last valid correction factor was obtained from the Crossflow system. Procedural guidance would direct that reactor power be reduced to a level less than or equal to the previously licensed power level (3411 MWt) if the Crossflow system cannot be restored to operation within 24 hours. Core power would be maintained at a level less than or equal to 3411 MWt until the Crossflow system was returned to service and a heat balance in accordance with SR 4.3.1.1.1 was performed with updated correction factors from the Crossflow system.

1.4.3 Maintenance And Calibration

Calibration and maintenance of the Crossflow system will be performed using site procedures developed from the Crossflow system technical manuals. All work is performed in accordance with site work control procedures. Verification of Crossflow System operation is provided by onboard system diagnostics.

Crossflow operation will be monitored on a periodic basis using an internal time delay check. In this way, the user is able to verify that the SCU, computer and software remain within the stated accuracy.

1.4.4 Training

Maintenance and Technical Support personnel will receive training on the Crossflow system before work or calibration may be performed. Initial training will be provided to site personnel by the Crossflow system vendor. Operations personnel will receive training on revised plant procedures before the proposed change is implemented.

1.4.5 Operations And Maintenance History At Salem 1 And 2

The Crossflow system will be installed before implementation of the proposed uprate. Therefore, plant specific maintenance and operations data is not available for evaluation However, significant operational experience has been accumulated from installations at several nuclear power plants. The cumulative operating history shows that the Crossflow system has proven to be reliable. To date, excluding dryout of a couplant that will not be used at Salem 1 and 2, no Crossflow installations have experienced failures which adversely impact the ability to provide the venturi recalibration function. This is over a period of approximately 136 effective years of operational flow measurements.

The Crossflow system that will be installed at Salem 1 and 2 is representative of the Crossflow UFM of the Topical Report CENPD-397-P-A, Rev. 01 and is bounded by the requirements set forth in the topical report.

1.4.6 Uncertainty Determination Methodology

CENP has completed the Salem 1 and 2 Crossflow uncertainty calculation indicating a mass flow accuracy of better than 0.5% of rated flow for the Salem 1 and 2 site specific installation (Calculation A-SA1-PS-0001, Rev 000 and Calculation A-SA2-PS-0001, Rev 000). The calculations are consistent with the methodology described in topical report CENPD-397-P-A, Rev.01. The uncertainty calculations specify requirements for 95% confidence interval flow measurement including:

- Inside pipe diameter measurement and associated uncertainty
- Transducer spacing measurement and associated uncertainty
- Velocity Profile Correction Factor (VPCF) and justification.
- Crossflow time delay calibration data and associated uncertainty.

The Crossflow flow uncertainty calculation supports an uncertainty in the reactor power measurement of 0.6% as discussed in section 6.7. The uncertainty is at a

95% confidence level (2σ). These calculations are based on accepted plant instrument uncertainty methodology.

Crossflow system operating procedures will ensure the assumptions and requirements of the uncertainty calculation remain valid.

1.4.7 Site Specific Piping Configuration

The Salem 1 and 2 Crossflow installation will be installed and calibrated to a site specific piping configuration (flow profile and meter factors are representaive of the plant –specific installation). The installation follows the guidelines in the Crossflow UFM topical report.

1.4.8 Monitoring, Verification And Error Reporting

Although use of the Crossflow system for this application is non-safety-related, the system is designed and manufactured under the vendor's quality control program, which provides for configuration control, deficiency reporting and correction, and maintenance. The current software was verified and validated under CENP's Verification and Validation Program. Specific examples of quality measures included in the design, fabrication and testing of the Crossflow system are provided in the Topical Report. CENP's Verification and Validation program provides procedures for deficiency reporting for engineering action and notification of holders of V&V software.

At Salem 1 and 2 the Crossflow system will be included in the preventive maintenance program. Technical Support personnel will monitor the Crossflow system's reliability. Equipment problems will be documented and corrected in accordance with PSEG Nuclear's corrective action program. Conditions that are adverse to quality are documented under the site corrective action program. The system software is subject to PSEG Nuclear's software quality assurance program.

1.4.9 Quality Control Standards Utilized By CENP

Quality control for the Crossflow meter is documented in section 3.2.5 of CENPD-397-P-A, Rev. 01 "Improved Flow Measurement Accuracy Using Crossflow Ultrasonic Flow Measurement Technology".

1.4.10 Hydraulic Modeling

The Crossflow meter discussed in the Topical report was calibrated at the Alden Research Laboratory (ARL) for a variety of Reynolds (Rd) numbers ranging from 0.8 million to 7 million. The ARL experimental data was used to establish a curve for VPCF as a function of Rd. This curve was then used to extend the VPCF to higher Rd numbers typical of those encountered in nuclear power plant feedwater systems A close agreement was found between the theoretical and experimental VPCF curves. The results of this comparison is included in CENPD-397-P-A, Rev. 01 and the differences between the measured and the predicted VPCF are well within the uncertainty of the ARL weigh tank test accuracy.

In addition to the ARL tests, the theoretical and experimental curves were validated on carbon steel and stainless steel pipes with pipe OD from 3 inches to 24 inches in different laboratories including ARL, NIST, Everest Laboratory (Chatou, France) National Research Council of Canada, and Ontario Hydro. The results of these tests and methodology of extrapolation to high Rd numbers is included in CENPD-397-P-A, Rev. 01

2 NUCLEAR STEAM SUPPLY SYSTEM DESIGN PARAMETERS

2.1 INTRODUCTION

The nuclear steam supply system (NSSS) design parameters are the fundamental parameters used as input in all the NSSS analyses. They provide the reactor coolant system (RCS) and secondary system conditions (temperatures, pressures, and flow) that are used as the basis for all the NSSS analyses and evaluations. Due to the 1.4 percent increase in licensed core power from 3411 MWt to 3459 MWt, it was necessary to revise these parameters. The new parameters are identified in Tables 2-1 and 2-2. These parameters have been incorporated, as required, into the applicable NSSS system and component evaluations, as well as safety analyses, performed in support of the uprate.

2.2 INPUT PARAMETERS AND ASSUMPTIONS

The NSSS design parameters are determined based on conservative inputs, such as a conservatively low thermal design flow (TDF) and bounding steam generator tube plugging (SGTP) levels, which yield primary- and secondary-side conditions that bound the way the plant operates.

An increased NSSS power level of 3471 MWt (3459 MWt core power) is the only input assumption that is changed from the current licensing basis.

2.3 RESULTS OF PARAMETER CASES

Table 2-1 provides the NSSS design parameter cases that were generated and used as the basis for the Unit 1 Model F SGs. Table 2-2 provides the NSSS design parameter cases generated and used as the basis for the 1.4 percent uprate for the Model 51 steam generators (SGs). The following cases are represented in both tables:

- Case 1 - Minimum Full Power T_{avg} and Minimum Tube Plugging
- Case 2 - Minimum Full Power T_{avg} and Maximum Tube Plugging
- Case 3 - Maximum Full Power T_{avg} and Minimum Tube Plugging
- Case 4 - Maximum Full Power T_{avg} and Maximum Tube Plugging

The 1.4 percent uprate resulted in changes to some of the NSSS design parameters, compared to the parameters that form the current licensing basis. The changes included the following RCS temperatures:

- T_{hot} increased by 0.5°F
- T_{cold} decreased by 0.5°F

These small changes occurred since the T_{avg} was maintained at the current design values (566.0°F and 577.9°F) while increasing the core power by 48 MWt to 3459 MWt. The temperature changes reflect the additional heat-up from the uprated core.

In addition, the 1.4 percent uprate resulted in the following changes to the secondary-side parameters:

- T_{steam} decreased by 0.7°F to 0.8°F
- P_{steam} decreased by 5 psi to 6 psi
- Q_{steam} increased by 1.4 percent

These small changes occurred based on a calculation of the steam generator and secondary-side performance resulting from the increased core power.

Table 2-1 NSSS Design Parameters for Salem Unit 1
1.4 percent Uprating (Model F SGs)

OWNER UTILITY: Public Service Electric & Gas
PLANT NAME: Salem
UNIT NUMBER: 1
BASIC COMPONENTS

Reactor Vessel, ID, in.	173	Isolation Valves	No
Core		Number of Loops	4
Number of Assemblies	193	Steam Generator	
Rod Array	17x17[1]	Model	F[2]
Rod OD, in.	0.374	Shell Design Pressure, psia	1200
Number of Grids	12	Reactor Coolant Pump	
Active Fuel Length, in.	144	Model/Weir	93A/No
Number of Control Rods, FL	53	Pump Motor, hp	6000
		Frequency, Hz	60

	1.4% Uprating			
THERMAL DESIGN PARAMETERS	**Case 1**	**Case 2**	**Case 3**	**Case 4**
NSSS Power, %	101.4	101.4	101.4	101.4
MWt	3471	3471	3471	3471
10^6 BTU/hr	11,844	11,844	11,844	11,844
Reactor Power, MWt	3459	3459	3459	3459
10^6 BTU/hr	11,803	11,803	11,803	11,803
Thermal Design Flow, Loop gpm	82,500	82,500	82,500	82,500
Reactor 10^6 lb/hr	127.3	127.3	125.3	125.3
Reactor Coolant Pressure, psia	2250	2250	2250	2250
Core Bypass, %	7.2	7.2	7.2	7.2
Reactor Coolant Temperature, °F				
Core Outlet	606.7	606.7	617.9	617.9
Vessel Outlet	601.8	601.8	613.1	613.1
Core Average	570.3	570.3	582.4	582.4
Vessel Average	566.0	566.0	577.9	577.9
Vessel/Core Inlet	530.2	530.2	542.7	542.7
Steam Generator Outlet	530.0	530.0	542.5	542.5
Steam Generator				
Steam Temperature, °F	515.0	512.7	527.8	525.5
Steam Pressure, psia	778	762	869	852
Steam Flow, 10^6 lb/hr total	15.05	15.04	15.10	15.09
Feed Temperature, °F	432.8	432.8	432.8	432.8
Moisture, % max.	0.25	0.25	0.25	0.25
Tube Plugging, %	0	10	0	10
Zero Load Temperature, °F	547	547	547	547

HYDRAULIC DESIGN PARAMETERS	
Mechanical Design Flow, gpm	99,600
Minimum Measured Flow, gpm total	337,920

FOOTNOTES:
(1) Parameters incorporate 17x17 RFA w/IFMs and the protective bottom grid.
(2) Unit 1 has Model F SGs.

Table 2-2 NSSS Design Parameters for Salem Unit 2 1.4 percent Uprating (Model 51 SGs)

OWNER UTILITY: Public Service Electric & Gas
PLANT NAME: Salem
UNIT NUMBER: 1 and 2
BASIC COMPONENTS

Reactor Vessel, ID, in.	173	Isolation Valves	No
Core		Number of Loops	4
Number of Assemblies	193	Steam Generator	
Rod Array	17x17[1]	Model	51[2]
Rod OD, in.	0.374	Shell Design Pressure, psia	1100
Number of Grids	12	Reactor Coolant Pump	
Active Fuel Length, in.	144	Model/Weir	93A/No
Number of Control Rods, FL	53	Pump Motor, hp	6000
		Frequency, Hz	60

	-------------------1.4% Uprating			----------------
THERMAL DESIGN PARAMETERS	**Case 1**	**Case 2**	**Case 3**	**Case 4**
NSSS Power, %	101.4	101.4	101.4	101.4
MWt	3471	3471	3471	3471
10^6 BTU/hr	11,844	11,844	11,844	11,844
Reactor Power, MWt	3459	3459	3459	3459
10^6 BTU/hr	11,803	11,803	11,803	11,803
Thermal Design Flow, Loop gpm	82,500	82,500	82,500	82,500
Reactor 10^6 lb/hr	127.3	127.3	125.3	125.3
Reactor Coolant Pressure, psia	2250	2250	2250	2250
Core Bypass, %	7.2	7.2	7.2	7.2
Reactor Coolant Temperature, °F				
Core Outlet	606.7	606.7	617.9	617.9
Vessel Outlet	601.8	601.8	613.1	613.1
Core Average	570.3	570.3	582.4	582.4
Vessel Average	566.0	566.0	577.9	577.9
Vessel/Core Inlet	530.2	530.2	542.7	542.7
Steam Generator Outlet	530.0	530.0	542.5	542.5
Steam Generator				
Steam Temperature, °F	508.5	501.1	521.4	514.0
Steam Pressure, psia	735	687	822	771
Steam Flow, 10^6 lb/hr total	15.03	15.01	15.08	15.05
Feed Temperature, °F	432.8	432.8	432.8	432.8
Moisture, % max.	0.25	0.25	0.25	0.25
Tube Plugging, %	0	20	0	20
Zero Load Temperature, °F	547	547	547	547

HYDRAULIC DESIGN PARAMETERS	
Mechanical Design Flow, gpm	99,600
Minimum Measured Flow, gpm total	337,920

FOOTNOTES:

(1) Parameters incorporate 17x17 Robust Fuel Assembly (RFA) with Intermediate Flow Mixing Grids (IFMs) and the protective bottom grid.

(2) Unit 2 has Model 51 SGs.

3 DESIGN TRANSIENTS

3.1 NUCLEAR STEAM SUPPLY SYSTEM DESIGN TRANSIENTS

3.1.1 Introduction

The revised nuclear steam supply system (NSSS) performance design conditions and the NSSS design transients applicable to the uprated conditions serve as primary inputs to the evaluation and analysis of the NSSS systems and components. Current primary- and secondary-side design transients were reviewed to determine their continued applicability for the revised design conditions.

Structural analyses exist for the components based on a full range of postulated design transients. These transients consist primarily of changes in temperatures and pressures resulting from postulated normal and abnormal events occurring during plant operation.

The purpose of this present evaluation was to review the current NSSS design transients to determine if they bound the uprated design conditions described in Section 2.

3.1.2 Discussion of Evaluation

For NSSS design transient purposes, the plant parameters considered to be most critical are the no-load temperature (T_{noload}), RCS hot leg temperature (T_{hot}), cold leg temperature (T_{cold}), and secondary-side steam temperature (T_{steam}). Any significant changes to these parameters impact the NSSS design transients. For the 1.4 percent uprating conditions, the nominal full-power T_{cold} at both high and low vessel average temperature (T_{avg}) conditions is within the window of the current NSSS design transient conditions. However, the nominal full power T_{hot} at low T_{avg} conditions is 1.1°F lower than the rerating T_{hot}. The T_{hot} at higher T_{avg} conditions is within the previous window. For design transient purposes, a deviation of 1.1°F would not have significant impact on the NSSS design transients. The steam temperature and pressure for the 1.4 percent uprating are within the current design transient conditions. Based on this evaluation, the current NSSS design transients remain applicable for the 1.4 percent power uprate analyses.

Two sets (high T_{avg} and low T_{avg}) of design transients are provided to bracket the T_{avg} operating window. For the component design fatigue and stress analyses and evaluations, the initial conditions are to be chosen based on either high or low T_{avg} conditions. This decision will be based on which one is deemed more conservative for the analysis or evaluation of the component under consideration.

3.1.3 Conclusions

The applicability of the current Salem NSSS design transients was confirmed. The transients remain valid for the uprated design conditions described in Section 2. These transients served as input to the component structural analyses and evaluations performed in support of this program.

3.2 AUXILIARY EQUIPMENT DESIGN TRANSIENTS

The review of the NSSS auxiliary equipment design transients was based on a comparison between the revised operating conditions described in Section 2 and the parameters that make up the current auxiliary equipment design transients. A review of the current auxiliary equipment transients determined that the only transients potentially impacted by the power uprate are those temperature transients impacted by full-load NSSS operating temperatures, namely T_{hot} and T_{cold}. These transients are currently based on an assumed full-load NSSS worst-case T_{hot} of 630°F and worst-case T_{cold} of 560°F. These NSSS temperatures were originally selected to ensure that the resulting design transients would be conservative for a wide range of NSSS operating temperatures.

A comparison of the limiting 1.4 percent uprate NSSS design temperature values for T_{hot} and T_{cold} of 613.1°F and 542.7°F, respectively, with the existing transient temperature values indicates that they are still well within the design. Thus, the actual temperature transients (that is, the change in temperature from T_{hot} or T_{cold} dictated by the power uprate parameters to a lower auxiliary system-related temperature or vice versa) are less severe than the current design temperature transients. The 1.4 percent uprate, therefore, does not require any changes to these transients.

4 NUCLEAR STEAM SUPPLY SYSTEMS

This chapter presents the results of the evaluations and analyses performed in the nuclear steam supply systems (NSSSs) area to support the revised design conditions described in Section 2. The systems addressed in this chapter include fluid systems, NSSS/balance-of-plant (BOP) interface systems, and control systems. The results and conclusions of each analysis are presented within each subsection.

4.1 NSSS FLUID SYSTEMS

4.1.1 Reactor Coolant System

The reactor coolant system (RCS) consists of four heat transfer loops connected in parallel to the reactor vessel. Each loop contains a reactor coolant pump (RCP), which circulates the water through the loops and reactor vessel, and a steam generator, where heat is transferred to the main steam system (MSS). In addition, the RCS contains a pressurizer that controls the RCS pressure through electrical heaters, water sprays, power-operated relief valves (PORVs), and spring-loaded safety/relief valves. The steam discharged from the PORVs and safety/relief valves flows through interconnecting piping to the pressurizer relief tank (PRT).

Various assessments were performed to help demonstrate that the RCS design basis functions could still be met at the revised design conditions.

It was demonstrated that the minimum required pressurizer spray flow of 800 gpm can be achieved for the 1.4 percent uprate conditions defined in Section 2. The maximum expected T_{hot} (613.1°F) at the revised design conditions is well below the RCS loop design temperature of 650°F. Therefore, all calculations performed using the RCS loop design temperature remain bounding.

With respect to the PRT discharge analysis, the nominal full-load pressurizer steam volume is essentially unaffected by the uprate since the maximum RCS average temperature of 577.9°F has not changed. Therefore, the existing discharge analysis is essentially unaffected.

All of the RCS assessments resulted in acceptable results for uprate conditions.

4.1.2 Chemical and Volume Control System

The chemical and volume control system (CVCS) provides for boric acid addition, chemical additions for corrosion control, reactor coolant cleanup and degasification, reactor coolant makeup, reprocessing of water letdown from the RCS, and RCP seal water injection. During plant operation, reactor coolant flows through the shell side of the regenerative heat exchanger and then through a letdown orifice. The regenerative heat exchanger reduces the temperature of the reactor coolant and the letdown orifice reduces the pressure. The cooled, low-pressure water leaves the reactor containment and enters the Auxiliary Building. A second temperature reduction occurs in the tube side of the letdown heat exchanger followed by a second pressure reduction due to the low-pressure letdown valve. After passing through one of the mixed bed demineralizers, where ionic impurities are removed, coolant flows through the reactor coolant filter and enters the volume control tank (VCT).

In the assessment of CVCS operation at revised RCS operating temperatures, the maximum expected RCS T_{cold} must be less than or equal to the applicable CVCS design temperature and less than or equal to the heat exchanger design inlet operating temperature. The former criterion supports the functional operability of the system and its components. The latter criterion confirms that the heat exchanger design operating conditions remain bounding.

With regards to the CVCS thermal performance, the maximum T_{cold} of 542.7°F is still lower than the design system inlet temperature of 560°F. Also, it is much lower than the shell side design temperature of 650°F for the regenerative heat exchanger. The excess letdown path is used to process excess effluents associated with fluid expansion during plant heatup. Therefore, it is unaffected by the revised T_{cold} at full-power conditions. If operated during power conditions, the excess letdown heat exchanger outlet flow is throttled to maintain the desired outlet temperature and flow. Therefore, operation of the CVCS is unaffected by the temperature change.

4.1.3 Safety Injection System

The safety injection system (SIS) is an engineered safeguards system used to mitigate the effects of postulated design basis events. The basic functions of this system include providing short- and long-term core cooling, and maintaining core shutdown reactivity margin. The SIS is made up of three subsystems. The passive portion of the system is the four accumulator vessels that are connected to each of the RCS cold leg pipes. Each accumulator contains borated water under pressure (nitrogen cover gas). The borated water automatically injects into the RCS when the pressure within the RCS drops below the operating pressure of each of the accumulators.

The "active" part of the SIS injects borated water into the reactor following a break in either the reactor or steam systems in order to cool the core and prevent an uncontrolled return to criticality. Two safety injection (SI) pumps and two residual heat removal (RHR) pumps take suction from the refueling water storage tank (RWST) and deliver borated water to four cold leg connections via the accumulator discharge lines. In addition, two centrifugal charging pumps take suction from the RWST on SI actuation and provide flow to the RCS via separate SI connections on each cold leg. This arrangement of SI pumps can provide safety injection flow at any RCS pressure up to the set pressure of the pressurizer safety valves.

The revised design conditions have no direct effect on the overall performance capability of the SIS. These systems will continue to deliver flow at the design basis RCS and containment pressures since there are no changes in the RCS operating pressure.

4.1.4 Residual Heat Removal System

The residual heat removal system (RHRS) is designed to remove sensible and decay heat from the core and reduces the temperature of the RCS during the second phase of plant cooldown. As a secondary function, the RHRS is used to transfer refueling water between the RWST and the refueling cavity at the beginning and end of refueling operations.

The RHRS consists of two residual heat exchangers, two RHR pumps, and associated piping, valves, and instrumentation. During system operation, coolant flows from one hot leg of the RCS to the RHR pumps, through the tube side of the residual heat exchangers, and back to four RCS cold legs. The RHR heat exchangers are of the shell and U-tube type. Reactor coolant circulates through the tubes, while component cooling water circulates through the shell.

Single train cooldown and normal cooldown cases were reviewed to address the uprated reactor power (3459 MWt). A single train cooldown is defined as cooling the RCS from 350°F at four hours after plant shutdown to 200°F by employing one RHR pump, one RHR heat exchanger, and one train of component cooling. A normal cooldown is defined as cooldown from 350°F at four hours after plant shutdown to 140°F using two trains of cooling equipment (two trains of RHR, component cooling water, and service water). The evaluation concluded that both the normal and single train cooldown can still be accomplished within 36 hours at the 1.4 percent uprate conditions.

4.1.5 Component Cooling Water System

The component cooling water system (CCWS) is an intermediate closed-cooling system between the ultimate heat sink (the SW system) and radioactive systems. The CCWS provides cooling to the RCP lube oil coolers, RCP seal barrier, RCP seal injection HX, ECCS pump seals, letdown and excess letdown HX's, and waste gas compressors. The uprate will increase the decay heat that is transferred from the RHR system to the CCWS during accident or normal cooldown. The uprate also increases the decay heat in the spent fuel pool transferred by the SFP cooling system to CCWS.

For the postulated limiting accident, the existing LOCA analysis assumes the plant was operating at the ESF design rating (102 percent of 3411 MWt). Therefore, the existing accident analysis bounds the proposed uprate.

The increase in decay heat can result is a small increase in the time required to cool down to 200°F after entry into mode 3. However, as discussed in section 4.1.4, plant cooldown can still be accomplished within the required time.

The review of the CCWS indicates that the system will perform adequately at the uprated conditions.

4.1.6 Waste Disposal System

The waste disposal system (WDS) consists of separate gaseous and liquid waste processing subsystems.

The review of the WDS concludes that the design of the WDS is adequate for the uprated conditions (3459 MWt).

4.1.7 Sampling System

The sampling system (SS) consists of various flow paths that provide means for samples from the RCS and selected auxiliary systems to be drawn and cooled for analyses.

The maximum hot leg temperature (613.1°F) for uprated conditions (3459 MWt) is well below the SS heat exchanger design temperature of 653°F. Therefore, the SS as designed is adequate for operation under uprated conditions.

4.1.8 Containment Spray System

The containment spray system (CSS) consists of two separate trains that can provide post-accident containment cooling, sump pH adjustment, and sump iodine retention.

Operation at the uprated conditions (3459 MWt) has no direct impact on CSS performance capability.

4.1.9 Pressurizer Overpressure Protection System

The pressurizer overpressure protection system (POPS) is designed to protect the RCS from overpressure events when the RCS temperature is below approximately 312°F. Changes to full-power operating parameters, such as NSSS power, do not impact POPS. Thus, the existing POPS analysis is unaffected. The revised Pressure-Temperature curve pressure limits have been reviewed against the POPS analysis results and verified to be still bounded using the criteria in ASME Code Case N-640.

4.2 NSSS/BOP FLUID INTERFACE

The following BOP fluid systems were reviewed to assess compliance with NSSSs/BOP interface guidelines at the revised design conditions described in Section 2. It was determined that these guidelines were met with the 1.4 percent uprated conditions.

4.2.1 Main Steam System

The following subsections summarize the evaluation of the major steam system components relative to the revised design conditions for the 1.4 percent power uprate. The major components of the MSS include the steam generator main steam safety valves (MSSVs), the steam generator power-operated relief valves (PORVs), and the main steam isolation valves (MSIVs).

Steam Generator Main Steam Safety Valves

The MSSVs must have sufficient capacity so that main steam pressure does not exceed 110 percent of the steam generator shell-side design pressure (the maximum pressure allowed by the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code) for the worst-case loss-of-heat-sink event.

Salem has 20 safety valves with a total capacity of 16.65 x 10^6 lb/hr in each operating unit. This provides about 110.3 percent of the maximum calculated steam flow of 15.10 x 10^6 lb/hr for the revised design conditions for Unit 1, and 110.4 percent of the maximum calculated steam flow of 15.08 x 10^6 lb/hr for Unit 2. Therefore, based on the range of NSSS performance parameters for the uprating, the capacity of the installed MSSVs meets the sizing criterion.

By letter dated September 26, 2000 (LCR S99-13) PSEG Nuclear proposed to amend TS 3/4.7.1, Plant Systems - Turbine Cycle - Safety Valves, and its associated basis to require a reduction of power based on the number of inoperable MSSVs with one or more MSSVs inoperable. The current TS requires a reduction in the power range neutron flux high trip setpoint based on the number of inoperable MSSVs. The proposed maximum allowable power levels in LCR S99-13 were listed in percent of the current rated thermal power. They were selected to ensure primary and secondary design pressure limits would not be exceeded during a loss of electrical load and/or turbine trip.

The maximum allowable power levels in TS Table 3.7-1 will be revised as follows to account for the increase in rated thermal power:

Maximum No. of Inoperable Safety Valves on Any Steam Generator	Maximum Allowable Power (Percent of Rated Thermal Power (3459 MWt))
1	87% (3009.3 MWt)
2	59% (2040.8 MWt)
3	39% (1349.0 MWt)

The revised values will ensure that all current analyses supporting the allowable power levels remain bounding for uprated conditions. The additional changes to TS Table 3.7-1 described in LCR S99-13 are included in the marked up TS pages in Attachment 4.

The Analysis of Record assumes a maximum flow limit of 1,100,000 lb/hr at 1000 psia for each MSSV (as well as each steam generator PORV and steam dump valve). Since the actual capacity of any single MSSV, PORV, or steam dump valve is less than the maximum flow limit per valve, the maximum capacity criteria is satisfied.

Steam Generator Power-Operated Relief Valves

The primary function of the PORVs is to provide a means for decay heat removal and plant cooldown by discharging steam to the atmosphere when either the condenser, the condenser circulating water pumps, or steam dump to the condenser is not available. Under such circumstances, the PORVs, in conjunction with the auxiliary feedwater (AFW) system, permit the plant to be cooled down from the pressure setpoint of the lowest-set MSSVs to the point where the RHRS can be placed in service. During cooldown, the PORVs are either automatically or manually controlled. In automatic, each PORV proportional and integral (P&I) controller compares steam line pressure to the pressure setpoint, which is manually set by the plant operator.

In the event of a tube rupture event in conjunction with loss of offsite power, the PORVs are used to cool down the RCS to a temperature that permits equalization of the primary and secondary pressures at a pressure below the lowest-set MSSV. Both RCS cooldown and depressurization are required to preclude steam generator overfill and to terminate activity release to the atmosphere.

The steam generator PORVs are sized to have a capacity equal to about 10 percent of the steam flow used for plant design, at no-load steam pressure. For the revised design conditions, each steam generator PORV is required to have a capacity at least equal to 382,794 lb/hr/valve at 1020 psia steam pressure. At these conditions, this capacity permits a plant cooldown to RHRS operating conditions in 4 hours (at an assumed cooldown rate of 50°F/hr) assuming a minimum of 2 hours at hot standby. This sizing is compatible with normal cooldown capability and minimizes the water supply required by the auxiliary feedwater system. Since the design capacity of the installed PORVs meets the sizing criteria, the valves are adequately sized for the 1.4 percent uprated conditions.

Main Steam Isolation Valves and Main Steam Isolation Bypass Valves

The MSIVs are located outside the containment and downstream of the MSSVs. The valves function to prevent the uncontrolled blowdown of more than one steam generator and to minimize the RCS cooldown and containment pressure to within acceptable limits following a main steam line break. To accomplish this function, the design requirements specified that the MSIVs must be capable of closure within 5 seconds of receipt of a closure signal against steam break flow conditions in either the forward or reverse direction.

