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DEC 0 1 2000

LR-N00-0405
LCR H00-05

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

Gentlemen:

REQUEST FOR LICENSE AMENDMENT
INCREASED LICENSED POWER LEVEL
HOPE CREEK GENERATING STATION
FACILITY OPERATING LICENSE NO. NPF-57
DOCKET NO. 50-354

In accordance with 10CFR50.90, PSEG Nuclear LLC hereby requests a change to Facility Operating License No. NPF-57 and to the Technical Specifications (TS) in Appendix A thereto for Hope Creek Generating Station. 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 3339 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.

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 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 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.

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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.

The proposed changes to the Technical Specifications include revised pressure-temperature limit curves. PSEG Nuclear requests two exemptions from the requirements of 10 CFR 50.60(a) and 10 CFR 50 Appendix G for use of the following American Society of Mechanical Engineers (ASME) Code Cases as alternatives to requirements described in 10 CFR 50 Appendix G:

- ASME Code Case N-640, "Alternative Reference Fracture Toughness for Development of P/T Limit Curves for ASME Section XI, Division I," and
- ASME Code Case N-588, "Alternative to Reference Flaw Orientation of Appendix G for Circumferential Welds in Reactor Vessels, Section XI, Division I."

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

Attachment 7 contains the calculation of heat balance uncertainty when the Crossflow system is used to correct the feedwater mass flow input.

A description of the inputs, methodology and results for the revised pressure-temperature curves is contained in Attachment 8.

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

1. A final evaluation of the impact of the proposed uprate on grid stability 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 June 1, 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,

Mark B. Bezilla
Vice President - Technical Support

Affidavit
Attachments (8)

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REF: LR-N00-0405
LCR H00-05

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

Mark B. Bezilla, 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 Hope Creek Generating Station are true to the best of my knowledge, information and belief.

Subscribed and Sworn to before me
this 1 day of December, 2000

Notary Public of New Jersey

My Commission expires on SHERI L. HUSTON
NOTARY PUBLIC OF NEW JERSEY
My Commission Expires 12/08/2003

**HOPE CREEK GENERATING STATION
FACILITY OPERATING LICENSE NO. NPF-57
DOCKET NO. 50-354
CHANGE TO TECHNICAL SPECIFICATIONS
INCREASED LICENSED POWER LEVEL**

DESCRIPTION OF THE PROPOSED CHANGE:

The proposed license amendment would revise the Hope Creek Generating Station Operating License and Technical Specification to increase licensed power level for operation to 3339 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 License NPF-57 is revised to authorize operation at a steady state reactor core power level not in excess of 3339 megawatts (one hundred percent of rated power).

2. The definition of RATED THERMAL POWER in Technical Specification (TS) 1.35 is revised to reflect the increase from 3293 MWt to 3339 MWt.

3. TS 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. New Heatup and Cooldown Curves

1. Technical Specification Figures 3.4.6.1-1, 3.4.6.1-2 and 3.4.6.1-3, pressure-temperature limit curves for hydrostatic testing, non-nuclear heatup and cooldown, and critical operation, and their associated Bases are revised to support the increase in core power based on uprated fluence projections.

2. Surveillance Requirement 4.4.6.1.4 is being revised to be made consistent with the limit on reactor vessel flange and head flange metal temperature in TS 3.4.6.1.d.

C. Editorial Changes

1. In TS Bases 3/4.4.6, 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. The TS Index is being revised to correctly show the locations for Figures 3.4.6.1-1, 3.4.6.1-2 and 3.4.6.1-3.

Along with the proposal to increase licensed power level to 3339 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 Hope Creek 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:

Hope Creek is currently licensed to operate at a maximum power level of 3293 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.

PSEG Nuclear intends to install Crossflow ultrasonic flow measurement (UFM) system for feedwater flow measurement in Hope Creek by January 15, 2001. 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% 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 Hope Creek. 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 Hope Creek Generating Station is presently licensed for a full core power rating of 3293 MWt. Through the use of more accurate feedwater flow measurement equipment, approval is sought to increase this core power by 1.4% to 3339 MWt. PSEG Nuclear evaluated the impact of the proposed core power uprate on nuclear steam supply system (NSSS) systems and components, balance of plant (BOP) systems, and safety analyses.

1.2 APPROACH

The evaluation of the proposed uprate covered the areas described in NEDO-31897, "Generic Guidelines for General Electric Boiling Water Reactor Power Uprate," issued in February 1992. The guidelines in NEDO-31897 describe the generic criteria, methodology and scope of evaluation to support applications for increases in authorized core power level.

The results of PSEG Nuclear's evaluation are summarized in the following sections of this attachment. Section 2 discusses the assessment of anticipated operational occurrences and postulated accidents described in Hope Creek Updated Final Safety Analysis Report (UFSAR) Chapter 15. Section 3 summarizes the evaluation of fuel mechanical design. Section 4 describes the assessment of the current thermal-hydraulic analysis. Sections 5 through 11 summarize the results of design bases reviews to assess the impact of the proposed uprate on plant systems structures and components. Section 12 provides a summary of effects on radiological consequences. Section 13 discusses the effect of the uprate on plant operations, and Sections 14 and 15 describe evaluations of other licensing requirements. The results of all of the analyses and evaluations performed demonstrate that all acceptance criteria will continue to be met.

1.3 GENERAL LICENSING APPROACH FOR PLANT ANALYSES USING PLANT POWER LEVEL

Rated thermal power is used as an input to most plant safety, component, and system analyses.

Analyses for which a 2% increase was applied to the initial power level to account solely for the power measurement uncertainty do not need to be re-performed for the 1.4% uprate conditions because the sum of increased core power level (1.4%) 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 1.4.6 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% power measurement uncertainty. Accordingly, the existing 2% uncertainty can be allocated such that 1.4% is applied to provide sufficient margin to address the uprate to 3339 MWt, and 0.6% is retained in the analysis to still account for the power measurement uncertainty.

Core and fuel performance analyses described in sections 2.3.1 and 2.3.5 of this Attachment will be reanalyzed or reevaluated on a cycle-specific basis as described in CENPD-300-P-A, "Reference Safety Report for Boiling Water Reactor Reload Fuel," July 1996, even if the current analysis included a 2% allowance for power measurement uncertainty.

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

Some analyses already employ a core power level greater than the proposed 3339 MWt. For these analyses, some of this available margin has been used to offset the 1.4% uprate, and the analyses have been evaluated to confirm that sufficient analysis margin exists to envelope the 1.4% uprate.

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 3339 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 (3293 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 3293 MWt until the Crossflow system was returned to service and a heat balance in accordance with SR 4.3.1.1 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 Hope Creek

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 Hope Creek, 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 Hope Creek 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 Hope Creek Crossflow uncertainty calculation indicating a mass flow accuracy of better than 0.5% of rated flow for the site specific installation. 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 shown in Attachment 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 Hope Creek Crossflow installation will be installed and calibrated to a site specific piping configuration (flow profile and meter factors are representative 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.

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 ACCIDENT ANALYSIS ASSESSMENT (UFSAR CHAPTER 15)

2.1 APPROACH FOR THE DISPOSITION OF EVENTS

Based on the relative event probabilities and failure assumptions, the events evaluated herein have been separated into two categories as defined in CE NPSD-839-P, Rev. 0, "Westinghouse BWR Reload Licensing Methodology Basis for Public Service Electric & Gas Hope Creek Generating Station," August 2000:

- Anticipated Operational Occurrences (AOOs), and
- Accidents.

The disposition of each of these categories is discussed below.

2.1.1 Anticipated Operational Occurrences

The AOOs identified in the updated Final Safety Analysis Report (UFSAR) were evaluated to determine the effect of a 1.4% power uprate on the Westinghouse BWR reload methodology application and AOO results. The evaluations have resulted in the disposition of each AOO as follows:

1. Reanalyze or reevaluate on a cycle-specific basis including the appropriate treatment of power measurement uncertainty as described in [CENPD-300-P-A, "Reference Safety Report for Boiling Water Reactor Reload Fuel"], or
2. Bounded by the results of the cycle-specific AOO analyses, or
3. Previously evaluated at a bounding power level, or
4. Unaffected by the proposed power uprate.

No methodology changes were identified as required for the application of the Westinghouse BWR reload methodology for the proposed power uprate.

From the uprated initial condition, both the Critical Power Ratio (CPR) and Linear Heat Generation Rate (LHGR) results during the AOO were evaluated. The acceptance limit for the CPR assessment is the Safety Limit Minimum CPR (SLMCPR) (i.e., MCPR > SLMCPR). The SLMCPR is evaluated on a cycle-specific basis. Consequently, the power uprate has no generic impact on the SLMCPR methodology or bases. For the LHGR assessment, the acceptance criterion is that the calculated LHGR during an AOO is less than or equal to an LHGR Overpower Limit (LHGR limit consistent with <1% plastic strain, no fuel melting). As discussed further in section 3, since the steady state operating limit LHGR and fuel rod burnup limits are not being increased due to the core power uprate, the uprate has no generic impact on the LHGR acceptance limit as determined by the Westinghouse BWR reload methodology.

2.1.2 Accidents

The accidents identified in the UFSAR were evaluated to determine the effect of a 1.4% power uprate on the Westinghouse BWR reload methodology application and accident results. The evaluations have resulted in the disposition of each accident as follows:

1. Reanalyze or reevaluate on a cycle-specific basis including the appropriate treatment of power measurement uncertainty as described in CENPD-300-P-A, "Reference Safety Report for Boiling Water Reactor Reload Fuel," or
2. Bounded by the results of the cycle-specific accident analysis, or
3. Previously evaluated at a bounding power level, or
4. Unaffected by the proposed power uprate.

No methodology changes were identified as required for the application of the Westinghouse BWR reload methodology for the proposed power uprate.

Of special note in the licensing approach for the loss of coolant accident evaluation is a recent change to the regulations regarding ECCS performance and analysis assumptions. Prior to July 31, 2000, in order to perform a small power uprate, a special exemption would have been required for the LOCA evaluation from Appendix K of Part 50 of Title 10 to the Code of Federal Regulations. This regulation imposed a 2% licensed power margin on ECCS evaluation models of light water power reactors licensed in accordance with the requirements of Appendix K. Since the issuance of Federal Register Notice, FR Doc. 00-13745 Filed May 31, 2000, which allows for decreased power measurement uncertainty based on the existence of improved feedwater flow measurement capabilities, an exemption to the power uncertainty required by 10CFR50 is no longer necessary.

The basis for the rule change is that the Crossflow instrumentation (and others like it) provides a more accurate indication of feedwater flow (and corresponding reactor thermal power) than assumed in the original Appendix K requirements. The improved thermal power measurement accuracy removes the need for the full 2% power margin originally required by Appendix K. This increases the thermal power available for electrical power generation, while improving the certainty that the actual reactor thermal power remains at or below the value used to analyze ECCS performance during a LOCA.

The remainder of the accidents have been evaluated and dispositioned as described above in accordance with standard Westinghouse BWR reload applications.

2.2 IMPACT OF UPRATE ON METHODOLOGY AND CORRELATIONS

Table 2.2-1 shows the magnitude of changes in plant parameters assumed for UFSAR Chapter 15 evaluations and analyses that are affected by the proposed power uprate. The values in the table, which were obtained by extrapolating current heat balance relationships, are approximate and correspond to a 2%

increase in rated thermal power. The table shows that the changes to the affected parameters are very small and will not affect the bases of the Westinghouse BWR reload methodology or correlations utilized for the Hope Creek safety analyses inputs.

TABLE 2.2-1

APPROXIMATE PLANT PARAMETER CHANGES AS A RESULT OF POWER UPRATE

PLANT PARAMETER (at full power)	*UNITS*	*CURRENT*	*POWER UPRATE ANALYSIS*
Rated Thermal Power	MWt	3293	3358.9
APRM High Flux Reactor Trip	%RTP[1]	120	122.4
Steam Line Flow Rate	Mlb/hr	14.159	14.485
Feedwater Flow Rate (excludes drive flow)	Mlb/hr	14.127	14.453
Turbine Inlet Pressure	Psia	985	983
Steam Dome Pressure	Psia	1020	1020
Feedwater Temperature	°F	420	421.6
Turbine Control Valve #4 Position	%	~24	~30

FOOTNOTES:
1. Values are in terms of current rated thermal power (3293 MWt).

In addition, since the proposed uprate will be accomplished using the existing maximum design core flow and without increasing the relative operating domain of the power-flow map, no new design and licensing configurations are introduced by the proposed uprate for the application of the Westinghouse BWR reload methodology and correlations. The power-flow map and the associated control and protection systems, which are based on the % reactor power and/or % reactor flow relationship, will maintain their current percent definitions with 100% power defined to be equal to the uprated MWt value for the purposes of UFSAR Chapter 15 Accident Analyses.

2.3 EVENT DISPOSITION

2.3.1 AOOs Dispositioned To Be Reanalyzed Or Reevaluated On A Cycle-Specific Basis

The AOOs listed below will be analyzed or evaluated on a cycle-specific basis in accordance with CENPD-300-P-A, "Reference Safety Report for Boiling Water Reactor Reload Fuel," at the uprated power level including the appropriate treatment of power measurement uncertainty. The changes in full power operating conditions that are of a magnitude that could be associated with the change in rated thermal power are shown in Table 2.2-1. No changes to the Westinghouse BWR reload methodology as described in CENPD-300-P-A, "Reference Safety Report for Boiling Water Reactor Reload Fuel," are required to address the magnitude of the changes illustrated in Table 2.2-1. Cycle specific reload safety analysis inputs will be specified for each cycle specific evaluation consistent with the uprated power level to reflect the actual HCGS plant configuration.

- Loss of Feedwater Heating
- Feedwater Controller Failure – Maximum Demand
- Generator Load Rejection
- Rod Withdrawal Error – At Power
- Recirculation Flow Control Failure with Increasing Flow
- Inadvertent High Pressure Coolant Injection Startup

2.3.2 AOOs Bounded By The Results Of The Cycle-Specific AOO Analyses Or Evaluations

The results and consequences of the AOOs listed below have been determined to be bounded by the results and the consequences of those AOOs which will be reanalyzed or reevaluated on a cycle specific basis to determine the appropriate reload core design and licensing limits. The relationship between the events that are bounded and the events that provide the cycle specific bounding results will not be changed by the proposed power uprate.

- Turbine Trip
- Pressure Regulator Failure – Open
- Inadvertent Main Steam Relief Valve Opening
- Inadvertent RHR Shutdown Cooling Operation
- Pressure Regulator Failure – Closed
- Main Steam Isolation Valve Closure (Note: AOO evaluation for power distribution limits only; the overpressurization evaluation is described in section 15.2)
- Loss of Condenser Vacuum
- Loss of AC Power
- Loss of Feedwater Flow
- Reactor Recirculation Pump(s) Trip
- Recirculation Flow Controller Failure – Decreasing Flow
- Abnormal Startup of Idle Recirculation Pump

2.3.3 AOOs Previously Evaluated At A Bounding Power Level

The AOOs listed below were evaluated in the current UFSAR Chapter 15 analyses at a power level greater than the proposed uprate power level including consideration of power measurement uncertainty.

- Failure of RHR Shutdown Cooling

2.3.4 AOOs Unaffected By The Proposed Power Uprate

The changes associated with the proposed power uprate have been evaluated to not have any impact on the application of the Westinghouse BWR reload methodology to the AOOs listed below. The consequences of the events as described in UFSAR Chapter 15 will not be affected.

- Rod Withdrawal Error – Low Power
- Control Rod Maloperation

2.3.5 Accidents Dispositioned To Be Reanalyzed Or Reevaluated On A Cycle-Specific Basis

The accidents listed below will be analyzed or evaluated on a cycle-specific basis in accordance with CENPD-300-P-A, "Reference Safety Report for Boiling Water Reactor Reload Fuel," at the uprated power level including the appropriate treatment of power measurement uncertainty. The changes in full power operating conditions that are of a magnitude that could be associated with the change in rated thermal power are shown in Table 2.2-1. No changes to the Westinghouse BWR reload methodology as described in CENPD-300-P-A are required to address the magnitude of the changes illustrated in Table 2.2-1. Cycle specific reload safety analysis inputs as required by CENPD-300-P-A will be specified for each cycle specific evaluation consistent with the uprated power level to reflect the actual HCGS plant configuration.

- Misplaced Bundle Accident
- Control Rod Drop Accident
- ECCS Performance Analysis

2.3.6 Accidents Bounded By The Results Of The Cycle-Specific Accident Analyses Or Evaluations

The results and consequences of the Accidents listed below have been determined to be bounded by the results and the consequences of those accidents which will be reanalyzed or reevaluated on a cycle specific basis to determine the appropriate reload core design and licensing limits. The relationship between the events that are bounded and the events that provide the cycle specific bounding results will not be changed by the proposed power uprate.

- Reactor Recirculation Pump Shaft Seizure
- Reactor Recirculation Pump Shaft Break

2.3.7 Accidents Previously Evaluated At A Bounding Power Level

The accidents listed below were evaluated in the current UFSAR Chapter 15 analyses at a power level greater than the proposed uprate power level including consideration of power measurement uncertainty.

- Loss-of-Coolant Accident Resulting from the Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary Inside Primary Containment (Radiological Consequences)
- Gaseous Radwaste Subsystem Leak or Failure
- Fuel Handling Accident

2.3.8 Accidents Unaffected By The Proposed Power Uprate

The changes associated with the proposed power uprate have been evaluated to not have any impact on the application of the Westinghouse BWR reload methodology to the Accidents listed below. The consequences of the events as described in UFSAR Chapter 15 will not be affected.

- Instrument Line Pipe Break
- Steam System Piping Break Outside Containment
- Feedwater Line Break – Outside Primary Containment
- Spent Fuel Cask Drop Accident

3 FUEL MECHANICAL DESIGN BASIS ASSESSMENT

The fuel assembly mechanical design methodology addresses fuel assembly and fuel rod mechanical performance relative to the following types of design criteria:

1. Criteria for Accidents,
2. Criteria for assembly components other than fuel rods during normal operation and Anticipated Operational Occurrences (AOOs), and
3. Criteria for fuel rods during normal operation and AOOs.

The methodology is applied in fuel design and in reload evaluations in accordance with CENPD-300-P-A, "Reference Safety Report for Boiling Water Reactor Reload Fuel."

3.1 MECHANICAL PERFORMANCE DURING ACCIDENTS

Mechanical characteristics of the fuel, such as yield and ultimate strength, corrosion, and hydriding are assumed in the accident evaluations. The assumptions regarding these characteristics during licensing basis accidents will not be affected by a 1.4% increase in rated power. Although the core average neutron flux at rated conditions will be increased, the mechanical properties assumed in the accident analyses depend on integrated fluence and burnup. Since the burnup limits for the assemblies and integrated neutron fluence will be unchanged, the increase in rated thermal power will not affect the mechanical properties assumed in the accident analysis. This conclusion applies to the GE fuel as well as the SVEA-96+ fuel.