Rapid closure of the MSIVs following postulated steam line breaks causes a significant differential pressure across the valve seats and a thrust load on the main steam system piping and piping supports in the area of the MSIVs. The worst cases for differential pressure increase and thrust loads are controlled by the steam line break area (i.e., mass flowrate and moisture content), throat area of the steam generator flow restrictors, valve seat bore, and no-load operating pressure. Since these variables and no-load operating pressure are not impacted by the uprating, the design loads and associated stresses resulting from rapid closure of the MSIVs will not change. Consequently, power uprate has no significant impact on the interface requirements for the MSIVs.

The MSIV bypass valves are used to warm up the main steam lines and equalize pressure across the MSIVs prior to opening the MSIVs. The MSIV bypass valves perform their function at no-load and low-power conditions where power uprate has no significant impact on main steam conditions (e.g., steam flow and steam pressure). Consequently, power uprate has no significant impact on the interface requirements for the MSIV bypass valves.

4.2.2 Steam Dump System

The steam dump system creates an artificial steam load by dumping steam from ahead of the turbine valves to the main condenser. The sizing criterion recommends that the steam dump system (valves and pipe) be capable of discharging 40 percent of the rated steam flow at full-load steam pressure to permit the NSSS to withstand an external load reduction of up to 50 percent of plant rated electrical load without a reactor trip. To prevent a trip, this transient requires all NSSS control systems to be in automatic, including the reactor control system, which accommodates 10 percent of the load reduction. A steam dump capacity of 40 percent of rated steam flow at full-load steam pressure also prevents MSSV lifting following a reactor trip from full power.

Steam Dump System Major Components

Each operating unit at Salem is provided with 12 condenser steam dump valves, which provide a total steam dump capacity of 6,600,000 lb/hr assuming a pressure of 615 psia at the inlet to the valves. This total capacity provides a steam dump capability of about 43.8 percent of the original maximum guaranteed steam flow (14.86 x 10^6 lb/hr), or 6.51 x 10^6 lb/hr at a full-load steam generator pressure of 805 psia versus the sizing criterion of 40 percent of rated steam flow.

Operation of the NSSS within the proposed range of operating parameters at increased steam flows will result in a small decrease in steam dump capacity. Based on the range of NSSS operating parameters approved for power uprate, an evaluation was performed and the results confirmed that total steam dump capacity continues to meet the design criterion. Therefore, the condenser steam dump capacity is adequate for 1.4 percent power uprate.

The NSSS controls system analysis provided in Section 4.3 demonstrates the adequacy of the steam dump control system at the uprated conditions.

4.2.3 Condensate and Feedwater System

The condensate and feedwater system (C&FS) must automatically maintain steam generator water levels during steady-state and transient operations. The range of NSSS performance parameters will result in a required feedwater volumetric flow increase of up to 1.5 percent during full-power operation. The higher feedwater flow will have an impact on system pressure drop, which may increase by as much as 3.0 percent..

The major components of the C&FS are the main feedwater isolation valves (MFIVs), the main feedwater regulator valves (MFRVs), and the C&FS pumps.

Main Feedwater Isolation Valves / Main Feedwater Regulator Valves

The MFIVs are located outside containment and downstream of the MFRVs. The valves function in conjunction with the primary isolation signals to the MFRVs and back up trip signals to the feedwater pumps to provide redundant isolation of feedwater flow to the steam generators following a steam line break or a malfunction in the steam generator level control system. Isolation of feedwater flow is required to prevent containment overpressurization and excessive RCS cooldowns. To accomplish this function, the MFRVs and the backup MFIVs must be capable of closure within 8 seconds and 30 seconds, respectively, following receipt of any feedwater isolation signal.

The quick-closure requirements imposed on the MFRVs will cause dynamic pressure changes that may be of large magnitude and must be considered in the design of the valves and associated piping. The worst loads occur following a steam line break from no-load conditions with the conservative assumption that all feedwater pumps are in service providing maximum flow following the break. Since these conservative assumptions are not impacted by the uprating, the design loads and associated stresses resulting from rapid closure of these valves will not change.

Condensate and Feedwater System Pumps

The C&FS available head, in conjunction with the MFRV characteristics, must provide sufficient margin for feed control to ensure adequate flow to the steam generators during steady-state and transient operation. A continuous steady feed flow should be maintained at all loads. To assure stable feedwater control, with variable speed feedwater pumps, the pressure drop across the MFRVs at rated flow (100 percent power) should be approximately equal to the dynamic losses from the feed pump discharge into the steam generator (i.e., equal to the frictional resistance of feed piping, MFIV, high-pressure feedwater heaters, feed flow meter, and steam generator feed nozzle). In addition, adequate margin should be available in the MFRVs at full-load conditions to permit a C&FS delivery of 96 percent of rated flow with a 100-psi pressure increase above the full-load pressure with the MFRVs fully open. However, based on the Salem MFRV design (maximum full open Cv of 1450 at 2-1/2 inches lift) and the system layout, the present pump speed control program was set to provide a MFRV pressure drop of about 60 psi to achieve about a 72.5 percent valve lift at full load.

For the range of revised NSSS performance parameters for the uprate, the present speed control program results in a negligible change in MFRV pressure drop (less than 2.5 psi) and a corresponding negligible change in valve lift (less than 3 percent) at 100 percent power. Therefore, based on the NSSS performance parameters for the 1.4 percent uprate, operation of the MFRVs (in conjunction with the present feedwater pump speed control program) is acceptable for both steady-state and transient operation.

To provide effective control of flow during normal operation, the MFRVs are required to stroke open or closed in 20 seconds over the anticipated inlet pressure control range (approximately 0 - 1600 psig). Additionally, rapid closure of the MFRVs is required in 8 seconds after receipt of a trip close signal in order to mitigate certain transients and accidents. These requirements are still applicable at the uprated conditions.

Auxiliary Feedwater System

The AFW system supplies feedwater to the secondary side of the steam generators at times when the normal feedwater system is not available, thereby maintaining the steam generator heat sink. The system provides feedwater to the steam generators during normal unit startup, hot standby, and cooldown operations and also functions as an engineered safeguards system. In the latter function, the AFW system is required to prevent core damage and system overpressurization during transients and accidents, such as a loss of normal feedwater or a secondary system pipe break. The design basis for the system is discussed in UFSAR section 10.4.7. The limiting accident for the AFW system flow rates is the double-ended Feedwater Line Break. The FSAR analyses for the Feedwater Line Break assumed 102% reactor power. Since the proposed increase in licensed power is offset by a reduction in the calorimetric error, reanalysis is not required and the required AFW flow rate does not change.

Auxiliary Feedwater Storage Requirements

The AFW pumps for each Salem Unit are normally aligned to take suction from the auxiliary feedwater storage tank (AFST). To fulfill the engineered safety features (ESF) design functions, sufficient feedwater must be available during transient or accident conditions to enable the plant to be placed in a safe shutdown condition.

The limiting transient with respect to AFST inventory requirements is the loss-of-offsite-power (LOOP) transient. The Salem Unit 1 and Unit 2 licensing basis dictates that in the event of a LOOP, sufficient AFST useable inventory must be available to bring the unit from full power to hot standby conditions, maintain the plant at hot standby for 4 hours, and then cool down the RCS to the RHRS cut-in temperature (350°F) in 4 hours. In light of these design bases requirements, the Salem AFST (dedicated to each unit) is designed to accommodate a minimum contained water inventory of 200,000 gallons. The minimum AFST contained inventory of 200,000 gallons is based on reactor trip from 102 percent of rated core power (3479 MWt). Since the proposed power uprate is based on improved calorimetric error, no change in the required inventory or the plant technical specifications is required for operation at the uprated power level.

Steam Generator Blowdown System

The steam generator blowdown system is used in conjunction with the chemical addition and sampling systems to control the chemical composition of the steam generator shell water within the specified limits. The blowdown system also controls the buildup of solids in the steam generator water.

The blowdown flowrates required during plant operation are based on chemistry control and tube-sheet sweep requirements to control the buildup of solids. The rate of addition of dissolved solids to the secondary systems is a function of condenser leakage and the quality of secondary makeup water, and the rate of generation of particulates is a function of erosion-corrosion (E/C) within the secondary systems. Since neither condenser leakage nor the quality of secondary makeup water is expected to be impacted by power uprate, the rate of blowdown required to address dissolved solids should not be impacted by power uprate.

The present range of NSSS operating parameters permits a maximum decrease in steam pressure from no load to full load of 274 psi (i.e. from 1020 psia to 746 psia). Since the inlet pressure to the steam generator blowdown system varies proportionally with operating steam pressure, the blowdown flow control valves must be designed to handle a corresponding range of inlet pressures. Based on the revised range of NSSS parameters for power uprate, the no-load steam pressure (1020 psia) remains the same and the full-load minimum steam pressure (746 psia) is within the present operating range. Therefore, the range of operating parameters revised for power uprate will not impact blowdown flow control.

4.3 NSSS CONTROL SYSTEMS

Condition I transients are evaluated to confirm that the plant can appropriately respond to these transients without generating a reactor trip or engineered safety feature actuation system (ESFAS) actuation. The transients of concern include:

- 10 percent step load increase
- 10 percent step load decrease
- 50 percent load rejection
- 5 percent per minute ramp load increase

The analysis methodology for these transients employs a 2 percent power calorimetric uncertainty to increase the power level to 102 percent. The improved thermal power measurement accuracy obviates the need for the full 2 percent power measurement margin assumed in the analysis.

Furthermore, the power measurement margin is only one of many conservative assumptions used in the analysis. Others include a minimum available steam dump capacity and more limiting beginning–of-life (BOL) fuel reactivity conditions (which provide the more severe reactivity response, and hence transient conditions). Together, the improved power measurement uncertainty and conservative assumptions provide substantial conservatism such that the transients noted above can be accommodated without resulting in a reactor trip or ESFAS actuation.

Likewise, the pressurizer PORV and spray valve capacity for response to key operational transients was determined to be unaffected by the power uprate due to the use of a 2 percent power uncertainty and other conservatisms.

The rod and steam dump control system stability for key operational transients was also examined. They are not a function of power level or full-load T_{avg}, but rather a function of the rod and steam dump control system deadbands and the reactor core kinetics. Since the 1.4 percent uprating does not include any change to the control systems deadbands or represent any significant change in the reactor core kinetics, the rod and steam dump control system stability are not affected by the 1.4 percent uprating.

5 NUCLEAR STEAM SUPPLY SYSTEM COMPONENTS

5.1 REACTOR VESSEL STRUCTURAL EVALUATION

An evaluation was performed to assess the effects that the 1.4 percent uprating conditions have on the most limiting locations with regard to ranges of stress intensity and fatigue usage factors in each of the regions as identified in the reactor vessel stress reports and addenda. The design inputs used to evaluate the reactor vessel structural analyses are either unchanged or are bounded by the parameters previously considered in the reactor vessel stress reports. These design inputs include the limiting values of T_{hot} and T_{cold} and the nuclear steam supply system (NSSS) design transients, which were demonstrated to be unaffected by the power uprate (see Section 3.1). Since the existing reactor vessel structural analyses remain bounding, the stress intensities and cumulative usage factors for the various regions of the Unit 1 and Unit 2 reactor vessels continue to satisfy the applicable limits of the 1965 edition of Section III of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code through the Winter 1965 Addenda (Unit 1) and the Winter 1966 Addenda (Unit 2).

An assessment was made to evaluate the change in the Alloy 600 primary water stress corrosion cracking (PWSCC) susceptibility for Salem Units 1 and 2 at the revised design conditions described in Section 2. It was concluded that the increase in the PWSCC susceptibility of the highest susceptible head penetration is not significant (approximately 2.25 percent).

5.2 REACTOR VESSEL INTEGRITY – NEUTRON IRRADIATION

The reactor vessel integrity analysis was evaluated for the 1.4 percent uprate by examining the revised design conditions (described in Section 2) and the increase in neutron fluences.

Neutron Fluence

An evaluation of the neutron exposure of the reactor vessel materials to determine the effects of the 1.4 percent increase in core power was performed. This evaluation included assessments not only at locations of maximum exposure at the inner diameter of the vessel, but also as a function of axial, azimuthal, and radial location throughout the vessel wall.

The fast neutron exposure levels were defined at depths within the vessel wall equal to 25 and 75 percent of the wall thickness for each of the materials constituting the beltline region. This was done to satisfy the requirements of 10 CFR 50, Appendix G, for the calculation of pressure/temperature limit curves for normal heatup and cooldown of the reactor coolant system. These locations are commonly referred to as the 1/4T and 3/4T positions in the vessel wall. The 1/4T exposure levels are also used in the determination of upper shelf fracture toughness as specified in 10 CFR 50, Appendix G. Maximum neutron exposure levels experienced by each of the beltline materials are required for determining the RT_{PTS} values. These RT_{PTS} values are compared with the applicable pressurized thermal shock screening criterion as defined in 10 CFR 50.61. The maximum exposure levels occur at the vessel inner radius.

The results of the fast neutron exposure evaluations for Salem Units 1 and 2 account for the uprated power level. The results are based on the conservative assumption that the power uprate was initiated coincident with the last surveillance capsule withdrawal from each unit. The resulting fast neutron (E > 1.0 MeV) exposure projections increased due to the power uprate. The new projections were used as input to the reactor vessel integrity evaluations. TS Bases Figure 3/4.4-1 shows predicted fluence as a function of Effective Full Power Years. The results of the fast neutron exposure evaluation are incorporated in the revised figures in Attachment 4.

Surveillance Capsule Withdrawal Schedule

A withdrawal schedule is developed to periodically remove surveillance capsules from the reactor vessel to effectively monitor the condition of the reactor vessel materials under actual operating conditions. The current withdrawal schedules were evaluated based on the revised fluence projections. It was determined that no change to the current withdrawal schedules is necessary.

Heatup and Cooldown Pressure - Temperature Limit Curves

New heatup and cooldown curves were developed for the Salem Units 1 and 2 reactor vessels based on the uprated fluence projection at 32 effective full-power years (EFPY). The maximum heatup rate for TS Figure 3.4-2 is being changed from 60°F/hr to 100°F/hr. This heatup rate change is consistent with the current Technical Specification LCO 3.4.9.1.a for Unit 1, LCO 3.4.10.1.a for Unit 2, the heatup cyclic limits of TS Table 5.7-1, and the Analysis of Record. The revised curves are being adjusted to account for pressure and temperature instrument uncertainties. Minimum boltup temperature is shown.

The heatup and cooldown curves were generated using the most limiting adjusted reference temperature (ART) values and the NRC-approved methodology documented in WCAP-14040-NP-A, Revision 2, with the following exceptions:

- The K_{Ic} critical stress intensities are used in place of the K_{Ia} critical stress intensities based on the approved methodology in ASME Code Case N-640.
- The reactor vessel flange pressure/temperature requirement has been eliminated consistent with the justification provided in WCAP-15315, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Operating PWR and BWR Plants.".
- The 1995 edition of the ASME Code through the 1996 Addenda (instead of the 1989 version) of Appendix G to Section XI was used.

The 1995 edition of the ASME Code through the 1996 Addenda of Appendix G to Section XI is the most recent version incorporated by reference in 10 CFR 50.55a. Use of Code Case N-640 and WCAP-15315 contributes to increasing the operating window by reflecting an updated understanding of material properties and operating conditions.

The increase in maximum heatup rate allows additional operating margin during RCS heatup, and so reduces the burden on the operator for control of the heatup rate. The proposed limits are applicable up to 32 EFPY, which corresponds to the end of the current 40-year license, assuming an 80% capacity factor. The limit curves are being adjusted for instrument uncertainties to ensure reactor coolant system pressure and temperature are maintained within applicable limits.

The curves have been developed in accordance with the methodology provided in Regulatory Guide 1.99, Revision 2, and ASME Code Case N-640. The use of Code Case N-640, K_{Ic} methodology will reduce the excess conservatism in the current Appendix G approach that could reduce overall plant safety by unnecessarily restricting plant operation. By changing from K_{IA} to K_{Ic} methodology, the operating window is made larger and the burden on the operator during plant heatup, cooldown and pressure testing is reduced.

The flange requirement of 10 CFR 50 Appendix G was originally developed using the K_{IA} fracture toughness. WCAP-15315 provides justification to show that the use of the newly accepted K_{Ic} fracture toughness for flange considerations leads to the conclusion that the flange requirement can be eliminated. It concluded that the elimination of the flange requirement would make a significant improvement in plant safety and ease the burden on the operators.

Pressurized Thermal Shock (PTS)

The RT_{PTS} screening criteria values were set (using conservative fracture mechanics analysis techniques) for beltline axial welds, plates, and beltline circumferential weld seams for end-of-life plant operation based on the Nuclear Regulatory Commission (NRC) screening criterion for pressurized thermal shock (10 CFR 50.61). The RT_{PTS} values for all beltline region materials of the Salem Units 1 and 2 reactor vessels for end of license (32 EFPY) were recalculated for the 1.4 percent uprate. These RT_{PTS} values increased due to the 1.4 percent uprating. However, other circumstances such as updated chemistry factor values and updated fluence values also had an effect on the results. The Salem Units 1 and 2 RT_{PTS} values remain below the NRC screening criteria values using projected fluence values through 32 EFPY.

Emergency Response Guideline (ERG) Limits

New RT_{NDT} values were determined for Salem Units 1 and 2 based on the revised fluence projections for the 1.4 percent uprate. A comparison of the current RT_{PTS} calculation (which is the RT_{NDT} value at the end-of-life (32 EFPY)) to the uprated RT_{PTS} values for Salem Units 1 and 2 was made to determine if the applicable ERG category (Westinghouse Owners Group Emergency Response Guidelines, Rev. 1C, September 30, 1997) would change.

The most limiting RT_{PTS} value for Salem Unit 1 is 264°F at 32 EFPY. The Salem Unit 1 limiting material is the lower shell longitudinal weld seam 3-042C and the ERG limit is only applicable to 250°F for a longitudinal weld. This result would place the Salem Unit 1 reactor vessel in Category II until the RT_{PTS} value for the lower longitudinal weld seam reaches 250°F at approximately 25.3 EFPY. The ERG limit for operation beyond 25.3 EFPY will need to be based on a plant-specific evaluation since no generic category currently exists for Salem Unit 1.

The Salem Unit 2 limiting RT_{PTS} value is 229°F at 32 EFPY, for the lower shell longitudinal weld seams 3-442A and C. Therefore, the Salem Unit 2 vessel will be in the ERG Limit Category II, until end of license (32 EFPY).

Upper Shelf Energy (USE)

Since the neutron fluence values for the 1.4 percent uprate have increased, the USE values were recalculated for Salem Units 1 and 2. It was determined that all reactor vessel beltline materials in the reactor vessel are expected to have a USE greater than 50 ft-lb through the end of license (32 EFPY) as required by 10 CFR 50, Appendix G.

5.3 REACTOR INTERNALS

The reactor internals support the fuel and control rod assemblies, absorb control rod assembly dynamic loads, and transmit these and other loads to the reactor vessel. The internals also direct flow through the fuel assemblies, provide adequate cooling to various internals structures, and support in-core instrumentation. The changes in the reactor coolant system (RCS) design temperatures, listed in Section 2, produce changes in the boundary conditions experienced by the reactor internals components. This section describes the analyses performed to demonstrate that the reactor internals can perform their intended design functions at the 1.4 percent uprated conditions.

5.3.1 Thermal-Hydraulic Systems Evaluations

A key area in evaluation of core performance is the determination of hydraulic behavior of coolant flow and its effect within the reactor internals system. The core bypass flows are required to ensure reactor performance and adequate vessel head cooling. The hydraulic lift forces are critical in the assessment of the structural integrity of the reactor internals. Baffle gap momentum flux/fuel stability is affected by pressure differences between the core and baffle former region. The results of the thermal-hydraulic evaluations are provided below.

Core Bypass Flow Calculation

Bypass flow is the total amount of reactor coolant flow bypassing the core region and is not considered effective in the core heat transfer process. The principal core bypass flows are the barrel-baffle region, vessel head cooling spray nozzles, vessel outlet nozzle gap, baffle plate cavity gap, and the thimble tubes. An analysis demonstrated that the core bypass flow with the revised design conditions remains less than the current design value, and is therefore acceptable.

Hydraulic Lift Forces

The reactor internals hold-down spring is essentially a large-diameter Belleville-type spring of rectangular cross-section. The purpose of this spring is to maintain a net clamping force between the reactor vessel head flange and upper internals flange, and the reactor vessel shell flange and the core barrel flange of the internals. An evaluation demonstrated that the hydraulic lift forces on the various reactor internals components were enveloped by the current Analysis of Record. It is concluded, therefore, that the spring would maintain a net clamping force and the reactor internals assembly would remain seated and stable for the 1.4 percent power uprate conditions.

Baffle Joint Momentum Flux and Fuel Rod Stability

Baffle jetting is a hydraulically induced instability or vibration of fuel rods caused by a high-velocity jet of water. This jet is created by high-pressure water being forced through gaps between the baffle plates, which surround the core.

To minimize the propensity for flow-induced vibration, the crossflow emanating from baffle joint gaps must be limited to a specific momentum flux, V^2h; that is, the product of the gap width, h, and the square of the baffle joint jet velocity, V^2. This momentum flux varies from point to point along the baffle plate due to changes in pressure differential across the plate and the local gap width variations. In addition, the modal response of the vibrating fuel rod must be considered. That is, a large value of local momentum flux impinging near a grid is much less effective in causing vibration than the same V^2h impinging near the mid span of a fuel rod.

The results showed that for all modal shapes, the momentum flux did not change as a result of the 1.4 percent power uprate conditions.

Rod Cluster Control Assembly Drop Time Analyses

Technical Specification 3.1.3.3 requires that the rod cluster control assembly (RCCA) drop time be less than or equal to 2.7 seconds. The revised design conditions, in particular the reduced T_{cold}, can increase the drop time due to the increased fluid density. An evaluation confirmed that the RCCA rod drop time is still within the current value of 2.7 seconds at the revised design conditions.

5.3.2 Mechanical Evaluations

The 1.4 percent uprate conditions do not affect the current design bases for seismic and loss-of-coolant-accident (LOCA) loads. Therefore, it was not necessary to re-evaluate the structural effects from seismic operating-basis earthquake (OBE) and safe shutdown earthquake (SSE) loads, and the LOCA hydraulic and dynamic loads.

With regards to flow-and pump-induced vibration, the current analysis uses a mechanical design flow, which did not change for the revised design conditions. The revised design conditions will slightly alter the T_{cold} and T_{hot} fluid densities, which will slightly change the forces induced by flow. However, these changes are enveloped by the current Analysis of Record. Therefore, the mechanical loads are not affected by the 1.4 percent uprated conditions.

5.3.3 Structural Evaluations

Evaluations are required to demonstrate that the structural integrity of the reactor components is not adversely affected by the 1.4 percent uprate conditions. The presence of heat generated in reactor internal components, along with the various fluid temperatures, results in thermal gradients within and between components. These thermal gradients result in thermal stresses and thermal growth, which must be accounted for in the design and analysis of various components. The core support structures affected by the revised design conditions are discussed in the following sections. The primary inputs to the evaluations are the NSSS design parameters described in Section 2 and the gamma heating rates. The gamma heating rates were modified, as required, to account for the 1.4 percent increase in core power.

Baffle-Barrel Region Evaluations

The baffle-barrel regions consist of a core barrel into which baffle plates are installed, supported by bolting interconnecting former plates to the baffle and core barrel. The baffle-to-former bolts restrain the motion of the baffle plates that surround the core. These bolts are subjected to primary loads consisting of deadweight, hydraulic pressure differentials, and seismic loads, as well as secondary loads consisting of preload, and thermal loads resulting from RCS temperatures and gamma heating rates. The baffle-to-former bolt thermal loads are induced by differences in the average metal temperature between the core barrel and baffle plate. In addition to providing structural restraint, the baffles also channel and direct coolant flow such that a coolable core geometry can be maintained.

The thermal stresses in the core barrel shell in the core active region are primarily due to temperature gradients through the thickness of the core barrel shell. These temperature gradients are caused by the fluid temperatures between the inside and outside surfaces and the contribution of gamma heating.

A structural assessment determined that the 1.4 percent uprate conditions had no impact on the current Analysis of Record for the baffle plate and core barrel. No changes have occurred in the gamma heating rates for the baffle plate and core barrel. In fact, the new gamma heating rates for the baffle barrel region are significantly reduced due to the fuel low leakage loading pattern being used in Salem Units 1 and 2. Thus, the ability to provide structural restraint and direct coolant flow (i.e., maintain coolable core geometry) of the baffle-barrel region is maintained.

Lower Core Plate Structural Analysis

The lower core plate is a perforated circular plate that supports and positions the fuel assemblies. The plate contains numerous holes to allow fluid flow through the plate. The fluid flow is provided to each fuel assembly and the baffle barrel region. The plate is bolted at the periphery to a ring welded to the inside diameter of the core barrel. The center span of the plate is supported by the lower support columns, which are attached at the lower end to the lower support plate.

Temperature differences between components of the lower support assembly induce thermal stresses in the lower core plate. In addition, due to the lower core plate's proximity to the core, the heat generation rates in the lower core plate due to gamma heating cause a significant temperature increase in this component. Thermal expansion of the lower core plate is restricted by the lower support columns, lower support plate, and core barrel. These restraining items are exposed to the inlet temperature and have heat generation rates much lower than those found in the lower core plate.

Structural evaluations were performed to demonstrate that the structural integrity of the lower core plates is not adversely affected by the revised design conditions. It was determined that the calculated fatigue usage factor remains less than 1.0 and the lower core plate is, therefore, structurally adequate at the revised design conditions for the 1.4 percent power uprate.

5.4 FUEL ASSEMBLY

The Salem Units 1 and 2 17X17 Vantage 5 Hybrid (V5H) without Intermediate Flow Mixers (IFMs), 17X17 Vantage+ (V+) without IFMs, and 17X17 Robust Fuel Assembly (RFA) with IFMs fuel designs were evaluated to determine the impact of the 1.4 percent uprate on the fuel assembly structural integrity. Since the core plate motions for the seismic and LOCA evaluations are not affected by the uprated conditions, there is no impact on the fuel assembly seismic/LOCA structural evaluation. The 1.4 percent uprate does not increase operating and transient loads such that they will adversely affect the fuel assembly functional requirements. Therefore, the fuel assembly structural integrity is not affected; and the homogeneous and mixed core seismic and LOCA evaluations of the 17X17 V5H without IFMs, 17X17 V+ without IFMs, and 17X17 RFA with IFMs fuel designs for Salem Units 1 and 2 are still applicable for the 1.4 percent uprate.

5.5 CONTROL ROD DRIVE MECHANISMS

The upper head of the Salem reactor vessel is exposed to fluid from the hot leg. Consequently, the control rod drive mechanisms (CRDMs) are also subjected to hot leg temperatures. Higher temperatures are more limiting for CRDM design. The maximum T_{hot} evaluated in the design basis analysis was 616.3°F. The maximum T_{hot}, shown earlier in Section 2, for the 1.4 percent uprate is 613.1°F and is, therefore, bounded by the previous analyses.

According to Section 3.1, the NSSS design transients used in the original stress analyses remain applicable to the 1.4 percent uprate program. Since these same transients remain applicable, the original stress analyses completed for the CRDMs remain applicable without change for the 1.4 percent uprating.

Based upon the above evaluation, it is concluded that the previous analyses performed and reported for the Salem CRDMs remain applicable for the 1.4 percent uprated conditions.

5.6 REACTOR COOLANT PIPING AND SUPPORTS

The 1.4 percent power uprate parameters were reviewed for impact on the existing design basis analysis for the reactor coolant loop (RCL) piping and supports. The revised RCS temperatures listed in Section 2, in particular T_{hot} and T_{cold}, can potentially alter the loads and stresses presently calculated for the RCL piping, primary loop nozzles, equipment nozzles and supports, and pressurizer surge line piping. The changes in RCS temperatures can also potentially impact the applicable fatigue usage values since these temperatures are used as initial conditions in some design transients.

5.6.1 Reactor Coolant Loop Piping, Equipment, and Branch Nozzles

As discussed in Section 3.1, the NSSS design transients are not affected by the uprating. Section 6.5 also indicates that that the 1.4 percent uprated conditions did not require a change to the LOCA hydraulic forcing functions. The loads on RCL piping and nozzles are, therefore, bounded by the loads in the existing analyses. An evaluation also confirmed that the existing fatigue usage factors for the RCL piping and nozzles remained bounding due to the conservative nature of the analysis (e.g. a conservative grouping of more severe transients). It is concluded that there is no effect on the existing loads, stresses, and fatigue usage factors.

As part of the Model F steam generator replacement for Unit 1, the reactor coolant system piping, components and supports were evaluated and the power uprate was found to have negligible effect on the resultant loads. The coefficients of thermal expansion, allowable stresses, steam generator primary nozzle stresses still remain bounded by the current analyses.

Pressurizer Surge Line Piping

The evaluation of pressurizer surge line stratification compared the change in T_{hot} resulting from the power uprate conditions. The 0.5°F increase in T_{hot} is a benefit to surge line stratification since it reduces the ΔT between the pressurizer and the hot leg. Therefore, the existing analysis remains bounding.

Equipment Supports

The equipment supports use a nominal zero gap between the equipment and the support structure. The increased primary side operating temperatures associated with the 1.4 percent uprated conditions may induce slight compression due to the potential thermal expansion. However, since the temperature increase is only 0.5°F, there would not be a measurable increase in displacements. Any change in displacements would be well within the measurement tolerance used in these gaps. As a result, the existing support analyses remain bounding.

5.7 LEAK BEFORE BREAK

The current leak-before-break (LBB) analysis, documented in WCAP-13659, justifies the elimination of large primary loop pipe rupture as the structural design basis. This applies to the primary loop piping. In order to demonstrate acceptability of the elimination of RCS primary loop pipe breaks, the following objectives must be achieved:

- Demonstrate that margin exists between the "critical" crack size and a postulated crack that yields a detectable leak rate
- Demonstrate that there is sufficient margin between the leakage through a postulated crack and the leak detection capability

- Demonstrate margin on applied load
- Demonstrate that fatigue crack growth is negligible

These were met in WCAP-13659.

As indicated in Section 5.6, there is no impact on the loads of the RCL piping due to the power uprate conditions. The effect on material properties due to the slight changes in temperature will have a negligible impact on the LBB margins documented in WCAP-13659. Therefore, the current LBB analysis remains applicable for the 1.4 percent power uprate conditions.

5.8 REACTOR COOLANT PUMPS AND MOTORS

5.8.1 Reactor Coolant Pump Structural Analysis

The reactor coolant pumps (RCPs) are located at the steam generator outlet in the reactor coolant loop. The maximum steam generator outlet temperature described in Section 2 is 542.5°F. This temperature is lower than the design basis temperature of 544.8°F, and, therefore, represents a less severe condition. Since the applicable NSSS design transients were also determined to be unaffected by the 1.4 percent uprate, the existing stress analyses remain applicable for the RCP pressure boundary components.

5.8.2 Reactor Coolant Pump Motor Evaluation

The limiting area of the RCP motor is the horsepower loading at continuous hot and cold operation. The current design basis analysis was based on a minimum steam generator outlet temperature of 528.7°F and associated best-estimate flow (BEF) of 89,200 gpm/loop. The lowest steam generator outlet temperature from Section 2 is 530.0°F, and the corresponding BEF for the 1.4 percent power uprate is 89,700 gpm/loop. The lower water density at 530.0°F will result in a slightly reduced horsepower load on the RCP motor during continuous hot operation. Likewise, the higher BEF will also result in a slightly lower horsepower load during both hot and cold operation.

Based upon the above evaluation, it is concluded that the current RCP motor evaluation is bounding for the 1.4 percent uprating parameters. Therefore, the Salem RCP motors are also acceptable for operation at the 1.4 percent uprated conditions.

5.9 STEAM GENERATORS

5.9.1 Model 51 Steam Generator Structural Integrity

An evaluation was performed to demonstrate that the design basis structural and fatigue analysis for the Model 51 steam generators in Salem Unit 2 is not affected by the 1.4 percent power uprate. The design conditions, such as the primary and secondary pressures and temperatures, that affect the structural performance of the steam generator components for the power uprate, were presented in Section 2. These uprated parameters were compared to the parameters for the existing steam generator structural and fatigue analysis. This comparison indicated that the design conditions for the 1.4 percent uprate are enveloped by the parameters for the current design basis analysis. In addition, Section 3.1 indicates that the NSSS design transients are not affected by the uprating. Therefore, the current structural and fatigue analysis for the Unit 2 steam generators remains valid.

5.9.2 Model F Steam Generator Structural Integrity

At Salem Unit 1, new Model F SGs have been in operation since cycle 13 (1998). The Model F SGs installed at Salem Unit 1 underwent the required design and licensing basis reviews before installation. The evaluations demonstrated that (1) the design parameters (e.g. pressures, temperatures, steam quality, primary system flow, steam flow and feedwater flow) specified for the Model F SGs were the same or more conservative than the parameters specified for the original SGs; and (2) the design transients and resulting fatigue cumulative usage factors remain valid. The proposed uprate results in slight changes to the SG operating parameters, but these are still enveloped by the original design parameters for the Model F SGs. The uprate effort reviewed the structural evaluations, including the NSSS design transients, performed for the Model F SGs. The results of this review concluded that the current design basis structural and fatigue analyses for the Unit 1, Model F SGs remain valid for the proposed 1.4% power uprate.

5.9.3 Steam Generator U-Bend Wear

An assessment of the steam generator U-bend wear was performed for the Model 51 steam generators in Salem Unit 2 to project the increase in the number of steam generator tubes that would require plugging as a result of the 1.4 percent power uprate. The highest power level and the lowest steam pressure characterize the limiting condition for U-bend wear. The 1.4 percent increase in power and minimum steam pressure associated with the power uprate are clearly bounded by the conditions analyzed in the current design basis analysis. The results of the design basis analysis concluded that only one additional tube per steam generator would be subject to plugging as a result of long-term operation at those conditions. Since the limiting conditions for the 1.4 percent uprate program are considerably less severe, increased U-bend wear for the Salem uprating program is not significant. It is expected that less than one additional tube per steam generator would be affected after long-term operation at the uprated conditions.

Salem Unit 1, new Model F SGs have been in operation since cycle 13 (1998). These SGs are designed for a 40 years life. It is standard practice to calculate normal operational tube wear for the design life of the SG to ensure adequate margin exists in the tube wall thickness. The original design parameters of the Unit 1 SGs bound the expected operating parameters for the uprated power condition. More specifically, the design steam flow for the SG, which is a major contributor of wear, bounds the analytical steam flows presented in Section 2. Therefore, power uprate has little to no effect on the expected wear rates of the tubing.

5.9.4 U-Bend Fatigue Evaluation

An evaluation was performed to determine the impact that the revised design conditions associated with the 1.4 percent uprating had on the steam generator U-bend fatigue. Key operating conditions used as input to the U-bend fatigue evaluation are steam flow, circulation ratio, steam pressure, and primary temperatures. The evaluation focused on the most susceptible steam generator tubes in the plant. Although additional tubes could potentially become affected at lower steam pressures, the analysis only considered the most susceptible tubes since it is unlikely that the necessary combination of steam pressure and power level required to affect any additional tubes would occur.

The evaluation found that some tubes would be susceptible to high cycle fatigue at the uprated conditions with the plant operating at lower steam pressures.

5.9.5 Evaluation of Steam Generator Tube Degradation Mechanisms

The revised design conditions will have a negligible impact on the existing and potential tube degradation mechanisms. Section 2 indicates that the design T_{hot} is expected to increase by 0.5°F for the 1.4 percent uprate and is considered to be the most sensitive operating parameter with respect to corrosion. The primary system pressure of 2250 psia is unchanged. Also, the reduction in steam pressure can have a secondary effect on corrosion. These changes are expected to have an insignificant effect on the tube corrosion mechanisms since they are relatively minor and are comparable to the range of uncertainties used in assessing corrosion.

5.10 PRESSURIZER

An analysis was performed to assess the impact of the revised NSSS parameters at the uprated conditions for Salem Units 1 and 2 on the pressurizer components. The limiting locations on the pressurizer from a structural standpoint are the surge nozzle, the spray nozzle, and the upper shell at the point of spray impingement. The conditions that affect the primary plus secondary stresses, and the primary plus secondary plus peak stresses, are the changes in the RCS hot leg temperature (T_{hot}), the RCS cold leg temperature (T_{cold}), and the pressurizer transients. A review of the revised temperature parameters in Section 2 showed that the changes in T_{hot} and T_{cold} are very small and are enveloped by the current stress analysis. Since the design transients (see Section 3.1) are also unaffected by the uprated conditions, the revised parameters do not impact the pressurizer stress and fatigue analysis. It is concluded that the pressurizer components meet the stress and fatigue analysis requirement of Section III of the ASME Code through the Winter 1965 Addenda for Unit 1, and the Summer 1966 Addenda for Unit 2, for the plant operation at the 1.4 percent uprated conditions.

5.11 NSSS AUXILIARY EQUIPMENT

The NSSS auxiliary equipment includes the heat exchangers, pumps, valves, and tanks in the auxiliary systems. An evaluation determined that the existing design conditions used in the fatigue analysis for these components envelop those reported in Section 2. The NSSS design transient evaluation presented in Section 3.1 also concluded that the power uprate design transients, which are applicable to some of the NSSS auxiliary valves, are bounded by the current design basis transients. Furthermore, as noted in Section 3.2, the current auxiliary equipment design transients, which apply to all the auxiliary heat exchangers, pumps, tanks, and the remaining valves, remain applicable for the 1.4 percent power uprate conditions. Therefore, the components will continue to meet their current design criteria since the fatigue usage values for each component will still be less than the allowable limit of 1.0.

6 NUCLEAR STEAM SUPPLY SYSTEM ACCIDENT EVALUATION

6.1 STEAM GENERATOR TUBE RUPTURE EVALUATION

The licensing basis steam generator tube rupture (SGTR) analysis for Salem is presented in the Updated Final Safety Analysis Report (UFSAR) Section 15.4.4. The SGTR analysis consists of a thermal and hydraulic analysis to determine the primary to secondary break flow and the steam released to the atmosphere, and a radiological consequences analysis to calculate the offsite radiation doses resulting from the event. The SGTR thermal and hydraulic analysis calculates the primary to secondary break flow and steam released to the atmosphere from the ruptured and intact steam generators for the time period before break flow termination. The analysis also calculates the long-term releases to the atmosphere from the intact steam generators after break flow termination. These results are then used to evaluate the offsite radiological consequences for an SGTR.

The current licensing basis SGTR thermal and hydraulic analysis was performed using a simplistic mass and energy balance method. The input parameters in the thermal and hydraulic analysis that are changing as a result of the power uprate are the nuclear steam supply system (NSSS) design parameters. These parameters include power, hot leg temperature, cold leg temperature, steam temperature, and steam pressure. The 1.4 percent increase in power results in a decrease in the steam pressure of about 5 psi. A decrease in steam pressure results in an increase in the primary to secondary break flow for all cases. The current licensing basis analysis included an 18 percent main steam safety valve (MSSV) blowdown to cover a 15 percent blowdown and 3 percent MSSV tolerance, by reducing the lowest safety valve setpoint by 18 percent. A 15 percent pressure reduction would be sufficient to cover both of these behaviors since the tolerance does not reduce the blowdown by an additional 3 percent. Therefore, margin exists in the MSSV setpoint assumed in the analysis. The MSSV setpoint is used in calculating the primary to secondary break flow post-reactor trip. This margin included in the calculation of the break flow in the reset pressure of the MSSV will cover the small increase of the break flow due to a decrease in steam pressure.

An increase in power, steam temperature, and hot leg temperature will also result in an increase in steam release due to an increase in the system energy. The methodology used in the current licensing basis analysis included a 4.5 percent increase in reactor power in the calculation of the feedwater flows and the steam releases. This 4.5 percent margin will cover the small increase in steam release due to the 1.4 percent power increase and minor changes to the design parameters.

Since the steam releases and the break flows determined in the current licensing basis analysis remain bounding, the input to the radiological consequences analysis is not affected by the power uprate.

6.2 STEAM LINE BREAK EVALUATION

The licensing basis safety analyses related to steam line break mass and energy releases were evaluated to determine the effect of a power uprate of up to 1.4 percent for Salem Units 1 and 2. The evaluation determined that the NSSS design parameters for Salem, as described in Section 2, remain unchanged or bounded by the safety analysis values. The nominal NSSS design parameters assumed in this evaluation are 330,000 gpm for the thermal design flow, 2250 psia for the reactor coolant system (RCS) pressure, 577.9°F for the RCS average temperature, and 432.8°F for the full-power feedwater temperature. The nominal steam temperatures assumed in the evaluation are 528.5°F for Unit 1 and 522.1°F for Unit 2.

6.2.1 Long-Term Steam Line Break Mass and Energy Releases Inside and Outside Containment

Critical parameters for the long-term steam line break event include the following conditions on the primary and secondary sides: NSSS power level, reactivity feedback characteristics including the minimum plant shutdown margin, initial and trip values for the steam generator water mass, main feedwater flow, auxiliary feedwater flow, main and auxiliary feedwater enthalpy, and the times at which steam line and feed line isolation occur. The input assumptions related to these critical parameters dictate the quantity of the mass and energy releases.

The power increase of up to 1.4 percent for the two Salem units will be offset by an equivalent reduction in the calorimetric uncertainty. The Analyses of Record applicable to both units for the inside and outside containment long-term steam line breaks assume a 2 percent power calorimetric uncertainty on a 3431 MWt NSSS power. A minimum 0.6 percent power calorimetric uncertainty applied to a maximum 1.4 percent power increase is equivalent to the licensing basis safety Analyses of Record for Salem. Therefore, as long as the sum of the power increase and power calorimetric uncertainty does not exceed 2 percent, there is no effect on either the current licensing basis long-term steam line break mass and energy release analyses or the UFSAR conclusions.

Since the mass energy released in postulated MS line breaks remains unchanged, the Equipment Qualification program is also not affected.

6.2.2 Short-Term Steam Line Break Mass and Energy Releases

Critical parameters for the short-term steam line break event are defined at no-load conditions. At this power level, the steam generator pressure is high, as is the steam enthalpy. Also, the steam generator inventory is greatest at no-load conditions. Since the power increase of up to 1.4 percent is not used as input to the short-term steam line break analysis, there is no effect on either the current licensing basis analysis or the UFSAR conclusions.

6.2.3 Radiological Steam Releases for Dose Calculations

Critical parameters for calculations of the radiological steam releases used as input to the dose evaluation model include the NSSS power, the RCS average temperature, and the steam temperature and pressure. Each of the primary side inputs is conservatively calculated assuming the engineered safeguards design power, which is equivalent to a 4.5 percent uprated power. The current Analysis of Record assumes primary and secondary side design parameters that are consistent with respect to the Salem Unit 2 operating conditions. Therefore, there is no effect on either the current licensing basis radiological steam release analysis or the UFSAR conclusions as a result of the power increase of up to 1.4 percent for Salem Unit 2.

The radiological doses consequences for a steam line break for the Model F SG were evaluated and found to be bounded by the Model 51 SG main steam line break because the mass inventory inside the Model F SG is less than the mass inside of the Model 51 SG. This mass relationship still holds for the proposed power uprate.

6.3 LOCA MASS AND ENERGY RELEASES

6.3.1 Long-Term LOCA/Containment Integrity Analysis

This analysis demonstrates the ability of the containment safeguards systems to mitigate the consequences of a hypothetical large-break loss-of-coolant accident (LOCA) (LBLOCA). The methodology for the most limiting Salem LOCA mass and energy release calculation is contained in WCAP-8264-PA. Based on this methodology, the Analysis of Record presently assumes an NSSS thermal power of 3570 MWt, which is about 4.3 percent greater than the current licensed NSSS power. In addition, the analysis applies an extra 2 percent to the 3570 MWt value to account for power measurement uncertainty. The improved thermal power measurement accuracy obviates the need for the full 2 percent power margin assumed in the analysis.

A subsequent LOCA containment integrity analysis was performed (after issuance of WCAP-8264-PA) as part of the fuel upgrade and margin recovery program for Units 1 and 2. In addition, a separate analysis was later performed for Unit 1 with replacement Model F steam generators. The methodology for these analyses is contained in WCAP-10325-PA. The analyses assumed a core power of 3411 MWt and also included the 2 percent power measurement uncertainty. The containment pressure response for this analysis is shown in Figure 15.4-91 in the UFSAR. Additional analyses were performed for both Units 1 and 2 to address an initial containment temperature of 122°F. Since the peak pressure for the double-ended pump suction (DEPS) LOCA containment integrity analysis for the margin recovery program for Unit 2 was lower (41.2 psig versus

45.8 psig), and the analysis for Unit 1 with the Model F steam generators was lower (38.6 psig versus 45.8 psig), the early vintage analysis with WCAP-8264-P-A methods remains bounding for both Salem units.

The power measurement margin is but one of many conservative assumptions used in the analysis. Taken together, the improved power measurement uncertainty and conservative assumptions provide substantial conservatism such that the margin of safety would not be reduced.

6.3.2 Short-Term LOCA Mass and Energy Release Analysis

Short-term LOCA mass and energy release calculations are performed to support the reactor cavity and loop subcompartment pressurization analyses. These analyses are performed to ensure that the walls in the immediate proximity of the break location can maintain their structural integrity during the short pressure pulse (generally less than 3 seconds) that accompanies a LOCA within the region.

The analysis inputs that may potentially change with the uprate are the initial RCS fluid temperatures. Since this event lasts for approximately 3 seconds, the single effect of power is not significant.

The short-term blowdown transients are characterized by a peak mass and energy release rate that occurs during a subcooled condition. The Zaloudek correlation, which models this condition, is currently used in the short-term LOCA mass and energy release analyses. This correlation was used to conservatively evaluate the impact of the changes in the RCS inlet and outlet temperatures for the 1.4 percent uprate relative to those used in the current Analysis of Record. The use of the lower temperatures maximizes the critical mass flux in the Zaloudek correlation.

The Salem Unit 2 short-term mass and energy releases used in the seven-loop compartment pressurization analysis cases and the reactor cavity pressure analysis cases were generated by Westinghouse. The double-ended cold leg guillotine releases were based on a core inlet temperature of 544°F. The double-ended hot leg guillotine releases were based on a vessel outlet temperature of about 606°F. The reactor cavity releases were based on a 100-in^2 cold leg break at an initial temperature of about 531°F.

The minimum core inlet temperature at the 1.4 percent maximum uprate conditions is 530.2°F and the minimum vessel outlet temperature is 601.8°F. The maximum instrument temperature uncertainty of 4.5°F is subtracted to further reduce the RCS conditions to be addressed. The lower core inlet and vessel outlet temperatures will result in slightly higher initial break flow rates into the reactor cavity and loop compartments.

The loop compartments and reactor cavity are now licensed under leak before break (WCAP-13659). The reduction in break area associated with assuming a break in the largest branch lines connected to the RCS primary loop, rather than a break in the main RCS piping, results in a decrease in mass and energy releases much greater than can occur due to any increase in RCS pressure or decrease in RCS temperature. Therefore, the current licensing basis, which is still based on breaks in the RCS main piping, remains bounding for the reactor loop compartment and reactor cavity region.

The current Salem Unit 1 short-term mass and energy release Analysis of Record is based on leak before break criteria as described above.

6.4 LOCA-RELATED ANALYSES

6.4.1 LBLOCA and SBLOCA

The current licensing basis LBLOCA and small-break LOCA (SBLOCA) analyses employ a nominal core power of 3411 MWt. The licensing basis analysis methodology employs a 2 percent calorimetric uncertainty (yielding an assumed core power of 3479 MWt) in accordance with the original requirements of 10 CFR 50, Appendix K. Consistent with the recent change to Appendix K, PSEG Nuclear, LLC proposes to reduce power measurement uncertainty to 0.6 percent based on the use of the Crossflow system. The existing 2 percent uncertainty margin in the LBLOCA and SBLOCA analyses would be reallocated with 1.4 percent applied to the increase in licensed power level and 0.6 percent retained to account for power measurement uncertainty. The total power assumed in the analyses remains 3479 MWt.

6.4.2 Post-LOCA Long-Term Core Cooling (LTTC)

The Westinghouse licensing position for satisfying the requirements of 10 CFR 50.46, Paragraph (b), Item (5), "Long-term cooling," (WCAP-8339) concludes that the reactor will remain shut down by borated emergency core cooling system (ECCS) water residing in the RCS/sump following a LOCA. Since credit for the control rods is not taken for a large-break LOCA, the borated ECCS water provided by the refueling water storage tank (RWST) and accumulators must have a concentration that, when mixed with other sources of water, will result in the reactor core remaining subcritical assuming all control rods out. The calculation is based upon the reactor steady-state conditions at the initiation of a LOCA and considers sources of both borated and unborated fluid in the post-LOCA containment sump. The other sources of water considered in the calculation of the sump boron concentration are the RCS, ECCS/residual heat removal (RHR) piping, and the boron injection tank (BIT) and piping. The water volumes and associated boric acid concentrations are not directly affected by the 1.4 percent power uprate. The core re-load licensing process will ultimately confirm that there are no required changes to these volumes and concentrations. Thus, there is no impact on the LTCC analysis.

6.4.3 Hot Leg Switchover

For a cold leg break post-LOCA, ECCS injection into the cold leg will circulate around the top of the full downcomer and out the broken cold leg. Flow stagnation in the core and the boiling off of near pure water will increase the boron concentration of the remaining water. As the boron concentration increases, the boron will eventually precipitate and potentially inhibit core cooling. Thus, at a designated time after a LOCA, the ECCS configuration is switched to hot leg injection to flush the core with water and keep the boron concentration below the precipitation point. The licensing basis analysis methodology employs a 2 percent calorimetric uncertainty in accordance with the original requirements of 10 CFR 50, Appendix K. Consistent with the recent change to Appendix K, PSEG Nuclear, LLC proposes to reduce power measurement uncertainty to 0.6 percent based on the use of the Crossflow system. The existing 2 percent uncertainty margin in hot leg switchover analysis would be reallocated with 1.4 percent applied to the increase in licensed power level and 0.6 percent retained to account for power measurement uncertainty. The total power assumed in the analysis remains 3479 MWt.

6.5 REACTOR VESSEL, LOOP, AND STEAM GENERATOR LOCA FORCES EVALUATION

The purpose of a LOCA hydraulic forces analysis is to generate the hydraulic forcing functions and hydraulic loads that occur on RCS components as a result of a postulated LOCA. These forcing functions and loads are considered in the structural design of the NSSS components.

In support of the 1.4 percent uprating conditions for Salem Units 1 and 2, an assessment of the impact of uprated RCS conditions from Section 2 on the LOCA forces was performed. This assessment demonstrated that the LOCA forces Analyses of Record for the vessel and Model 51 steam generator were based on more limiting RCS conditions than those conditions defined for the 1.4 percent uprate program.

Break area reduction margin was used, as allowed with leak-before-break (LBB) methodology (WCAP-13659), to estimate the change in reactor coolant loop forces for the uprate program. The estimated increase to the LOCA loop forces due to the change in RCS temperatures for the uprate was then compared to the estimated decrease in LOCA loop forces due to the break area reduction. The comparison showed that the loop force reduction from the break area margin more than offset the increase in loop forces associated with the uprated conditions.

Thermal hydraulic analyses were performed as part of the Model F SG replacement project to determine the forcing functions, mass and energy releases, and Asymmetric Cavity Pressurization (ACP). All of these evaluations were based on the Leak Before Break criteria.

As part of the proposed power uprate PSEG Nuclear reviewed the RCS and RSG blowdown and ACP analyses, jet impingement and thrust loads and concluded that the results obtained from the analyses of record remain bounding for the proposed power uprate conditions

Therefore, it is concluded that the existing LOCA hydraulic forces Analyses of Record supporting Salem Units 1 and 2 remain conservative.

6.6 NON-LOCA/TRANSIENT ANALYSES

The 1.4 percent uprate can potentially impact a number of different areas related to the non-loss-of-coolant-accident (LOCA) safety analyses. These include the reload related inputs (i.e., reactivity assumptions), protection system setpoints, and initial condition uncertainties. The following non-LOCA evaluation assumes that the reload related inputs will not be impacted and will be verified as part of the normal reload process, prior to the implementation of the power uprate. As discussed below, the protection system setpoints are not impacted (with the exception of a small change to the overtemperature Delta T (OTΔT) $f_{\Delta I}$ penalties).

Initial Power Conditions Assumed in the Safety Analyses (UFSAR Section 15.1.2)

The non-LOCA safety analyses can be divided into those events that account for uncertainties in the RCS temperature, pressure, power, and flow deterministically by applying the uncertainties to the initial conditions, and those events that statistically convolute the uncertainties into the departure from nucleate boiling (DNB) design limit (i.e., those events analyzed with the Revised Thermal Design Procedure (RTDP)).

With the use of the Crossflow device, the power measurement uncertainty is now $\pm$0.6 percent, as noted in Section 6.7 of this report. All of the other initial condition uncertainties (i.e., average RCS temperature, pressurizer pressure and RCS flow) remain unaffected.

The effect of the revised power measurement uncertainty has been accounted for in the evaluations of the various non-LOCA accidents discussed below. For analyses that utilize RTDP methods for the calculation of the minimum departure from nucleate boiling ratio (DNBR), the uncertainties are accounted for in the minimum DNBR safety analysis limit rather than being accounted for explicitly in the analyses.

Trip Points and Time Delays to Trip Assumed in Accident Analyses (UFSAR Section 15.1.3)

The protection setpoints remain unchanged. However, the nuclear instrumentation system (NIS) trips (power range neutron flux), which are based on a fraction of nominal, are effectively increased by the amount of the power increase. That is, 118 percent of the current power is different than 118 percent of the increased nominal power. In general, this is not a concern since transients, such as rod ejection, which rely on the high flux protection, have a rapid increase in the nuclear power and would be unaffected by an effective increase in the setpoint of less than 2 percent.

With respect to the OT and overpower (OP)ΔT setpoints, an increase in the nominal power affects the core thermal limits (exit boiling limits and DNB limits). The DNB margin can be allocated such that the core thermal limits remain unchanged, but the exit boiling limits change. Even though DNB margin can be allocated such that the DNB limits remain unchanged, the limits are affected by the increased power when they are converted from T_{in} versus power space to

T_{avg} versus power space, since the OTΔT and OPΔT setpoints are T_{avg}-based setpoints.

Based on the 1.4 percent increased core power, a revised set of core thermal limits was prepared using the RTDP methodology. It was not necessary to change the DNB design basis since existing analysis margin was used to offset the reduction in margin from the increased core power. Using the revised set of core thermal limits, it was determined that the OTΔT and OPΔT setpoints did not need to be modified to accommodate the increased core power. However, the OTΔT $f_{\Delta I}$ penalties presented in the Salem Units 1 and 2 Technical Specifications need to be changed slightly. The OTΔT and OPΔT setpoints and the corresponding $f_{\Delta I}$ penalties that support the Salem uprating are as follows:

K1 = 1.35 safety analysis limit (SAL)	K2 = 0.02037	K3 = 0.00102
K4 = 1.157 (SAL)	K6 = 0.00149	

$f_{\Delta I}$ penalty:

Positive side = 2.37%/% for all ΔIs greater than +11%

Negative side = 2.34%/% for all ΔIs less than -33%

These setpoints are applicable to both Salem units and all current fuel types (17X17 V5H without IFMs, 17X17 V+ without IFMs, and RFA with IFMs).

6.6.1 Non-LOCA/Transient Analyses Performed With Statistical Methods

Note that the evaluations that follow for DNB events discuss safety analysis margin. This is defined as the difference between the SAL and the actual LOFTRAN calculated value. The margin maintained between the design limit DNBR and the SAL DNBR is unaffected by the use of accident-specific safety analysis margin.

Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power (UFSAR Section 15.2.2)

This event is defined as an inadvertent addition of reactivity to the core caused by the withdrawal of rod cluster control assembly (RCCA) banks when the core is above the no-load condition. The event is analyzed at 10 percent, 60 percent and 100 percent of rated thermal power assuming beginning-of-life (BOL) and end-of-life (EOL) reactivity conditions and a spectrum of reactivity insertion rates. Unless terminated by manual or automatic action, the power mismatch between the reactor core power generation and the steam generator heat extraction results in a coolant temperature increase that could potentially lead to a violation of the DNBR limits. Therefore, to prevent damage to the fuel cladding, the reactor protection system is designed to terminate the transient before the DNBR limit is violated. The purpose of the analysis is to demonstrate that the minimum DNBR remains above the limit value.

The current limiting case (with respect to the DNB acceptance criteria) is based on an initial core power level equal to 100 percent of the rated thermal power. For the limiting cases, the 1.4 percent power increase would result in an increase in the peak core heat flux of approximately 1.4 percent with no appreciable increase in the reactor trip time. An increase in power of 614.0 MWt will result in a reactor trip for the pre-uprated conditions (18 percent of 3411 MWt) and an increase of 622.6 MWt will result in a reactor trip for the uprated conditions. This is a difference of only 8.6 MWt and even slow reactivity insertion rate cases (i.e., < 10 pcm/sec) result in power increases of this magnitude over very small time intervals (~0.1 seconds). The current Analysis of Record shows approximately 14 percent safety analysis DNB margin, which is sufficient to offset an increase of approximately 1.4 percent in peak power (1.4 percent power corresponds to an expected upper bound 3.5 percent DNBR reduction). Thus, the results of this evaluation show that the DNB design basis continues to be met and the conclusions presented in the UFSAR remain valid.

Rod Cluster Control Assembly Misalignment (UFSAR Section 15.2.3)

The RCCA misalignment analysis includes the following events:

- One or more dropped RCCAs within the same group
- A dropped RCCA bank
- A statically misaligned RCCA

These transients are investigated to demonstrate that the DNB design basis is met.

An evaluation confirmed that the current statepoints were still applicable for use at uprated conditions. It was also verified that there is sufficient DNB margin to accommodate the 1.4 percent uprating. Thus, the conclusions presented in the UFSAR remain valid.

Partial and Complete Loss of Forced Reactor Coolant Flow (UFSAR Sections 15.2.5 and 15.3.4)

The partial/complete loss of forced reactor coolant flow events may result from mechanical or electrical failure(s) in the reactor coolant pump(s) (RCP(s)). These faults may occur from an undervoltage condition in the electrical supply to the RCP(s) or from a reduction in motor supply frequency to the RCP(s) due to a frequency disturbance on the power grid. These analyses demonstrate that the minimum DNBR remains above the limit value. The limiting results are obtained at full-power conditions and occur very quickly following initiation of the event.