3.2 ASSEMBLY COMPONENTS OTHER THAN FUEL RODS

An increase in core power will cause a change in the hydraulic conditions that potentially could affect the mechanical performance of the assembly during normal operations and AOOs. Specifically, an increase in power will affect the assembly lift forces, the wear on assembly components, and the differential pressures on the channel used to evaluate channel stress and strain, creep rate, and fatigue. Since the peak burnup on the assembly will not change, and very wide margins to mechanical limits are available for all assembly components other than the channel and the fuel rods, no other effects that could adversely impact assembly performance require consideration. Furthermore, an increase in core power of 1.4% will not affect the axial growth of the SVEA-96+ or resident GE9B fuel since the burnup limits will not be changed. Therefore, the current compatibility evaluation depending on axial growth and relative axial growth will continue to be adequate for a 1.4% increase in core power.

The SVEA-96+ assembly lift evaluation for normal operation and AOOs was performed at core powers and flows that will bound the conditions expected for the proposed uprate to 3339 MWt. Since a 1.4% increase in core power will have a minimal effect on fuel assembly lift and the margin to assembly lift is in excess of 25%, continued wide margins to fuel assembly lift will continue to exist for an increase in rated core power of 1.4%.

This conclusion is also valid for the lift evaluation of the GE9B fuel assembly at uprated conditions for normal operation and AOOs. The GE9B fuel assembly is heavier than the SVEA-96+ assembly, and therefore, more resistant to lift. The geometry of the bottom nozzle and the pressure drop are approximately the same for the two bundles such that a 1.4% increase in core power will not penalize the GE9B bundle (relative to the SVEA-96+ bundle) in terms of a lift evaluation. Thus, by way of comparison to the conservative SVEA-96+ lift evaluation, it can be concluded that an increase in core power of 1.4% will not cause lifting of the GE9B assembly for normal conditions and AOOs. The impact on fuel assembly lift of a 1.4% increase in core power under accident conditions is discussed in section 15.3.

Endurance testing for hydraulic conditions bounding those expected in BWRs with substantially higher power density than Hope Creek, as well as operation in such high power density plants, has demonstrated acceptable wear performance of SVEA-96+ assembly components. Acceptable wear performance has been demonstrated for GE9B assembly components in BWR/6 plants, which have higher relative power densities and flows than Hope Creek. Therefore, the 1.4% power increase will not cause unacceptable wear.

An increase of 2% power would result in an increased differential pressure across the channel wall of about 1%. Therefore, the impact of a 1.4% increase in power on channel creep, stress and strain in the channel, and channel fatigue is evaluated conservatively by considering the effect of a 1% increase in differential pressure across the channel wall.

An increase of 1% in differential pressure across the channel wall will cause an increase of 1% in channel wall stress. Since the current evaluation of peak channel stress during normal operations and AOOs indicates margin greater than 1% to stress limits in the SVEA-96+ channel, an increase of 1.4% power will not alter the conclusion that adequate margins to channel stress limits are available. Similarly, since the GE9B fuel has a thicker channel wall than the SVEA-96+ fuel, an increase of 1.4% power will not alter the conclusion that adequate margins to channel stress limits are available. Instantaneous strain deflections during normal operation and AOOs are bounded by the maximum creep deflection discussed below.

The SVEA-96+ channel creep analysis and an evaluation of the GE9B channel creep performance conservatively indicate the maximum outward channel creep for the 1.4% core power uprated conditions to be lower than the deflection required for control rod interference. Therefore, a 1.4% increase in rated core power will not have an adverse affect on the interface between the fuel channels and the control rods.

Current fatigue analysis for the SVEA-96+ channel conservatively assumes a bounding history of pressure loading cycles on the channel and an assembly power in excess of the power at which an assembly would operate for a core power of 3339 MWt. Even with conservative wall thinning assumptions due to corrosion and an additional increase in pressure loading of 10%, a usage factor less than 0.2 was determined in the design analysis. Therefore, an increase of 1% in channel wall differential pressure corresponding to a 2% power increase would not alter the conclusion that substantial margin to fatigue limits are available for the SVEA-96+ channel. Since the GE9B fuel has a thicker channel wall than the SVEA-96+ fuel, and since significant margin to fatigue limits exists for the SVEA-96+ channel, a 2% increase in core power would not exceed the GE9B fuel channel fatigue limits.

For the proposed 1.4% increase in core power, adequate margins are available to all non-fuel design criteria for both the SVEA-96+ and GE9B assemblies.

3.3 COMPLIANCE WITH DESIGN CRITERIA FOR FUEL RODS

The mechanical design and licensing criteria for SVEA-96+ and GE9B fuel rods in HCGS have been demonstrated to be satisfied provided that:

1. Fuel rod Linear Heat Generation Rates (LHGRs) are maintained below the steady state operating limits,
2. Transient fuel rod LHGR overpowers are limited to a certain percent overpower above the steady state limits,
3. Fuel rod burnups are limited to existing NRC approved design or application burnup limitations.

The steady state operating limits, the transient overpower limits, and the fuel rod burnup limits will be unchanged by the core power increase of 1.4%. Therefore, the only potential impact of the proposed increase in rated core power on the current fuel rod evaluation is the impact that it might have on the validity of the assumptions of the design analyses that demonstrate acceptable performance within the existing steady state operating, overpower, and fuel rod burnup limits.

Fuel rod characteristics evaluated in the licensing analyses can be divided into those characteristics depending on the fuel rod power histories and those characteristics that do not depend on fuel rod power histories. Characteristics such as cladding corrosion and hydriding can be assumed to be independent of rod power history and, therefore, are strongly dependent only on the peak burnup, which does not change. While there might be some dependence of surface heat flux on corrosion rate, the corrosion and hydriding database used to demonstrate adequate performance to the burnup limit includes data from fuel rods operated at substantially higher surface heat fluxes than those that will be operated in Hope Creek with a 1.4% increase in rated power.

Design characteristics such as fuel rod internal pressure, clad strain, and fuel temperature do depend on fuel rod power history. The licensing analysis, however, is performed with a very conservative representation of a power history that envelops actual power histories anticipated during operation at the increased core power.

3.4 CONCLUSION

The existing mechanical design analyses/evaluations performed for both SVEA-96+ and GE9B fuel will continue to demonstrate adherence with the mechanical design criteria for an increase in rated thermal power of 1.4%.

4 THERMAL HYDRAULIC DESIGN BASIS ASSESSMENT

A 1.4% increase in rated core power could potentially affect the following aspects of the current thermal-hydraulic design evaluation:

1. Validity of the thermal-hydraulic design models,
2. Validity of Critical Power Ratio (CPR) correlations,
3. Applicability of the thermal-hydraulic design evaluation, and
4. Validity of SLMCPR evaluations.

4.1 THERMAL-HYDRAULIC DESIGN MODELS

Steady-state thermal and hydraulic design models describing the Hope Creek core and the resident GE fuel were established based on plant-specific data. These models describe the core region from the inlet orifice to the fuel channel exit. The fuel assemblies are described in sufficient detail to accurately predict the pressure, enthalpy, and mass flow rate distributions in the fuel assemblies, the inter-assembly bypass, and the bypass regions within the fuel assemblies given the following input conditions:

1. Core Power
2. Core Flow
3. Core Inlet Enthalpy or Feedwater Temperature
4. Dome Pressure
5. Axial and Radial Power Distributions

Thermal-hydraulic models were established for the Westinghouse BWR thermal-hydraulics code, CONDOR, to match as nearly as possible the pressure and flow rate distributions from the Hope Creek database. The CONDOR model was then used as a basis to establish Hope Creek plant specific hydraulic models for other design codes such as POLCA, BISON, and RAMONA. The database included core power and flow conditions on the perimeter of the current Hope Creek power-to-flow map to assure that all conditions required for the licensing analyses would be addressed. The database included an extrapolated point at 108% Power/104% Flow, where 100% power is 3293 MWt and 100%

flow is 100 Mlb/hr. Therefore, the steady-state hydraulic models were established over a range that includes the 1.4% power uprate. In addition, the CONDOR model showed good agreement with Hope Creek database over the entire range of conditions. Since these models capture physical bundle characteristics, which have been benchmarked over an entire range of power/flow points, a 1.4% increase in rated power would not negatively affect the acceptability of the calculated results.

4.2 VALIDITY OF CORRELATIONS

For the current core design and licensing, two CPR correlations are used:

- ABBD2.0 for the SVEA-96+ fuel, and
- US96G9 for the GE9 fuel.

The ABBD2.0 CPR correlation is valid over the range shown in Table 5.3 of CENPD-389-P-A, "10x10 SVEA Fuel Critical Power Experiments and CPR Correlations: SVEA-96+," September 1999. The US96G9 correlation is valid over the range described in Public Service Electric & Gas Company letter LR-N00105, dated March 27, 2000. The ranges of validity for the ABBD2.0 and US96G9 correlations bound anticipated transient and normal operation conditions. Therefore, the CPR correlations used to evaluate margins to dryout for the fuel in Cycle 10 and subsequent cycles will not be affected by a 1.4% increase in initial power level. Similarly, the ranges of correlations used to evaluate two-phase pressure drops and determine void fractions from hydraulic conditions are sufficiently broad that they will continue to be adequate. Therefore, the current SVEA-96+ and GE9B thermal hydraulic models in the Westinghouse BWR design and licensing analysis methodologies will continue to be valid for a 1.4% increase in rated thermal power.

4.3 VALIDITY OF THERMAL-HYDRAULIC DESIGN EVALUATION

The thermal-hydraulic design evaluation of SVEA-96+ fuel considered the following areas:

1. compatibility with the GE9B fuel,
2. fuel assembly lift,
3. hydraulic loads for mechanical design evaluations,
4. MCPR performance, and
5. treatment of crud.

The current hydraulic compatibility assessment evaluated core performance for simulations of Cycle 9 as well as transition cores through a full SVEA-96+ core for conditions ranging from 100% power (3293 MWt) and 100% core flow (100 Mlb/hr) to 60% power and 45% core flow. The hydraulic compatibility evaluations performed for the introduction of SVEA-96+ fuel will remain valid for the power uprate based on the minor differences in statepoint parameter/power distribution changes. Hydraulic compatibility is verified on a cycle specific basis in accordance with the methodology described in CENPD-300-P-A.

The lift and mechanical evaluation of the power uprate are described in section 3. The MCPR performance evaluation is described in section 4.4.

Conservative assumptions regarding crud buildup are included in the MCPR and mechanical evaluation relative to hydraulic effects in the Westinghouse BWR methodology. A 1.4% power uprate does not invalidate this methodology.

4.4 VALIDITY OF SLMCPR EVALUATION

The SLMCPR is established to protect the fuel from boiling transition during steady state operation and anticipated transients. The SLMCPR is established to provide that at least 99.9% of the fuel rods avoid boiling transition. The Westinghouse BWR SLMCPR evaluation conservatively utilizes power distributions preserving margins to thermal limits at rated conditions for the SLMCPR evaluation. This methodology will be unchanged by the proposed increase in rated thermal power.

A 1.4% increase in rated power could have a minor impact on initial power distributions. Implementation of the Crossflow system will permit a reduction in the assumed feedwater flow uncertainty. The combined effects of a power uprate on power distribution and reduced feedwater flow uncertainty (lower core power uncertainty) will not invalidate the current Cycle 10 SLMCPR conclusions. Furthermore, as required by CENPD-300-P-A, the SLMCPR for both SVEA-96+ and GE9B fuel will continue to be evaluated on a per cycle basis.

4.5 CONCLUSION

The current thermal-hydraulic analyses will continue to be applicable for a 1.4% increase in core power. Cycle-specific evaluations will continue to be performed in accordance with CENPD-300-P-A without any revised methodology.

5 REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

5.1 NUCLEAR SYSTEM PRESSURE RELIEF

Main Steam Line Safety/Relief Valves (SRVs) provide overpressure protection of the nuclear steam supply system (NSSS) during abnormal operational transients. For the 1.4% power uprate, power uprate operation is achieved through wider turbine control valve position. The maximum operating reactor dome pressure is unchanged from the current power operation. Therefore, there are no changes to the nuclear system pressure relief system for power uprate, and the nominal setpoints of the SRV's are also unchanged from the setpoints for current power level. The SRV operating pressure and temperature conditions for power uprate remain unchanged because the reactor operating pressure remains unchanged. Therefore, there is no impact on the SRV and the SRV discharge line due to power uprate. Additionally, plant operation with the current SRV setpoints has shown that an adequate difference exists between operating pressure and SRV setpoints (simmer margin). Since there is no change in the maximum operating reactor dome pressure specified in TS 3.4.6.2, uprate operation should not result in an increase in inadvertent SRV actuation.

5.2 REACTOR OVERPRESSURE PROTECTION

The ASME Code allowable maximum transient pressure limit for pressurization events is 1375 psig (110% of design value). The overpressure analysis performed in support of increased Technical Specification limits on SRV setpoint tolerances assumed initial core thermal power at 102% of rated. For the most limiting pressurization event (main steam line isolation valve closure with flux scram), the peak vessel bottom pressure is 1331 psig which is below the 1375 psig ASME limit. Since the nominal SRV setpoints and the reactor operating parameters are not changed as part of the 1.4% power uprate, the proposed change has no impact on the reactor overpressure protection analysis.

5.3 REACTOR VESSEL AND INTERNALS

5.3.1 Reactor Vessel Fracture Toughness

Revised pressure-temperature (P-T) curves were developed for pressure test, core not critical, and core critical conditions. A report describing the inputs, methodology and results for the revised curves is provided in Attachment 8. The revised curves are applicable for 32 effective full power years (EFPY). Curves applicable for 48 EFPY are included in the report for information only.

The curves were developed using the methodology specified in ASME Code Cases N-588 and N-640, as well as 1989 ASME Code, Section XI, Appendix G, 10CFR50 Appendix G, and WRC-175. The version of Appendix G used is the same as the 1989 version. The improvement realized from the Code Case methodology is as much as 60°F, and is primarily obtained from using the critical fracture toughness, K_{IC}, in accordance with Code Case N-640. Pressure and temperature instrument uncertainties are included in the revised curves.

Adjusted reference temperature (ART_{NDT}) values were developed for the Hope Creek reactor pressure vessel (RPV) materials in accordance with Regulatory Guide 1.99, Revision 2 based on projected fluence values which were increased in proportion to the increase in rated power. The calculated increase was based on the conservative assumption that the power uprate was initiated coincident with the beginning of the current fuel cycle.

Three regions of the reactor pressure vessel (RPV) were evaluated to develop the revised P-T curves: (1) the beltline region, (2) the bottom head region, and (3) the feedwater nozzle/upper vessel region. These regions bound all other regions with respect to brittle fracture.

Upper shelf energy (USE) calculations were performed and confirmed that all USE values are greater than 50 ft-lb throughout RPV life as required by 10 CFR 50 Appendix G.

Surveillance Requirement 4.4.6.1.4 is being revised to be made consistent with the TS limit on reactor vessel flange and head flange metal temperature. TS 3.4.6.1.d requires that reactor vessel flange and head flange metal temperatures be maintained greater than or equal to 79°F when reactor vessel head bolting studs are under tension. SR 4.4.6.1.4.a requires flange temperatures to be verified to be greater than 70°F in Operational Condition 4 once per 12 hours when RCS temperature is less than or equal to 100°F and once per 30 minutes when RCS temperature is 80°F. SR 4.4.6.1.4 is being revised to refer to the TS limit for minimum flange temperature, instead of 70°F. The limits on RCS temperature are also being revised to assure margins to temperature-pressure limits are maintained.

5.3.2 Reactor Internals and Pressure Differentials

The planned approach to achieve a 1.4% increase in rated power requires no increase in the maximum design core flow of 105%. Reactor operation will be accomplished within the existing flow domain by maintaining the relative rod lines consistent with the uprated power. RPV operating parameters (pressure and temperature) are not changed. Higher average power inside the core will tend to increase core average void fraction slightly thereby causing a slight change in the two phase flow quality and flow resistance. However, any change in the hydraulic state due to the 1.4% uprate is within the variation that may be experienced during normal operation over a cycle due to axial shifts in power distribution, crudding of the fuel or operation in the extended load line limit domain. Therefore, no change in the recirculation drive flow design bases is required for the 1.4% uprate. Hope Creek is analyzed for operation at 105% core flow. The evaluation of operation in the increased core flow (ICF) and extended load line limit (ELLA) regions was submitted by letter dated December 14, 1987 and approved in Hope Creek TS Amendment 15 on March 15, 1988. The evaluation concluded the impact on reactor internals due to acoustic and flow-induced loads were within design limits.

Seismic loads are not affected by power uprate. Dynamic loads due to LOCA and SRV were analyzed at 102% power level, which bounds the proposed 1.4% power uprate. Since the reactor pressure, temperature and recirculation design flow are not changing, the most limiting annulus pressurization and jet reaction loads are not expected to be impacted. Therefore the seismic and hydrodynamic loads are bounded by the original design basis.

The maximum internal pressure loads can be considered to be composed of two parts: steady state and transient pressure differentials. Core flow essentially affects only the steady state part. Design core flow is not being changed as a result of the proposed 1.4% power uprate. Core power affects both steady state and the transient differential pressure. It is noted that increased void fraction will also occur when the reactor pressure is below the maximum operating pressure, therefore higher transient differential pressure will be experienced by the reactor internals at below maximum pressure operation or during reactor transient. Since there is no change in design core flow and the transient differential pressure is covered by existing operating transients, the reactor internal differential pressure is not affected. In addition, the ICF and ELLA analysis described above evaluated the reactor internal pressure difference loads impact at maximum core flow at normal, upset and faulted conditions and concluded the stresses in reactor internals components remain within allowable design limits.

Current structural evaluation results for reactor internal and associated equipment show that there is at least a 17% margin between the calculated stress and the allowable value. Also, cumulative usage factors are less than 0.5 for the most critical components. Therefore, the impact of the 1.4% is expected to be insignificant given the available margin to the allowable stresses and the cumulative usage factors. In addition, the proposed uprate is still within the bounds of the current licensed 2% uncertainty operating limit.

Based on the above evaluation, it is concluded that 1.4% power uprate has minimal impact on the integrity of the reactor vessel internals.

5.3.3 Reactor Vessel Integrity

The reactor vessel design pressure is 1250 psig and the design temperature is 575°F. The maximum installed test pressure is 1563 psig. For the proposed 1.4% power uprate, RPV normal operating parameters (pressure and temperature) remain unchanged from the current operating condition. The planned approach to achieve 1.4% increase in rated power requires no increase in the maximum core flow.

The seismic loads are not affected by power uprate. There is no change in the SRV hydrodynamic load because there is no change in the SRV set pressures and the reactor operating pressure and temperature. Hydrodynamic loads associated with the most severe Loss Of Coolant Accident (LOCA) were developed based on 102% power level. Therefore reactor vessel design is not impacted by the proposed power uprate.