An analysis determined that the 1.4 percent uprating has a negligible effect on the transient statepoints. As such, the current transient statepoints remain applicable and can be used with the increased nominal heat flux (by 1.4 percent) when evaluating the DNB acceptance criteria. This analysis concluded that the DNB design basis continues to be met. The conclusions documented in the UFSAR remain valid.

Loss of External Electrical Load and/or Turbine Trip (UFSAR Section 15.2.7)

This event is defined as a complete loss of steam load from full power without a direct reactor trip, or a turbine trip with or without a direct reactor trip. It is analyzed to demonstrate that: 1) primary and secondary pressures remain below

110 percent of design, and 2) the minimum DNBR remains above the safety analysis limit value.

The loss of load/turbine trip analysis includes cases both with and without automatic pressure control. Although cases have historically been analyzed with both minimum and maximum reactivity feedback conditions, this accident, as an RCS heatup event, is limiting at minimum feedback conditions. Maximum feedback cases are bounded by the minimum feedback cases and, therefore, do not need to be addressed separately. The case with pressure control is analyzed to investigate the RCS heatup effect on the DNBR response. The licensing basis analysis shows that there is 38 percent analysis margin for Unit 1 and 59 percent analysis margin for Unit 2. This is sufficient safety analysis margin to offset the penalty associated with a 1.4 percent uprating (1.4 percent power corresponds to an expected upper bound 3.5 percent DNBR reduction).

The case performed without pressure control is used to investigate RCS peak pressure and is performed with a 2 percent power uncertainty, as noted in Section 6.6.2. This case remains applicable and bounds the 1.4 percent uprating.

The results of the evaluation show that the DNB design basis continues to be met, the peak primary and secondary pressures remain below their respective limits, and the conclusions presented in the UFSAR remain valid.

Excessive Heat Removal Due to Feedwater System Malfunctions (UFSAR Section 15.2.10)

Reductions in the feedwater temperature or the addition of large amounts of feedwater to the steam generators result in excessive heat removal from the plant primary coolant system. Analyses are performed under both full-power and no-load conditions to demonstrate that the DNB design basis is met. Both single-loop and multiple-loop feedwater malfunctions are considered, as well as operation with both manual and automatic rod control.

The cases initiated at hot zero power are unaffected by the uprating, so the hot zero power licensing basis cases remain applicable and bounding.

For the full-power cases for Salem Unit 2, the most limiting case is the single loop feedwater malfunction with automatic rod control. There is currently 20 percent margin to the DNB limit for this case. For Unit 1, the most limiting full-power case is the multi-loop automatic rod control case. In the analysis, several variations of this case were considered and the most limiting case shows over 8 percent margin to the SAL. Thus, for both units there is sufficient safety analysis margin to offset the penalty associated with a 1.4 percent uprating (1.4 percent power corresponds to an expected upper bound 3.5 percent DNBR reduction).

Accidental Depressurization of the Reactor Coolant System (UFSAR Section 15.2.12)

An accidental depressurization of the RCS could occur as a result of an inadvertent opening of a pressurizer relief or safety valve. The purpose of the analysis is to demonstrate that the minimum DNBR remains above the limit value.

The current licensing basis analysis indicates that there is 29 percent analysis margin. This is sufficient safety analysis margin to offset the penalty associated with a 1.4 percent uprating (1.4 percent power corresponds to an expected upper bound 3.5 percent DNBR reduction). Thus, the DNB design basis continues to be met and the conclusions presented in the UFSAR remain valid.

Inadvertent Operation of Emergency Core Cooling System (UFSAR Section 15.2.14)

This analysis assumes that the safety injection system is inadvertently actuated. Two separate cases are considered for this event. One case that assumes no direct reactor trip as a result of ECCS actuation is investigated to verify that the DNB design basis is satisfied. This case is inherently non-limiting as the DNBR increases throughout the duration of the transient. The minimum DNBR never falls below its initial value. The DNBR design basis continues to be met for the 1.4 percent uprating and the conclusions presented in the UFSAR remain valid.

The other case is analyzed to investigate the potential for pressurizer filling due to continued ECCS injection and reactor coolant expansion resulting from residual heat generation. This case assumes a reactor trip coincident with event initiation and is performed at 102 percent power (i.e., 2 percent power uncertainty is included in the analysis). Therefore, as is discussed in Section 6.6.2, this case remains bounding for the 1.4 percent uprating.

Single Reactor Coolant Pump Locked Rotor (UFSAR Section 15.4.4)

A single RCP locked rotor event is based on the sudden seizure of an RCP impeller or failure of the RCP shaft. A reactor trip via the low RCS flow protection function terminates this event very quickly. Two cases are considered. The first case is done to determine the percentage of fuel rods expected to experience DNB. The second case investigates the peak primary and secondary pressure transients with respect to RCS and main steam system (MSS) pressure limits.

The DNB case is analyzed using RTDP assumptions and the initial power level is defined as the nominal full-power rating. The power level in the transient statepoints generated is in the form of fraction of the initial power level. It was determined that the 1.4 percent uprating has a negligible effect on the transient statepoints. As such, the current transient statepoints remain applicable and can be used with the increased nominal heat flux (by 1.4 percent) when evaluating the DNB acceptance criteria. This analysis concluded that the percentage of fuel rods expected to be in DNB is less than the percentage of fuel rods assumed to have failed in the locked rotor dose calculations.

The pressure case is analyzed with a 2 percent uncertainty included in the initial power level as described in Section 6.6.2. Thus, the current analyses remain bounding for the 1.4 percent uprating and the conclusions presented in the UFSAR remain valid.

Excessive Load Increase Incident (UFSAR Section 15.2.11)

This transient is defined as a rapid increase in the steam flow that causes a power mismatch between the reactor core power and the steam generator load demand. Cases are evaluated at BOL and EOL conditions with and without rod control to demonstrate that the DNB design basis is met. The transient response to this accident is that the reactor stabilizes at a new equilibrium condition corresponding to conditions well below that which would challenge the DNBR limit without generating a reactor trip.

The analysis includes four different cases: minimum and maximum reactivity feedback with and without automatic rod control. The most limiting of these cases is the minimum feedback/automatic rod control case. This case shows 38 percent safety analysis margin to the DNBR limit value. This is sufficient safety analysis margin to offset the penalty associated with a 1.4 percent uprating (1.4 percent power corresponds to an expected upper bound 3.5 percent DNBR reduction). Thus, the DNB design basis continues to be met and the conclusions presented in the UFSAR remain valid.

6.6.2 Non-LOCA/Transient Analyses Performed with Non-Statistical Methods

The following non-LOCA/transient analyses are currently analyzed with an explicit 2 percent power measurement uncertainty to increase the initial power level to 102 percent. This explicit 2 percent power uncertainty remains bounding for the 1.4 percent power uprate since the power uncertainty has been reduced to 0.6 percent.

- Loss of External Electrical Load and/or Turbine Trip – overpressure analysis (UFSAR Section 15.2.7)
- Loss of Normal Feedwater (UFSAR Section 15.2.8)
- Loss of Offsite Power to the Station Auxiliaries (UFSAR Section 15.2.9)
- Inadvertent Operation of Emergency Core Cooling System – Overfill Analysis (UFSAR Section 15.2.14)
- Major Rupture of a Main Feedwater Pipe – 102 percent power case (UFSAR Section 15.4.2.2)
- Single Reactor Coolant Pump Locked Rotor – overpressure, maximum cladding temperature, and maximum zirconium-water reaction analysis (UFSAR Section 15.4.4)

In addition to these transients, the following events require some additional explanation regarding why they are acceptable with the 1.4 percent uprating.

Uncontrolled Rod Cluster Control Assembly Bank Withdrawal from a Subcritical Condition (UFSAR Section 15.2.1)

This event is defined as an uncontrolled addition of reactivity to the reactor core caused by withdrawal of one or more RCCA banks, resulting in a rapid power excursion. This transient is promptly terminated by a reactor trip on the power range high neutron flux - low setpoint. Due to the inherent thermal lag in the fuel pellet, heat transfer to the RCS is relatively slow. The purpose of the analysis is to demonstrate that the DNB design basis is met.

This event is analyzed at zero power, so the initial conditions are unaffected by the 1.4 percent uprating. The statepoints, which are based upon fraction of nominal conditions, are unaffected by the 1.4 percent uprating. This is because the reactor trip, which occurs on the power range high neutron flux - low setpoint, is based on a fraction of nominal conditions (i.e., 35 percent). Thus, the time of trip is negligibly impacted. To address the 1.4 percent uprating, the limiting statepoints were evaluated with the increased heat flux associated with the 1.4 percent uprating. This evaluation showed that the DNB design basis is satisfied and the conclusions presented in the UFSAR remain valid.

Rupture of Control Rod Drive Mechanism Housing (UFSAR Section 15.4.6)

The Rupture of a Control Rod Drive Mechanism Housing event models the power range high neutron flux setpoints for primary protection. The event is the result of the assumed mechanical failure of a control rod mechanism pressure housing such that the RCS would eject the control rod and drive shaft to the fully withdrawn position. The transient responses for the hypothetical RCCA ejection event are analyzed at beginning-of-life and end-of-life for both hot full-power (HFP) and hot zero-power (HZP) operation in order to bound the entire fuel cycle and expected operating conditions. The analyses are performed to show that the fuel and cladding limits are not exceeded.

For the full-power cases, the 2 percent power uncertainty bounds the initial condition associated with the 1.4 percent uprating. The high neutron flux setpoint assumed in the analysis does not bound the setpoint associated with the 1.4 percent uprating, which would be approximately 1.4 percent higher. However, this difference would have a negligible impact on the results because of the rapid increase in the nuclear power. Thus, the HFP rod ejection analysis is bounded for the 1.4 percent uprate. The same argument applies for the HZP rod ejection cases.

Furthermore, the current licensing basis analyses show 6 percent margin to the fuel enthalpy limit (BOL/HFP case), 10 percent margin to the fuel melt limit (EOL/HFP case), 49 percent margin to the reacted Zirc limit (BOL/HZP case) and 67°F margin to the peak cladding temperature (PCT) limit (BOL/HZP case). These margins are sufficient to offset any penalties associated with the small delay that could occur on the reactor trip time. Thus, the conclusions presented in the UFSAR remain valid.

Accidental Depressurization of the Main Steam System and Major Rupture of a Main Steam Line (UFSAR Sections 15.2.13 and 15.4.2)

For these events, excessive steam relief is assumed to cause an RCS cooldown that results in a positive reactivity excursion. The safety analyses are performed under zero-power initial conditions and show that the minimum DNBR limit is not exceeded as a result of any potential recriticality. The results of the major rupture of a main steam line analysis bound the results of the accidental depressurization analysis.

The transient statepoints remain unaffected by the 1.4 percent uprating. A detailed DNB evaluation with the increased nominal heat flux associated with the 1.4 percent uprating concluded that the DNB design basis continues to be met. As such, the conclusions presented in the UFSAR remain valid. Note that the accidental depressurization of the MSS case (i.e., credible break) is always bounded by the major rupture of a main steam line case (i.e., hypothetical break). As such, the credible break case is no longer analyzed and did not need to be considered as part of this evaluation.

Uncontrolled Boron Dilution (UFSAR Section 15.2.4)

The boron dilution event is analyzed to demonstrate that the operator has at least 15 minutes (30 minutes in Mode 6) to terminate the RCS dilution before a complete loss of shutdown margin occurs. The critical parameters in the determination of the time available to terminate the dilution include the overall RCS active volume, the dilution flow rate, and the initial and critical boron concentrations. The analysis does not explicitly model or consider the initial power level.

The Mode 1 analysis (manual rod control case) uses the reactor trip time (via OTΔT) from the Rod Withdrawal at Power analysis in part of the calculation. The change in the time of reactor trip would be expected to be negligible (i.e., much less than 1 second). The licensing basis analysis assumes a conservative trip time of 120 seconds when only 89 seconds was needed. The existing analysis remains conservative and bounding. None of the other cases would be impacted by the uprated conditions. The conclusions documented in the UFSAR remain valid.

6.7 REACTOR TRIP AND ENGINEERED SAFETY FEATURE ACTUATION SYSTEM SETPOINTS

The impact of the proposed uprate was evaluated for setpoints which could be affected by changes in process conditions. This evaluation included the steam generator water level low-low reactor trip function and high-high turbine trip. The maximum additional process error was found to be negligible. No changes are required to the existing TS setpoints and allowable values for these functions.

6.8 REVISED THERMAL DESIGN PROCEDURE UNCERTAINTIES

Westinghouse WCAP-13651 provides the basis for the RTDP uncertainties that are used in the plant safety analyses. These include T_{avg} (rod) control, pressurizer pressure control, RCS flow measurement (calorimetric) and indication, and power measurement (calorimetric). The effect of the power uprating on these uncertainties is discussed in the following subsections.

6.8.1 Power Calorimetric

Typical plant safety analysis evaluations assume a power calorimetric uncertainty of 2.0 percent rated thermal power (RTP). This power uprate concept is based on a reduction in the power calorimetric uncertainties such that the calculated uncertainties plus the magnitude of the power uprate remains within the original plant analysis assumptions. Therefore, the final calculated uncertainties are used to determine the magnitude of the power uprate. The primary source in reducing the power calorimetric uncertainties is a reduction in the uncertainties associated with the measurement of secondary side feedwater mass flow. New calculations were performed to determine the uncertainties for the daily power calorimetric assuming the use of the Crossflow measurement system to determine total feedwater mass flow. The calorimetric uncertainty calculation assumed an uncertainty for total feedwater system mass flow of 0.5 percent. This assumption is conservative since the calculated uncertainties for the site specific installations are actually less than 0.5 percent. The 0.5 percent feedwater mass flow error in combination with the remaining uncertainty components results in a total 95/95 power measurement uncertainty of ±0.6 percent RTP. A power measurement uncertainty of ±0.6 percent justifies a power uprate of 1.4 percent RTP. The methodology used to determine the power calorimetric uncertainties is documented in Attachment 7.

6.8.2 T_{avg} (Rod) Control and Pressurizer Pressure Control

The uncertainties associated with the T_{avg} and pressurizer pressure control systems are not affected by changes in plant parameters for the 1.4 percent power uprate conditions. Therefore, the uprate does not necessitate changes to the uncertainties documented in WCAP-13651 for these controllers.

6.8.3 Reactor Coolant System Flow Calorimetric

The RCS flow calorimetric calculation uses nominal plant conditions for feedwater temperature and steam pressure as part of the input assumptions for the uncertainty calculations. The small changes in these plant parameters due to the power uprate conditions do not change the calculated RCS flow uncertainties. Therefore, the uprate does not necessitate changes to these uncertainties documented in WCAP-13651.

7 NUCLEAR FUEL

This section summarizes the evaluations performed to determine the effect of the 1.4 percent uprating on the nuclear fuel. The core design for Salem Units 1 and 2 is performed for each specific fuel cycle and varies according to the needs and specifications for each cycle. However, some fuel-related analyses are not cycle-specific. The nuclear fuel review for the 1.4 percent uprate evaluated the fuel core design, thermal-hydraulic design, and fuel rod performance.

7.1 FUEL CORE DESIGN

A representative equilibrium cycle model was developed to evaluate the effects of the 1.4 percent uprate conditions on the fuel core design. Since the power uprate is relatively small, the representative cycle is adequate to demonstrate the sensitivity of reload parameters to the power uprate conditions. The expected ranges of variation in key parameters were determined.

The methods and core models used in the uprate analyses are consistent with those presented in the Salem Updated Final Safety Analysis Report (UFSAR).

No changes to the nuclear design philosophy, methods, or models are necessary due to the uprating. The core analyses for the uprating were performed primarily to determine if the values previously used for the key safety parameters remain applicable prior to the cycle-specific reload design.

The core analyses show that the implementation of the power uprate will not result in changes to the current nuclear design basis documented in the UFSAR. The impact of the uprate on peaking factors, rod worths, reactivity coefficients, shutdown margin, and kinetics parameters is either well within normal cycle-to-cycle variation of these values or controlled by the core design and will be addressed on a cycle-specific basis consistent with reload methodology.

7.2 CORE THERMAL-HYDRAULIC DESIGN

The core thermal-hydraulic analyses and evaluations were performed at the uprated core power level of 3459 MWt. The analyses assumed that the uprated core designs are composed of Robust Fuel Assemblies (RFAs) with Intermediate Flow Mixer (IFM) grids and VANTAGE 5H (V5H) fuel assemblies without IFMs. A reduced $F_{\Delta H}$ was credited for the burnt V5H assemblies without IFMs. Because these assemblies are at least once burned and are typically placed in relatively low power locations, this is a reasonable assumption. As a result of this peaking factor assumption, separate peaking factors will be defined in the Core Operating Limits Report for each fuel type.

The thermal-hydraulic design criteria and methods for the 1.4 percent uprating are consistent with those presented in the Salem UFSAR. The design method employed to meet the departure from nucleate boiling (DNB) design basis is the Revised Thermal Design Procedure (RTDP). The WRB-2 DNB correlation is used for the 17X17 RFA with IFMs fuel assemblies and the WRB-1 DNB correlation is used for the 17X17 V5H without IFMs and 17X17 V+ without IFMs fuel assemblies.

To support operation of the Salem units at the uprated conditions, DNBR re-analysis was performed to define new core limits, axial offset limits, and Condition II accident acceptability. The impact on the protection setpoints and specific events are discussed in Section 6.5. The analyses demonstrated that the DNB design basis continues to be met for the 1.4 percent uprating and the conclusions presented in the UFSAR remain valid.

7.3 FUEL ROD DESIGN

All fuel rod design criteria evaluated for a standard reload design have been evaluated for an equilibrium cycle model of the Salem units at a 1.4 percent uprated power (3459 MWt core power). Conservative conditions were utilized. Rod burnups up to 62,000 MWD/MTU were considered for three-cycle fuel. The current feed product (RFA design identical to Salem Unit 2 Region 14) was assumed for all fuel in the core, and integral fuel burnable absorber (IFBA) loadings as high as 1.5X were evaluated. The results of these evaluations demonstrated that this fuel meets all fuel rod design limits with margin.

8 ELECTRICAL POWER

8.1 ELECTRICAL DISTRIBUTION SYSTEM

As a result of this uprate, no ac or dc auxiliary load ratings are expected to change, and the loads are not expected to experience additional demands above their ratings. The main generator electrical parameters remain the same, and the uprate capacity remains within the generator rating. The voltage controls and grid source impedance at the PJM 500 kV grid will not be affected by this uprate; therefore, the evaluated voltages and short circuit values at different levels of station auxiliary electrical distribution system will not change as a result of this uprate.

8.2 TURBINE/GENERATOR

The generator is a steam turbine-driven polyphase 4 pole machine, rated at 1300 MVA, and a 0.9 power factor. This rating is based on 75-psig hydrogen pressure, which is supplemented with water cooling for the stator and rotor.

At the present core thermal power rating of 3411 MWt, Salem Units 1 and 2 main generator gross electrical outputs are 1162 and1166 MWe respectively. System operating procedures will be revised as necessary to ensure that the anticipated net increase of 16 MWe will lie within the limits of the generator capability curves. Therefore, there will be no generator limitations to prevent operation at the uprated core thermal power of 3459 MWt.

PSEG Nuclear has not identified any changes to equipment protection relay settings for the generator. Process alarm setpoints for the generator and exciter will be revised as required.

To deliver electrical power provided by the generator to the transmission system, each Salem unit is equipped with an isolated phase bus, three main transformers and switchyard breakers and switches. The components are rated to deliver electrical power at or in excess of the main generator nameplate rating of 1300 MVA.

The electrical systems associated with the turbine auxiliary systems are not affected by the uprate.

8.3 ISOPHASE BUS

The isophase bus is designed with forced cooling rating of 32000 amps. These ratings are greater than the main generator rating of 30022 stator amps at 1300 MVA and are well in excess of the anticipated generator output. The isophase bus will support the power increase with no modifications.

8.4 MAIN TRANSFORMERS

System operating procedures will be revised as necessary to ensure that operation of the generator remains within applicable limits for the main transformers at the 1.4 % uprated power.

8.5 SWITCHYARD

The switchyard equipment exceeds the nameplate rating of the main generator. All 500 kV switches and breakers are rated a minimum of 2000 amperes, which exceeds the main generator maximum output current (at the 500 kV switchyard) of approximately 1400 amperes at its nameplate rating of 1300 MVA. The switchyard will accept the additional load without the need for any hardware modifications.

8.6 500 KV GRID STABILITY

No stability issues were identified during a feasibility study performed in support of the proposed uprate. An impact study including stability analysis will be completed before implementation of the proposed change.

9 <u>BALANCE OF PLANT</u>

The balance of plant (BOP) systems were designed for the turbine valves wide open condition, corresponding to a NSSS power of approximately 3570 MWt, a power in excess of the proposed uprate. No BOP hardware changes and no significant setpoint changes are anticipated since the uprate should be accommodated within the excess capacity of the as-designed BOP equipment.

BOP systems used to perform safety-related and normal operation functions were reviewed for the uprate. The review included all or portions of the main steam, feedwater, turbine, condenser, condensate, heater drains, service water, circulating water, turbine auxiliary cooling, HVAC, and support systems.

The BOP systems evaluation demonstrated the following:

1. There will be no changes in the primary side systems other than those directly caused by increasing the reactor thermal power by 1.4%. The higher core power results in about a 1°F increase in core delta-T.
2. The T_{avg} program will remain the same. For a given value of T_{avg} from the allowable range of 566.0°F through 577.9°F, T_{hot} will go up by approximately 0.5°F, and T_{cold} will go down by about the same amount.
3. There will be no changes in the ESF system requirements. This includes no changes in setpoints and system flows.
4. The only systems/components subject to increased flow are in the turbine cycle – main steam, bleed steam, reheat steam, main turbine, condenser, condensate, feedwater, and heater drains. These are the systems typically included in the BOP heat balance.
5. No increased flows are required for any intermediate cooling systems or ultimate heat sinks. This includes the RHR Shutdown cooling and Spent Fuel Pool Cooling. The increased heat load can be accommodated by the existing design.
6. The only hardware changes are those required for adding a more accurate, calorimetric feedwater flow measurement system. This system does not replace the present feedwater flow measurement system that provides continuous control room indication and feedwater flow control. The new,

Crossflow measurement system is used solely to periodically perform the calorimetric calculation and calibrate the power range meters.

7. The normal ambient containment temperature is not expected to increase. None of the electrical loads (e.g. RCP motors) will change. With a constant T_{avg} and a slight increase in power, T_{hot} will be higher but T_{cold}, $T_{feedwater}$, and T_{steam} will all be slightly lower. Although some piping and component heat loads will go up, it is expected that they will be offset by the reductions.

10 RADIOLOGICAL CONSEQUENCES

10.1 POST-ACCIDENT DOSES

Control Rod Ejection Accident, Fuel Handling Accident, LOCA, and Locked Rotor Accident source terms are based on the core thermal power level of 3600 MWt; therefore, the 1.4% uprate is bounded by the current analyses. This also satisfies Regulatory Guide 1.49 which requires that analyses of possible offsite radiological consequences of postulated design bases accidents be performed at 102% of rated power.

Main Steam Line Break Outside Containment and Instrument Line Pipe Break source terms are based on the design basis source terms (1% failed fuel for noble gas and concurrent/pre-accident spikes for iodine); therefore, they are not impacted by the proposed power uprate.

10.2 DOSES FROM NORMAL EFFLUENT RELEASES

The assumed offsite doses (10 CFR 50, Appendix I) resulting from the liquid and gaseous effluent releases are based on a core power of 3558 MWt. Therefore, the current analyses bound the proposed uprate, 3459 MWt core power.

Radiation monitor setpoints are based on the various regulatory requirements and they are independent of the core thermal power. The technical specifications limit the primary activity and the primary to secondary leakage. These will not be changed with the proposed uprate. Therefore, radiation monitor setpoints are not affected by the proposed power uprate.

The design basis source terms used for shielding design are very conservative. They are not impacted by 1.4% power uprate.

11 PLANT OPERATIONS

11.1 PROCEDURES

Plant procedures will not require significant changes for the uprate. The same steps and sequence of steps will be maintained. Procedural limitations on power operation due to BOP equipment unavailability will be revised as necessary to account for the increase in NSSS power to 3471 MWt.

The only new procedures required are for operation and maintenance of the Crossflow system.

Specific operator actions to be taken when the Crossflow system is inoperable are discussed in Section 1.4.2 and will be addressed in procedural guidance.

11.2 EFFECT ON OPERATOR ACTIONS

ESF System design and setpoints, and procedural requirements already bound the proposed uprate. The uprate will not change the time available for the operator to respond, or add additional steps.

Similarly, although the uprate will reduce the margin available during the limiting BOP transients, it does not change the required operator response. The most limiting transient, a SGFP trip, does not impose any new requirements for operator response.

11.3 ALARMS, CONTROLS AND DISPLAYS

There will be minimal impact on alarms, controls and displays for a 1.4% uprate.

11.3.1 Indicated Power

Reactor power 100% power will be scaled to the new uprated power. Therefore the increased megawatt rating will indicate at 100% power.

11.3.2 Alarms

The Crossflow system will have alarms in the control room to alert Operators to conditions that impair its availability or accuracy.

No other alarm impacts are expected. It is not anticipated that any existing alarms will be modified or deleted. Alarms will be recalibrated as necessary to reflect small setpoint changes; however, no significant or fundamental setpoint changes are anticipated. Also, the operator response to existing alarms is anticipated to remain as before.

11.4 SAFETY PARAMETER DISPLAY SYSTEM

Process parameter scaling changes will be made as required for the Safety Parameter Display System (SPDS). There are no other impacts to the SPDS from the proposed uprate. Implementation of scaling changes will be controlled under PSEG Nuclear's software configuration change control program.

11.5 OPERATOR TRAINING

Since the power uprate is nominal and there is no change to how the plant will be operated, the impact on operator training is minimal. Plant operators will be briefed on:

- Offsetting the increased nominal reactor power by reducing the error margin for the calorimetric.
- Minor setpoint changes in the BOP systems.
- New procedures specific to the Crossflow improved flow measurement system used for the calorimetric calculation.

The effect on the plant simulator will be minimal. The simulator initial conditions will be revised to account for the increase from 3411 to 3459 MWt as 100%

power. The simulator OTΔT neutron flux penalties will be revised to reflect the revised flux penalties described in the proposed changes to the Technical Specifications. An additional overhead annunciator window will be added to alert operators to Crossflow trouble. No other changes to the simulator are required.

12 OTHER EVALUATIONS

12.1 10 CFR 50 APPENDIX R

The proposed increase in licensed power level will have negligible impact on the way the unit is operated, shutdown, and maintained. The uprate will not cause any area, piping or component temperature to increase by more than a fraction of a degree. The uprate will not cause electrical equipment to be operated beyond its rated capacity. Therefore, the uprate is not anticipated to increase the probability of a fire.

No equipment is being added, removed, or modified for the uprate with the exception of the Crossflow system. The modification to install this flow meter will verify that the existing requirements for combustible loadings are met. No change to the combustible loading assumptions is anticipated.

The uprate will not change any fire mitigation barrier or suppression requirements since these are not based on power level.

Appendix R safe shutdown calculations that use reactor power as an input (for example, decay heat removal and cooldown from 350°F to 200°F) have been reviewed. The impact of the 1.4% increase is minor and can be accommodated without changing the conclusions.

Since there is no change in the way the plant is operated or shutdown, the safe shutdown analysis is not changed.

Based on the above, it is concluded that there is no impact to Appendix R evaluations.

12.2 ENVIRONMENTAL QUALIFICATION

The uprate does not affect the environmental qualification of equipment for the following reasons:

- The integrated dose inside and outside containment during normal operation is based on 3558 MWt core power which bounds the proposed uprate.
- The post-LOCA dose calculations assumed source terms based on 3600 MWt core power. These calculations remain bounding.
- Temperature and pressure transients due to pipe breaks outside containment are unchanged as discussed in the section 12.4.
- The limiting temperature and pressure analysis for accidents inside the containment are unchanged because the present limiting accidents still bound the uprated condition.
- Normal ambient temperature is not impacted as discussed in Section 9.

12.3 STATION BLACKOUT

The only potential impact to the ability to withstand and recover from a station blackout (SBO) is the increased decay heat that must be removed from the RCS to keep the unit in hot standby. This is done by using the turbine driven AFW pump to supply water to the steam generators and exhausting steam through the main steam PORV's. Sections 4.2.1 and 4.2.3 discuss the MS PORV and AFW, respectively. They show that the MS PORV and AFW can meet SBO requirements at the uprated power. Area and room temperature transients are not expected to change as a result of the uprate since the initial temperatures and heat loads do not change.

12.4 HIGH ENERGY LINE BREAK

Design basis line breaks are postulated in the following lines:

- RCS including the pressurizer surge line
- Letdown
- Charging including the RCP seal injection
- MS including the branch line to the TDAFW pump
- Feedwater
- Steam generator blowdown
- Heating Steam
- Heating Water

The uprate does not increase the design temperature and pressure in any line, and it does not increase the duration that exempted lines are operated. Accordingly, there are no changes in the lines subject to design basis breaks.

Postulated break locations in these lines are based on (1) specified locations (terminal points) and (2) at high stress points. The uprate has been evaluated as having a negligible impact on pipe stress. The uprate will not require any pipe stress reanalysis. Accordingly, the postulated pipe break locations will not change.

The mass and energy blowdown from an isolatable postulated break is based on the volume, temperature and pressure in the line. Again, there are no changes to these parameters. The mass energy used for the limiting RCS break assumed a core power of 3479 MWt (1.02 times present power) which bounds the uprate.

Since there are no changes to break locations and no changes to assumed blowdown from the postulated breaks, it follows that there is no impact to the high energy line break analyses for the Salem units.

12.5 EROSION/CORROSION

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. This program will be updated to incorporate the increased process flow values for the main steam, condensate and feedwater systems and their sub-systems. Results will be factored in to the future inspection schedules.

12.6 SAFETY RELATED MOTOR OPERATED VALVES

There is no change in the limiting temperature, pressure, flow in any Emergency Safeguard System (ESF). The ESF systems bound the proposed uprate since the calculations for these systems assumed (1) a 2% calorimetric error and (2) the ESF design rating.

All feedwater and main steam MOV calculations were reviewed. They were based on the limiting condition (highest pressure differential), which occur at the no-load condition. Therefore, the nominal uprate does not impact these calculations.

12.7 IMPACT ON PROBABALISTIC SAFETY ASSESSMENT RESULTS

The proposed increase in core power is not expected to significantly change the results of the plant Probabilistic Safety Assessment (PSA).

System success criteria including containment heat removal and pressure control capacity are not affected by the proposed uprate. Potential effects on time sensitive operator actions due to the uprate were evaluated and found to have a negligible impact on risk from anticipated transients without scram (ATWS) or on the overall PSA results.

ENVIRONMENTAL IMPACT:

The proposed TS changes were reviewed against the criteria of 10CFR51.22 for environmental considerations. The proposed changes do not involve a significant hazards consideration, a significant change in the types or a significant increase in the amounts of effluents that may be released offsite, or a significant increase in the individual or cumulative occupational radiation exposures. Based on the foregoing, PSEG Nuclear concludes that the proposed TS changes meet the criteria given in 10CFR51.22(c)(9) for a categorical exclusion from the requirements for an Environmental Impact Statement.

SALEM GENERATING STATION
UNIT NOS. 1 AND 2
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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 and WCAP-15315, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Operating PWR and BWR Plants." 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.

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FACILITY OPERATING LICENSE PAGES WITH PROPOSED CHANGES

The following section of Facility Operating License No. DPR-70 is affected by this change request:

FOL Paragraph	Page
2.C.(1)	4

The following section of Facility Operating License No. DPR-75 is affected by this change request:

FOL Paragraph	Page
2.C.(1)	3

(1) Maximum Power Level

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PSEG Nuclear LLC is authorized to operate the facility at a steady state reactor core power level not in excess of ~~3411~~ megawatts (one hundred percent of rated core power). ~~Prior to attaining the one hundred percent power level, Public Service Electric and Gas Company shall complete the preoperational tests, startup tests and other items identified in Attachment 1 to this amended license in the sequence specified. Attachment 1 is an integral part of this amended license.~~

(2) Technical Specifications

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

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

(4) Less than Four Loop Operation

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

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

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

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

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

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

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

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

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

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

PSEG Nuclear LLC is authorized to operate the facility at steady state reactor core power levels not in excess of 3411 megawatts (thermal).

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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

TECHNICAL SPECIFICATION PAGES WITH PROPOSED CHANGES

The following Technical Specifications for Facility Operating License No. DPR-70 are affected by this change request:

Technical Specification	Page
1.25	1-5
Figure 2.1-1	2-2
Table 2.2-1	2-8
Figure 3.4-2	3/4 4-26
Figure 3.4-3	3/4 4-27
Table 3.7-1	3/4 7-2
6.9.1.9	6-24a
Bases 3/4.4.9	B 3/4 4-6 B 3/4 4-7 B 3/4 4-8 B 3/4 4-9 B 3/4 4-10 B 3/4 4-11
Table B 3/4.4-1	B 3/4 4-12
Figure B 3/4.4-1	B 3/4 4-15
Bases 3/4.7.1.1	B 3/4 7-1

TECHNICAL SPECIFICATION PAGES WITH PROPOSED CHANGES (cont'd)

The following Technical Specifications for Facility Operating License No. DPR-75 are affected by this change request:

Technical Specification	Page
1.25	1-5
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Figure 3.4-3	3/4 4-29
Table 3.7-1	3/4 7-2
6.9.1.9	6-24a
Bases 3/4.4.9	B 3/4 4-7 B 3/4 4-8 B 3/4 4-9 B 3/4 4-10 B 3/4 4-11 B 3/4 4-12
Table B 3/4.4-1	B 3/4 4-13
Figure B 3/4.4-1	B 3/4 4-16
Bases 3/4.7.1.1	B 3/4 7-1

PHYSICS TESTS

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

PRESSURE BOUNDARY LEAKAGE

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

PROCESS CONTROL PROGRAM (PCP)

1.22 The PROCESS CONTROL PROGRAM shall be that program which contains the current formula, sampling, analyses, test, and determinations to be made to ensure that the processing and packaging of solid radioactive wastes, based on demonstrated processing of actual or simulated wet solid wastes, will be accomplished in such a way as to assure compliance with 10 CFR Part 20, 10 CFR Part 71 and Federal and State regulations and other requirements governing the disposal of the radioactive waste.

PURGE - PURGING

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

QUADRANT POWER TILT RATIO

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

RATED THERMAL POWER

1.25 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of ~~3411 MWt.~~

3459 MWt.

Insert
A1
Here

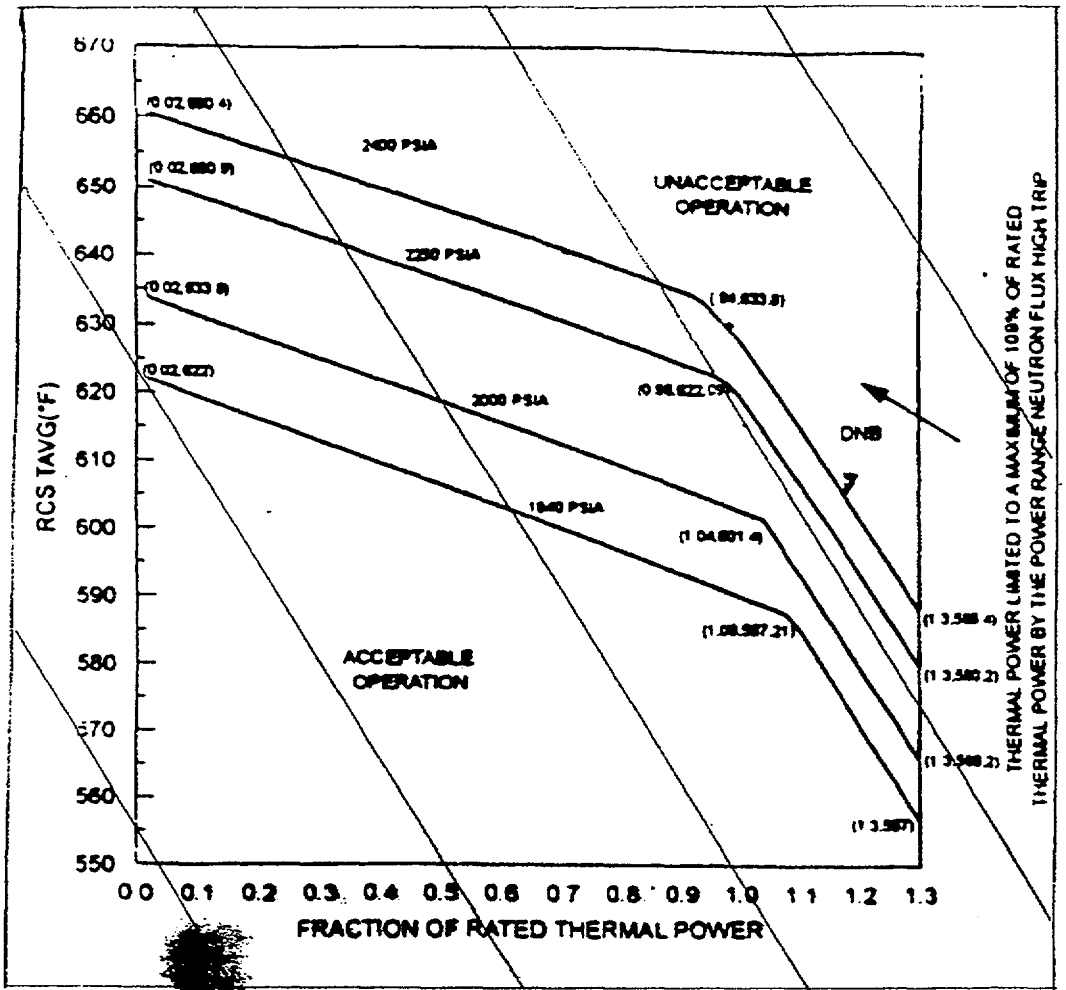
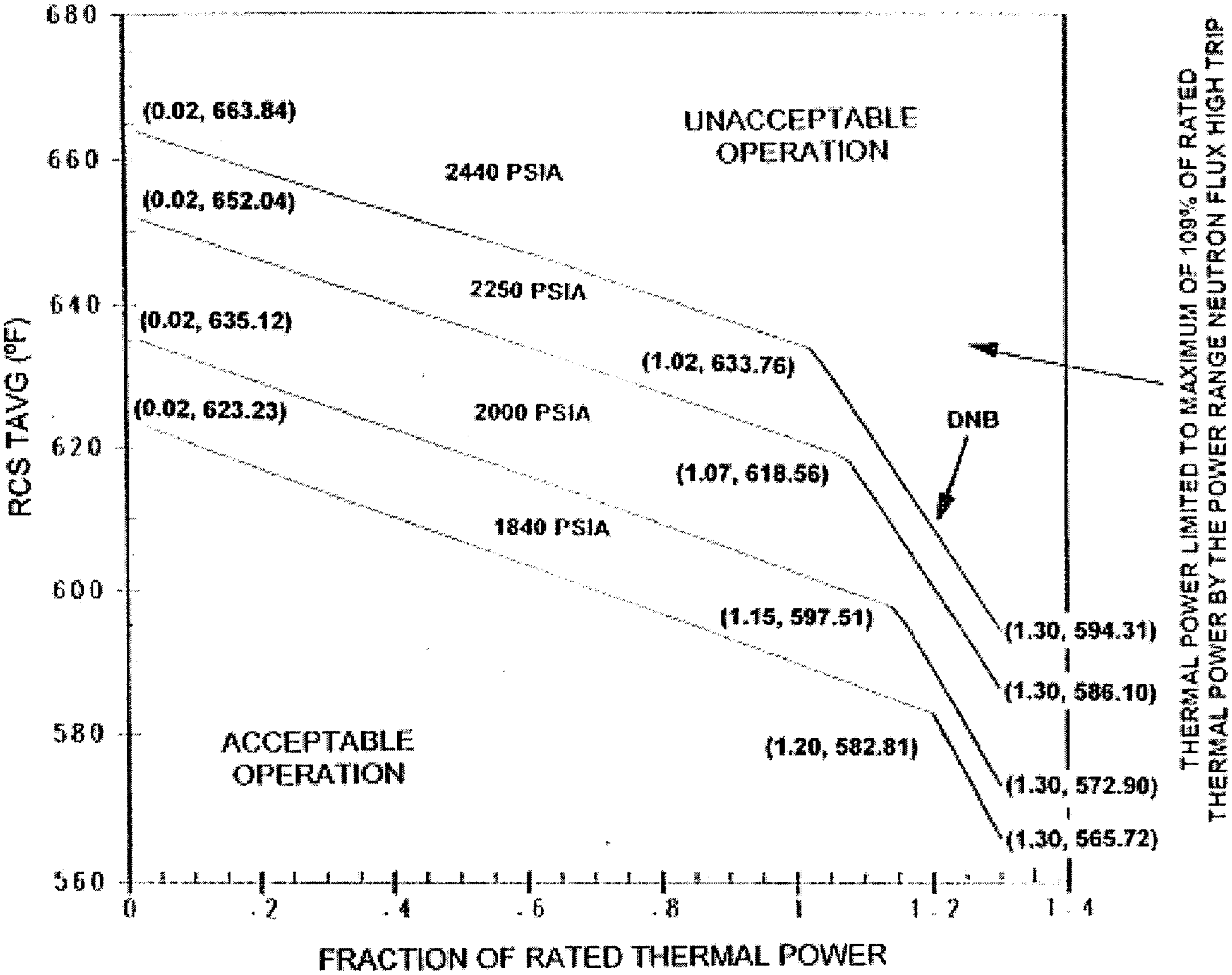


FIGURE 2.1-1
REACTOR CORE SAFETY LIMIT - FOUR LOOPS IN OPERATION

INSERT A1



680
660
640
620
600
580
560
RCS TAVG (ºF)
UNACCEPTABLE OPERATION
ACCEPTABLE OPERATION
2440 PSIA
2250 PSIA
2000 PSIA
1840 PSIA
(0.02, 663.84)
(0.02, 652.04)
(0.02, 635.12)
(0.02, 623.23)
(1.02, 633.76)
(1.07, 618.56)
(1.15, 597.51)
(1.20, 582.81)
(1.30, 594.31)
(1.30, 586.10)
(1.30, 572.90)
(1.30, 565.72)
DNB
THERMAL POWER LIMITED TO MAXIMUM OF 109% OF RATED THERMAL POWER BY THE POWER RANGE NEUTRON FLUX HIGH TRIP
0
.2
.4
.6
.8
1
1.2
1.4
FRACTION OF RATED THERMAL POWER

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

Operation with 4 Loops

K1 = 1.22
K2 = 0.02037
K3 = 0.001020

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

(i) for $q_t - q_b$ between ~~-23~~ -33 percent and ~~+13~~ +11 percent, f_1 (ΔI) = 0 (where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER).

(ii) for each percent that the magnitude of ($q_t - q_b$) exceeds ~~-23~~ -33 percent, the ΔT trip setpoint shall be automatically reduced by ~~1.26~~ 2.34 percent of its value at RATED THERMAL POWER.

(iii) for each percent that the magnitude of ($q_t - q_b$) exceeds ~~+13~~ +11 percent, the ΔT trip setpoint shall be automatically reduced by ~~2.63~~ 2.37 percent of its value at RATED THERMAL POWER.

Insert B1 Here

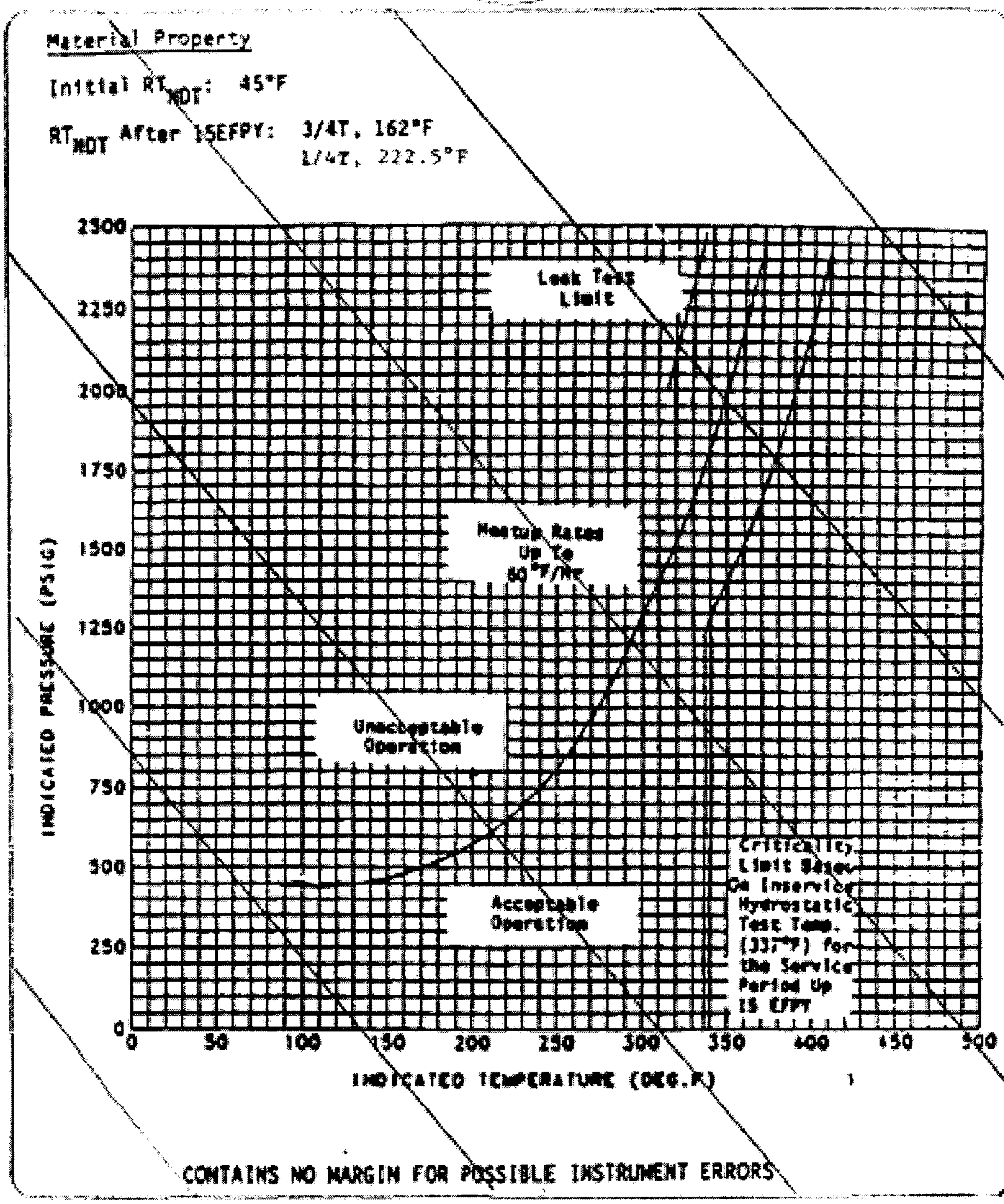


Figure 3.4-2 . Salem Unit 1 Reactor Coolant System Heatup Limitations Applicable for Heatup Rates up to $\boxed{60}$°F/HR for the Service Period up to $\boxed{15}$ EFPY

100

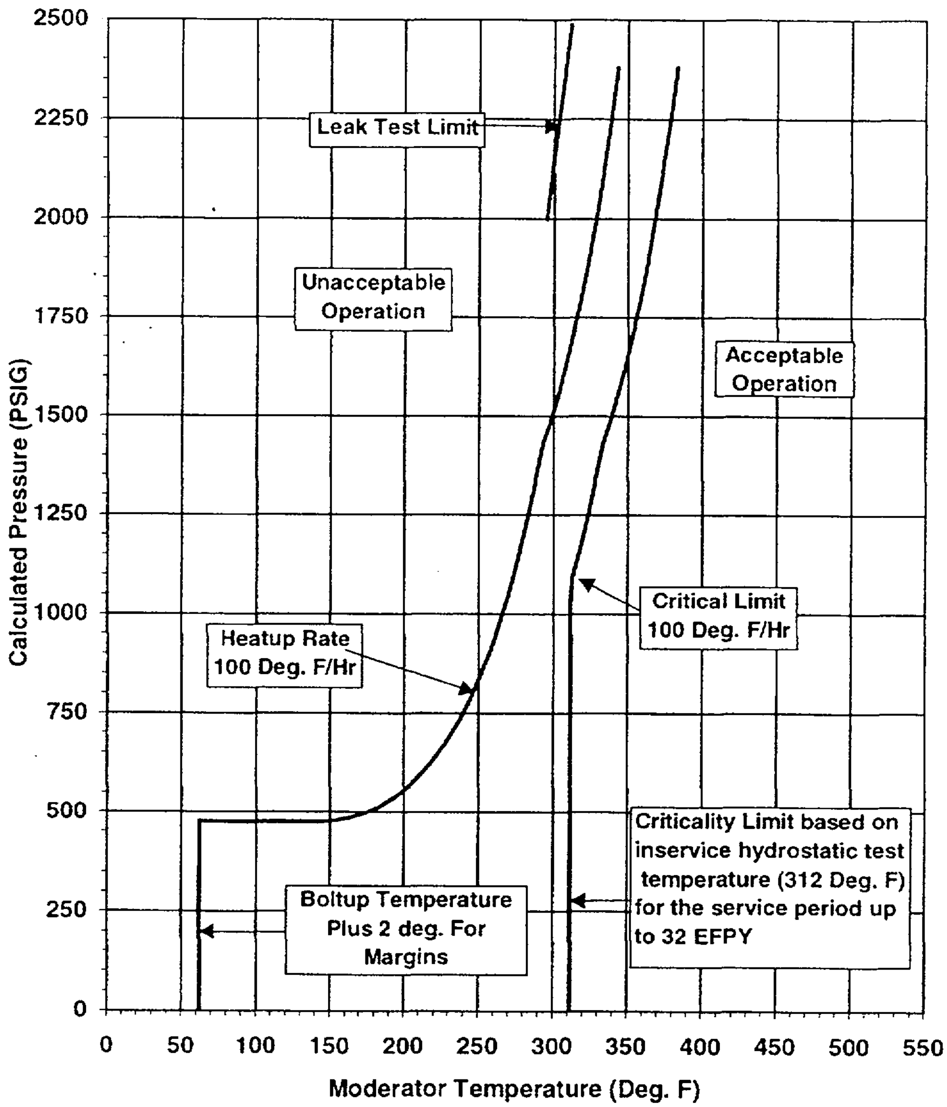
32

(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



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C1
Here

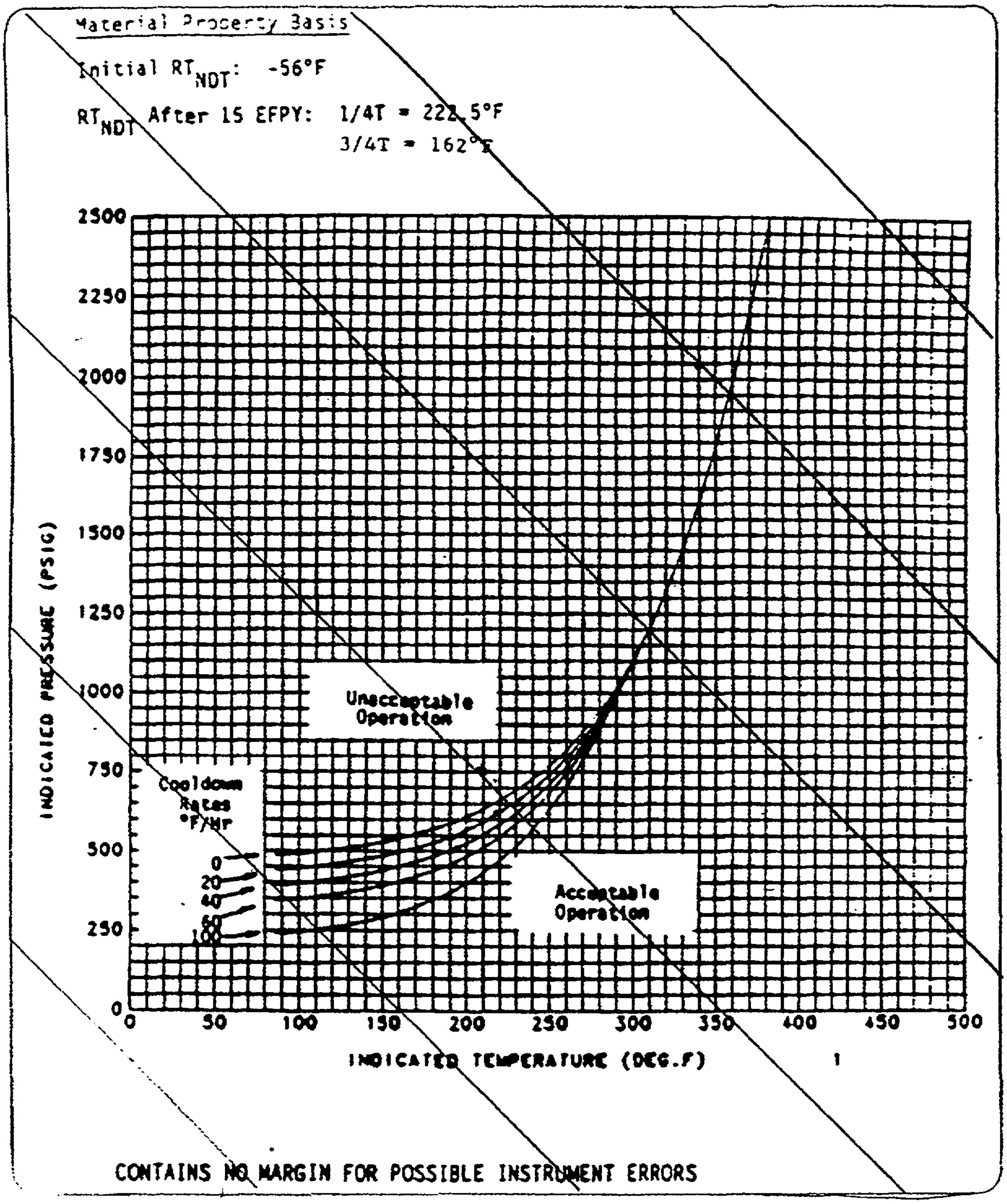


Figure 3.4-3 Salem Unit 1 Reactor Coolant System Cooldown Limitations Applicable for Cooldown Rates up to 100°F/HR for the Service Period up to $\boxed{\cancel{15}}$ EFPY (with uncertainties for instrumentation errors)

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

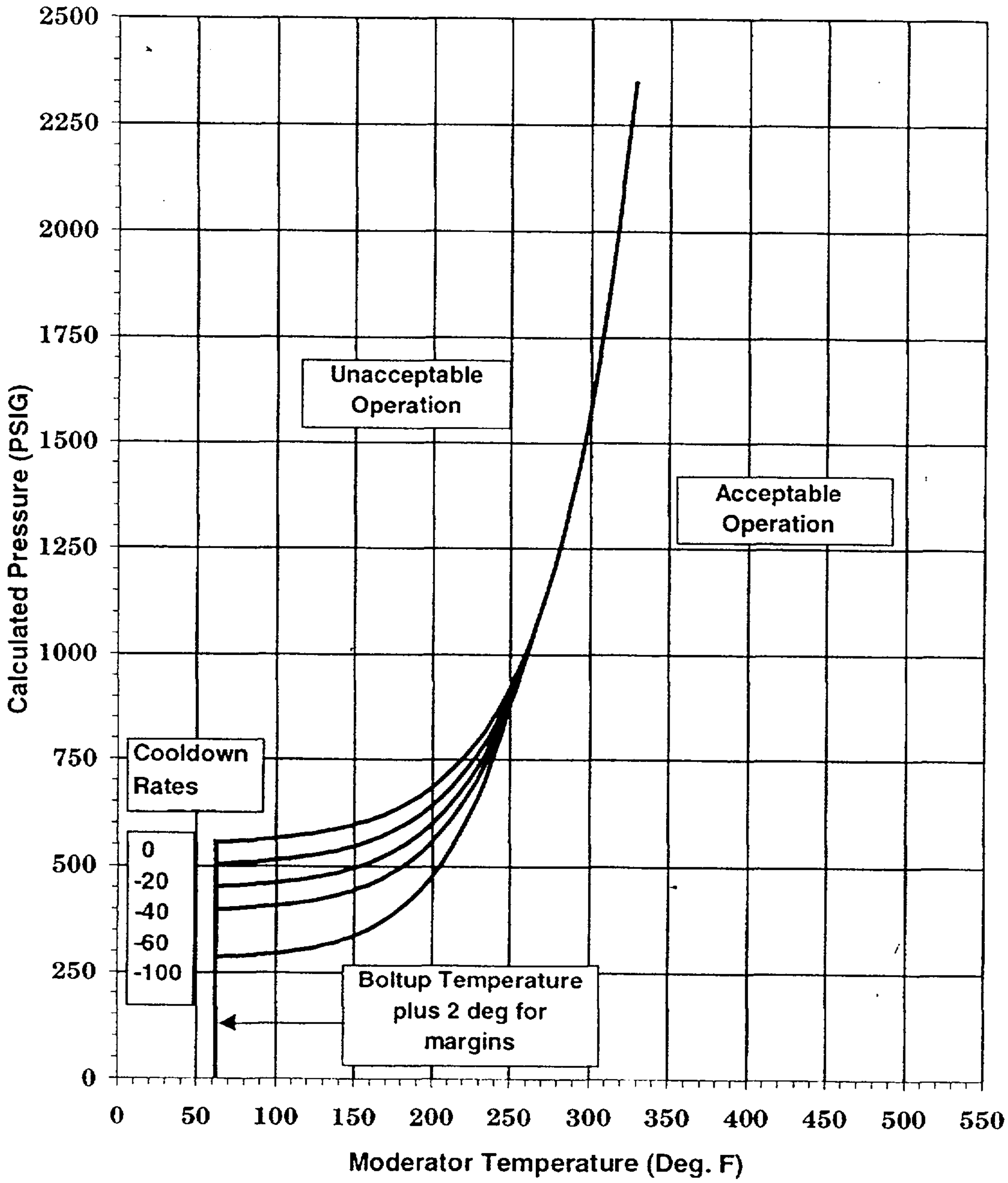


TABLE 3.7-1

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

Maximum Number of Inoperable Safety Valves on Any ~~Operating~~ Steam Generator	Maximum Allowable Power* ~~Range Neutron Flux High Setpoint~~ (Percent of RATED THERMAL POWER)
1	87
2	~~64~~ 59
3	~~42~~ 39

These proposed changes are described in PSEG NUCLEAR letter dated 9/26/2000 (LCR S99-13).