The proposed power uprate will have no impact on the recirculation inlet and out nozzles and the main steam nozzles because there is no change to the reactor and the main steam operating pressure and temperature. The revised heat balance at 101.4% power shows a slight increase in the feedwater temperature at the outlet of the No. 6 feedwater heater, the operating temperature would rise from about 420°F to 421°F. Design and normal operating temperatures used in the feedwater nozzle analyses are 575°F and 547°F respectively. Feedwater piping analyses have used 425°F as the operating temperature; therefore the nozzle loads from attached piping are not adversely affected by the proposed 1.4% power uprate. The only occasion where Feedwater operating temperature was used in design analyses is the thermal histogram during startup, power increase and reduction for use in the thermal discontinuity stress and fatigue evaluation. The temperature used in design analyses was at 420°F, the proposed 1.4% power uprate would raise this temperature to about 421°F. The change is only 1°F; the effect of a 1°F change on discontinuity stress contribution and fatigue usage factor is considered negligible. Therefore it is concluded that feedwater nozzle is also not impacted by the proposed power uprate.

Additionally, current design assessments show significant design margin in the reactor vessel integrity analyses. For the design load combinations and criteria for normal, upset, emergency and faulted conditions, the design margins range from 28% for vessel support to 8% for feedwater nozzle. These margins are not affected by the proposed power uprate, because the loading conditions are either unchanged or are bounded by the analyzed loading conditions.

Based on the above evaluation, it is concluded that the integrity of the reactor vessel and its appurtenances are not impacted by the proposed 1.4% power uprate.

5.4 REACTOR RECIRCULATION SYSTEM

The proposed power uprate will be achieved by expanding the power/flow map within existing relative rod and core flow control lines. Hope Creek is analyzed for operation at 105% core flow. No increase in the maximum core flow is required to achieve the 1.4% increase in rated thermal power. Reactor recirculation flow will be maintained along the existing power-flow map with 100% power defined as the uprated power level. Higher average power inside the core will tend to increase core average void fraction slightly thereby causing a slight change in the two phase flow quality and flow resistance. However, any change in the hydraulic state due to the 1.4% uprate is within the variation that may be experienced during normal operation over a cycle due to axial shifts in power distribution, crudding of the fuel or operation in the extended load line limit domain. This type of variation is already considered in the design bases of the recirculation system.

Each recirculation pump motor is vertical, variable speed ac electric motor that can drive the pump over a controlled range of 20 to 115% of rated pump speed. Therefore no change in the recirculation flow design bases is required for the 1.4% power uprate.

5.5 REACTOR COOLANT PIPING AND BALANCE-OF-PLANT PIPING

The 1.4% power uprate requires no change in the reactor design basis operating conditions. No changes are required to the design core flow or rated reactor pressure. There are no changes in the Reactor Coolant system operating and design pressures and temperatures. Nor are there any changes in the Main Steam operating and design pressures (1250 psig down to MS stop valve/1042 psig down to turbine stop valve) and temperatures (575°F). There is a slight increase in the Feedwater system operating pressure and temperature but no change in the design pressure (1500 psig) and temperature (425°F). The power uprate will result in about 1.8% flow increase in those system associated with the turbine cycle (i.e., Condensate, Feedwater, Main steam, etc.). However, they are still bounded by the 105% design flows for these systems as shown on the valves-wide-open heat balance. Also there is no impact to the main steam flow restrictor since its design flow bounds the 1.8% increase in the steam flow. Design of the NSSS and BOP piping and its support components have been reviewed. There are no changes in the piping design pressures and temperatures, and there are no changes in the seismic and hydrodynamic design loads. The only load that may be affected by the proposed 1.4% power uprate is the fluid transient load on steam lines due to main steam stop valve closure. The effect of this fluid transient load on piping design is evaluated in Section 5.10 and is found to have no impact on the piping and support design. Therefore it is concluded that the existing design bounds the 1.4% uprate conditions.

Flow accelerated corrosion program monitors 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 as required to the future inspection schedules.

5.6 MAIN STEAM ISOLATION VALVES

The MSIV's must close within specified limits under all design and operating conditions when receiving the signal to close. Existing design pressure (1250 psig), temperature (575°F) and flow (3.72×10^6 lbs/hr) for the MSIV's bounds the maximum operating conditions for the power uprate condition.

MSIV's are designed to fully close within the time limits set forth in the plant Technical Specifications at maximum flow and maximum differential pressure conditions. The maximum flow rate and the maximum pressure differential across the MSIV depend on maximum reactor dome pressure during the postulated steam line break event and the flow restrictor design. Neither the maximum dome pressure nor the flow restrictor design will be changed for power uprate. Additionally, high flow differential pressure switch setpoints, which provide MSIV closure, are not being changed. Therefore, the closure function and closure time will be unaffected.

5.7 REACTOR CORE ISOLATION COOLING SYSTEM

The RCIC system provides core cooling when the RPV is isolated from the main condenser. Transient analyses performed in UFSAR chapter 15 which require operation of RCIC system have been performed at initial condition of 104.3% power (UFSAR Table 15.0-3). Maximum operating dome pressure and SRV opening setpoints will not be changed for power uprate. Therefore, RCIC system will continue to maintain adequate water level at the proposed power uprate conditions.

5.8 RESIDUAL HEAT REMOVAL SYSTEM

The RHR system is designed to restore and maintain the reactor coolant inventory and provide for decay heat removal following reactor shutdown for both normal and post-accident conditions. Post accident inventory makeup and decay heat removal modes of the RHR system are discussed under the ECCS systems evaluation in section 6.2.

Shutdown cooling mode has a functional design requirement to reduce the reactor coolant temperature to 125°F within 20 hours after the control rods have been inserted. This can be achieved in less than 7 hours. Thus the added decay heat due to 1.4% power uprate will not compromise the shutdown cooling mode functional design to reduce the reactor coolant temperature to 125°F within 20 hours.

The functional design basis for the suppression pool cooling mode is to ensure that the suppression pool temperature does not exceed its maximum temperature limit immediately after blowdown. The safety related design basis requirements in the suppression pool cooling mode were performed at 102% power level in existing evaluations, therefore 1.4% power uprate has no impact on this design function.

5.9 REACTOR WATER CLEANUP SYSTEM

The RWCU system is designed to remove solid and dissolved impurities from the recirculated reactor coolant, thereby reducing the concentration of radioactive and corrosive species in the reactor coolant system. System temperature, pressure and flow during operation is not changed as a result of the 1.4% power uprate. Thus, RWCU system piping and component integrity is not affected.

As a result of the 1.4% power uprate, there will be a small increase in feedwater flow causing a slight increase in the chemical impurity concentration factor for the reactor vessel. This increase will be minor and is not expected to impact the RWCU system operation. Since the reactor water quality requirements, and the RWCU quality requirements are not a function of power level, the RWCU water chemistry is not impacted.

5.10 MAIN STEAM AND FEEDWATER PIPING.

Main Steam and Feedwater piping has been designed to Turbine Valves-Wide-Open (VWO) conditions which is at 105% steam flow (UFSAR Fig. 10.1-2). The proposed power uprate is expected to increase the main steam and feedwater system flow by about 1.8%, which is still bounded by the original VWO design basis flows. At uprated power level, normal operating temperature and pressure of the main steam line do not change from the current operating values. Feedwater temperature and pressure increase slightly. However they are also bounded by the VWO parameters. The existing balance of plant (BOP) design bounds the 1.4% uprated power conditions.

Fluid transient load due to turbine stop valve closure will increase as a result of the 1.8% increase in main steam flow. However, fluid transient loads are considered as occasional loads and are combined with other dynamic loads and with pressure and dead weight loads in piping evaluations. The resultant load increase is negligible and has no material impact on the piping and support design.

5.11 CONTROL ROD DRIVE HYDRAULIC SYSTEM

The control rod drive system (CRD) controls gross changes in core reactivity by incrementally positioning neutron absorbing control rods within the reactor core. The system is also required to quickly shut down the reactor (scram) by rapidly inserting withdrawn control rods into the core. The proposed power uprate does not change reactor operating pressure, temperature and design core flow. The system components within the reactor coolant pressure boundary have been designed to 1250 psig and 575°F. Therefore, the calculated stresses are unchanged. Since the reactor operating pressure does not change for the proposed 1.4% power uprate, current fatigue analyses remain valid. Therefore the CRD Hydraulic System is not impacted by the proposed 1.4% power uprate.

6 ENGINEERED SAFETY FEATURES

6.1 CONTAINMENT SYSTEMS

6.1.1 Containment Response Analyses

The current licensing basis analyses for both short term and long term containment pressure and temperature responses following a LOCA assume an initial core power of 3359 MWt (102% RTP) to determine peak containment and torus (suppression pool) temperature and pressure values. Thus, containment response analyses for the 1.4% power uprate are still bounded by the existing analyses.

A separate suppression pool temperature analysis was performed to demonstrate local pool temperatures were within the condensation stability limits specified in NUREG-0783 (Suppression Pool Temperature Limits for BWR Containments). The power level used for this analysis was 3436 MWt (104.3% of RTP). Therefore, the existing analysis bounds the 1.4% power uprate.

6.1.2 Containment Dynamic Loads

Analyses of containment dynamic loads as a result of a LOCA and SRV blowdown were performed to address each of the applicable requirements of NUREG 0661, "Mark I Containment Long Term Program." Results of the plant specific evaluation were submitted to the NRC as "HCGS Plant Unique Analysis Report (HCGS PUAR), dated 1984" as referenced in the UFSAR.

The initial power level used to determine the design basis LOCA hydrodynamic load definitions, was at 102% power level (3359 MWt). There is no change in SRV set pressure, therefore there is no impact on the SRV load definition for the initial SRV actuations. For subsequent actuations, the only parameter changed with power uprate is the time between SRV actuations. The SRV hydrodynamic design basis ensures that subsequent SRV actuation occurs only after the water level oscillations have damped out and the level has stabilized at a point determined by the drywell-to-wetwell differential pressure minus the vacuum breaker setpoint (HCGS PUAR). Primary system transient analyses are used to confirm that more than the minimum required time is available for the SRV discharge line (SRVDL) water leg to return to the equilibrium position. To further insure that the SRVDL water leg will be at its equilibrium height for all subsequent actuation cases, Hope Creek has delay logic on the 2 lowest-set SRVs to allow the water leg to clear after initial actuation (HCGS PUAR 1-4.2). Therefore there is no impact on the SRV hydrodynamic loads, and the existing design for the LOCA hydrodynamic loads bounds the 1.4% power uprate.

Power uprate may cause an increase in the suppression pool temperature response, thereby affect the local temperature limit as defined in NUREG-0783. Suppression pool temperature response was evaluated at 3436 MWt (104.3%), therefore the proposed power uprate has no impact on the suppression pool temperature limit.

6.1.3 Subcompartment Pressurization

The RPV bioshield annulus area pressurization considered both a reactor recirculation line and a feedwater line break. For the proposed 1.4% power uprate, reactor pressure and temperature are not changing. Therefore pressure and temperature parameters used for the recirculation line break case still bound the 1.4% power uprate. Feedwater temperature and pressure inputs were based on 105% steam flow conditions which bound the uprated operating temperature and pressure. Therefore the mass and energy releases associated with the rupture of feedwater piping are conservative and bound the 1.4% power uprated condition.

Drywell head area pressurization analysis is based on the rupture of the RPV head spray piping. The reactor operating parameters are not changed by the 1.4% power uprate. Thus, the current analysis is still valid for the 1.4% power uprate.

6.1.4 Containment Isolation

Reactor operating parameters are not changing for the 1.4% power uprate and the current containment response analyses have been performed at 102% power level, therefore there is no change in the resultant post accident short-term containment pressure and temperature responses that occur during the isolation of the majority of the containment isolation valves. Therefore containment isolation associated with systems connected to the reactor coolant pressure boundary and the containment air space are not affected by the power uprate. Also there are no changes in the operating parameters of the closed loop BOP systems for the 1.4% power uprate, therefore their isolations are also not affected.

In addition, Generic Letter 89-10 program was reviewed. The BWR MOV Program guidelines stipulate that worst-case pressure, temperature and flow parameters be used for determining the differential pressure and flow conditions. The only systems within the GL 89-10 program where the normal operating pressure, temperature and flow increase slightly are the Feedwater and the Main Steam System. Main Steam system will only experience a slight increase in flow but no increase in operating pressure and temperature.

For the Main Steam System valves, safety related MOVs include drain line isolation valves, main steam drain valves, steam header downstream drain isolation valve, startup drain valves, main steam stop valve and the pneumatically operated MSIVs. Only the MSIVs will experience flow rate change. Since closures of MSIV's are flow assisted; the increased flow will help in the closure function. The main steam stop valves are closed under no flow conditions to initiate MSIV sealing system. No other operating parameters are changed for other isolation valves. Therefore there is no impact on the MOV program for the Main Steam System.

For the Feedwater system, safety related MOVs include Feedwater inlet check valves and crosstie isolation valve. There is no differential pressure across the Feedwater inlet check valves (1AE-HV-FO32A/B) for normal and abnormal operations of the inlet check valve. Differential pressure for the crosstie isolation valve (1AE-HV-4144) has been conservatively calculated using the pump shutoff head and low water temperature to maximize the contribution from both pump head and the momentum head due to larger water density. Therefore the MOV program for the Feedwater system is also not impacted.

Based on the above evaluation it is concluded that 1.4% power uprate does not affect the capability of valves to operate under the GL 89-10 program. Additionally, check valve integrity to maintain containment isolation capability has been demonstrated for the feedwater system for the postulated break outside containment. The critical parameters for the pipe break transient are the reactor pressure and temperature. Since there are no changes to both the reactor pressure and temperature, there is no impact on the containment isolation function for the postulated break outside containment event.

6.2 EMERGENCY CORE COOLING SYSTEMS

The emergency core cooling system (ECCS) consists of the following systems:

- High pressure injection system (HPCI)
- Automatic depressurization system (ADS)
- Core spray system
- Low pressure coolant injection system (LPCI)

The ECCS systems are designed to mitigate the consequences of design basis accidents. The system functional requirements (e.g., coolant delivery rates) are such that the system performance under all LOCA conditions postulated in the design satisfies the requirements of 10 CFR 50.46. The ECCS systems are designed to provide adequate core cooling for an accident postulated to occur at 3430 MWt (104.2% power and 105% steam flow). Therefore, the existing design basis bounds the design requirements of the ECCS at 101.4% power with 0.6% uncertainty in reactor power determination.

NPSH for the ECCS pumps has been calculated in accordance with Regulatory Guide 1.1, "Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps." The suppression pool temperature analysis has been performed based on 3436 MWt (104.3% reactor thermal power), as such the ECCS operations are not affected by the 1.4% power uprate.

6.3 ECCS PERFORMANCE EVALUATION

Input parameters for the design basis LOCA analyses used 104.2% of rated thermal power as an initial condition, and the rod heatup analysis was performed at 102% of rated power. With the reduced power measurement uncertainty, the current analysis of record bounds the proposed increase in rated thermal power.

6.4 POST-LOCA COMBUSTIBLE GAS CONTROL

The post-LOCA combustible gas control system (CGCS) includes hydrogen recombiners that are utilized following a LOCA to maintain containment atmosphere in a noncombustible mixture. The power level used in the current licensing basis evaluation of hydrogen recombiner performance is 3440 MWt, which bounds the 1.4% power uprate. As a result the CGCS remains fully capable of maintaining a noncombustible environment at the uprated conditions.

6.5 FILTERATION RECIRCULATION AND VENTILATION SYSTEM

The Filtration Recirculation Ventilation System (FRVS) removes fission products from Reactor Building enclosure air following an accident resulting in the release of radioactivity in either the primary containment or the Reactor Building. The FRVS also maintains a negative pressure of approximately 0.25 inches water gauge inside the Reactor Building enclosure in order to minimize the potential for unfiltered release of fission products to the environment. Current radiological off-site dose assessment is based on greater than 102% power level. This evaluation will still be valid and bounding for the proposed power uprate. Therefore, operation of the FRVS system is not affected by the proposed 1.4% power uprate.

6.6 CONTROL ROOM AND CONTROL AREA HVAC SYSTEMS

Control room HVAC system, which includes Control Room Emergency Filter system (CREF), is designed to maintain the control room envelope at a slightly positive pressure relative to the outside atmosphere to minimize unfiltered in-leakage of contaminated outside air into the control room following a design basis LOCA. There are no plant modifications associated with the proposed 1.4% power uprate. Therefore, control room heat load is not expected to be impacted as a result of the proposed power uprate. For response to a toxic chemical spill the system is designed to go into full recirculation mode, thus will not be affected by the uprate. Current radiological assessment for control room habitability is based on greater than 102% power level. This evaluation will still be valid and bounding for the proposed uprate. Therefore, operation of the control room HVAC system is not affected by the proposed 1.4% power uprate

7 INSTRUMENTATION AND CONTROL SYSTEMS

7.1 NUCLEAR INSTRUMENTATION

The safety-related subsystems of the Neutron Monitoring System (NMS) are the Intermediate Range Monitor (IRM), and the Average Power Range Monitor (APRM) including the Local Power Range Monitors (LPRMs). The safety-related NMS instrumentation and controls are designed to monitor reactor power (neutron flux), and to trip the Reactor Protection System (RPS) when predetermined limits are reached. The NMS also provides the operator with real time information about the core power level and flux distribution during normal operation and during and following an accident.

An Oscillation Power Range Monitor (OPRM) subsystem is also provided. This system detects power oscillations that can result from thermal-hydraulic reactor core instabilities, and provides alarms that alert Control Room operators to their occurrence.

The LPRM subsystem provides localized neutron flux detection over the full power range for input to the APRM and OPRM subsystems. The current signals from the LPRM detectors are transmitted to the LPRM flux amplifiers in the main control room. The amplifier is a linear current amplifier whose voltage output is proportional to the current input and therefore proportional to the magnitude of the neutron flux.

The Average Power Range Monitor (APRM) subsystem monitors neutron flux from approximately 1 percent to above 100 percent power. Each APRM channel averages the neutron flux signals from the Local Power Range Monitors (LPRMs) assigned to it and generates a signal representing core average power. This signal is used to drive a local meter and a remote recorder located on the main control room vertical board. It is also applied to a trip unit to provide APRM downscale, inoperative and upscale alarms, and upscale reactor trip signals for use in the Reactor Protection System (RPS) and Reactor Manual Control System (RMCS).

The LPRM and APRM circuits will be rescaled such that 100% indicated power is equal to the uprated licensed Rated Thermal Power. The scaling change is accomplished by calculating the new neutron flux to detector current relationship at 100% RTP based on the uprated neutron flux. The new detector current data will be input into the LPRM flux amplifiers for calibration. Also, the Core Thermal Power calculation used to adjust the APRM amplifier gains will be revised to calculate 100% RTP based on the uprated thermal power.

An evaluation of the impact of the uprate on the accuracy of the neutron monitoring system determined that the uprate will not affect the results of the existing APRM, LPRM and rod block monitor (RBM) uncertainty evaluations.

The Intermediate Range Monitor (IRM) subsystem monitors neutron flux from the upper portion of the source range to the lower portion of the power range. The 1.4% uprate has no impact on the operation or calibration of IRM sensors or signal conditioning equipment.