* The values do not provide any allowance for calorimetric error.

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

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

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

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

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

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

SPECIAL REPORTS

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

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

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

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

The temperature and pressure changes during heatup and cooldown are limited to be consistent with the requirements given in the ASME Boiler and Pressure Vessel Code, Section ~~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.I 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, WCAP-15315, “Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Operating PWR and BWR Plants”, October 1999, and ASME Boiler and Pressure Vessel Code Case N-640, “Alternative Reference Fracture Toughness for Development of P-T Limit Curves for Section XI, Division 1”, approved March 1999.

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

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

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

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

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

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

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

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

where "f surface" is from Figure B3/4.4-1, and X (in inches) is the depth into the vessel wall.

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

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

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

The heatup and cooldown limit curves of Figures 3.4-2 and 3.4-3 include predicted adjustments for this shift in RT_{NDT} at the end of ~~15~~ 32 EFPY.

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

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

Insert DI Here

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

The ASME approach for calculating the allowable limit curves for various heatup and cooldown rates specifies that the total stress intensity factor, K_I, for the combined thermal and pressure stresses at any time during heatup or cooldown cannot be greater than the reference stress intensity factor, ~~K_{IR}~~ K_{IC}, for the metal temperature at that time. ~~K_{IR}~~ K_{IC} is obtained from the reference fracture toughness curve, defined in ~~Appendix G to the~~ ASME Code Case N-640. The ~~K_{IR}~~ K_{IC} curve is given by the equation:

$$\cancel{K_{IR}}\ K_{IC} = \cancel{26.78}\ 20.734 + \cancel{1.223}\ 33.2 \exp\left[\cancel{0.0145}\ 0.02(T - RT_{NDT}\ \cancel{+160})\right] \qquad (1)$$

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

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

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

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

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

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

C = 1.5 for inservice hydrostatic and leak test operations. K_{IC}

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

COOLDOWN

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

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

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

HEATUP

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

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

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

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

~~Finally, the new 10CFR50 rule which addresses the metal temperature of the closure head flange is considered. This 10CFR50 rule states that the metal temperature of the closure flange regions must exceed the material RT_{NDT} by at least 120°F for normal operation when the pressure exceeds 20 percent of the preservice hydrostatic test pressure (621 psig for Salem). Table B3/4 4-1 indicates that the limiting RT_{NDT} of 28°F occurs in the closure head flange of Salem Unit 1, and the minimum allowable temperature of this region is 148°F at pressure 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 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.

TABLE B 3/4.4-1

SALEM UNIT 1 REACTOR VESSEL TOUGHNESS DATA

Component	Plate No. or Weld No.	Material Type	Cu (%)	Ni (%)	T (°F)	50 ft lb 35-Mil Temp (°F)	RT (°F)	Average Upper Shelf Energy: Normal to Principal Working Direction (ft-lb)	Average Upper Shelf Energy: Principal Working Direction (ft-lb)
Cl Hd Dome	B2407-1	A533B, Cl.1	0.20	0.50	-30	99*	39	71.5*	110
Cl Hd Segment	B2406-1	A533B, Cl.1	0.13	0.52	-20	89*	29	97*	125
Cl Hd Segment	B2406-2	A533B, Cl.1	0.16	0.50	-30	85*	25	79*	122
Cl Hd Segment	B2406-3	A533B, Cl.1	0.10	0.53	-50	66*	6	86*	132
Cl Hd Flange	B2811	A508, Cl.2	-	0.72	28*	22*	28	129*	199
Vessel Flange	B2410	A508, Cl.2	-	0.67	60*	0*	60	94*	145
Inlet Nozzle	B2408-1	A508, Cl.2	-	0.68	50*	43*	50	94*	144
Inlet Nozzle	B2408-2	A508, Cl.2	-	0.71	46*	26*	46	102*	157
Inlet Nozzle	B2408-3	A508, Cl.2	-	0.66	47*	37*	47	105*	161
Inlet Nozzle	B2408-4	A508, Cl.2	-	0.65	9*	17*	9	108.5*	167
Outlet Nozzle	B2409-1	A508, Cl.2	-	0.69	60*	95*	60	48*	75
Outlet Nozzle	B2409-2	A508, Cl.2	-	0.69	60*	95*	60	51*	78
Outlet Nozzle	B2409-3	A508, Cl.2	-	0.74	60*	10*	60	79*	121
Outlet Nozzle	B2409-4	A508, Cl.2	-	0.74	60*	11*	60	82*	126
Upper Shell	B2401-1	A533B, Cl.1	0.22	0.48	-30	87*	27	74*	114
Upper Shell	B2401-2	A533B, Cl.1	0.19	0.48	0	80*	20	79*	122
Upper Shell	B2401-3	A533B, Cl.1	0.24	0.51	-10	114*	34	62*	96
Inter Shell	B2402-1	A533B, Cl.1	0.24	0.~~52~~ .53	-30	105	45	~~73~~ 91	97
Inter Shell	B2402-2	A533B, Cl.1	0.24	0.~~50~~ .53	-30	55	-5	~~91~~ 98	112
Inter Shell	B2402-3	A533B, Cl.1	0.~~12~~ .22	0.~~50~~ .51	-40	57	-3	104	127
Lower Shell	B2403-1	A533B, Cl.1	0.19	0.48	-40	70	~~10~~ 4	~~99~~ 93	143
Lower Shell	B2403-2	A533B, Cl.1	0.19	0.49	-70	86	~~26~~ 18	~~94~~ 83	128
Lower Shell	B2403-3	A533B, Cl.1	0.19	0.48	-40	90	~~30~~ 6	~~102~~ 85	131
Bot Hd Segment	B2404-1	A533B, Cl.1	0.10	0.52	10	48*	10	78*	120
Bot Hd Segment	B2404-2	A533B, Cl.1	0.11	0.53	-50	60*	0	86*	132
Bot Hd Segment	B2404-3	A533B, Cl.1	0.12	0.52	10	47*	10	82*	126
Bot Hd Dome	B2405-1	A533B, Cl.1	0.15	0.50	-20	57*	-3	69*	106
Cirum. Weld Bet Nozzle Shell & Int. Shell	8-042	-	0.22	1.02 .73	-	-	-56***	-	-
Cirum. Weld Bet. Int. and Lower Shell	9-042	-	0.~~25~~ .22	0.~~72~~ 1.04	-	-	-56***	~~-~~ 112	-
Int. Shell Vertical Weld	2-042 (A,B,C)	-	0.18	~~1.00**~~ 1.04	-	-	-56***	~~-~~ 96.2	-
Lower Shell Vertical Weld	3-042 (A,B,C)	-	0.19	~~1.00**~~ 1.04	-	-	-56***	~~-~~ 112	-

* Estimated per NRC Standard Review Plan Section 5.3.2.

~~** Estimated per Regulatory Guide 1.99, Rev. 2~~

*** Estimated per Pressurized Thermal Shock Rule, 10 [illegible].61

Insert E1 Here

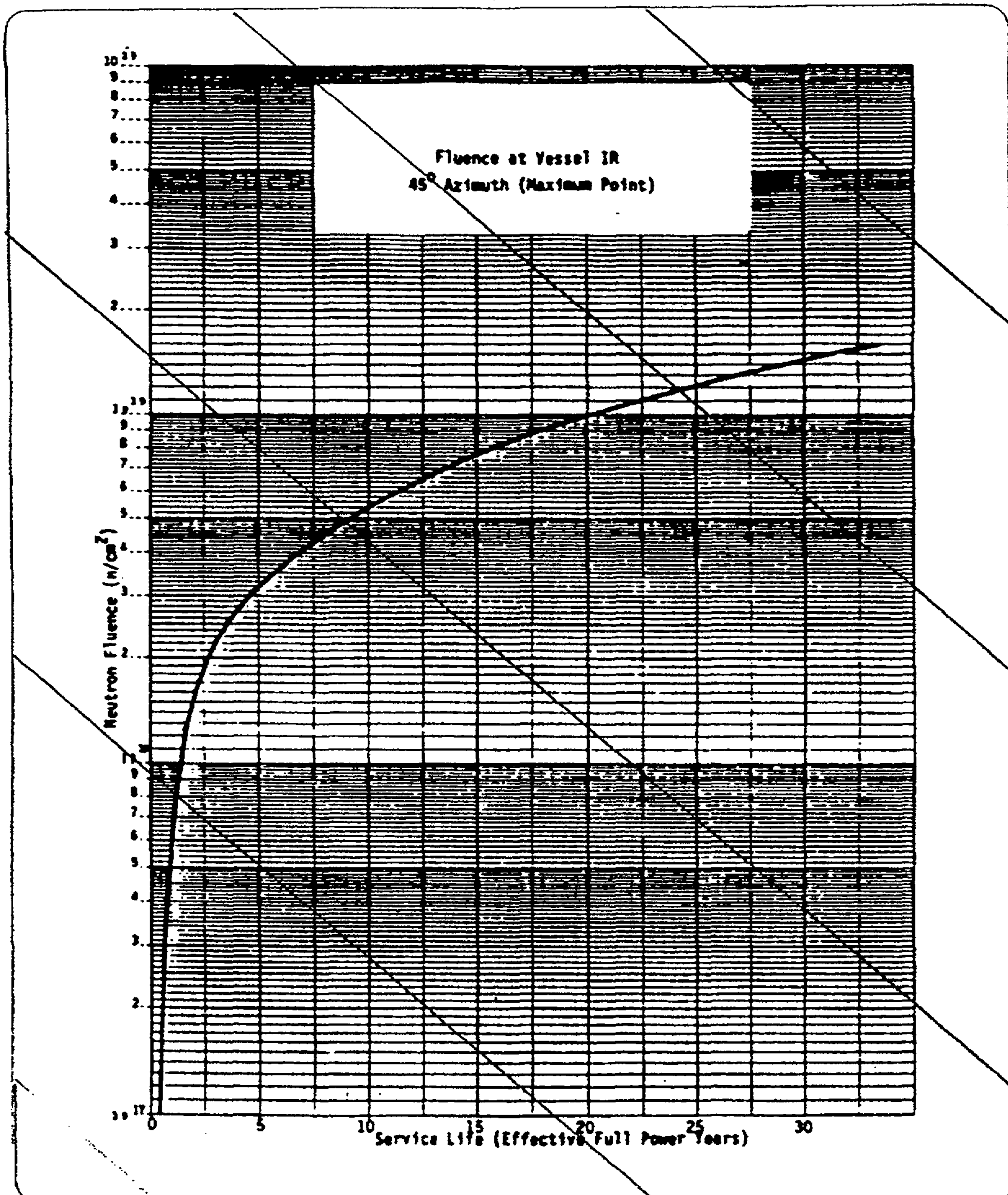
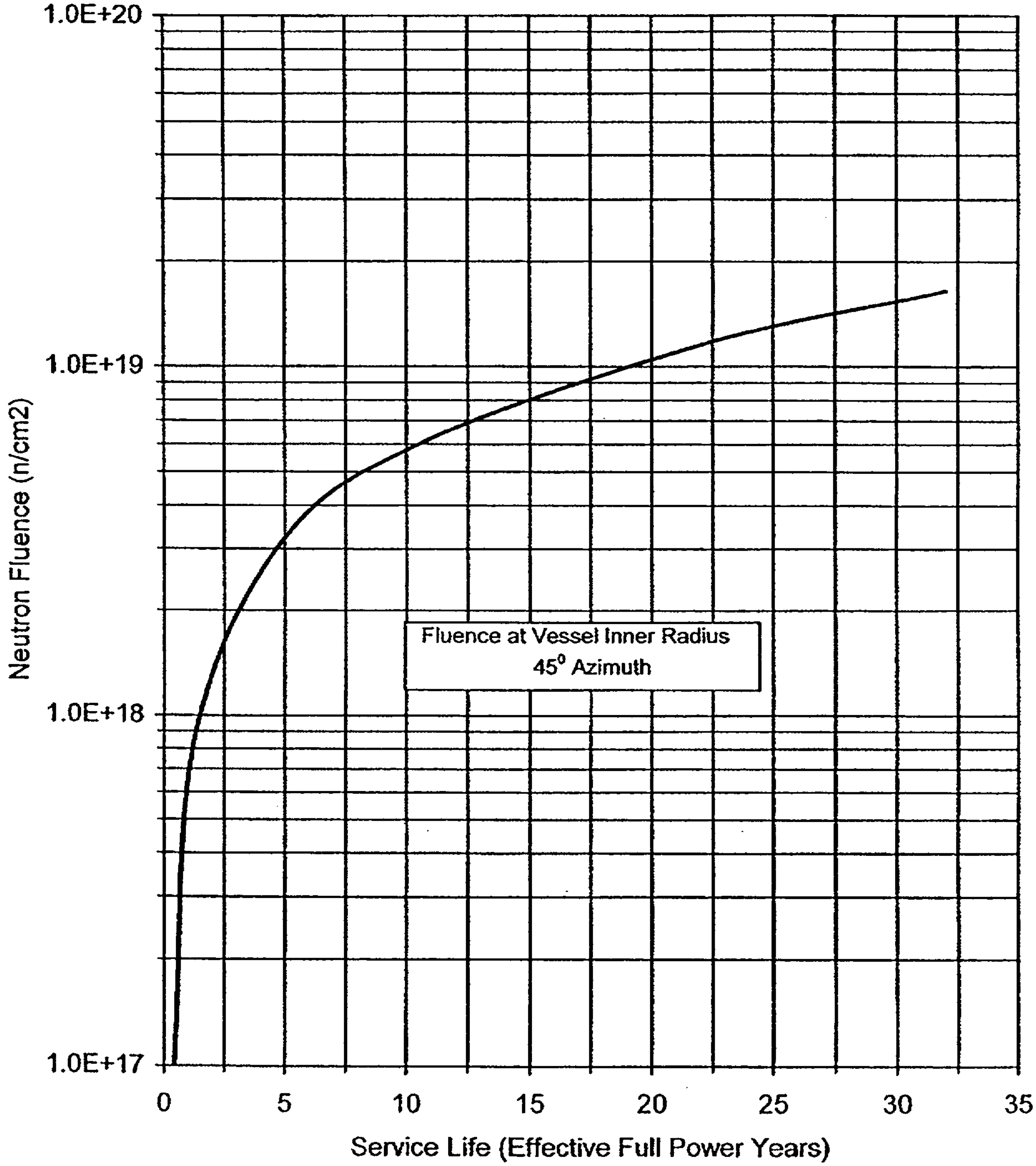


Figure B 3/4.4-1 Fast neutron fluence (E > 1MeV) as a function of of full power service life (EPPY)

INSERT E1



1.0E+20
1.0E+19
1.0E+18
1.0E+17
Neutron Fluence (n/cm2)
0
5
10
15
20
25
30
35
Service Life (Effective Full Power Years)
Fluence at Vessel Inner Radius
45^0 Azimuth

PROPOSED CHANGE DESCRIBED IN PSEG NUCLEAR letter dated 10/06/2000 (LCR S99-13)

3/4.7 PLANT SYSTEMS

BASES

3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES

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

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Code, 1971 Edition. The total relieving capacity for all valves on all of the steam lines is ~~16,655,268~~ 16.66x10^6 lbs/hr which is ~~115~~ 110.3 percent of the ~~total secondary~~ maximum calculated steam flow of ~~14,458,360~~ 15.10x10^6 lbs/hr at 100% RATED THERMAL POWER. A minimum of 2 OPERABLE safety valves per ~~OPERABLE~~ steam generator ensures that sufficient relieving capacity is available for the allowable THERMAL POWER restriction in Table 3.7-~~2~~ 1.

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

For 4 loop operation

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

For 3 loop operation

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

Where:

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

V = maximum number of inoperable safety valves per steam line

PROPOSED CHANGE DESCRIBED IN PSEG NUCLEAR letter DATED 9/26/2000 (LCR S99-13.)

PHYSICS TESTS

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

PRESSURE BOUNDARY LEAKAGE

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

PROCESS CONTROL PROGRAM (PCP)

1.22 The PROCESS CONTROL PROGRAM shall be that program which contains the current formula, sampling, analyses, test, and determinations to be made to ensure that the processing and packaging of solid radioactive wastes, based on demonstrated processing of actual or simulated wet solid wastes, will be accomplished in such a way as to assure compliance with 10 CFR Part 20, 10 CFR Part 71 and Federal and State regulations and other requirements governing the disposal of the radioactive waste.

PURGE - PURGING

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

QUADRANT POWER TILT RATIO

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

RATED THERMAL POWER

1.25 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of ~~3411 MWt.~~

3459 MWt.

Insert
A2
Here

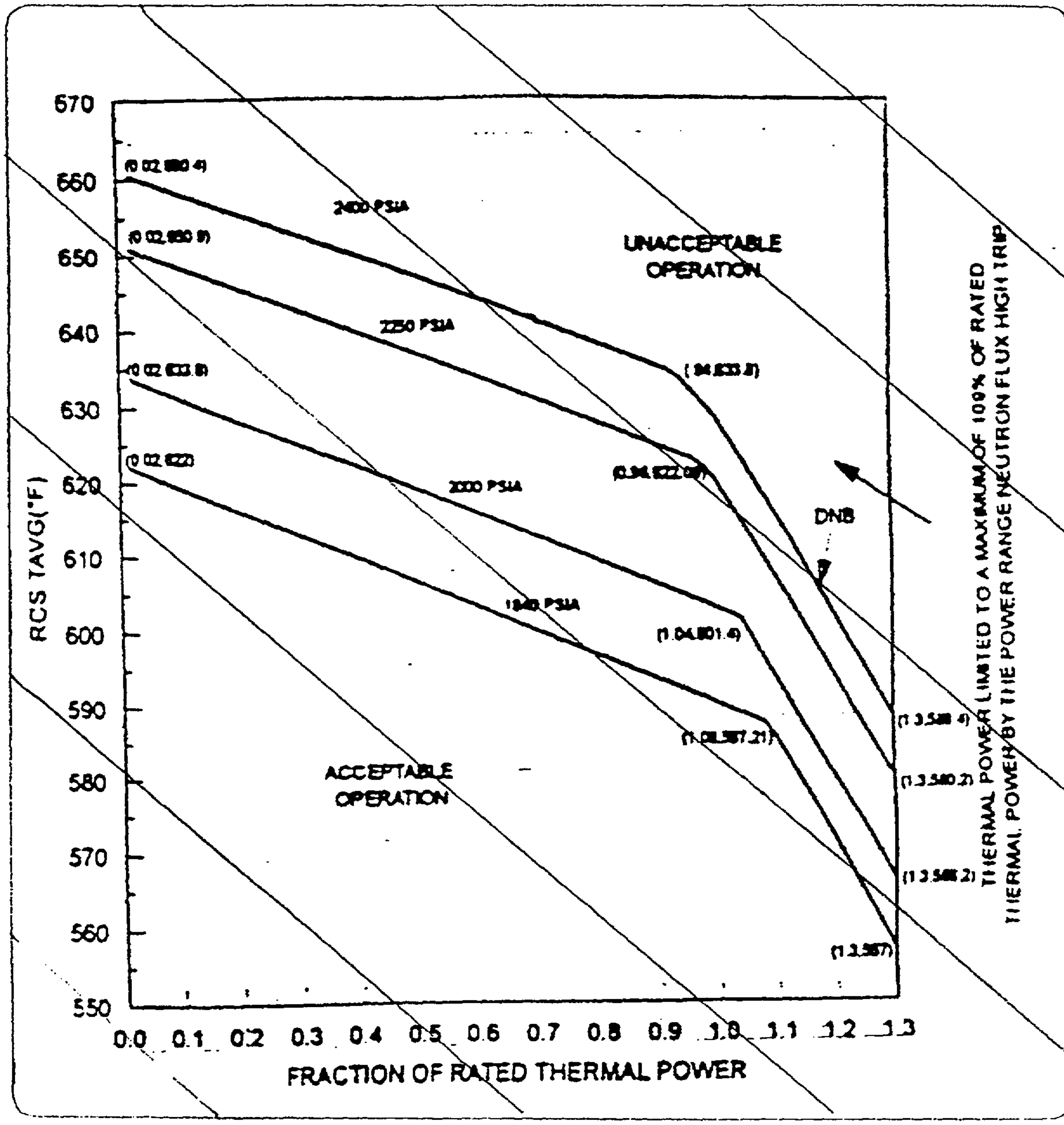


FIGURE 2.1-1 REACTOR CORE SAFETY LIMIT - FOUR LOOPS IN OPERATION

INSERT A2

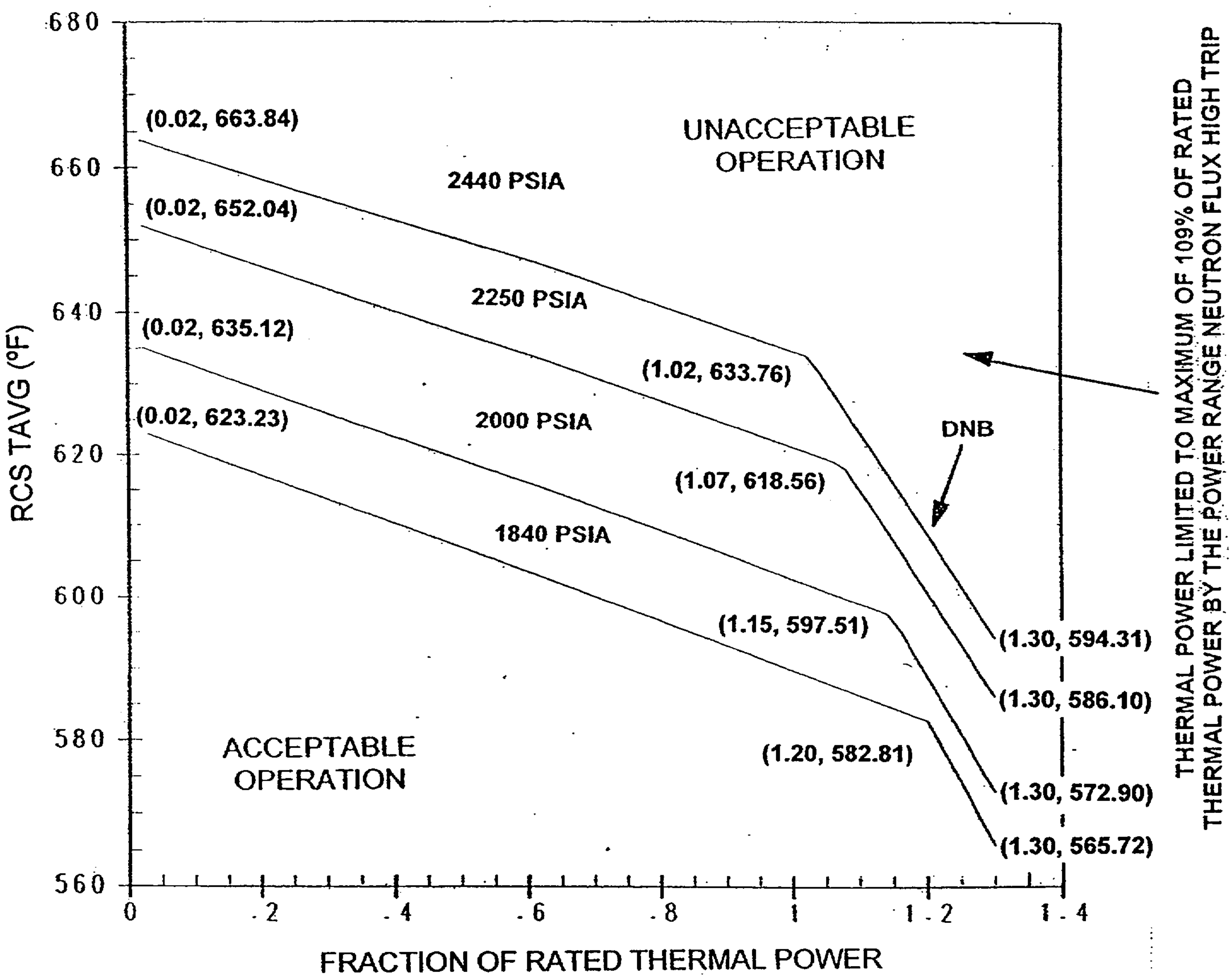

RCS TAVG (ºF)
FRACTION OF RATED THERMAL POWER
UNACCEPTABLE OPERATION
ACCEPTABLE OPERATION
2440 PSIA
2250 PSIA
2000 PSIA
1840 PSIA
DNB
(0.02, 663.84)
(0.02, 652.04)
(0.02, 635.12)
(0.02, 623.23)
(1.02, 633.76)
(1.07, 618.56)
(1.15, 597.51)
(1.20, 582.81)
(1.30, 594.31)
(1.30, 586.10)
(1.30, 572.90)
(1.30, 565.72)
THERMAL POWER LIMITED TO MAXIMUM OF 109% OF RATED THERMAL POWER BY THE POWER RANGE NEUTRON FLUX HIGH TRIP
680
660
640
620
600
580
560
0
.2
.4
.6
.8
1
1.2
1.4

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

Operation with 4 Loops

$K_1 = 1.22$
$K_2 = 0.02037$
$K_3 = 0.001020$

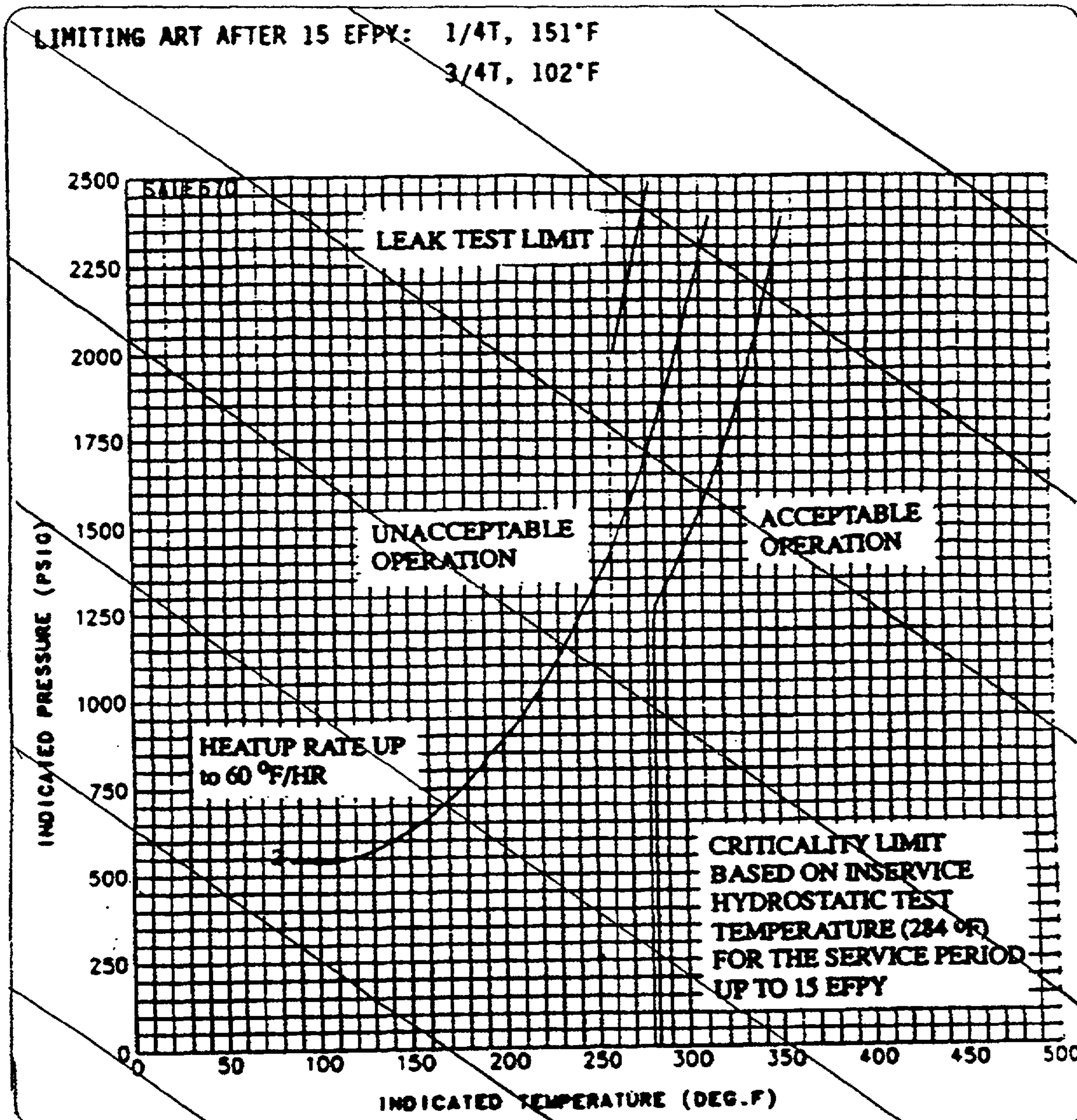
and f_1 (ΔI) is a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

(i) for $q_t - q_b$ between ~~-23~~ -33 percent and ~~+13~~ +11 percent, f_1 (ΔI) = 0 (where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER).

(ii) for each percent that the magnitude of ($q_t - q_b$) exceeds ~~-23~~ -33 percent, the ΔT trip setpoint shall be automatically reduced by ~~1.26~~ 2.34 percent of its value at RATED THERMAL POWER.

(iii) for each percent that the magnitude of ($q_t - q_b$) exceeds ~~+13~~ +11 percent, the ΔT trip setpoint shall be automatically reduced by ~~2.63~~ 2.37 percent of its value at RATED THERMAL POWER.

Insert B2 Here



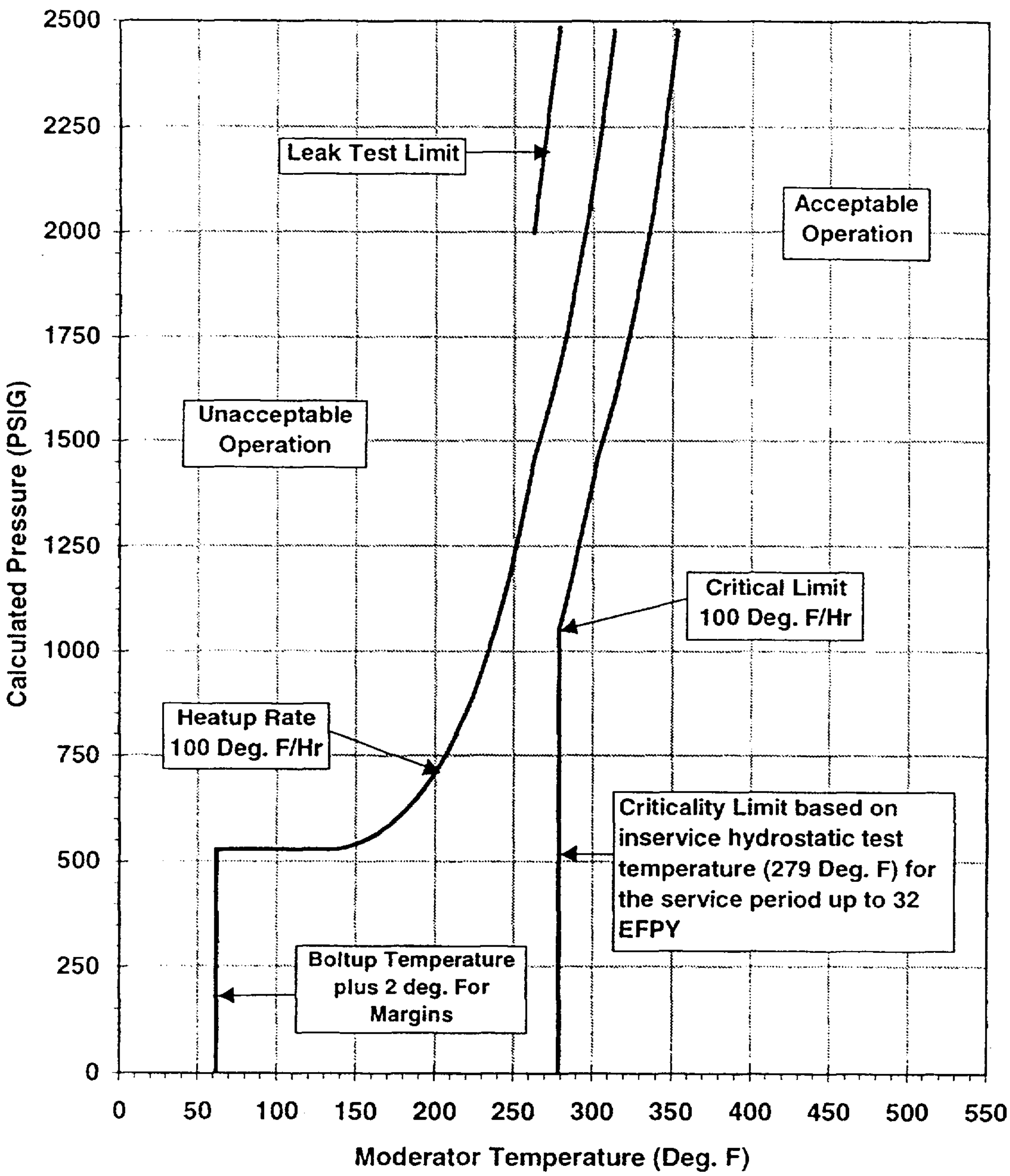
SALEM UNIT 2 REACTOR COOLANT SYSTEM HEATUP LIMITATIONS APPLICABLE FOR THE FIRST $\boxed{15}$ EFPY (32) WITH MAXIMUM HEATUP RATE OF $\boxed{60}$ F/HR. CURVE CONTAINS $\boxed{NO}$ MARGIN FOR INSTRUMENT ERRORS. (100)

Figure 3.4-2

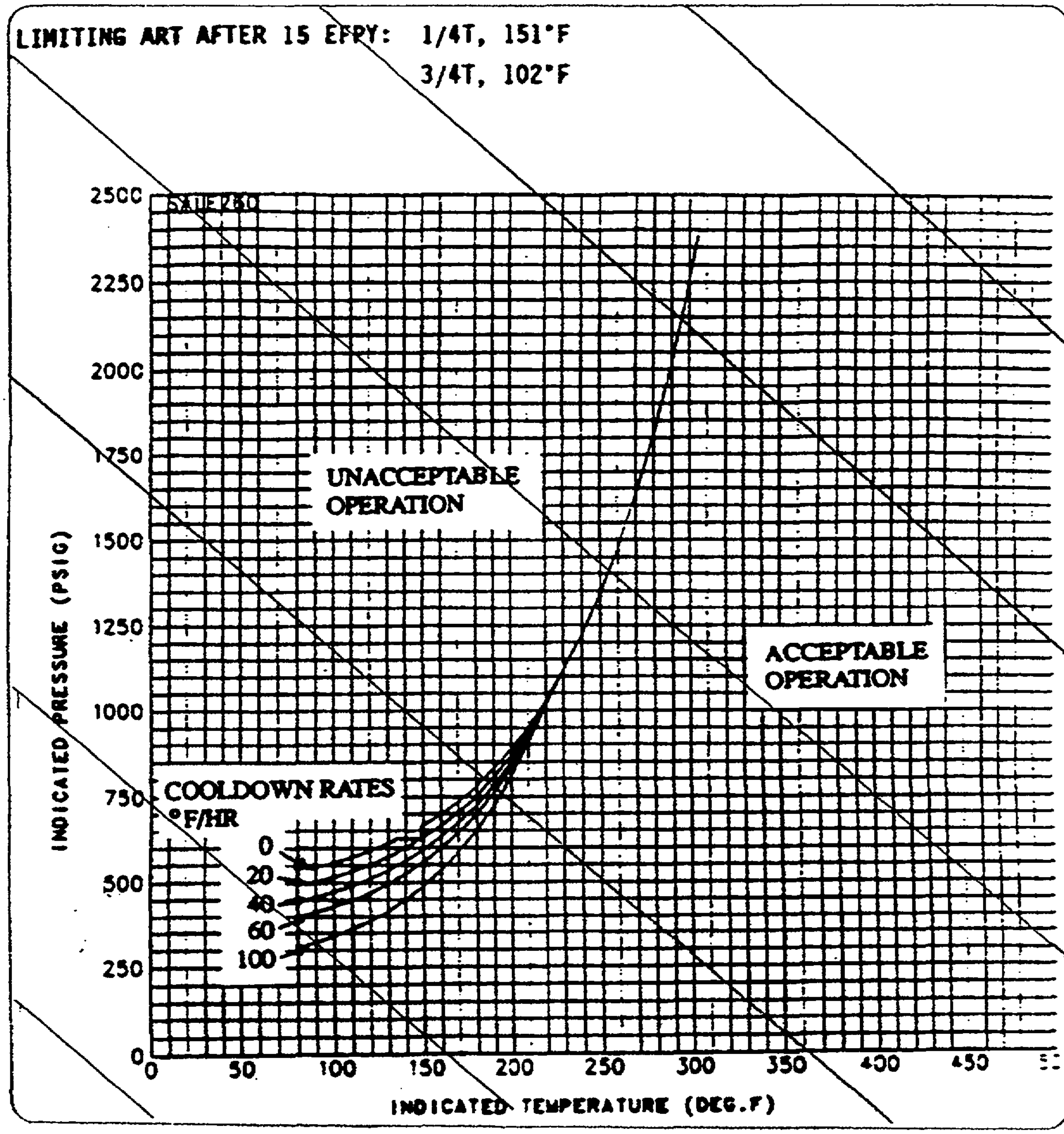
INSERT B2

Limiting Material Property

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



Insert C2 Here



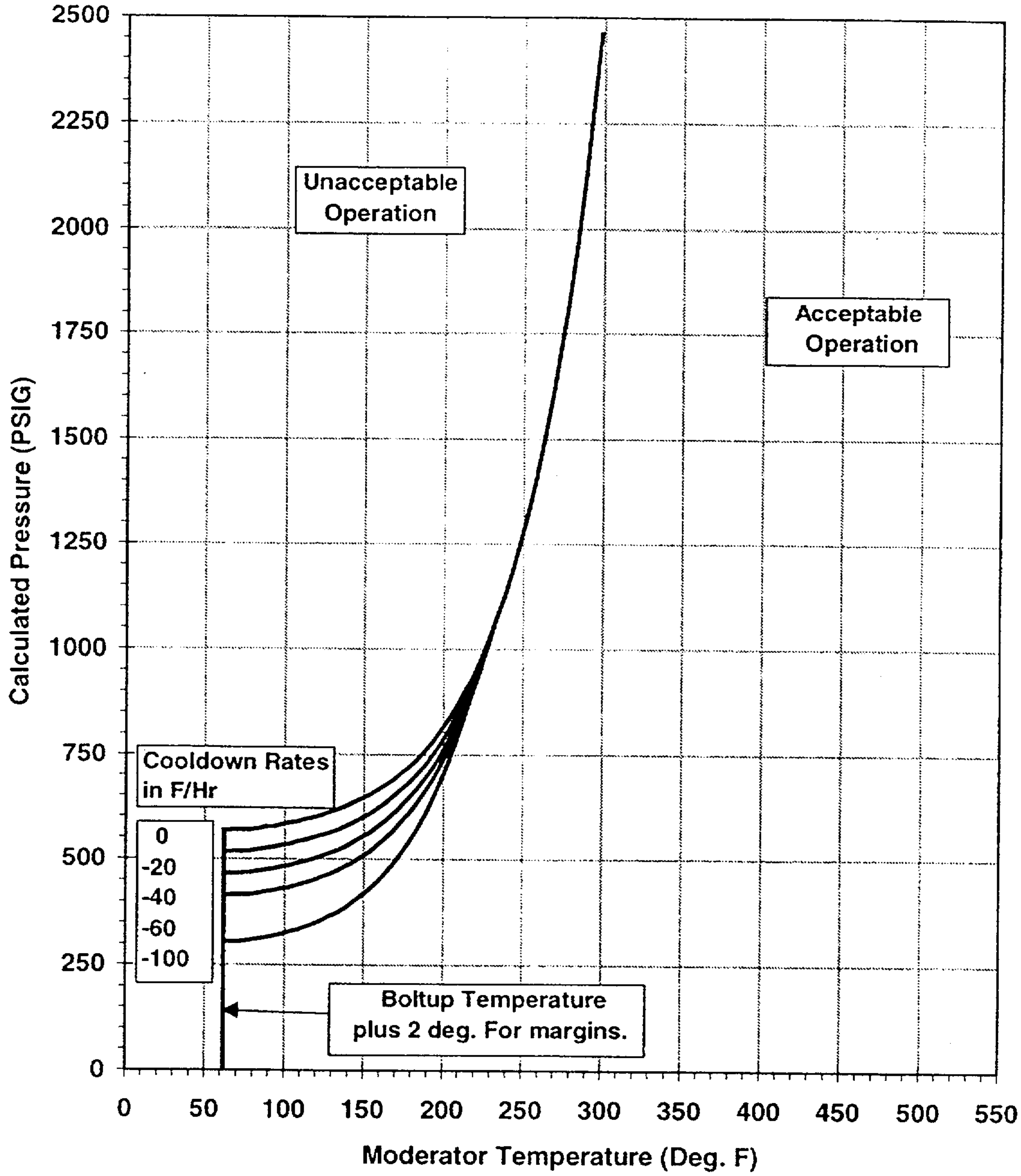
SALEM UNIT 2 REACTOR COOLANT SYSTEM COOLDOWN (32)
LIMITATIONS APPLICABLE FOR THE FIRST $\boxed{15}$ EFPY
WITH MAXIMUM COOLDOWN RATE OF 100 F/HR. CURVE
CONTAINS $\boxed{NO}$ MARGIN FOR INSTRUMENT ERRORS.

Figure 3.4-3

INSERT C2

Limiting Material Property

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



Thermal

TABLE 3.7-1

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

Maximum Number of Inoperable Safety Valves on Any ~~Operating~~ Steam Generator	* Maximum Allowable Power ~~Range Neutron Flux High Setpoint~~ (Percent of RATED THERMAL POWER)
1	87
2	~~64~~ 59
3	~~42~~ 39

These proposed changes are described in PSEG NUCLEAR letter dated 9/26/2000 (LCR S99-13)

* The values do not provide any allowance for calorimetric error

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

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

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

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

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

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

SPECIAL REPORTS

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

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

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

3/4.4.10 PRESSURE/TEMPERATURE LIMITS

The temperature and pressure changes during heatup and cooldown are limited to be consistent with the requirements given in the ASME Boiler and Pressure Vessel Code, Section ~~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 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~~ 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, WCAP-15315, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Operating PWR and BWR Plants", October 1999, and ASME Boiler and Pressure Vessel Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves for Section XI, Division 1", approved March 1999.

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

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

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

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

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

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

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

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

4

where "f surface" is from Figure B3/4.$\boxed{4}$-1, and X (in inches) is the depth into the vessel wall.

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

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

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

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

32

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

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

Insert D2 Here

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

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

$$\cancel{K_{IR}}\ K_{IC} = \cancel{26.78}\ 20.734 + \cancel{1.223}\ 33.2 \exp\left[\cancel{0.0145}\ 0.02\,(T - RT_{NDT}\ \cancel{+160})\right] \quad (1)$$

Case N-640

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

$$C\,K_{IM} + K_{IT} \le \cancel{K_{IR}}\ K_{IC} \quad (2)$$

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

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

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

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

C = 1.5 for inservice hydrostatic and leak test operations. K_{IC}

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

COOLDOWN

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

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

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

HEATUP

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

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

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

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

~~Finally, the new 10CFR50 rule which addresses the metal temperature of the closure head flange regions is considered. This 10CFR50 rule states that the metal temperature of the closure flange regions must exceed the material 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.

TABLE B 3/4.4-1

SALEM UNIT 2 REACTOR VESSEL TOUGHNESS DATA

Component	Plate No. or Weld No.	Material Type	Cu (%)	Ni (%)	T (°F)	50 ft lb 35-Mil Temp (°F)	RT (°F)	Average Upper Shelf Energy: Normal to Principal Working Direction (ft-lb)	Average Upper Shelf Energy: Principal Working Direction (ft-lb)
Closure Hd Dome	B4708	A533BCL.1	0.11	0.70	-40	45*	-15*	82.5	127
Closure Hd Peel	B5007-3	A533BCL.1	0.12	0.57	-20	15*	-20*	97*	149
Closure Hd Peel	B4707-1	A533BCL.1	0.10	0.55	0	51*	0*	84*	129
Closure Hd Peel	B4707-3	A533BCL.1	0.13	0.63	0	66*	6*	84*	129.5
Closure Hd Flng	B4702-1	A508CL.2	-	0.68	28*	39*	28*	104*	160
Vessel Flange	B5001	A508CL.2	-	0.70	12*	4*	12*	107*	164
Inlet Nozzle	B4703-1	A508CL.2	-	0.69	60*	62*	60*	>72*	>111**
Inlet Nozzle	B4703-2	A508CL.2	-	0.69	60*	25*	60*	>61*	> 94**
Inlet Nozzle	B4703-3	A508CL.2	-	0.68	60*	32*	60*	>71*	>109**
Inlet Nozzle	B4703-4	A508CL.2	-	0.81	60*	40*	60*	80*	121.5
Outlet Nozzle	B4704-1	A508CL.2	-	0.84	60*	8*	60*	82*	126
Outlet Nozzle	B4704-2	A508CL.2	-	0.77	60*	20*	60*	75*	116
Outlet Nozzle	B4704-3	A508CL.2	-	0.69	28*	8*	28*	82*	126
Outlet Nozzle	B4704-4	A508CL.2	-	0.71	60*	40*	60*	77*	119
Upper Shell	B4711-1	A533BCL.1	0.11	0.55	0*	50*	0*	87*	134
Upper Shell	B4711-2	A533BCL.1	0.14	0.56	-10	60*	0*	79*	122
Upper Shell	B4711-3	A533BCL.1	0.12	0.58	-10	88*	28*	69*	107
Inter. Shell	B4712-1	A533BCL.1	0.13	0.56	0	<60	0	~~105~~ 106	138
Inter. Shell	B4712-2	A533BCL.1	~~0.14~~ .12	~~0.60~~ .62	-20	72	12	97	127.5
Inter. Shell	B4712-3	A533BCL.1	0.11	0.57	-50	70	10	107	116
Lower Shell	B4713-1	A533BCL.1	0.12	0.60	-10	68	8	98	127
Lower Shell	B4713-2	A533BCL.1	0.12	0.57	-20	68	8	103	135.5
Lower Shell	B4713-3	A533BCL.1	0.12	0.58	-10	70	10	~~122~~ 121	135.5
Bottom Hd Peel	B4709-1	A533BCL.1	0.12	0.60	-30	54*	-6*	90*	139
Bottom Hd Peel	B4709-2	A533BCL.1	0.12	0.58	-20	42*	-18*	89*	137.5
Bottom Hd Peel	B4709-3	A533BCL.1	0.11	0.56	-20	71*	11*	93*	143
Bottom Head	B4710	A533BCL.1	0.12	0.60	-30	60*	0*	77*	118
Circum. Weld Bet Nozzle Shell & Int. Shell	8-442	-	0.28	0.74	-	-	-56***	- 99.7	-
Circum. Weld Bet Int. Shell & Lower Shell	9-442	-	~~0.17~~ 0.197	~~0.20****~~ 0.060	-	-	-56***	~~[illegible]~~ 96.2	-
Int. Shell Vertical Weld	2-442 [A,B,C]	-	~~0.22~~ 0.219	~~0.71~~ 0.735	-	-	-56***	~~[illegible]~~	-
Lower Shell Vertical Weld	3-442 [A,B,C]	-	~~0.20~~ 0.213	~~0.86~~ 0.867	-	-	-56***	~~[illegible]~~ 114	-

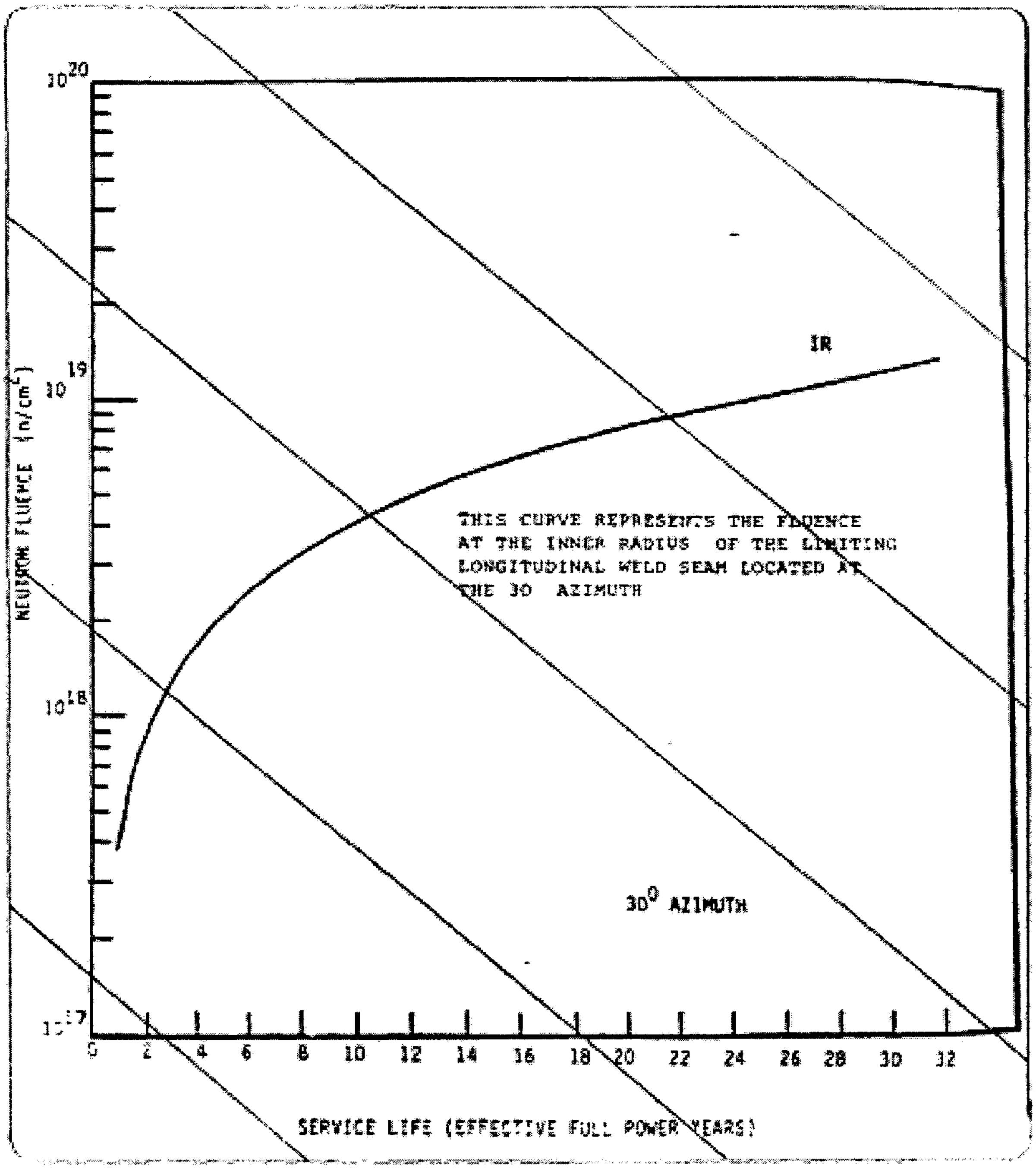
* Estimated per NRC Standard Review Plan Section 5.3.2.

** 100% Shear not reached

*** Estimated per Pressurized Thermal Shock Rule, 10CFR50.61

~~**** Estimated value for type MIL B-4 wire heats.~~

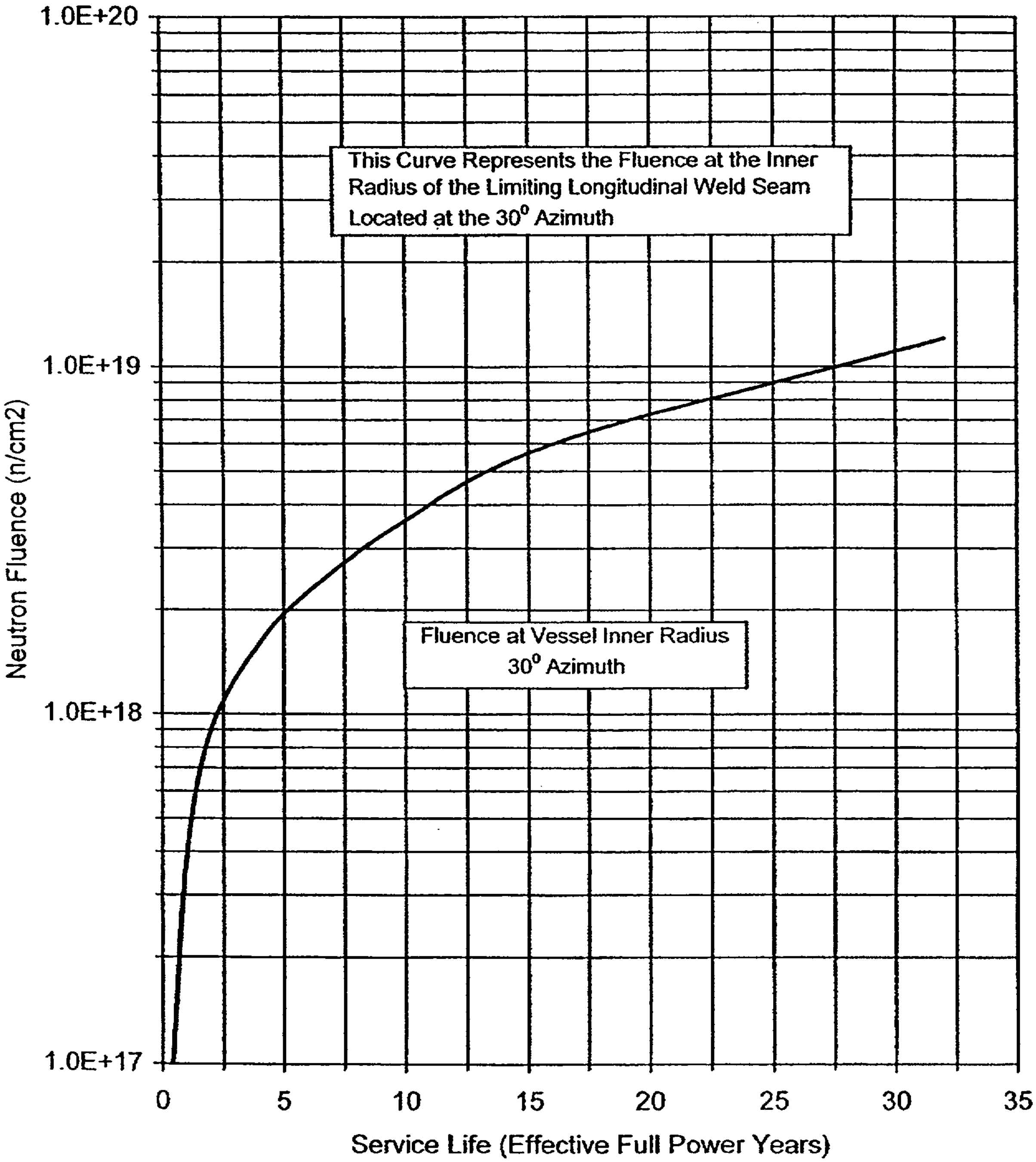
Insert E 2 Here



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

FIGURE B 3/4.4-1

INSERT E2



1.0E+20
1.0E+19
1.0E+18
1.0E+17
Neutron Fluence (n/cm2)
This Curve Represents the Fluence at the Inner Radius of the Limiting Longitudinal Weld Seam Located at the 30° Azimuth
Fluence at Vessel Inner Radius 30° Azimuth
0
5
10
15
20
25
30
35
Service Life (Effective Full Power Years)

3/4.7 PLANT SYSTEMS

BASES

3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES

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

16.66x10^6

15.08x10^6

110.4

maximum calculated

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Code, 1971 Edition. The total relieving capacity for all valves on all of the steam lines is ~~16,655,268~~ lbs/hr which is ~~118%~~ of the ~~total secondary~~ steam flow of ~~14,465,360~~ lbs/hr at 100% RATED THERMAL POWER. A minimum of 2 OPERABLE safety valves per OPERABLE steam generator ensures that sufficient relieving capacity is available for the allowable THERMAL POWER restriction in Table 3.7-~~2~~. 1

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

For 4 loop operation

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

For 3 loop operation

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

Where:

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

V = maximum number of inoperable safety valves per steam line

PROPOSED CHANGE DESCRIBED IN PSEG NUCLEAR LETTER DATED 10/06/2000 (LCR S99-13)

Proposed change described in PSEG Nuclear LLC ltr dated 9/26/2000. (LCR S99-13).

Justification for ASME Code Case N-640 Exemption Request

The following information provides the basis for the exemption request to 10 CFR 50.60 for use of American Society of Mechanical Engineers (ASME) Code Case N-640, "Alternative Reference Fracture Toughness for Development of P/T Limit Curves for ASME Section XI, Division I," in lieu of 10 CFR 50 Appendix G.

The requested exemption meets the criteria 10 CFR 50.12 as discussed below. 10 CFR 50.12 states that the Commission may grant exemptions from the requirements of 10 CFR 50 provided that:

1. **The requested exemption is authorized by law.**

   No law exists which precludes the activities covered by this exemption request. 10 CFR 50.60(b) permits the use of alternatives to the requirements in 10 CFR 50, Appendices G and H, when an exemption is granted under 10 CFR 50.12.