7.2 SAFETY SYSTEM SETPOINTS

The power-flow map and the associated control and protection systems, which are based on the % reactor power and/or % reactor flow relationship, will maintain their current percent definitions with 100% power defined to be equal to the uprated MWt value for the purposes of UFSAR Chapter 15 Accident Analyses. Reactor protection system trip setpoints are selected to ensure reactor core and reactor coolant system safety limits are not exceeded during normal operation and design basis anticipated operational occurrences and to assist in mitigating the consequences of postulated accidents. Operating limits developed in accordance with the methodology described in CENPD-300-P-A will ensure the current reactor protection system setpoints provide sufficient margin to the safety limits with the 1.4% increase in rated thermal power.

7.2.1 Reactor Vessel Steam Dome Pressure

Reactor vessel steam dome pressure will not increase due to the power uprate and there are no requirements to change the Reactor Protection System (RPS) reactor pressure trip setpoints. Therefore, the power uprate will have no impact on the scaling or trip setpoints for the RPS reactor pressure trips.

7.2.2 Reactor Vessel Water Level

Reactor Vessel Water Level is monitored to provide operator indication at the control board, input to the Reactor Protection System (RPS), Emergency Core Cooling System (ECCS) and Reactor Vessel Level control system.

Feedwater temperature will increase by about 1°F. Core flows will remain within previously analyzed and licensed limits. The increase in feedwater temperature will decrease the density of water in the region of the reactor vessel level sensing taps and therefore, will impact the level sensed by the reactor vessel level transmitters.

The indicated level will be less than the actual level due to the temperature increase. The maximum level error due to this density change will be less than 0.15%. This error is insignificant on its impact of level indication because it is less than the readability of the main control board level indication.

When considering decreasing level trips, this error will cause the trip function to occur at a higher level (low level trip will occur sooner in a decreasing level transient) than was originally designed and therefore this is a conservative error. This error is also too small to present any significant increase in the probability of spurious level trips due to loss of operating margin between operating levels and trip setpoints.

When considering increasing level trips (Level 8 only) this error would result in a slightly non-conservative error (high level trip will occur later in an increasing level transient). There is sufficient margin between the analytical value and the trip setpoint identified in the existing uncertainty calculation to bound this error increase.

7.2.3 Redundant Reactivity Control System (RRCS)

The RRCS is a system designed to mitigate the potential consequences of an anticipated transient without scram (ATWS) event. The RRCS logic monitors reactor dome pressure and RPV water level. The logic will immediately energize the Alternate Rod Insertion (ARI) valves when either the reactor high-pressure trip setpoint or low water (L2) setpoint is reached, or manual initiation is actuated. In addition to immediate ARI initiation an immediate initiation of a recirculation pump trip (RPT) is provided by the RRCS.

The 1.4% power uprate for Hope Creek will not require an increase in steam pressure. Therefore, there will be no impact on the reactor pressure setpoint trip functions for the RRCS.

Feedwater inlet temperature will increase approximately 1°F and therefore the reactor vessel water temperature will increase and will impact the level sensed by the reactor vessel level sensors. However, as discussed in the Reactor Vessel Water Level section, the impact on low-level trip settings is conservative.

7.2.4 Automatic Removal of the Turbine Control Valve Fast Closure (TCVFC) and Main Stop Valve Closure (MSVC) Reactor Protection System (RPS) and Recirculation Pump Trip System (RPT) Trip Bypass

The RPS will trip the reactor upon sensing a turbine control valve fast closure or a main stop valve closure. This RPS trip circuitry also provides the trip logic input for the end of cycle recirculation pump trip (EOC-RPT) system to trip the recirculation pumps.

The MSVC and TCVFC RPS and EOC-RPT trips are automatically bypassed when turbine first stage pressure is less than or equal to 157.7 psig, equivalent to core thermal power less than 30% of rated thermal power. The current bypass setpoint was chosen to ensure that it is conservative relative to a steam pressure equivalent to 30% of the current rated thermal power. The setpoint is further reduced in accordance with TS requirements to account for instrument inaccuracy, calibration and drift.

The proposed increase in rated core power will result in a slightly higher turbine steam flow and turbine first stage pressure equivalent to 30% of rated thermal power. The current TS setpoints for turbine first stage pressure therefore are adequate to ensure the automatic bypass is removed when core power is greater than or equal to 30% of rated thermal power.

7.2.5 Main Steam Line High Flow Isolation

The purpose of this function is to isolate the main steam lines by providing a high steam flow signal to the Isolation Actuation Instrumentation during a steam-line break. Hope Creek Technical Specification Table 3.3.2-2 requires the high main steam line flow isolation setpoint to be ≤ to 108.7 psid. The analytical value listed in Table IV of the GE Nuclear Boiler Design Specification is 114.7 psid. Per Table IV of the GE Nuclear Boiler Design Specification this corresponds to 140% of Nuclear Boiler rated steam flow.

The power uprate will result in a small increase in Nuclear Boiler Rated steam flow. This will result in steady state operations closer to the existing isolation setpoint. Sufficient margin still exists between the maximum expected normal operational steam flow and the isolation setpoint. The existing setpoint uncertainty calculation bounds the effects of small changes in process conditions due to the proposed uprate. Therefore, there are no changes required to the current setpoint.

8 ELECTRICAL DISTRIBUTION

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. Therefore, the plant auxiliary AC/DC electrical load will not change. 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 Hope Creek Unit steam turbine-driven polyphase generator is a 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.

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

At the current thermal rating of Hope Creek Unit of 3293 MWt, the Hope Creek Unit main generator gross electrical output is 1118 MWe. Generator capability curves will be revised, ensuring that the anticipated net increase of 15 MWe will lie well within the limits of the generator. Therefore, there will be no generator limitations to prevent operation at a core power of 3339 MWt.

PSEG Nuclear has not identified any changes to equipment protection relay settings for the generator, although some process alarm setpoints for the generator and exciter may require adjustment.

To deliver electrical power provided by the generator to the transmission system, the 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.

8.3 ISOPHASE BUS

The isophase bus is designed with forced cooling rating of 32000 amperes. These ratings are greater than the Hope Creek Unit 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 required 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 equipment was designed for transmission of the full 1300 MVA generator rating. 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>AUXILIARY SYSTEMS</u>

9.1 FUEL POOL COOLING

The fuel pool cooling and cleanup system (FPCCS) is designed to remove heat that is released from the spent fuel assemblies stored in the spent fuel pool (SFP) to maintain the SFP water temperature at or below its design temperature during plant operations, to reduce activity and maintain water clarity and to maintain its cooling function during and after a seismic event.

The FPCCS is designed to maintain pool temperature at a maximum of 135°F under design load of 16.1E+6 Btu/hr. In addition system is designed to permit the Residual Heat Removal (RHR) system to be operated in parallel to remove the maximum anticipated heat load of 34.2E+6 Btu/hr to maintain the SFP at or below 150°F.

Spent fuel pool heat load calculations were reviewed and found to have adequate margin for the expected decay heat load increase that will be proportional to the power increase.

9.2 COOLING WATER SYSTEMS

Station Service Water System (SSWS) provides river water (Ultimate Heat Sink) to cool the Safety Auxiliary Cooling System (SACS) heat exchangers and the Reactor Auxiliary Cooling System (RACS) heat exchangers during normal operating conditions and loss of offsite power (LOP) conditions. The SACS heat exchangers service both the SACS and the Turbine Auxiliary Cooling System (TACS) during normal operating conditions. However during a LOP and/or a loss-of-coolant accident (LOCA) event, the TACS is isolated so that only the SACS is cooled by the SACS heat exchangers. Additionally, during a loss-of-coolant accident (LOCA) and other design basis accidents (DBAs) the RACS heat exchangers are isolated, and the SSWS provides cooling water only to the SACS heat exchangers.

Since the accident and transient analyses have been performed at 102% power or higher as discussed previously, the heat removal capability of SSWS bounds the 1.4% power uprate. During normal operation SSWS also provides cooling water to the RACS heat exchangers. RACS system is a closed loop system, which provides cooling to the non-safety related components (reactor recirculation pump seal and motor oil cooler, reactor water cleanup system pump seal cooler and nonregenerative heat exchanger, control rod drive pump seal cooler and miscellaneous condensers and coolers). Reactor operating temperature and pressure are not changed for the 1.4% uprate, and reactor coolant flow will still be within the original design limits. Therefore 1.4% power uprate does not impact any of the heat loads associated with these systems and will not impact the design and operation of the RACS.

SACS system is part of the Safety and Turbine Auxiliaries Cooling System (STACS). SACS system provides the safety related post accident decay heat removal function. The non-safety related TACS provides cooling to the turbine auxiliaries. On an accident initiation signal TACS is automatically isolated from the SACS. SACS system has been sized to mitigate the consequences of accidents or transients, which have been assumed to occur at a power level of 102% or greater. Therefore, the design basis heat removal capacity of the SACS bounds the 1.4% power uprate conditions.

The TACS provides the cooling for the turbine/generator cooling equipment, turbine building chillers, non-safety related instrument air compressors, reactor feed pump turbine lube oil, condensate pump motor bearings and various other non-safety room and equipment coolers. Design basis heat loads used were based on turbine load at valves wide open condition which is at about 105% steam flow. At 101.4% power additional heat input will still be bounded by the original design basis for the system.

Therefore, the design of the HCGS cooling water systems bounds the 1.4% uprate.

9.3 STANDBY LIQUID CONTROL SYSTEM

The function of Standby Liquid Control System is to provide the capability of bringing the reactor from full power to a cold xenon free shutdown assuming that none of the withdrawn control rods can be inserted. This function is met by the injection of a quantity of boron in the form of sodium pentaborate, which produces an equivalent concentration of at least 660 ppm of natural boron in the reactor core. A 25% additional concentration (825 ppm) is maintained to ensure the required 660 ppm is injected to the core. In addition, SLCS shutdown capability in terms of required boron concentration is evaluated for each core reload. For the 1.4% power uprate, SLCS required boron concentration is not changed.

The SLC system pumps have sufficient pressure margin, up to the system relief valve setting of approximately 1400 psig, to ensure solution injection into the reactor at all reactor operating pressures. The main steam safety/relief valves

(SRVs) begin to relieve pressure above approximately 1100 psig. The reactor operating pressure and the nominal SRV setpoints are not changed. Therefore, the SLC system positive displacement pumps will continue to function as designed.

Based on the above evaluation, the capability of the SLCS to provide its backup shutdown and anticipated transients without scram (ATWS) functions is not affected by this power uprate.

9.4 HEATING, VENTILATION AND AIR CONDITIONING SYSTEMS (HVAC)

HVAC systems that could be impacted by a power uprate at HCGS are drywell cooling, reactor building HVAC, turbine building HVAC, auxiliary building service and radwaste area ventilation. The function of the HVAC systems is to prevent extreme thermal environmental conditions from personnel and equipment by ensuring that the design temperature limits are not exceeded. In general heating and maintaining the minimum area temperatures during normal operation are not adversely affected by a power uprate. Also, ability of these systems to minimize spread of contamination is not adversely impacted since operation at uprated condition does not change the flow path or the differential pressure requirements of the HVAC systems.

Reactor vessel operating temperature and the design recirculation flow are not changed by the power uprate. There will be a small increase in feedwater system operating temperature, approximately 1°F at the outlet of the sixth heater. However the piping heat load has been calculated using a conservative feedwater temperature of 425°F based on valve wide open (VWO) heat balance. Since this temperature is higher than the expected operating temperature of 421°F at the 1.4% power uprate condition, the calculated piping heat loads bound the uprated condition. Additionally, a 20% margin was added to the calculated drywell heat loads. Therefore, drywell cooling is not impacted by the power uprate.

During normal operation reactor building HVAC with its various subsystems supplies the reactor building areas including the ECCS pump rooms, refuel floor, RWCU system equipment, steam tunnel. As a result of the power uprate only the feedwater/condensate temperature is increasing slightly. All other reactor coolant and balance of plant system parameters are not impacted by the power uprate. Design basis heat load calculations used maximum pipe temperatures when computing heat loads. The maximum increase in feedwater/condensate operating temperature has been determined to be about 1°F. Piping heat loads were calculated based on conservative piping temperature specifications, which bounds uprated operating temperatures. Post accident heat loads are expected to be less than normal operational heat loads, because non-safety systems would not be operating. Therefore, reactor building HVAC system, including the steam tunnel cooling, can accommodate the power uprate conditions without any impact to system performance.

Post accident heat loads in the ECCS pump room are not expected to change since the containment response analysis was determined based on 102% power. Therefore there will be no change in the peak suppression pool temperature, and the ECCS room cooling loads will not be affected.

Turbine building HVAC heat load calculation has included 15% margin over the calculated total heat loads. This is more than enough to offset the small heat load increase coming from the piping and other turbine/generator cooling systems. Therefore the 1.4% power uprate is not expected to have any impact effect on the turbine building HVAC system.

9.5 FIRE PROTECTION

There are no physical plant configuration or combustible load changes due to the 1.4% power uprate. As a result fire detection or suppression systems will not be impacted. Safe shutdown systems and equipment are not changing for the uprated condition. Actions taken to mitigate the consequences of a fire are also not impacted by power uprate. Therefore, power uprate has no adverse affect on the safe shutdown systems and procedures to mitigate the consequences of a fire.

9.6 CHEMISTRY AND RADIOCHEMISTRY SYSTEMS

Water chemistry programs are designed to control conductivity and chloride levels to minimize oxidizing conditions and to provide protection against corrosion. The chemistry and radiochemistry systems at HCGS consist of condensate pre-filter, condensate demineralizer, reactor water cleanup (RWCU), zinc and iron injection and hydrogen water chemistry. An evaluation was performed to identify the chemistry parameter changes, if any, which would occur in the chemistry systems due to operation at the higher power level and to assess their impact.

The condensate demineralizer and pre-filter systems have no safety-related functions. The pre-filter system is designed to remove insoluble impurities from the condensate upstream of the condensate demineralizers. The condensate demineralizer purifies condensate continuously. Evaluation indicates that condensate flow at the uprated condition is still within the design flow rate, and the temperature and pressure remain essentially unchanged. Therefore the condensate pre-filter and demineralizer systems are not impacted by the 1.4% power uprate.

The primary function of the non-safety related RWCU is to reduce solid and dissolved impurities (corrosion and fission products) within the reactor coolant system. The RWCU system has been evaluated in Section 5.9, it was determined that RWCU water chemistry is not impacted by the 1.4% power uprate.

The Zinc Injection System (GEZIP) is installed to add zinc oxide and iron oxide to minimize plant radiation fields controlled by chemistry. Since there is no change in the water chemistry, the effects of 1.4% power uprate on the zinc and iron concentration are negligible and within the uncertainty band of GEZIP. Therefore no impact on the GEZIP is expected.

The primary function of the non-safety related Hydrogen Water Chemistry System (HWC) is to reduce the potential damage due to intergranular stress corrosion (IGSCC) cracking. By controlling the reactor water oxygen concentration and the conductivity, the potential of IGSCC is reduced. The reactor water oxygen concentration is maintained less than 2 ppb and the conductivity is controlled to less than or equal to 0.2 μs/cm. Although radiolytic decomposition of the water in the core is proportional to the core thermal power, the effect of 1.4% power uprate on oxygen concentration is negligible and within the uncertainty band of the oxygen concentration. Therefore the HWC is not expected to be impacted by the proposed 1.4% power uprate.

Based on the above evaluations, it is concluded that the Chemistry and Radiochemistry systems are not impacted by the 1.4% power uprate.

10 STEAM AND POWER CONVERSION SYSTEMS

The power conversion systems at HCGS are designed to 105% of rated steam flow, which is the flow condition at valves wide open. The proposed power uprate conditions increase the rated steam and feedwater flows by about 1.8 percent, which are still bounded by the conditions at 105% steam flow determined for valves wide open. This is also applicable to the power conversion support systems, i.e. condenser air removal, steam jet air ejectors, turbine steam bypass, condensate system. Therefore, the proposed 1.4% power uprate has no impact on the power conversion systems.

11 RADIOACTIVE WASTE MANAGEMENT

11.1 LIQUID RADWASTE MANAGEMENT

The radioactive waste management systems are designed to provide for the controlled handling and treatment of liquid, gaseous, and solid wastes. The liquid radwaste management system collects, processes, monitors, and recycles or disposes radioactive liquid wastes.

The liquid radwaste system has been designed with sufficient capacity to accommodate flexible operation. The equipment drain subsystem has two 29,000 gallon tanks, and the process flow rate at 180 gpm is sufficient to treat the daily input estimated at 30,500 gal/day in less than 3 hours. The floor drain system has a design process capacity of approximately 170 gpm, this flow rate is sufficient to process the daily inputs in less than half the acceptable time duration of one hour. The system design margins are unaffected by the proposed power uprate because there are no hardware nor operation changes. The estimated release of radioactive materials in liquid effluents was calculated based on 3458 MWt (105% power) Since there is no change in the system design and operation and the radioactive effluent has been determined conservatively at 105% power, it is therefore concluded that the liquid radwaste management system is not impacted by the proposed 1.4% power uprate.

11.2 GASEOUS RADWASTE MANAGEMENT

The gaseous radwaste management systems include all systems that process potential sources of airborne releases of radioactive materials during normal operation and anticipated operational occurrences. The gaseous radwaste management systems include the offgas system and various ventilation systems. These reduce radioactive gaseous releases from the plant by filtration or delay, which allows decay of radioisotopes prior to release. The GWMS are designed to limit offsite doses from routine plant releases to significantly less than the limits specified in 10 CFR 20 and to operate within the dose objectives established in 10 CFR 50, Appendix I.

The gaseous source terms used for Hope Creek are calculated based on 3458 MWt or 105% thermal power. The design of the system and the structure housing the system meets the criteria in Regulatory Guide 1.143 and referenced in the Standard Review Plan. Since there are no changes in the system design and operation, there is no change in the seismic design, and the gaseous source terms are conservatively determined at 105% power, it is concluded that the gaseous radwaste management is not impacted by the proposed 1.4% power uprate.

Maintaining the concentration of hydrogen below the flammability limit provides assurance that the release of radioactive materials will be controlled in conformance with the requirements of General Design Criterion 60 of Appendix A to 10 CFR Part 50.

The offgas system is designed to maintain the concentration of hydrogen in the gases exhausted from the main condenser below flammable limits. Radiolytic production of hydrogen in the reactor vessel is proportional to the reactor power level; however, the increase in the radiolysis process does not alter the hydrogen-oxygen composition. Therefore the capability of the offgas system to maintain the concentration of hydrogen in the gases exhausted from the main condenser below flammable limits is not impacted by the proposed 1.4% power uprate.