2. **The requested exemption will not present an undue risk to the public health and safety.**

   The revised Pressure-Temperature Limit Curves use a lower stress intensity factor, K_{IC}, as specified in ASME Code Case N-640, instead K_{IR}, which results in higher allowable pressures. K_{IR} is a reference stress intensity factor and is based on the lower band values of K_{IC} and K_{IA}. Use of K_{IR} would make the Pressure-Temperature Limit Curves overly conservative, since, the K_{IR} stress intensity is based on both static and dynamic fracture toughness data, while the K_{IC} stress intensity is based on only static fracture toughness data.

   Use of the K_{IC} in determining the lower bound fracture toughness in the development of Pressure-Temperature Limit Curves is more technically correct than the K_{IA} curve since the rate of loading during a heatup or cooldown is slow and is more representative of a static than a dynamic condition. The K_{IC} curve appropriately implements the use of static initiation fracture toughness behavior to evaluate the integrity of the reactor vessel. The conservatism of the K_{IA} curve was necessary due to limited knowledge of reactor pressure vessel (RPV) materials when the curve was first codified in 1974. Since then, however, knowledge gained about RPV materials demonstrated that the lower bound on fracture toughness provided by the K_{IA} curve is well beyond the margin of safety required to protect the public health and safety from potential RPV failure.

   Pressure-Temperature limit curves based on the K_{IC} curve will enhance overall plant safety by reducing the burden on plant operations and by improving margins to fuel damage. Heatup and cooldown curves based on ASME Section XI, Appendix G requirements would significantly restrict the ability to perform plant heatup and cooldown and create an unnecessary burden to plant operations. In addition, excessive shielding to provide an acceptable operating window with the current requirements can result in higher fuel peaking and less margin to fuel damage during an accident condition.

3. The requested exemption is consistent with the common defense and security.

The common defense and security are not endangered by this exemption request.

4. Special circumstances are present which necessitate the request for an exemption to the requirements of 10 CFR 50.60.

Pursuant to 10 CFR 50.12(a)(2), the NRC will consider granting an exemption if special circumstances are present. This exemption meets the special circumstances in 10 CFR 50.12(a)(2)(ii):

> Application of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule;

The underlying purpose of 10 CFR 50.60(a) and 10 CFR 50 Appendix G is to protect the integrity of the reactor coolant pressure boundary.

The ASME Section XI, Appendix G procedure was conservatively developed based on the level of knowledge existing in 1974 concerning RPV materials and the estimated effects of plant operation. Since then, the level of knowledge of these topics has been greatly expanded. Application of Code Case N-640 in the development of Pressure-Temperature limits curves is a more technically correct method than the current requirement. Use of the KIC curve in accordance with Code Case N-640 achieves the underlying intent of the applicable regulations.

Justification for Reactor Head and Vessel Flange Requirements Exemption Request

The following information provides the basis for the exemption request to 10 CFR 50.60 for use of WCAP-15315, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Operating PWR and BWR Plants." in lieu of 10 CFR 50 Appendix G.

The requested exemption meets the criteria 10 CFR 50.12 as discussed below. 10 CFR 50.12 states that the Commission may grant exemptions from the requirements of 10 CFR 50 provided that:

1. **The requested exemption is authorized by law.**

   No law exists which precludes the activities covered by this exemption request. 10 CFR 50.60(b) permits the use of alternatives to the requirements in 10 CFR 50, Appendices G and H, when an exemption is granted under 10 CFR 50.12.

2. **The requested exemption will not present an undue risk to the public health and safety.**

   The revised Pressure-Temperature Limit Curves being proposed rely in part on ASME Code Case N-640 which allows use of a lower stress intensity factor, K_{IC}, instead K_{IR}, which results in higher allowable pressures. 10 CFR Part 50, Appendix G addresses the metal temperature of the closure head flange and vessel flange regions. The regulation 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 preservice hydrostatic test pressure of 3106 psig. Implementing the K_{IC} stress intensity factors allowed by ASME Code Case N-640 without eliminating the flange requirement of 10 CFR 50 Appendix G would eliminate the benefit of ASME Code Case N-640 at temperatures below (flange RT_{NDT} + 120°F). In accordance with WCAP-15315, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation For Operating PWR and BWR Plants", the 10 CFR 50 Appendix G flange requirement is no longer necessary when using the methodology of Code Case N-640. Therefore, the Pressure-Temperature Limit Curves were generated without flange requirements included.

3. **The requested exemption is consistent with the common defense and security.**

   The common defense and security are not endangered by this exemption request.

4. **Special circumstances are present which necessitate the request for an exemption to the requirements of 10 CFR 50.60.**

   Pursuant to 10 CFR 50.12(a)(2), the NRC will consider granting an exemption if special circumstances are present. This exemption meets the special circumstances in 10 CFR 50.12(a)(2)(ii):

   > Application of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule;

The underlying purpose of 10 CFR 50.60(a) and 10 CFR 50 Appendix G is to protect the integrity of the reactor coolant pressure boundary.

WCAP-15315, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation For Operating PWR and BWR Plants," concluded that there is significant margin between the applied stress intensity factor and the fracture toughness at virtually all crack depths when using the K_{IC} toughness, which has been adopted by Section XI of the ASME Code for developing Pressure-Temperature Limit Curves. Another objective of the requirements in 10 CFR 50 Appendix G is to assure that fracture margins are maintained to protect against service induced cracking due to environmental effects. Since the governing flaw is on the outside surface (the inside is in compression) where there are no environmental effects, there is even greater assurance of fracture margin. Therefore, it can be concluded that the integrity of the closure head/flange region is not a concern for any of the operating plants using the K_{IC} toughness. In addition, there are no known mechanisms of degradation for this region, other than fatigue. The calculated design fatigue usage for this region is less than 0.1, so it may be concluded that flaws are unlikely to initiate in this region.

Therefore, use of WCAP-15315 together with Code Case N-640 achieves the underlying intent of the applicable regulations.

WCAP-15553, "Power Calorimetric for the 1.4% Uprating for Public Service Electric and Gas Company Salem Units 1 and 2"
(proprietary)

Application and Affidavit by Westinghouse Electric Company LLC for Withholding Proprietary Information Contained in Attachment 7 From Public Disclosure In Accordance With 10 CFR 2.790

CAW-00-1428

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

Before me, the undersigned authority, personally appeared Henry A. Sepp, who, being by me duly sworn according to law, deposes and says that he is authorized to execute this Affidavit on behalf of Westinghouse Electric Company LLC ("Westinghouse"), and that the averments of fact set forth in this Affidavit are true and correct to the best of his knowledge, information, and belief:

Henry A. Sepp, Manager

Regulatory and Licensing Engineering

Sworn to and subscribed

before me this 3RD day

of November, 2000

Lorraine M. Piplica

Notary Public

LORRAINE M. PIPLICA
COMMONWEALTH OF PENNSYLVANIA
NOTARY PUBLIC

Notarial Seal
Lorraine M. Piplica, Notary Public
Monroeville Boro, Allegheny County
My Commission Expires Dec. 14, 2003

Member, Pennsylvania Association of Notaries

(1) I am Manager, Regulatory and Licensing Engineering, in the Nuclear Services Division, of the Westinghouse Electric Company LLC ("Westinghouse"), and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rulemaking proceedings, and am authorized to apply for its withholding on behalf of Westinghouse.

(2) I am making this Affidavit in conformance with the provisions of 10CFR Section 2.790 of the Commission's regulations and in conjunction with the Westinghouse application for withholding accompanying this Affidavit.

(3) I have personal knowledge of the criteria and procedures utilized by Westinghouse Electric Company LLC in designating information as a trade secret, privileged or as confidential commercial or financial information.

(4) Pursuant to the provisions of paragraph (b)(4) of Section 2.790 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

(i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Westinghouse.

(ii) The information is of a type customarily held in confidence by Westinghouse and not customarily disclosed to the public. Westinghouse has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitutes Westinghouse policy and provides the rational basis required.

Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential competitive advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage, e.g., by optimization or improved marketability.

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.

(e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

(f) It contains patentable ideas, for which patent protection may be desirable.

There are sound policy reasons behind the Westinghouse system which include the following:

(a) The use of such information by Westinghouse gives Westinghouse a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Westinghouse competitive position.

(b) It is information which is marketable in many ways. The extent to which such information is available to competitors diminishes the Westinghouse ability to sell products and services involving the use of the information.

(c) Use by our competitor would put Westinghouse at a competitive disadvantage by reducing his expenditure of resources at our expense.

(d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component may be the key to the entire puzzle, thereby depriving Westinghouse of a competitive advantage.

(e) Unrestricted disclosure would jeopardize the position of prominence of Westinghouse in the world market, and thereby give a market advantage to the competition of those countries.

(f) The Westinghouse capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.

(iii) The information is being transmitted to the Commission in confidence and, under the provisions of 10CFR Section 2.790, it is to be received in confidence by the Commission.

(iv) The information sought to be protected is not available in public sources or available information has not been previously employed in the same original manner or method to the best of our knowledge and belief.

(v) The proprietary information sought to be withheld in this submittal is that which is appropriately marked in WCAP-15553, "Power Calorimetric for the 1.4% Uprating for Public Service Electric and Gas Company Salem Units 1 and 2". This information is being transmitted by Public Service Enterprise Group, LLC letter and Application for Withholding Proprietary Information from Public Disclosure, to the Document Control Desk, Attention: Mr. Samuel J. Collins. The proprietary information as submitted for use by the Public Service Enterprise Group, LLC, Salem Units 1 and 2 is expected to be applicable in other licensee submittals in response to certain NRC requirements for licensing of a 1.4% power uprate to 3459 MWt.

This information is part of that which will enable Westinghouse to:

(a) Provide documentation supporting the determination of power measurement uncertainty associated with the 1.4% uprate.

(b) Provide the applicable engineering evaluations which establish the technical basis for the 1.4% power uprate.

(c) Provide licensing information to support license amendments.

Further, this information has substantial commercial value as follows:

(a) Westinghouse plans to sell the use of similar information to its customers for purposes of obtaining power uprates.

(b) Westinghouse can sell support and defense of the methodology in the licensing process.

Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar services and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

The development of the technology described in part by the information is the result of applying the results of many years of experience in an intensive Westinghouse effort and the expenditure of a considerable sum of money.

In order for competitors of Westinghouse to duplicate this information, similar technical programs would have to be performed and a significant manpower effort, having the requisite talent and experience, would have to be expended for developing testing and analytical methods and performing tests.

Further the deponent sayeth not.

PROPRIETARY INFORMATION NOTICE

Transmitted herewith are proprietary and/or non-proprietary versions of documents furnished to the NRC in connection with requests for generic and/or plant-specific review and approval.

In order to conform to the requirements of 10 CFR 2.790 of the Commission's regulations concerning the protection of proprietary information so submitted to the NRC, the information which is proprietary in the proprietary versions is contained within brackets, and where the proprietary information has been deleted in the non-proprietary versions, only the brackets remain (the information that was contained within the brackets in the proprietary versions having been deleted). The justification for claiming the information so designated as proprietary is indicated in both versions by means of lower case letters (a) through (f) contained within parentheses located as a superscript immediately following the brackets enclosing each item of information being identified as proprietary or in the margin opposite such information. These lower case letters refer to the types of information Westinghouse customarily holds in confidence identified in Sections (4)(ii)(a) through (4)(ii)(f) of the affidavit accompanying this transmittal pursuant to 10 CFR 2.790(b)(1).

COPYRIGHT NOTICE

**WCAP-15553, "Power Calorimetric for the 1.4% Uprating for Public Service Electric and Gas Company Salem Units 1 and 2"
(non-proprietary)**

POWER CALORIMETRIC FOR THE 1.4 % UPRATING FOR PUBLIC SERVICE ELECTRIC AND GAS COMPANY SALEM UNITS 1 AND 2

September, 2000

T. P. Williams
M. D. Coury
C. R. Tuley

Westinghouse Electric Company LLC
4350 Northern Pike
Monroeville, Pennsylvania 15146-2886

FOREWORD

This document contains material that is proprietary to the Westinghouse Electric Company LLC. The information contained within brackets is considered to be proprietary information. The basis for making the information proprietary and the basis on which the information may be withheld from public disclosure is set forth in the affidavit of H. A. Sepp. Pursuant to the provisions of Section 2.790 of the Commission's regulations, this affidavit is attached to the application for withholding from public disclosure which accompanies this document.

This information is for your internal use only and should not be released to any persons or organizations outside Public Service Electric & Gas Company, Salem Units 1 and 2 without the prior written approval of Westinghouse Electric Company LLC, Nuclear Services Division. Should it become necessary to obtain such approval, please contact B. W. Bevilacqua, Manager, Equipment Design and Regulatory Engineering, Westinghouse Electric Company LLC, Nuclear Services Division, 4350 Northern Pike, Monroeville, Pennsylvania 15146-2886.

TABLE OF CONTENTS

POWER CALORIMETRIC FOR THE 1.4 % UPRATING
INSTRUMENT UNCERTAINTY METHODOLOGY

I. INTRODUCTION

The purpose of this analysis is to determine the uncertainty in the Daily Power Calorimetric for the 1.4% Uprating. Reactor power is monitored by the performance of a secondary side heat balance (power calorimetric) at least once every 24 hours. The Daily Power Calorimetric uncertainty must be a value significantly small enough to account for the increase in nominal operating power.

Westinghouse has been involved with the development of several techniques to treat instrumentation uncertainties. An early version used the methodology outlined in WCAP-8567 "Improved Thermal Design Procedure",[1,2,3] which is based on the conservative assumption that the uncertainties can be described with uniform probability distributions. Another approach is based on the more realistic assumption that the uncertainties can be described with random, normal, two sided probability distributions.[4] This approach is used to substantiate the acceptability of the protection system setpoints for many Westinghouse plants, e.g., Millstone Unit 3, Diablo Canyon, Farley and others. The second approach is now utilized for the determination of all instrumentation uncertainties for the RTDP parameters and protection functions.

II. METHODOLOGY

The methodology used to combine the error components for a channel is the square root of the sum of the squares (SRSS) of those groups of components which are statistically independent. Those uncertainties that are dependent are combined arithmetically into independent groups, which are then systematically combined. The uncertainties used are considered to be random, two sided distributions. This technique has been utilized before as noted above, and has been endorsed by the NRC staff[(6,7,8,9)] and various industry standards[(10,11)].

The relationships between the error components and the channel instrument error allowance are variations of the basic Westinghouse Setpoint Methodology[(12)] and are based on Salem Units 1 & 2 specific procedures and processes and are defined as follows:

For parameter indication utilizing the plant process computer:

$$\begin{aligned} CSA = \{&(PMA)^2 + (PEA)^2 + (SMTE+SCA)^2 + (SPE)^2 + (STE)^2 + (SRA)^2 + \\ &(SMTE + SD)^2 + (RMTE + RCA)^2 + (RTE)^2 + (RMTE + RD)^2 + (COMPREF)^2 + \\ &(COMPMTE + COMPCAL)^2 + (COMPTE)^2 + (COMPMTE + COMPDRIFT)^2\}^{1/2} + \\ &BIAS \end{aligned} \quad \text{Eq. 1}$$

where:

CSA	=	Channel Statistical Allowance
PMA	=	Process Measurement Accuracy
PEA	=	Primary Element Accuracy
SRA	=	Sensor Reference Accuracy
SCA	=	Sensor Calibration Accuracy
SMTE	=	Sensor Measurement and Test Equipment Accuracy
SPE	=	Sensor Pressure Effects
STE	=	Sensor Temperature Effects
SD	=	Sensor Drift
RCA	=	Rack Calibration Accuracy
RMTE	=	Rack Measurement and Test Equipment Accuracy
RTE	=	Rack Temperature Effects
RD	=	Rack Drift
COMPCAL	=	Plant Computer Calibration Accuracy

COMPTE	=	Plant Computer Temperature Effects
COMPDRIFT	=	Plant Computer Drift
COMPREF	=	Plant Computer Reference Accuracy
COMPMTE	=	Plant Computer Measurement and Test Equipment Accuracy

Many of the parameters above are defined in Reference 12 and are based on ANSI/ISA 51.1-1979 (Reaffirmed 1993)[13]. However, for ease in understanding they are paraphrased below:

PMA - non-instrument related measurement errors, e.g., temperature stratification of a fluid in a pipe

PEA - errors due to a metering device, e.g., elbow, venturi, orifice

SRA - reference (calibration) accuracy for a sensor/transmitter

SCA - calibration tolerance for a sensor/transmitter

SMTE - measurement and test equipment used to calibrate a sensor/transmitter

SPE - change in input-output relationship due to a change in static pressure for a differential pressure (d/p) cell.

STE - change in input-output relationship due to a change in ambient temperature for a sensor or transmitter

SD - change in input-output relationship over a period of time at reference conditions for a sensor or transmitter

RCA - calibration accuracy for all rack modules in loop or channel assuming the loop or channel is string calibrated, or tuned, to this accuracy

RMTE - measurement and test equipment used to calibrate rack modules

RTE - change in input-output relationship due to a change in ambient temperature for the rack modules

RD - change in input-output relationship over a period of time at reference conditions for the rack modules

COMPCAL - calibration accuracy for plant computer in loop or channel assuming the loop or channel is string calibrated, or tuned, to this accuracy

COMPDRIFT - change in input-output relationship over a period of time at reference conditions for the plant computer

COMPREF - Allowance encompassing the effects of linearity, hysterisis, and repeatability for the plant computer.

COMPTE - change in input-output relationship due to a change in ambient temperature for the plant computer

COMPMTE - measurement and test equipment used to calibrate plant computer

BIAS - a one directional uncertainty for a sensor/transmitter or a process parameter with a known magnitude

A more detailed explanation of the Westinghouse methodology noting the interaction of several parameters is provided in Reference 12.

III. INSTRUMENTATION UNCERTAINTIES

The Reactor Power Measurement algorithm will be discussed first, followed by the results of the power calorimetric calculations.

Reactor Power Measurement

The daily power measurement assumes the measurement of the feedwater flow using the CROSSFLOW system. The results of this measurement are used to adjust the feedwater flow venturi measurement as indicated in the plant process computer.

Assuming that the primary and secondary sides are in equilibrium; the core power is determined by summing the thermal output of the steam generators, correcting the total secondary power for Steam Generator blowdown, subtracting the RCP heat addition, adding the primary side system losses, and dividing by the core Btu/hr at rated full power. The equation for this calculation is:

$$RP = \frac{\{(\sum Q_{SG}) + Q_L - Q_P\}(100)}{H} \qquad \text{Eq. 2}$$

Where:

RP = Core power (% RTP)
Q_{SG} = Steam generator thermal output (BTU / hr)
Q_P = RCP heat addition (BTU / hr)
Q_L = Primary system net heat losses (BTU / hr)
H = Rated core power (BTU / hr).

For the purposes of this uncertainty analysis (and based on H noted above) it is assumed that the plant is at 100 % RTP when the measurement is taken. Measurements performed at lower power levels will result in different uncertainty values.

The thermal output of the Steam Generator is determined by a secondary side calorimetric measurement, which is defined as:

$$Q_{SG} = (h_s - h_f)*W_f - (h_s - h_{bd})*W_{bd} \qquad \text{Eq. 3}$$

Where:

h_s	=	Steam enthalpy (BTU/lb)
h_f	=	Feedwater enthalpy (BTU/lb)
h_{bd}	=	Steam generator blowdown enthalpy (BTU/lb)
W_f	=	Feedwater flow (lb/hr)
W_{bd}	=	Steam generator blowdown flow (lb/hr)

The Steam enthalpy is based on the measurement of Steam Generator outlet Steam pressure assuming saturated conditions. The Feedwater enthalpy is based on the measurement of Feedwater temperature and Feedwater pressure. Blowdown enthalpy is based on the measurement of Steam Generator outlet steam pressure assuming saturated conditions.

The feedwater flow is determined by a single measurement and the following calculation:

$$W_f = (C_o)(A_p)(\rho_{fw})(L/\Delta t) \qquad \text{Eq. 4}$$

where:

W_f = Feedwater loop flow (lb/hr)
C_o = CROSSFLOW flow profile correction factor
A_p = Cross sectional area of pipe flow path
ρ_{fw} = Feedwater density (lb/ft^3)
L = Length of pipe between transducer points
Δt = Time required for signature to travel length of L

- The feedwater flow profile correction factor is the product of a number of constants including as-built dimensions of the CROSSFLOW and calibration tests performed by the vendor.
- Feedwater density is based on the measurement of feedwater temperature and feedwater pressure.
- The pipe length between transducer points is a fixed value once CROSSFLOW system is installed.
- Time required for signature to travel between transducers is obtained from the CROSSFLOW electronics.

The power measurement is thus based on the following plant measurements:

Steamline pressure (P_s)
Feedwater temperature (T_f)

Feedwater pressure (P_f)
Steam generator blowdown
Feedwater flow (W_f) (from CROSSFLOW system)

and on the following calculated values:

Feedwater density (ρ_f)
Feedwater enthalpy (h_f)
Steam enthalpy (h_s)
Moisture carryover (affects h_s)
Primary system net heat losses (Q_L)
RCP heat addition (Q_p)

Uncertainties

The secondary side uncertainties are in four principal areas, Feedwater flow, Feedwater enthalpy, Steam enthalpy and net pump heat addition. These areas are specifically identified on Table 3.

For the measurement of feedwater flow, the CROSSFLOW has a stated accuracy of [$]^{+a,c}$ which the utility provided to Westinghouse to use in the calculations. Since the calculated steam generator thermal output is proportional to feedwater flow, the flow coefficient uncertainty is expressed as [$]^{+a,c}$.

An allowance of [$]^{+a,c}$ was used for the Steam Generator Blowdown venturi flow coefficient. This resulted in an uncertainty of [$]^{+a,c}$ power.

The uncertainty applied to the Steam Generator Blowdown venturi thermal expansion correction (F_a) is based on the uncertainties of the temperature and the coefficient of thermal expansion for the venturi material, 304 stainless steel. For this material, a change of $\pm$ 1.0 °F in the nominal temperature range changes F_a by [$]^{+a,c}$ but the change in steam generator thermal output is negligible.

An uncertainty of 5.0 % in F_a for 304 stainless steel is used in this analysis. This results in an additional uncertainty bounded by [$]^{+a,c}$ power. This allowance is included to account for the variations in material composition that could exist for the venturi.

Using the NBS/NRC Steam Tables it is possible to determine the sensitivities of various parameters to changes in feedwater temperature and pressure. Table 1 notes the instrument uncertainties for the hardware used to perform the parameter measurements. Table 2 lists the various parameter sensitivities. As can be seen on Table 2, feedwater temperature uncertainties have an effect on feedwater density and feedwater enthalpy. Feedwater pressure uncertainties affect feedwater density and feedwater enthalpy.

Steam Generator Blowdown venturi d/p uncertainties are converted to % Steam Generator Blowdown flow using the following conversion factor:

$$\% \text{ flow} = (\text{d/p uncertainty})(1/2)(\text{transmitter span} / 100)^2. \qquad \text{Eq. 5}$$

Using the NBS/NRC Steam Tables, it is possible to determine the sensitivity of Steam enthalpy to changes in Steam pressure and Steam quality. Table 1 notes the uncertainty in Steam pressure and Table 2 provides the sensitivity. For Steam quality, the Steam Tables were used to determine the sensitivity at a moisture content of [ ]$^{+a,c}$. This value is noted on Table 2.

The net pump heat addition uncertainty is derived from the combination of the primary system net heat losses and pump heat addition and are summarized for a four loop plant as follows:

System heat losses	- 2.0 MWt
Component conduction and convection losses	- 1.4 MWt
Pump heat adder	+ 15.4 MWt
Net Heat input to RCS	+ 12.0 MWt

The uncertainty on system heat losses, which is essentially all due to charging and letdown flows, has been estimated to be [ ]$^{+a,c}$ of the calculated value. Since direct measurements are not possible, the uncertainty on component conduction and convection losses has been assumed to be [ ]$^{+a,c}$ of the calculated value. Reactor coolant pump hydraulics are known to a relatively high confidence level, supported by system hydraulics tests performed at Prairie Island Unit 2 and by input power measurements from several other plants. Therefore, the uncertainty for the pump heat addition is estimated to be [ ]$^{+a,c}$ of the best estimate value. Considering these parameters as one quantity, which is designated the net pump heat addition uncertainty, the combined uncertainties are less than [ ]$^{+a,c}$ of the total, which is less than [ ]$^{+a,c}$ of core power.

The calorimetric power measurement determination is performed using the plant computer or a manual calculation. As noted in Table 3, Westinghouse has determined the dependent sets in the calculation and the direction of interaction. The same was performed for the bias values.

Using the power uncertainty values noted on Table 3, the 4-loop uncertainty equation is as follows:

+a,c

Power = [ ] Eq. 6

Where:

- CF = Feedwater Flow (mass flow accuracy of CROSSFLOW system)
- $SGBF_{\Delta P}$ = Steam Generator Blowdown flow Delta P
- $SGBF_V$ = Steam Generator Blowdown flow venturi (basic accuracy)
- ρ_t = Feedwater flow density (as a function of temperature)
- h_t = Feedwater flow enthalpy (as a function of temperature)
- Fa_t = Steam Generator Blowdown flow F_a (as a function of temperature, inferred from steam pressure)
- Fa_m = Steam Generator Blowdown flow F_a (as a function of material)
- ρ_p = Feedwater flow density (as a function of pressure)
- h_p = Feedwater flow enthalpy (as a function of pressure)
- h_{SP} = Steam enthalpy (as a function of pressure)
- $h_{s\ moist}$ = Steam enthalpy (as a function of moisture)
- h_{SG_LIQ} = Steam Generator Blowdown flow enthalpy (as a function of steam pressure)
- ρ_{SG_P} = Steam Generator Blowdown flow density (as a function of steam pressure)
- NPHA = Net pump heat addition
- N = Number of primary side loops

+a,c

Power = [ ]

Based on the number of loops and the instrument uncertainties for the four parameters, the uncertainty for the secondary side power calorimetric measurement is:

# of loops	power uncertainty (% RTP)
4	[ ]$^{+a,c}$

TABLE 1
POWER CALORIMETRIC INSTRUMENTATION UNCERTAINTIES

+a,c

	FW TEMP °F	FW PRESS % Span ***	FW HDR. %Mass Flow	S/G BLDN % D/P Span	STM PRESS %Span
CF					
SRA					
SCA					
SMTE					
SPE					
STE					
SD					
BIAS					
COMPREF					
RCA					
COMPCAL					
RMTE					
COMPMTE					
RTE					
COMPTE					
RD					
COMPDRFT					
SQRTEXTR					
CSA					
# Inst Used					
Units	°F	Psi	Mass Flow	% d/p	psi
Inst Span	480	2000		120,000 lb/hr	1200
Inst Unc. (Random)					
Nominal	434.6°F	969 psia		35,000 lb/hr	869 psia

+a,c

* Provided by PSE&G

** Provided by CROSSFLOW

*** Rosemount Transmitter

TABLE 2
POWER CALORIMETRIC SENSITIVITIES AT 3459 MW THERMAL

+a,c

FEEDWATER FLOW	=	
FEEDWATER DENSITY		
TEMPERATURE	=	
PRESSURE	=	
FEEDWATER ENTHALPY		
TEMPERATURE	=	
PRESSURE	=	
h_s	=	
h_f	=	
Δh (SG)	=	
STEAM ENTHALPY		
PRESSURE	=	
MOISTURE	=	
SG BLOWDOWN ENTHALPY		
PRESSURE	=	
SG BLOWDOWN FLOW		
Fa		
TEMPERATURE	=	
MATERIAL	=	
DENSITY		
PRESSURE	=	
DELTA P	=	

TABLE 3
SECONDARY SIDE POWER CALORIMETRIC MEASUREMENT UNCERTAINTY

+a,c

COMPONENT	INSTRUMENT UNCERTAINTY	POWER UNCERTAINTY
FEEDWATER FLOW		
CF (CROSSFLOW)		
SG BLOWDOWN FLOW		
VENTURI ($SGBF_V$)		
THERMAL EXPANSION COEFFICIENT		
TEMPERATURE (Fa_t)		
MATERIAL (Fa_m)		
DENSITY		
PRESSURE (ρ_{SG_P})		
DELTA P ($SGBF_{\Delta P}$)		
SG BLOWDOWN LIQUID ENTHALPY		
PRESSURE (h_{SG_LIQ})		
FEEDWATER DENSITY		
TEMPERATURE (ρ_t)		
PRESSURE (ρ_p)		
FEEDWATER ENTHALPY		
TEMPERATURE (h_t)		
PRESSURE (h_p)		
STEAM ENTHALPY		
PRESSURE (h_{sp})		
MOISTURE ($h_{s\ moist}$)		
NET PUMP HEAT ADDITION (NPHA)		
SINGLE LOOP UNCERTAINTY		
3 LOOP UNCERTAINTY		

*, * *, * * * Indicates sets of dependent parameters

IV. RESULTS/CONCLUSIONS

The preceding sections provide the methodology to account for the Power Calorimetric uncertainties for the 1.4 % Uprating. The uncertainty calculations have been performed for Salem Units 1 and 2 utilizing plant specific instrumentation and calibration procedures. A power calorimetric uncertainty value of [ ]$^{+a,c}$ will be used in the Salem Units 1 and 2 safety analysis.

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