12 <u>RADIOLOGICAL CONSEQUENCES</u>

12.1 PLANT EXPOSURE AND SOURCE TERMS DURING NORMAL OPERATION

The plant exposure and expected source terms subsequent to implementation of 1.4% power uprate will increase less than 1% due to conservatism existing in the current normal operation source term. The offsite doses (10 CFR 50, Appendix I) resulting from the liquid and gaseous effluent releases are not impacted by 1.4% power uprate because the uprated power is less than the core power used for the source term development. The increase in the annual occupational exposure is insignificant and is well below the estimated value. Radiation monitor setpoints are based on the various regulatory requirements, and they are independent of the core thermal power. Therefore, radiation monitor setpoints are not impacted by the proposed power uprate. The design basis source terms used for shielding design are conservative and are not impacted by 1.4% power uprate.

12.2 RADIATION ENVIRONMENTAL QUALIFICATION

The integrated dose inside the containment during normal operation is insignificantly increased due to 1.4% power uprate. Normal operational integrated doses outside the containment (Reactor Building, Turbine Building, Control Building, Service building, and Radwaste Building) are not impacted by 1.4% power uprate because they are established based on the conservative reactor coolant and main steam source terms and conservative use of measured radiation exposures at other operating plants. The post-LOCA doses are not impacted by 1.4% power uprate because the post-LOCA source terms are based on a core thermal power level, which is greater than the uprated power level.

12.3 CONSEQUENCES OF DESIGN BASIS ACCIDENTS

Control Rod Drop Accident, LOCA, and Fuel Handling Accident source terms are based on the core thermal power level of 3,458 MWt (105% of current rated thermal power); therefore, the 1.4% uprate condition is bounded by the current analyses. Instrument Line Pipe Break and Steam System Piping Beak Outside Containment source terms are based on the core thermal power level of 3,435 MWt (104% of current rated thermal power), therefore, these analyses bound the proposed power uprate condition. Gaseous Radwaste Subsystem failure source terms are based on the core power of 3,400 MWt, therefore, the current analysis bounds the proposed power uprate.

13 PLANT OPERATIONS

13.1 PROCEDURES

Plant procedures will not require significant changes for the uprate. The same steps and sequence of steps will be maintained. 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.

13.2 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.

13.3 ALARMS, CONTROLS AND DISPLAYS

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

13.3.1 Indicated Power

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

13.3.2 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.

13.4 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.

13.5 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 3293 to 3339 MWt as 100% power. An additional annunciator point will be added to alert operators to Crossflow trouble. No other changes to the simulator are required.

14 OTHER EVALUATIONS

14.1 ENVIRONMENTAL QUALIFICATION

Safety related equipment have been evaluated for the normal and accident conditions associated with the proposed 1.4% power uprate.

The current normal conditions for temperature, pressure, and humidity are unchanged for the proposed power uprate operating condition. There is no change in the High Energy Line Break (HELB) profile. The only high energy system with operating pressure and temperature change is the feedwater system. There, the change is minor, the operating pressure increases by only 1% from 1185 psia to 1188 psia and less than 0.5% for the design operating temperature from 420°F to 421°F. Additionally, the HELB profile had been generated using 1185 psia and 425°F as the initial condition for Feedwater break analyses. Thus the uprated feedwater temperature is bounded by the conservatism embedded in the analyses and the pressure change is only 0.25%. It is noted that the net change would be further reduced when the pressure loss is considered at the postulated break location. Therefore no impacts on the HELB profiles are expected for the proposed 1.4% power uprate.

Containment pressure and temperature responses during LOCA are not impacted by the proposed 1.4% power uprate, because the Hope Creek LOCA and containment response analyses were performed assuming the reactor power was at 102% or greater. Therefore, for equipment inside the containment, the pressure, temperature and humidity profiles are bounded by the existing analyses results.

14.1.1 Pressure, Temperature, Humidity

Safety related equipment was evaluated for the normal and accident conditions associated with a 1.4% uprate.

The current normal conditions for temperature, pressure, and humidity are unchanged for the proposed power uprate operating condition. There is no change in the High Energy Line Break (HELB) profile.

Containment pressure and temperature responses during LOCA is not impacted by the proposed uprate, because the Hope Creek LOCA and containment response analyses were performed assuming the reactor power was at 102% or greater. Therefore, for equipment inside the containment, the pressure, temperature and humidity profiles are bounded by the existing analyses results.

Since the temperature, pressure and humidity is not affected by the proposed uprate of 1.4%, the uprate will not affect the environmental qualification of the safety related equipment.

14.1.2 Radiation

Normal radiation inside containment (drywell) will increase approximately 2% for a 1.4% power uprate. The accident radiation inside containment (drywell) is not affected by the power uprate.

The qualified radiation value envelopes the postulated normal radiation dose (including 2% radiation increase) plus the postulated accident radiation dose with margin required per IEEE 323-1974.

Therefore, the proposed power 1.4% uprate does not have any impact on the environmental qualification of the equipment.

14.2 STATION BLACKOUT

Station blackout (SBO) is not a design basis event, but is required by 10CFR 50.63 to demonstrate that the reactor core and associated coolant control, and protection systems have sufficient capacity to cool the core and maintain containment integrity in the event of a SBO for the specified duration. HCGS has performed evaluation for SBO in accordance with NUMARC 87-00 and successfully demonstrated that the SBO procedure and plant responses satisfy the requirements of 10CFR 50.63.

The SBO plant response and coping evaluations have been reviewed for the 1.4% power uprate. There are no changes to the systems and equipment used to respond to an SBO. Also, the SBO coping duration of 4 hours is not affected. Slight increase in the decay heat may have an impact on the suppression pool and drywell temperature and CST inventory. Results of the evaluation demonstrate that there is sufficient inventory in the CST to provide RPV makeup. Increase in the drywell and suppression pool temperatures are insignificant (about 1°F). Computer simulations also demonstrate that the minimum volume maintained in the CST is adequate for providing makeup to the RPV for four hours.

Temperatures in the RCIC and HPCI rooms are not affected since the steam temperature remains the same for turbine operations. Temperatures in the other dominant areas with coping equipment and the electrical power requirements are not affected by the power uprate.

Therefore, HCGS continues to meet the requirements of 10 CFR 50.63.

14.3 HIGH ENERGY LINE BREAK ANALYSES

The initial pressure and temperature used in the original feedwater break analyses were 1185 psia and 425°F. The proposed 1.4% power uprate would change the operating conditions to about 1188 psia and 421°F. The uprated feedwater temperature is bounded by the current design input. While the initial pressure is about 0.25% higher for the uprated operating condition, the difference is insignificant. Flow losses in the piping system would further reduce the line pressure at the break location; therefore no change in the feedwater break analyses are expected for the 1.4% power uprate.

No other high-energy lines are affected by the proposed power uprate.

14.4 IMPACT ON IPE RESULTS

The main contribution to the CDF (Core Damage Frequency) is due to transients and LOP (Loss of Power)/SBO (Station Blackout) events. The most limiting scenario for these types of sequences is an SBO sequence. PSEG has evaluated the impact of reactor power increase to 102% on the SBO through simulation of various SBO related scenarios, and compared the results with similar cases at 100% power. The results showed no noticeable difference, both from the CDF determination (Level I) and the containment performance and release of fission products to the environment (Level II) points of views. In general, probababalistic safety analyses (PSAs) are not sensitive to a very small percentage difference in results, especially since many conservatisms are built into the Hope Creek PSA models, assumptions, and results interpretations.

Initiators, such as ATWS and LOCA, do not make significant contribution to the CDF, and will not have any impact on the PSA results.

Based on the above analyses, it is determined that the proposed RPV power increase from 100% to 101.4% will not affect the plant PSA.

14.5 ANTICIPATED TRANSIENTS WITHOUT SCRAM (ATWS)

An anticipated transient without scram (ATWS) is defined as an anticipated operational occurrence during which an automatic reactor scram is required, but fails to occur. Because an ATWS event would require multiple failures, it is not considered a design-basis event; ATWS events are evaluated to demonstrate compliance with 10 CFR 50.62. Hope Creek meets the requirements of 10CFR50.62 with automatic tripping of the reactor recirculation pumps (ATWS-RPT) to produce negative reactivity and with Alternate Rod Injection (ARI) and Standby Liquid Control (SLC) to shutdown the reactor in the unlikely event of an ATWS. ARI and auto tripping of the recirculation pumps are unaffected by power uprate. The SLC system is discussed in section 9.3 and is not impacted by the proposed power uprate.

For the MSIV closure without scram event, the calculated maximum vessel bottom pressure was 1425 psig, well within the ATWS design criterion of 1500 psig. There is ample margin to accommodate the 1.4% power uprate. In addition, the ATWS analysis conservatively assumed initial operation at 100% power/87% core flow and a single upper limit SRV setpoint of 1250 psig for the available 13 SRVs with one SRV out of service. At the maximum power/minimum core flow condition, effectiveness of the ATWS-RPT is less pronounced than at higher core flows. The actual SRV setpoints are 1108 psig (4 SRVs), 1120 psig (5 SRVs) and 1130 psig (5 SRVs). These conservatisms would more than offset the effect of 1.4% power uprate. Therefore it is concluded that the ability to mitigate the consequence of ATWS is not impacted by the proposed 1.4% power uprate.

15 OTHER CORE-RELATED EVALUATIONS

15.1 CORE THERMAL-HYDRAULIC STABILITY

As a result of generic concerns following several stability events at operating reactors, PSEG Nuclear installed the Oscillation Power Range Monitor (OPRM) that will automatically detect and suppress instability. This design modification is in accordance with Option III of the BWR Owners' Group recommendation (NEDO-32465-A, "BWR Owners' Group Reactor Stability Detect and Suppress Solution Licensing Bases Methodology for Reload Applications," August 1996). Until this system is completely tested, the BWR Owners' Group Interim Corrective Actions (ICAs) are being implemented. The ICAs include identifying an exclusion region on the power-flow map.

Analyses are performed to confirm the stability boundary defined in Technical Specification Figure 3.4.1.1-1 at several power/flow statepoints and burnups throughout the cycle. No explicit power measurement uncertainty are applied to any of the cases in the stability analysis. Since the objective of the analysis is to confirm the existing stability boundary, cases that correspond to the power/flow statepoints on the boundary and inside the boundary are evaluated. Enough conservatism is present in the analysis methodology (described in CENPD-295-P-A, "Thermal-Hydraulic Stability Methodology for Boiling Water Reactors," July 1996) such that power measurement uncertainty is unnecessary in the evaluation. For example:

1. Core-wide decay ratio calculations are conservatively set to a calculated decay ratio of 0.8.
2. Channel thermal-hydraulic decay ratio calculations are conservatively set to a calculated decay ratio of 0.8.
3. Out-of-phase instability-threshold power calculations are set to either:
   a) The actual threshold power for out-of-phase instabilities calculated minus an uncertainty margin that is calculated as the power required to reduce by 0.2 the core-wide decay ratio under those operating conditions, or
   b) The power at which the core-wide decay ratio is 1.0 (i.e., 20% higher than the core-wide acceptance criteria) if out-of-phase instabilities are not observed following an appropriate out-of-phase perturbation.

Based on the above, the Westinghouse BWR stability analysis methodology is unaffected by the proposed power uprate and the current analysis remains bounding. The stability boundary or inputs to the OPRM will continue to be confirmed on a cycle-by-cycle basis.

15.2 OVERPRESSURE PROTECTION

The fuel related overpressure protection analysis is performed to confirm the design bases requirement that the SRVs are sized adequately and open at a setpoint such that the ASME Code overpressure requirements (peak pressure is less than 110% of design pressure) are not violated due to changes in the reload core response. The overpressure protection analysis is the simulation of the most severe pressurization event with no credit for the scram associated with the initiating event (e.g., for the MSIV closure event, the reactor trip on MSIV position is not credited).

The AOO resulting in the largest increase in reactor pressure for HCGS is analyzed on a cycle-specific basis. Although the MSIV closure event is expected to remain limiting, other pressurization events (e.g., generator load rejection without bypass) are evaluated for each reload cycle to confirm that the limiting event is identified.

The analysis of record for HCGS, which was performed in support of Cycle 10 operation, was initiated at 102% of rated power. Further, since the original heat balance correlations (for the current rated power of 3293 MWt) remain applicable, the cross sections and subsequent analyses that were generated for the 102% power cases are valid for uprated operating conditions.

In addition, to address the fact that the high flux setpoint will be preserved at 118% of rated thermal power for the uprate, the following support the conclusion that there will be no quantifiable impact on the analysis of record:

1. The analysis predicts a very rapid increase in nuclear flux (211% rated at ~2 seconds) and the maximum vessel pressure occurs at ≈ 3 seconds.
2. There is more than 100 psi margin to the ASME 110% peak pressure design criteria in the analysis of record.

Consequently, preserving the high flux setpoint at 118% of the new rated thermal power will not produce a more severe pressurization of the vessel due to the rapid rate at which the neutron flux increases and the slower thermal response of the fuel to add energy to the coolant.

Therefore, it is concluded that the Westinghouse BWR methodology for analyzing the ASME overpressurization event remains valid for the proposed power uprate. The ASME overpressurization event will continue to be analyzed on a cycle-by-cycle basis in accordance with the requirements of CENPD-300-P-A, "Reference Safety Report for Boiling Water Reactor Reload Fuel."

15.3 SEISMIC – LOCA

The fuel assembly is classified as a Seismic Category I component. The fuel assembly is designed to withstand a Safe Shutdown Earthquake (SSE) in conjunction with structural and hydraulic loads from the worst case LOCA event. A SSE is an earthquake that is based upon an evaluation of the maximum earthquake potential considering the regional and local geology and seismology and specific characteristics of local subsurface material. The following design criteria are chosen to assure compliance with seismic-LOCA requirements for the fuel:

1. Fuel rod fragmentation will not occur as a result of combined seismic and LOCA loads,
2. Control rod insertability must not be impaired, and
3. Fuel rod coolability must be maintained.

The Seismic-LOCA analysis for SVEA-96+ fuel in HCGS was performed in support of Cycle 10 operation. The seismic loads used in the analysis of record are unchanged for the proposed power uprate and remain valid. The LOCA hydraulic loads used in the analysis of record were evaluated at rated conditions consistent with a rated power of 3293 MWt. The LOCA hydraulic loads are a small contributor to the overall fuel component stresses resulting from seismic and LOCA loads. Thus, the secondary influence of power level, along with numerous conservative modeling assumptions, justify the current analysis of record for a power uprate of ≤ 2%.

Examples of conservative modeling assumptions in the current analysis of record include:

1. The channel pressure loads include a conservative margin of 10%.
2. The assembly lift forces from the analysis of record include a margin of 5%.
3. The LOCA loads analysis assumed 106% initial core flow relative to the HCGS actual maximum flow of 105%.

The influence of initial core flow on the total loads is greater than that of initial power level. Therefore, the influence of both the conservatively calculated LOCA loads and the conservative initial core flow offset any potential impact from the proposed power uprate. No further evaluation is required for the SVEA-96+ fuel for this event.

By way of comparison with the analysis for the SVEA-96+ fuel, no reevaluation of the licensing analysis is required for the resident GE9B fuel for the uprate for the following reasons:

1. The seismic loads are unchanged for the proposed power uprate, independent of the fuel type being evaluated.
2. The hydraulic loads are an insignificant contributor to the fuel component stresses resulting from seismic and LOCA loads. This is due to the significant difference in the frequencies associated with peak hydraulic loads and the typical range of fundamental frequencies for BWR fuel. Hence, any impact due to power uprate is judged to also be insignificant.
3. The GE9B bundle is heavier than the SVEA-96+ bundle, and therefore, more resistant to lift, although the LOCA hydraulic loads are a small contributor to the total seismic and LOCA loads.

Therefore, the current analysis of record remains bounding for the proposed increase in rated thermal power and no further evaluation is required for this event for the power uprate.

15.4 ANTICIPATED TRANSIENTS WITHOUT SCRAM (ATWS)

Anticipated transients without scram (ATWS) are defined as the postulated occurrences of an anticipated transient that reaches a reactor protection system setpoint (or requires a manual scram to terminate the event) and for which there is a failure of sufficient control rods to insert to shut the reactor down. For the purpose of this set of events, anticipated transients are generally defined as those conditions of operation expected to occur one or more times during the service life of the plant. Because an ATWS event would require multiple failures, it is considered beyond the plant design bases and is analyzed to demonstrate conformance to 10CFR50.62.

The SVEA-96+ fuel has mechanical design features that result in larger margins to the fuel integrity limits (i.e., lower linear heat generation rate than for larger diameter fuel rod designs) than the GE9B fuel. Consequently, the heatup characteristics of a mixed core of SVEA-96+ and the resident fuel or a full core of SVEA-96+ fuel are less severe than the heatup response produced by a full core of the resident fuel when subjected to ATWS conditions. An ATWS evaluation that was performed in support of Cycle 10 operation showed that the SVEA-96+ fuel design and Cycle 10 mixed core met the ATWS criteria by demonstrating that the heatup characteristics for the mixed core was less limiting than that assumed in the plant licensing basis ATWS analysis. Furthermore, this conclusion was deemed to be applicable for all future cycles with the same conditions for design since additional margin to the plant licensing bases will continue to be gained as the resident fuel is discharged and a full core of SVEA-96+ is achieved.

The analysis and evaluations performed for the introduction of SVEA-96+ fuel and Cycle 10 operation have been reviewed to determine the effect of an uprate on the conclusions. The conclusions of the analysis and evaluations remain applicable for power uprates as much as 101.5%.

15.5 REACTOR SHUTDOWN WITHOUT CONTROL RODS

For the shutdown without control rods event, the standby liquid control system (SLCS) capability analysis is performed to demonstrate that the core can be made subcritical in the cold condition without insertion of the control rods.

The SLCS analysis of record for HCGS fuel was performed in support of Cycle 10 operation. The acceptance criterion for this event is that the SLCS shall be capable of shutting the reactor down from the most reactive reactor operating state at any time in cycle life. As an alternative to performing calculations throughout the cycle, a single, bounding calculation at the most reactive point in the cycle with all rods removed is performed.

The SLCS shutdown capability analysis is performed by calculating the minimum shutdown margin with no control rods in the core for core conditions from cold clean to hot zero power with 660 ppm of natural boron. Since this event is bounding at zero power conditions, no power measurement uncertainty is required in this analysis. Therefore, the reduction in core power uncertainty and the proposed power uprate do not impact this analysis.

15.6 FUEL-RELATED EMERGENCY OPERATING PROCEDURE PARAMETERS

Certain Emergency Operating Procedure (EOP) parameters depend on the post-accident fuel heat up. The fuel-related parameters include items that depend on fuel assembly geometry and items that depend on operating limits. The parameters that depend on fuel assembly geometry are not affected by the proposed uprate. The parameters that depend on operating limits include:

1. Minimum steam cooling RPV water level,
2. Minimum zero-injection RPV water level,
3. Minimum core flooding interval, and
4. Minimum steam cooling flow rate.

Items 1, 2, and 3 above are based on the fuel being operated at the limiting peak linear heat generation rate which remains bounding at the uprated conditions. The minimum steam cooling flow rate (item 4 above) is used to calculate the following three parameters:

A. The minimum number of SRVs needed to perform emergency RPV depressurization,
B. The minimum alternate flooding pressure, and
C. The RPV pressure as a function of the number of SRVs opened.

The system analysis that was used to determine item (A) was initiated at a power level that bounds the proposed power uprate. Therefore, this parameter remains bounding.

The analysis to support items (B) and (C) above used a bounding peak linear heat generation rate of 12 KW/ft. This value was shown to be bounding for Cycle 10 and was reviewed to confirm that the value will continue to be bounding for cycle 10 operation after the uprate.

The Westinghouse BWR methodology for evaluating the fuel-related EOP parameters remains valid for the proposed power uprate and will continue to be implemented on a cycle-by-cycle basis for future cycles..

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 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.

HOPE CREEK GENERATING STATION
FACILITY OPERATING LICENSE NO. NPF-57
DOCKET NO. 50-354
CHANGE TO TECHNICAL SPECIFICATIONS
INCREASED LICENSED POWER LEVEL

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATION

PSEG Nuclear LLC has determined that operation of Hope Creek Generating Station 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 3339 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 License NPF-57 is revised to authorize operation at a steady state reactor core power level not in excess of 3339 megawatts (one hundred percent of rated power).
2. The definition of RATED THERMAL POWER in Technical Specification (TS) 1.35 is revised to reflect the increase from 3293 MWt to 3339 MWt.
3. TS 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. **New Heatup and Cooldown Curves**

1. Technical Specification Figures 3.4.6.1-1, 3.4.6.1-2 and 3.4.6.1-3, pressure-temperature limit curves for hydrostatic testing, non-nuclear heatup and cooldown, and critical operation, and their associated Bases are revised to support the increase in core power based on uprated fluence projections.
2. Surveillance Requirement 4.4.6.1.4 is being revised to be made consistent with the limit on reactor vessel flange and head flange metal temperature in TS 3.4.6.1.d.

C. **Editorial Changes**

1. In TS Bases 3/4.4.6, 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. The TS Index is being revised to correctly show the locations for Figures 3.4.6.1-1, 3.4.6.1-2 and 3.4.6.1-3.

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 that could be affected by this change. Evaluation of accident analyses 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. 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. 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 evaluated 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.

C. 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.

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

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. 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.

C. 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.

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 a margin of safety.

B. 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 Cases N-640 and N-588 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 a margin of safety.

C. 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.

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.

HOPE CREEK GENERATING STATION
FACILITY OPERATING LICENSE NPF-57
DOCKET NO. 50-354
CHANGE TO FACILITY OPERATING LICENSE

FACILITY OPERATING LICENSE PAGES WITH PROPOSED CHANGES

The following section of Facility Operating License No. NPF-57 is affected by this change request:

FOL Paragraph	Page
2.C.(1)	3

(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 reactor core power levels not in excess of 3339 ~~3293~~ megawatts thermal (100 percent rated power) in accordance with the conditions specified herein.

(2) <u>Technical Specifications and Environmental Protection Plan</u>

The Technical Specifications contained in Appendix A, as revised through Amendment No. , and the Environmental Protection Plan contained in Appendix B, are hereby incorporated into the license. PSEG Nuclear LLC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) <u>Inservice Testing of Pumps and Valves (Section 3.9.6, SSER No. 4)*</u>

This License Condition was satisfied as documented in the letter from W. R. Butler (NRC) to C. A. McNeill, Jr. (PSE&G) dated December 7, 1987. Accordingly, this condition has been deleted.

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*The parenthetical notation following the title of many license conditions denotes the section of the Safety Evaluation Report and/or its supplements wherein the license condition is discussed.

HOPE CREEK GENERATING STATION
FACILITY OPERATING LICENSE NO. NPF-57
DOCKET NO. 50-354
CHANGE TO TECHNICAL SPECIFICATIONS

TECHNICAL SPECIFICATION PAGES WITH PROPOSED CHANGES

The following Technical Specifications for Facility Operating License No. NPF-57 are affected by this change request:

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insert

PROCESS CONTROL PROGRAM

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

PURGE - PURGING

1.34 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 manner that replacement air or gas is required to purify the confinement.

RATED THERMAL POWER

1.35 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of ~~3293~~ 3339 MWT.

REACTOR PROTECTION SYSTEM RESPONSE TIME

1.36 REACTOR PROTECTION SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until de-energization of the scram pilot valve solenoids. The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

REPORTABLE EVENT

1.37 A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 to 10 CFR Part 50.

ROD DENSITY

1.38 ROD DENSITY shall be the number of control rod notches inserted as a fraction of the total number of control rod notches. All rods fully inserted is equivalent to 100% ROD DENSITY.

Insert 5 here

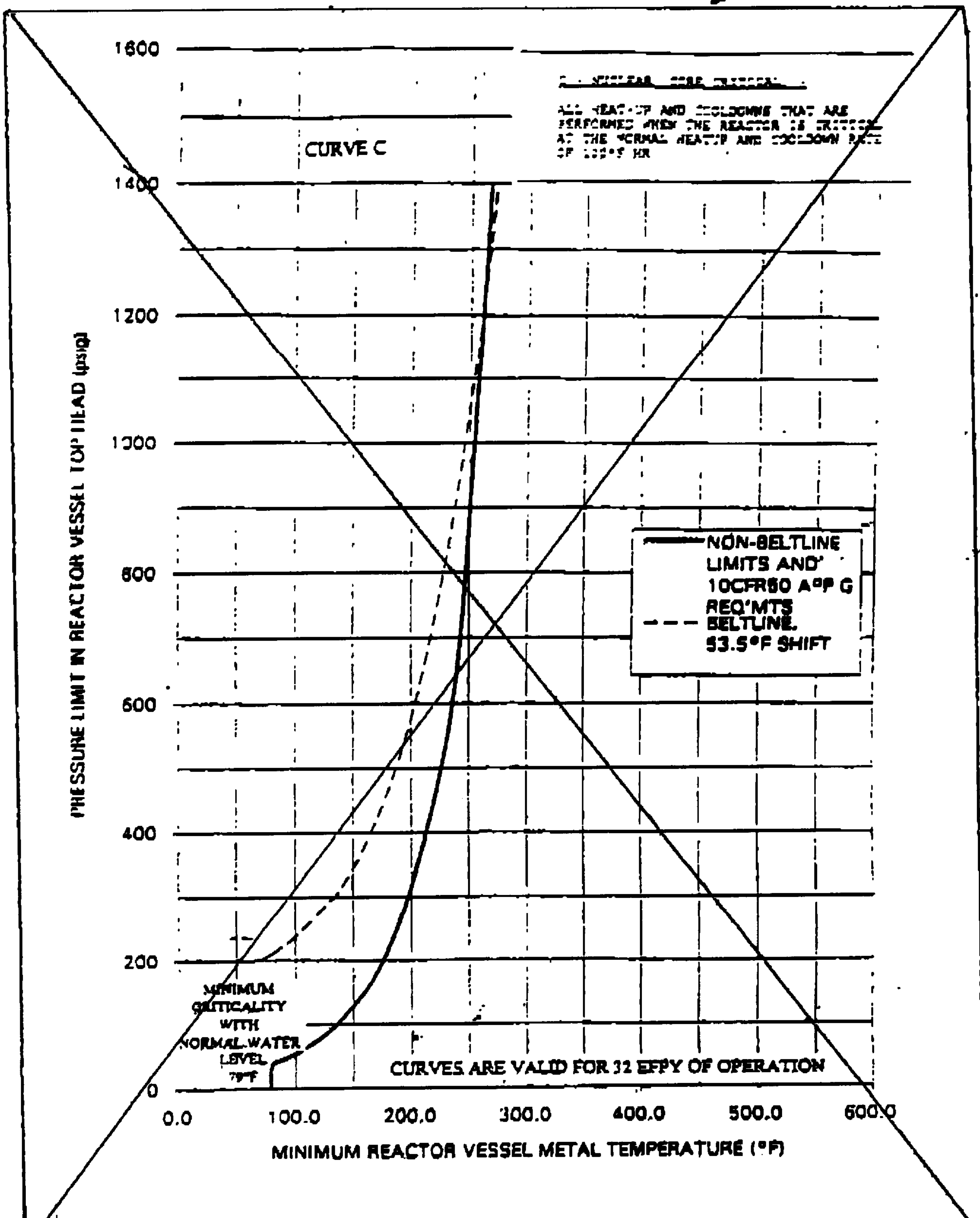
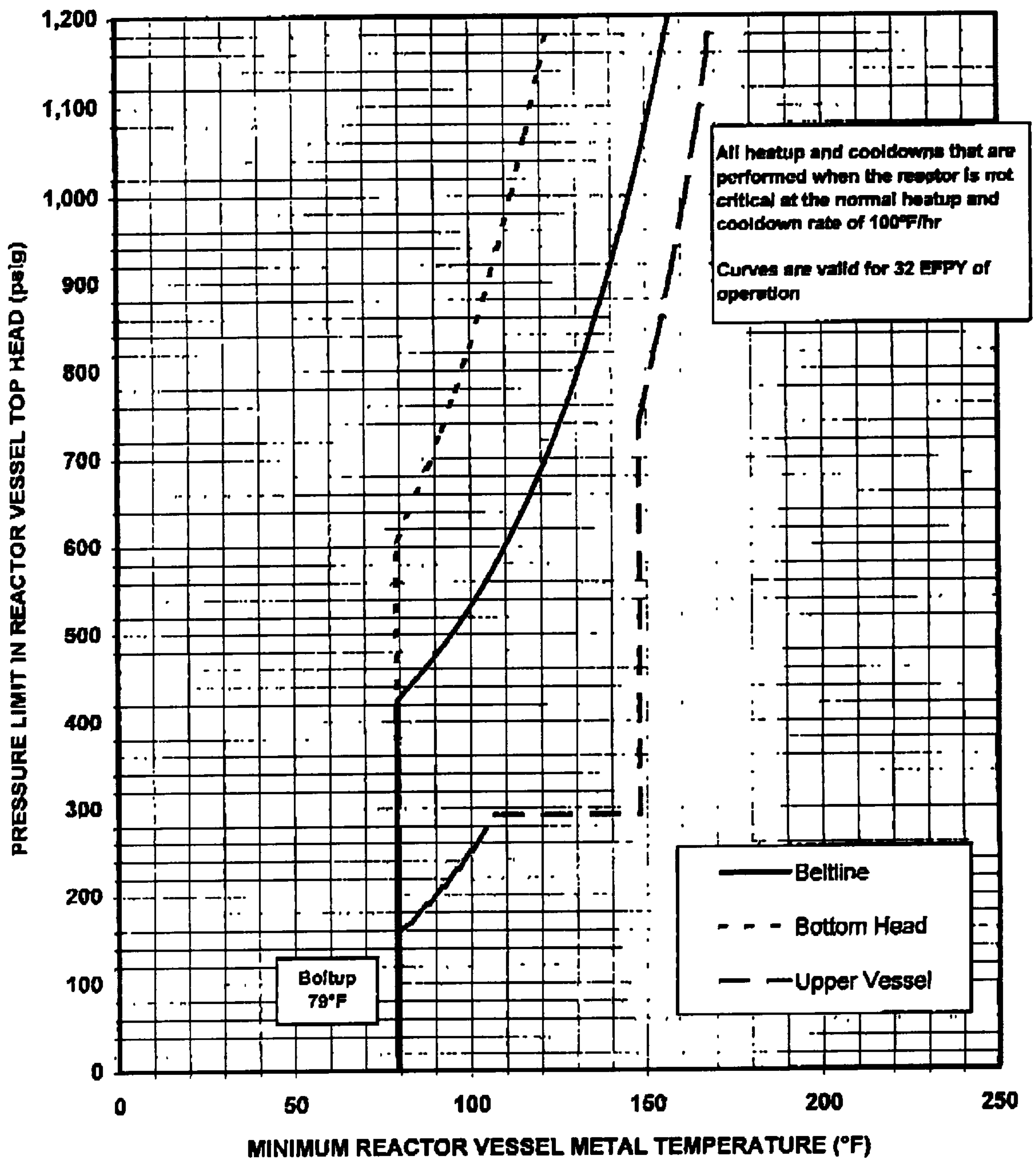


FIGURE 3.4.6.1-3

Insert 2

Figure 3.4.6.1-2

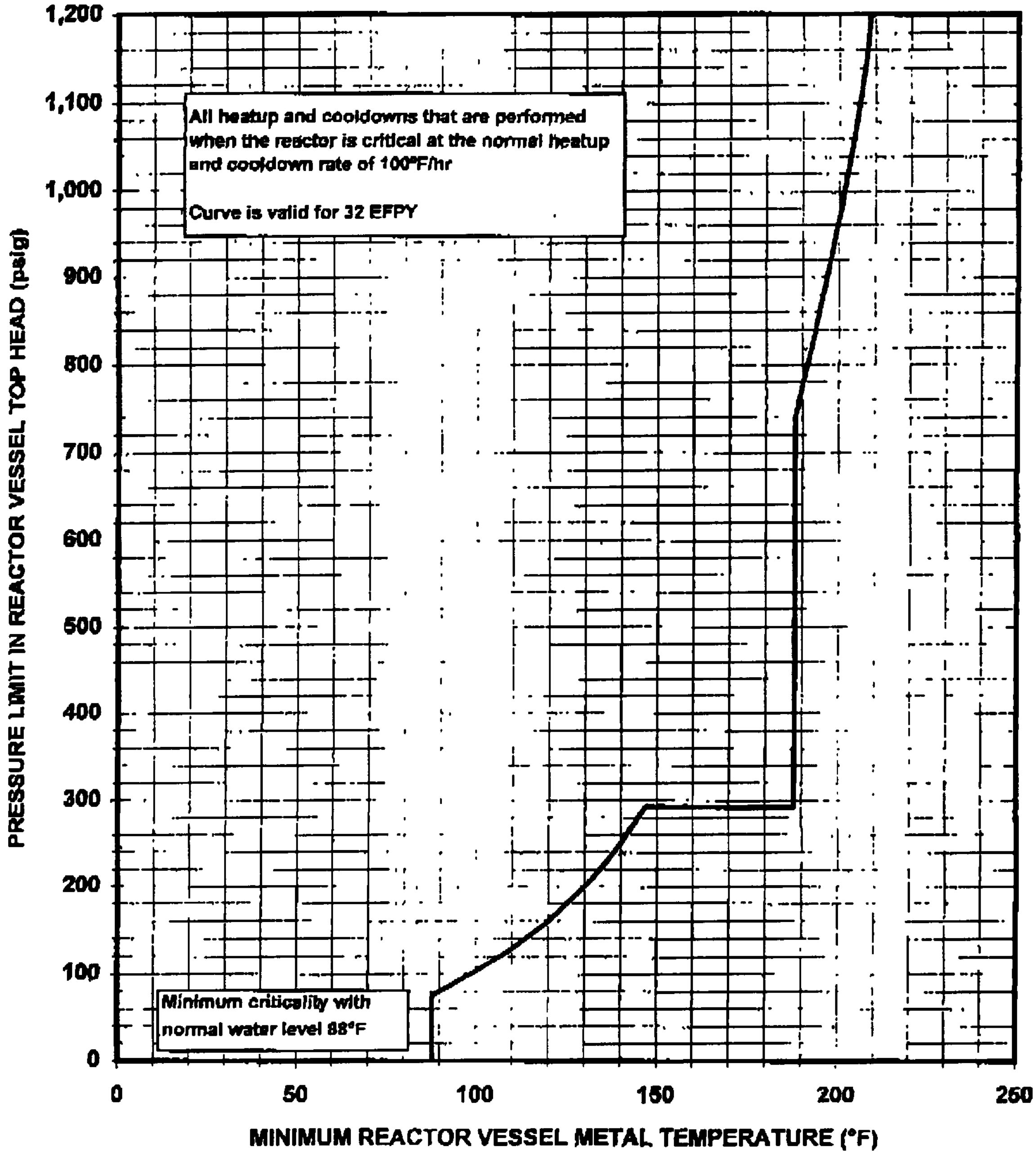
Non-Nuclear Heatup and Cooldown Pressure/Temperature Limits - Curve B



Insert 7

Figure 3.4.6.1-3

Core Critical Heatup and Cooldown Pressure/Temperature Limits - Curve C



3/4.4.6 PRESSURE/TEMPERATURE LIMITS

All components in the reactor coolant system are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in Section (3.9) of the UFSAR. During startup and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

XI — and ASME Code Cases N-588 and N-640.

The operating limit curves of Figures 3.4.6.1-1, 3.4.6.1-2, and 3.4.6.1-3 are derived from the fracture toughness requirements of 10 CFR 50 Appendix G, and ASME Code Section III Appendix G. The curves are based on the RT_{NDT} and stress intensity factor information for the reactor vessel components. Fracture toughness limits and the basis for compliance are more fully discussed in UFSAR Chapter 5, Paragraph 5.3.1.5, "Fracture Toughness."

The reactor vessel materials have been tested to determine their initial RT_{NDT}. The results of some of these tests are shown in Table B 3/4.4.6-1. Reactor operation and resultant fast neutron, E greater than 1 MeV, irradiation will cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence, nickel content and copper content of the material in question, can be predicted using Bases Figure B 3/4.4.6-1 and the recommendations of Regulatory Guide 1.99, Rev. 2, "Radiation Embrittlement of Reactor Vessel Material". The pressure/temperature limit curves, Figures 3.4.6.1-1, 3.4.6.1-2, and 3.4.6.1-3, includes an assumed shift in RT_{NDT} for the end of life fluence.

The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating, irradiated flux wires installed near the inside wall of the reactor vessel in the core area. Since the neutron spectra at the flux wires and vessel inside radius are essentially identical, the irradiated flux wires can be used with confidence in predicting reactor vessel material transition temperature shift. The operating limit curves of Figures 3.4.6.1-1, 3.4.6.1-2, and 3.4.6.1-3 shall be adjusted, as required, on the basis of the flux wire data and recommendations of Regulatory Guide 1.99, Rev. 2.

BASES TABLE B 3/4.4.6-1

REACTOR VESSEL TOUGHNESS

BELTLINE COMPONENT	WELD SEAM I.D. OR MAT'L TYPE	HEAT/SLAB OR HEAT/LOT	CU(%)	NI(%)	HIGHEST RT_{NDT}(°F)	ΔRT_{NDT}(°F)	PREDICTED EOL UPPER SHELF (FT-LBS)	MAX. EOL RT_{NDT}(°F)
Plate	SA-533 GR B CL.1*	5K3025-1	.15	0.71	+19	~~53.5~~ 53.8	67	~~72.5~~ 72.8
Weld	Vert. seams for shells 4&5	D53040/ 1125-02205	.08	0.59	-30	~~62.8~~ 63.1	120	~~33.8~~ 33.1

NOTE: * These values are given only for the benefit of calculating the end-of-life (EOL) RT_{NDT}.

NON-BELTLINE COMPONENT	MT'L TYPE OR WELD SEAM I.D.	HEAT/SLAB OR HEAT/LOT	HIGHEST REFERENCE TEMPERATURE RT_{NDT} (°F)
Shell Ring Connected to Vessel Flange	SA 533, GR.B, Cl.1	All Heats	+19
Bottom Head Dome	SA 533, GR.B, Cl.1	All Heats	+30
Bottom Head Torus	SA 533, GR.B, Cl.1	All Heats	+30
LPCI Nozzles(1)	SA 508, Cl.2	All Heats	-20
Top Head Torus	SA 533, GR.B, Cl.1	All Heats	+19
Top Head Flange	SA 508, Cl.2	All Heats	+10
Vessel Flange	SA 508, Cl.2	All Heats	+10
Feedwater Nozzle	SA 508, Cl.2	All Heats	-20
Weld Metal	All RPV Welds	All Heats	0
Closure Studs	SA 540, GR.B, 24	All Heats	Meet 45 ft-lbs & 25 mils lateral expansion at +10°F

(1) The design of the Hope Creek vessel results in these nozzles experiencing a predicted EOL fluence at 1/4T of the vessel thickness of ~~2.87~~ 2.83 x 10^{17} n/cm^2. Therefore, these nozzles are predicted to have an EOL RT_{NDT} of ~~+24~~ +24.6°F.

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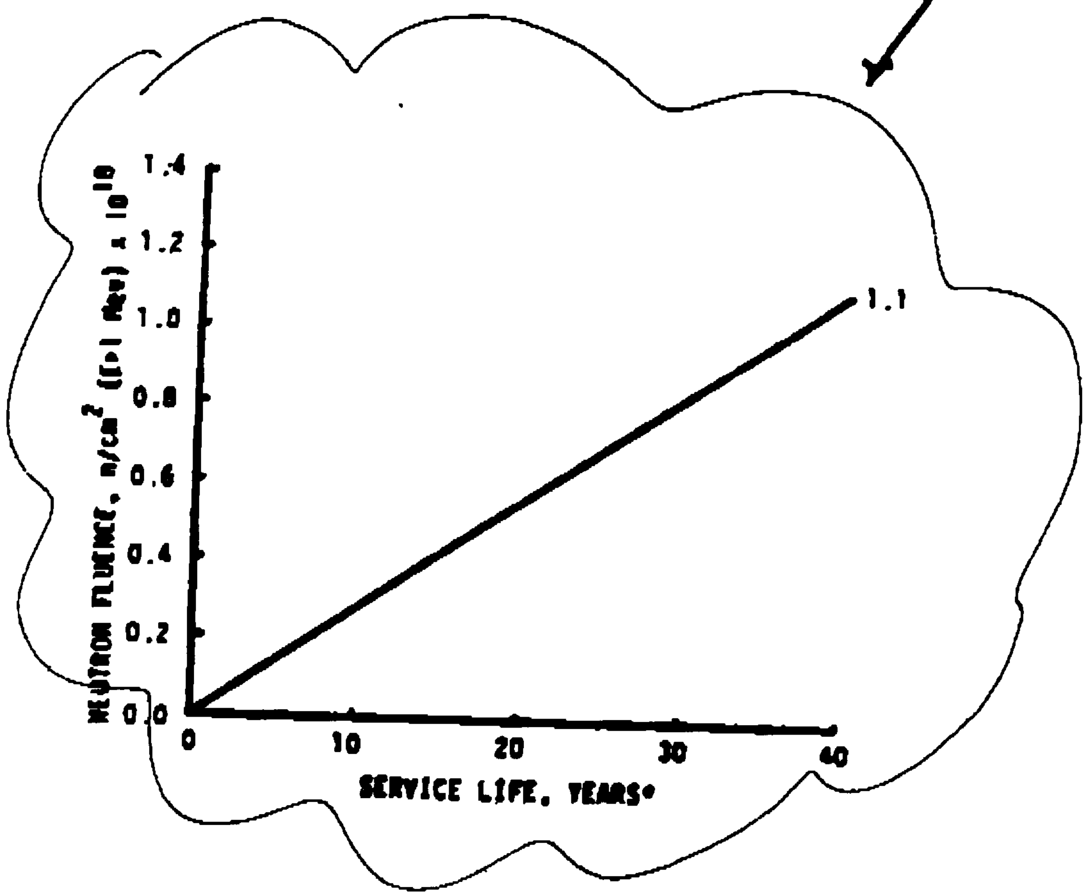


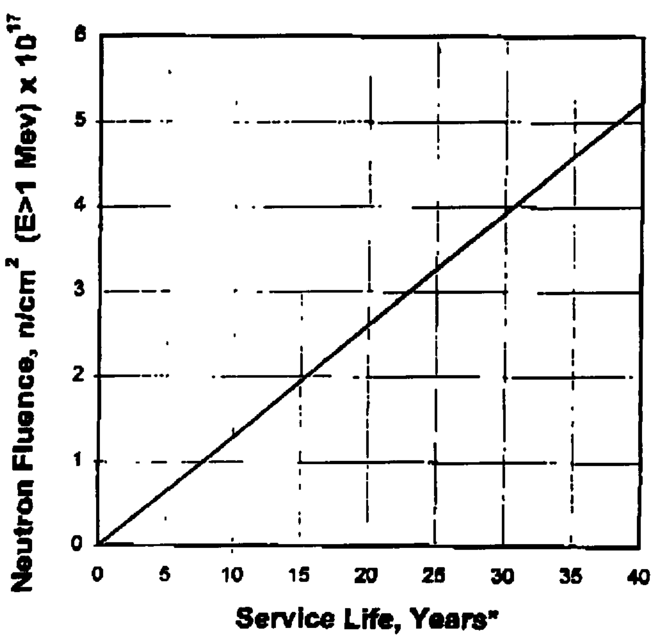
lower-intermediate shell

FIGURE B 3/4 4.6-1 FAST NEUTRON FLUENCE (E>1 Mev) AT 1/4 T AS A FUNCTION OF SERVICE LIFE*

Bases Figure B 3/4.4.6-1

~~* At 90% of RATED THERMAL POWER and 90% availability~~
At 80% capacity factor (40years = 32EFPY)

INSERT 4



Neutron Fluence, n/cm^2 (E>1 Mev) x 10^{17}
0
1
2
3
4
5
6
0
5
10
15
20
25
30
35
40
Service Life, Years*

CORE OPERATING LIMITS REPORT (Continued)

The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by NRC, as applicable in References 1 ~~and~~ 2, and 3

The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the U.S. Nuclear Regulatory Commission, Document Control Desk, Washington, DC 20555, with a copy to the USNRC Administrator, Region 1, within the time period specified for each report.

6.9.3 Violations of the requirements of the fire protection program described in the 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 USNRC Administrator, Region 1, via the Licensee Event Report System within 30 days.

6.10 RECORD RETENTION

6.10.1 In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.

SPECIAL REPORTS

6.10.2 The following records shall be retained for at least 5 years:

a. Records and logs of unit operation covering time interval at each power level.

b. Records and logs of principal maintenance activities, inspections, repair, and replacement of principal items of equipment related to nuclear safety.

c. All REPORTABLE EVENTS submitted to the Commission.

d. Records of surveillance activities, inspections, and calibrations required by these Technical Specifications.

e. Records of changes made to the procedures required by Specification 6.8.1.

f. Records of radioactive shipments.

g. Records of sealed source and fission detector leak tests and results.

REFERENCES

1. CENPD-300-P-A, "Reference Safety Report for Boiling Water Reactors Reload Fuel," (latest approved revision)

2. NEDE-24011-P-A (latest approved revision), "General Electric Standard Application for Reactor Fuel (GESTAR-II)"

3. CENPD-397-P-A, "Improved Flow Measurement Accuracy Using Cross Flow Ultrasonic Flow Measurement Technology," latest approved revision

Justification for ASME Code Case N-588 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-588, "Alternative to Reference Flaw Orientation of Appendix G for Circumferential Welds in Reactor Vessels, 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.**

   Appendix G, requires that Article G-2120 of ASME Code, Section XI, Appendix G, be used to determine the maximum postulated defects in RPVs for the vessel P/T limits. These limits are determined for normal operation and pressure/leak test conditions. Article G-2120 specifies, in part, that the postulated defect be in the surface of the vessel material and normal (perpendicular in the plane of the material) to the direction of maximum stress. ASME Code, Section XI, Appendix G, also provides methodology for determining the stress intensity factors for a maximum postulated defect normal to the maximum stress. The purpose of this article is, in part, to ensure the prevention of nonductile fractures by providing procedures to identify the most limiting postulated fractures to be considered in the development of pressure-temperature limits.

   Code Case N-588 provides benefits in terms of calculating P/T limits by revising the Article G-2120 reference flaw orientation for circumferential welds in reactor vessels. The reference flaw is a postulated flaw that accounts for the possibility of a prior existing defect that may have gone undetected during the fabrication process. Thus, the intended application of a reference flaw is to account for defects that could physically exist within the geometry of the weldment. The current ASME Code, Section XI, Appendix G approach mandates the consideration of an axial reference flaw in circumferential welds for purposes of calculating the P/T limits. Postulating the Appendix G reference flaw in a circumferential weld is physically unrealistic and overly conservative, because the length of the flaw is 1.5 times the vessel wall thickness, which is much longer than the width of circumferential welds. The possibility that an axial flaw may extend from a circumferential weld into a plate/forging or axial weld is already adequately covered by the requirement that defects be postulated in plates/forgings and axial welds. The fabrication of RPVs for nuclear power plant operation involved precise welding procedures and controls designed to optimize the resulting weld microstructure and provide the required material properties.

These controls are also designed to minimize defects that could be introduced into the weld during the fabrication process. Industry experience with the repair of weld indications found during preservice inspection, inservice nondestructive examinations, and data taken from destructive examination of actual vessel welds confirms that any remaining defects are small, laminar in nature, and do not cross transverse to the weld bead. Therefore, any postulated defects introduced during the fabrication process and not detected during subsequent nondestructive examinations would only be expected to be oriented in the direction of weld fabrication. For circumferential welds, this indicates a postulated defect with a circumferential orientation.

ASME Code Case N-588 addresses this issue by allowing consideration of maximum postulated defects oriented circumferentially in circumferential welds. ASME Code Case N-588 also provides appropriate procedures for determining the stress intensity factors for use in developing RPV P/T limits per ASME Code, Section XI, Appendix G procedures. The procedures allowed by ASME Code Case N-588 are conservative and provide a margin of safety in the development of RPV P/T operating and pressure test limits that will prevent nonductile fracture of the vessel.

The proposed P/T limits include restrictions on allowable operating conditions and equipment operability requirements to ensure operating conditions are consistent with the assumptions of the accident analysis. Specifically, RCS pressure and temperature must be maintained within the heatup and cooldown P/T limits specified in Technical Specification 3.4.6.1. Therefore, this exemption does not present an undue risk to the public health and safety.

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, Appendix G and ASME Code, Section XI, Appendix G, is to satisfy the underlying requirement that:

   1. The reactor coolant pressure boundary be operated in a regime having sufficient margin to ensure that when stressed the vessel boundary behaves in a non-brittle manner and the probability of a rapidly propagating fracture is minimized; and

2. P/T operating and test curves provide margin in consideration of uncertainties in determining the effects of irradiation on material properties.

Application of ASME Code Case N-588 when determining P/T operating and test limit curves per ASME Code, Section XI, Appendix G, provides appropriate procedures for determining limiting maximum postulated defects and considering those defects in the P/T limits. This application of the Code Case maintains the margin of safety originally contemplated when ASME Code, Section XI, Appendix G was developed. Therefore, use of ASME Code Case N-588, as described above, satisfies the underlying purpose of the ASME Code and the NRC regulations to ensure an acceptable level of safety.

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, 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 proposed revision to the P/T limits relies, in part, on the requested exemption. The revised P/T limits were developed using the K_{IC} fracture toughness curve shown on ASME Code, Section XI, Appendix A, Figure A-2200-1, in lieu of the K_{IA} fracture toughness curve of ASME Code, Section XI, Appendix G, Figure G-2210-1, as the lower bound for fracture toughness. The other margins involved with the ASME Code, Section XI, Appendix G process of determining P/T limit curves remain unchanged.

   Use of the K_{IC} curve in determining the lower bound fracture toughness in the development of P/T operating limits curve is more technically correct than the K_{IA} curve. The K_{IC} curve models the slow heatup and cooldown process of a reactor vessel. Use of this approach is justified by the initial conservatism of the K_{IA} curve when the curve was codified in 1974. This initial conservatism was necessary due to limited knowledge of reactor pressure vessel (RPV) material fracture toughness.

   Since 1974, additional knowledge about the fracture toughness of vessel materials and their fracture response to applied loads has been gained. The additional knowledge demonstrates the lower bound fracture toughness provided by the K_{IA} curve is well beyond the margin of safety required to protect against potential RPV failure. The lower bound K_{IC} fracture toughness provides an adequate margin of safety to protect against potential RPV failure and does not present an undue risk to public health and safety.

   P/T curves based upon the K_{IC} toughness limits will enhance overall plant safety by opening the P/T operating window, especially in the region of low-temperature operations. The two primary benefits occurring during the pressure test are a reduction in the duration of the pressure test and personnel safety while conducting inspections in primary containment at elevated temperatures with no decrease to the margin of safety.

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;

   ASME Code, Section XI, Appendix G, provides procedures for determining allowable loading on the RPV and is approved for that purpose by 10 CFR 50, Appendix G. Application of these procedures in the determination of P/T operating and test curves satisfies the underlying requirement that:

   1. The reactor coolant pressure boundary be operated in a regime having sufficient margin to ensure, when stressed, the vessel boundary behaves in a non-brittle manner and the probability of a rapidly propagating fracture is minimized; and
   2. P/T operating and test limit curves provide adequate margin in consideration of uncertainties in determining the effects of irradiation on material properties.

   The ASME Code, Section XI, Appendix G, procedure was conservatively developed based upon the level of knowledge existing in 1974 concerning RPV materials and the estimated effects of operation. Since 1974, the level of knowledge concerning these topics has greatly expanded. This increased knowledge permits relaxation of the ASME Code, Section XI, Appendix G, requirements via application of ASME Code Case N-640, while maintaining the underlying purpose of the ASME Code and NRC regulations to ensure an acceptable margin of safety.

HOPE CREEK HEAT BALANCE UNCERTAINTY

FORM 1
Page 2 of 2
(Page 1 contains the instructions)

,ALC NO.: SC-BB-0525 **REVISION: 0IR2**	**CALCULATION COVER SHEET**	**Page 1 of 38**

CALC. TITLE:	Hope Creek Heat Balance Uncertainty Calculation						
# SHTS (CALC):	**38**	**# ATT / # SHTS:**	**1/2**	**# IDV/50.59 SHTS:**	**5/--**	**# TOTAL SHTS:**	**45**

CHECK ONE:

☐ FINAL ☒ INTERIM (Proposed Plant Change) ☐ FINAL (Future Confirmation Req'd) ☐ VOID

SALEM OR HOPE CREEK: ☐ Q – LIST ☒ IMPORTANT TO SAFETY ☐ NON-SAFETY RELATED
HOPE CREEK ONLY: ☐Q ☐Qs ☐Qsh ☐F ☐R

☒ STATION PROCEDURES IMPACTED, IF SO CONTACT SYSTEM MANAGER
☐ CDs INCORPORATED (IF ANY): ______

DESCRIPTION OF CALCULATION REVISION (IF APPL.):

0IR0 - Initial issue. Suports DCP # 80010289. No 50.59 was performed for the calculation, the 50.59 was performed for the subject DCP.
0IR1 - Revises Main Feedwater Flow Error. Changes indicated with Rev. Bar.
0IR2 – Removes Crossflow accuracy assumption 6.1, range 100%-80%; added Attachment 1

PURPOSE:

0IR0 - Determine the Heat Balance Uncertainty Calculation for HC unit in support of the DCP.
ΠIR1 – Determine the eror with the new Main Feedwater Flow Error
R2 – Clearly document the Crossflow range

CONCLUSIONS:

0IR0 - The Heat Balance Uncertainty at 100% power, 3293 MWt, is 0.7% per Section 7/8 of the calculation.
0IR1 - The Heat Balance Uncertainty at 100% power, 3293 MWt, is 0.6% per Section 7/8 of the calculation.
0IR2 - The Heat Balance Uncertainty at 100% power, 3293 MWt, is 0.6% per Section 7/8 of the calculation.

	Printed Name / Signature	Date
ORIGINATOR/COMPANY NAME:	Luis Gonzalez/PSEG *L Gonzalez*	11/21/2000
PEER REVIEWER/COMPANY NAME:	Robert Mann/PSEG *Robert Mann*	11/21/2000
VERIFIER/COMPANY NAME:	Robert Mann/PSEG *Robert Mann*	11/21/2000
PSEG SUPERVISOR APPROVAL:	Terry McCool/PSEG *T McCool*	11-21-00

LIST OF EFFECTIVE PAGES

Page No.	Revision	Page No.	Revision	IDV	Revision
1	0IR2	21	0IR1	1	N/A
2	0IR1	22	0IR1	2	N/A
3	0IR2	23	0IR1	3	N/A
4	0IR1	24	0IR1	4	N/A
5	0IR1	25	0IR1	5	N/A
6	0IR1	26	0IR1		
7	0IR1	27	0IR1	Attachment 1	Revision
8	0IR2	28	0IR1	1	N/A
9	0IR1	29	0IR1	2	N/A
10	0IR2	30	0IR1		
11	0IR1	31	0IR1		
12	0IR1	32	0IR1		
13	0IR1	33	0IR1		
14	0IR1	34	0IR1		
15	0IR1	35	0IR1		
16	0IR1	36	0IR1		
17	0IR1	37	0IR1		
18	0IR1	38	0IR1		
19	0IR1				
20	0IR1				

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Attachments:

Attachment 1. Westinghouse Letter HC-PS-2000-0009, Rev 000, Operating Range for the Crossflow Meter (2 pages) 0IR2
Attachment IDV (4 pages)

1. PURPOSE

The purpose of this calculation is to determine the uncertainty in the heat balance calculation performed by the plant computer.

2. FUNCTIONAL DESCRIPTION/DESIGN BASIS

2.1 Functional Description

The Core Thermal Power is determined by a heat balance calculation performed in the secondary system.
The heat balance accounts for heat added and loses into the "system" as depicted in the loop diagram in Section 4.0.
The calculated power by the secondary heat balance is utilized to calibrate the Neutron Monitoring System.

2.2 Design Basis

Hope Creek current design basis, for the most part, is based on reactor power greater than or equal to 102% of the licensed reactor thermal power for the Nuclear Steam Supply System (NSSS) and Emergency Core Cooling System (ECCS) design and 105% steam flow for the Balance of Plant (BOP) design (UFSAR Chapter 1.1, 5.4, 6.2, 6.3, 10 and 15). Additionally, the plant has been licensed to operate within the 2% power uncertainty at 100% power.

The accuracy of the calculated Core Thermal Power is used to determine the plant operation power relative to the Licensed Power Limit.

The calculated heat balance uncertainties are applicable for 100% power rate of 3293MWt. Uncertainties at other power level might change.

3.0 REFERENCES

3.1 Updated Final Safety Analysis Report

3.1.1 Figure 1.1-1, UFSAR Rev. 10 September 30, 1999, Heat Balance at Rated Power
3.1.2 Table 4.4-1, UFSAR Rev. 0 April 11, 1988, Thermal Hydraulic Design Characteristics of the Reactor Core
3.1.3 Section 5.1.11, Reactor Water Cleanup System
3.1.4 Section 5.4.8, Reactor Water Cleanup System
3.1.5 Section 4.1.1, Information for the CRD System
3.1.6 Section 7.5.1.3.3, Plant Computer System
3.1.7 Section 7.7.13, Feedwater Control System
3.1.8 Section 7.7.1.6, Reactor Water Cleanup System
3.1.9 Section 10.4.7, Condensate and Feedwater

3.2 Technical Specifications

3.2.1 Section 2.1, Safety Limits
3.2.2 Section 2.2, Limiting Safety Limits Settings

3.3 Drawings

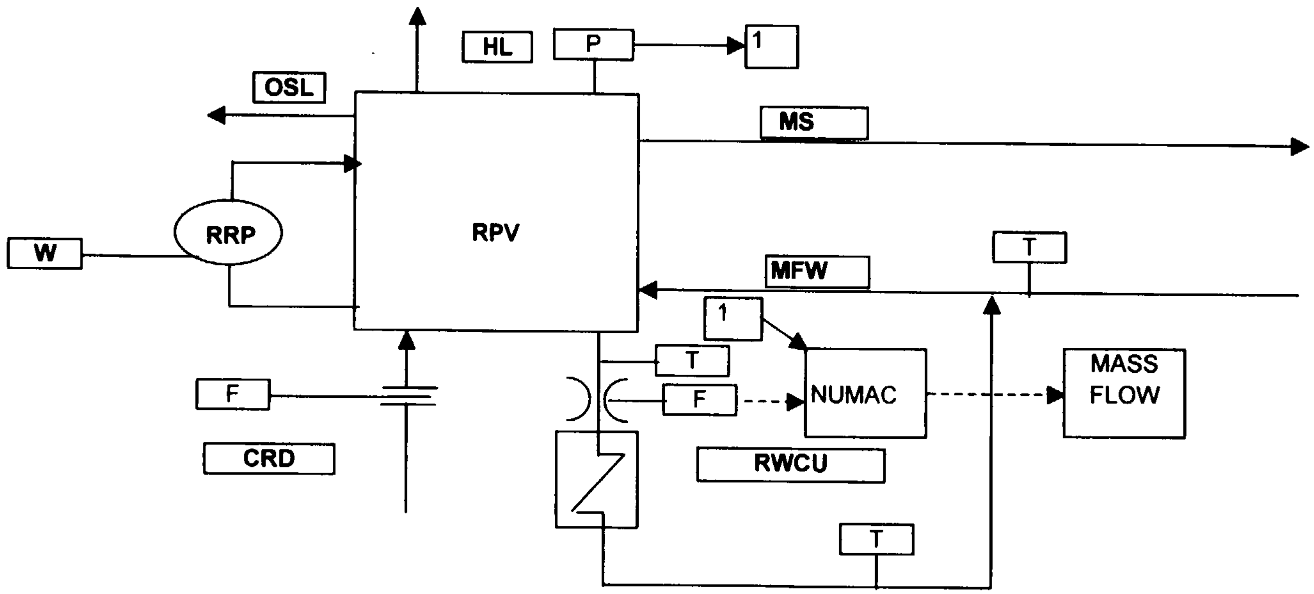
3.3.1 M-44-1, Rev. 27, Reactor Water Clean-up P&ID
3.3.2 M-41-1, Sht. 1, Rev. 29, Nuclear Boiler P&ID
3.3.3 M-42-1, Sht. 1, Rev. 30, Nuclear Boiler Vessel Instrumentation P&ID
3.3.4 M-43-1, Sht. 1, Rev. 26, Reactor Recirculation System P&ID
3.3.5 M-46-1, Rev. 21, Control Rod Drive Hydraulic Part A

3.4 Support Documents
3.4.1 H-1-RJ-ECS-0190(07), Rev. 1, Software Design Specification NSSS Process Computer Replacement Heat Balance Program
3.4.2 PN0-A41-5050-0009, Rev. 3, GE Reactor System Heat Balance Rated
3.4.3 HC.RE-RA.ZZ-0001, Core Thermal Power Evaluation Application Results 11/22/99
3.4.4 SC-BB-0355, Rev. 1, Reactor Vessel Pressure 1BBPT-N005-C32
3.4.5 SC-BF-0511, Rev. 1, Control Rod Drive 1BFFT-N004-C11
3.4.6 SC-BG-0515, Rev. 1, Reactor Water Cleanup Temperature
3.4.7 VTD 324602, Rev. 0, Calculation For Feedwater Flow Measurement Using AMAG Crossflow Meter at Hope Creek |0IR1
3.4.8 SC-AE-0541, Rev. 1, Feedwater Temperature 1AETT-N602A-D-C32
3.4.9 Fluid Meters their Theory and Application, Sixth Edition (ASME)
3.4.10 SC-BG-0516, Rev. 0, Reactor Water Cleanup System Inlet Flow 1BGFT-N03A-G33
3.4.11 SC-BB-0526, Rev. 0, Reactor Recirculation Pump Motor Power
3.4.12 ISA-RP67.04, Par II, Methodologies for the Determination of Setpoints for Nuclear Safety-Related Instrumentation, September 1994
3.4.13 Regulatory Guide 1.105, Rev. 3, Setpoints for Safety-Related Instrumentation
3..4.14 NEDC-31336, Class III, October 1986, General Electric Instrument Setpoint Methodology
3.4.15 ASME Steam Tables, Sixth Edition

3.5 Procedures

3.5.1 HC.RE-RA.ZZ-0001, Rev. 10 -Core Thermal Power Evaluation

4.0 LOOP DIAGRAM



Power = MFW(MSh-FWh)-CRDF(hin-hout)+RWCU(hout-hin)-RRP+HL+Miscellaneous

5.0 DESIGN INPUTS

5.1 Rated Power Conditions

100% rated Power Conditions are listed below. Actual 100% operation values might be used in lieu of rated nominal values, for other than Rated MWt, Feedwater Flow and Main Steam parameters as deemed appropriate. This is found acceptable, since for the purpose of error determination to rated MWt "heat differential errors", actual heat contribution deviation from measured/calculated heat to rated MWt due to instrumentation uncertainties, is required not the actual deviation value.

Rated MWt =	**3293MW**	(Ref. 3.4.1)
Rated FW flow =	**1.4260E+07lbm/hr**	(Ref. 3.4.2)
Rated FW temperature =	**420 F**	(Ref. 3.4.2)
Rated MS pressure =	**985 psia**	(Ref. 3.4.2)
Rated MS quality =	**0.999**	(Ref. 3.4.1)
Rated RWCU flow =	**145000.0 lb/hr**	(Ref. 3.4.10)
Rated RWCU temperature =	**532 F**	(Ref. 3.4.2)
Rated RWCU return temperature =	**435 F**	(Ref. 3.4.2)
Rated CRD flow =	**35900.0 lb/hr**	(Ref. 3.4.3)
CRD Calibration pressure =	**1474.0 psia**	(Ref. 3.4.5)
Rated CRD temperature =	**80 F**	(Ref. 3.4.1)
Radiation Loses =	**1.10MW**	(Ref. 3.4.2)
Other System Loses =	**0.84MW**	(Ref. 3.4.2) Note: These loses are not included in UFSAR Heat Balance, Ref. 3.1.1
MWt/BTU/hr =	**2.9300E-07**	(Ref. 3.4.1)

0IR1

6.0 ASSUMPTIONS

0IR2

See body of calculation for specific assumptions.

7.0 CALCULATIONS

7.1 Methodology

The methodology used to combine the uncertainties for the different contributors to the heat balance calculation is the square root of the sum of the squares of those uncertainties which are statistically independent. Then algebraically combined with the those errors that are systematic, or bias. The uncertainties are considered to be random, two sided distributions. This methodology has been utilized before, and has been endorsed by the NRC and various industry standards (Ref. 3.4.12-3.4.14).

The uncertainty calculation combines the different errors from the different parameters contributing to the heat balance in accordance to the heat balance equation. The contributing parameters errors are taken from the specific system uncertainty calculation.

These errors are generally classified in two groups:

a) Instrument loop(s) uncertainty
b) Process effects

The individual uncertainties affecting the heat balance calculation are determined by the application of the corresponding process algorithm, as described below:

a) the appropriate algorithm for the specific process parameter is set in the form to its specific contribution to the heat balance at specified rated conditions,
b) subsequently, the instrumentation loop error is factored in the process algorithm to calculate the corresponding process parameter with the error built-in,
c) the difference of the calculated contributed process parameter heat, with error, to the rated process rated heat is then calculated,
d) the resultant heat error contribution is then divided by the rated 100% power thermal megawatts. The result is the error contribution by the specific process to the total heat balance uncertainty,
e) finally, all the calculated heat errors are combined in accordance to their parameter function in the heat balance equation. The resultant combination of all contributing errors is the Heat Balance Uncertainty.

7.2 Uncertainties Calculation

7.2.1 Main Feedwater Uncertainty(ies)

Referring to the schematic drawing in section 4.0, it can be seen that the Main Feedwater heat contribution is affected by the following parameters:

a) Mass Flow Measurement affected by: 1) flow, 2) temperature, and 3) pressure instrumentation loops error.

The error provided by the vendor for the ultrasonic flow meter already factors the corresponding temperature and pressure loops effect.

b) Feedwater Enthalpy determination affected by: 1) temperature, and 2) pressure, instrumentation loops error.

The main feedwater enthalpy is calculated using the following signals:

- Main Steam Pressure and Main Feedwater Temperature

7.2.1.1 Main Feedwater Mass Flow Heat Error due to Flow Element Uncertainty (FWm)

The main feedwater mass error is provided as percentage of reading from 100% down to 80% rated flow, documented in:

Ref.: 3.4.7, Attachment 1 |0IR2

The heat error is the difference in the MFW heat content at rated flow conditions minus the heat error content at rated conditions plus flow error:

FWm = [hrated (Flowrated - Flowrated+err)] x 2.93E-07/3293 x100%

F Rtd	F+err	P Rtd	TRtd	h rated
14,260,000	14,311,336	985	420	397.53

Error % reading **0.36%** (for a FW temp error of 2F) |0IR1

TTL MFW Heat
5,689,235,953 BTU/hr |0IR1

5,668,828,171 BTU/hr at rated conditions

Error
20,407,781 **BTU/hr at rated main feedwater flow** |0IR1

Uncertainty in Rated MWT
FWm = 0.1816% |0IR1

7.2.1.2 Main Feedwater Heat Enthalpy Error due to Pressure Loop Uncertainty (**FWhp**)

The main feedwater enthalpy is determined from reactor loop pressure, that is affected by the loop uncertainty.
The pressure loop uncertainty is documented in:

Ref.: 3.4.4

The heat error is the difference in the MFW heat content at rated flow and enthalpy conditions minus the heat content at rated flow with enthalpy at rated pressure plus pressure induced error:

FWhp = [Flow (hrated - hrated+err)] x 2.93E-07 / 3293 x 100%

P Rtd	P+err	TRtd	h rated	h+err	Flow
985	1005.5	420	397.53	397.55	14,260,000
Error (psi) =	**20.5**				

TTL MFW Heat
5,669,114,129 BTU/hr

5,668,828,171 BTU/hr at rated conditions

Error
285,958 BTU/hr at rated main feedwater flow

Uncertainty in Rated MWt
FWhp = 0.0025%

7.2.1.3 Main Feedwater Heat Enthalpy Error due to Temperature Loop Uncertainty **(FWht)**

The main feedwater enthalpy determination is affected by the loop temperature error, documented in:

Ref.: 3.4.8

The heat error is the difference in the MFW heat content at rated flow and enthalpy conditions minus the heat content at rated flow conditions with enthalpy at rated temperature plus temperature induced error:

FWm = [Flow (hrated - hrated+err)] x 2.93E-07 / 3293 x 100%

TRtd	T+err	P Rtd	h rated	h+err	Flow
420	422	985	397.53	399.71	14,260,000
Error (F) =	2				

TTL MFW Heat
5,699,852,808 BTU/hr

5,668,828,171 BTU/hr at rated conditions

Error
31,024,637 BTU/hr error at rated main feedwater flow

Uncertainty in Rated MWt
FWht = 0.2760%

7.2.2 Main Steam Flow Uncertainty(ies)

Referring to the schematic drawing in section 4.0, It can be seen that the Main Steam heat contribution is affected by the following parameters:

a) The calculated Main Steam Heat is affected by the mass flow measurement error (see section 7.2.1.1 Main Feedwater Mass Error).

b) Main Steam Enthalpy determination is affected by the pressure instrumentation loop error, documented in calculation:

Ref. 3.4.4

7.2.2.1 Main Steam Mass Flow Heat Error due to Main Feedwater Flow Uncertainty (**MSm**)

The main feedwater mass error is provided from Section 7.2.1.1:

The heat error is the difference in the Main Steam flow heat content at rated flow conditions minus the heat content at rated conditions plus flow error:

MSm = [hrated (Flowrated - Flowrated+err)] x 2.93E-07 / 3293 x 100%

FWFEerr = 0.36% Span Mass Flow |0IR1

F Rtd	**F+err**	**P Rtd**	**Moist Rtd**	**hs rated**
14,260,000	14,311,336	985	0.999	1192.83

TTL MFW Heat
17,070,980,169 BTU/hr |0IR1

17,009,745,087 BTU/hr at rated conditions

Error
61,235,082 BTU/hr at rated main feedwater flow |0IR1

Uncertainty in Rated MWT
MSm = 0.5448% |0IR1

7.2.2.2 Main Steam Heat Enthalpy Error due to Pressure Loop Uncertainty (**MShp**)

This Section calculates the Main Heat Steam enthalpy error due to the loop pressure uncertainty, documented in:

Ref.: 3.4.4

The heat error is the difference in the Main Steam flow heat content at rated flow and enthalpy conditions minus the heat content at rated flow with enthalpy at rated pressure plus pressure induced error:

MShp = [Flow (hrated - hrated+err)] x 2.93E-07 / 3293 x 100%

P Rtd	**P+err**	**Moist Rtd**	**Moist+err**	**h rated**	**h+err**	**Flow**
985	1005.5	0.999	0.999	1192.83	1192.08	14,260,000
Error (psi) =	20.5		0			

TTL MS Heat
16,999,119,561 BTU/hr

17,009,745,087 BTU/hr at rated conditions

Error
10,625,526 BTU/hr error at rated main feedwater flow

Error in Rated MWt
MShp = 0.0945%

7.2.2.3. Main Steam Moisture Heat Enthalpy Error due to Steam Moisture Uncertainty (**MSmoist**)

This section calculates the Main Steam enthalpy error due to MS moisture uncertainty. The uncertainty is conservatively set to 50% of rated moisture content of 0.1%.

The heat error is the difference in the Main Steam flow heat content at rated flow and moisture conditions minus the heat content at rated moisture conditions plus moisture error:

MSmoist = [Flowrated hrated (hmoist-rated - hmoist-rated+err)] x 2.93E-07 / 3293 x 100%

P Rtd	P+err	Moist Rtd	Moist+err	h rated	h+err	Flow
985	985	0.999	0.9995	1192.83	1193.16	14,260,000
	0	0.05%	5			

TTL MS Heat
17,014,402,763 BTU/hr

17,009,745,087 BTU/hr at rated conditions

Error
4,657,676 BTU/hr error at rated main feedwater flow

Error in Rated MWt
MSmoist = 0.0414%

7.2.3 Control Rod Drive Flow

Referring to the schematic drawing in section 4.0, it can be seen that CRD flow is not monitored for pressure or temperature.

The CRD calculated heat is affected by the following effects:

a) Mass Flow Measurement affected by: 1) flow, instrumentation loop effect and 2) temperature, and 3) pressure deviation from calibration values

b) The CRD Enthalpy determination is affected by: 1) temperature, from applied constants, and 2) pressure, from reactor pressure loop error

The CRD flow loop errors are documented in calculation:

Ref. 3.4.5

The flow formula is derived from the ASME (Ref. 3.4.9) as follows:

Flow = C*Fa*K*(DP*r)^0.5

where K is calculated below:

K = Calib Flow/(Calib inWC * Calib ρ)^0.5

7.2.3.1 Control Rod Drive Flow Heat Error due to Flow Element and Fluid Specific Weight Uncertainty due to Temperature Error (**CRDt**)

The uncertainty is dependent for the following variables:

Fa = FE Thermal Expansion

ρ = Fluid Specific Weight

The plant computer calculates the CRD flow with a constant flow K factor. However, the actual temperature could vary as much as 40F from the expected 80F, and this impacts the Fa and ρ impacting the calculated flow; therefore, the effect due to this temperature deviation is:

CRDt = [(hs rated - CRDh rated) (Flowrated - Flowrated+err)] x 2.93E-07 / 3293 x 100%

Ref.: 3.4.5

K = 448.38
Calib Flow = 50131 lb/hr   100 GPM
Calib inWC = **200**
Calib Temp = **80**
Calib Press = **1474.7**
ρ = 62.501
Assumed Rated Flow = **35900.0 lb/hr**
Rated Press = 985 psia
Rated Temp = **80 F**
Rated h = 50.72 btu/lb

Fa Error

Rated 80F	1.0003
@140 F	1.0013
@40 F	0.9995
Fa/F	1.8E-05

C Rtd	C+err	DP Rtd	DP+err	Fa Rtd	Fa+err	T Rtd	T+err
1.0000	1.0000	102.5061	102.5061	1.0003	1.0010	80	120
0.00%	0	0.00%	0				40

Temp error = Calib Temp - Min Temp

ρ	h Rtd	Flow	hs Rtd
61.8964	50.72	35,751.731	1192.83

TTL CRD Heat
40,832,495 BTU/hr

41,001,835 BTU/hr at rated conditions

Error
(169,340) BTU/hr error at rated CRD flow

Error in Rated MWt
CRDt = -0.0015%

This error is a bias, not a random instrument induced uncertainty.

SURVEILLANCE REQUIREMENTS (Continued)

4.4.6.1.2 The reactor coolant system temperature and pressure shall be determined to be to the right of the criticality limit line of Figure 3.4.6.1-3 within 15 minutes prior to the withdrawal of control rods to bring the reactor to criticality and at least once per 30 minutes during system heatup.

4.4.6.1.3 The reactor vessel material surveillance specimens shall be removed and examined, to determine changes in reactor pressure vessel material properties, as required by 10 CFR 50, Appendix H. The results of these examinations shall be used to update the curves of Figures 3.4.6.1-1, 3.4.6.1-2, and 3.4.6.1-3.

4.4.6.1.4 The reactor vessel flange and head flange temperature shall be verified to be greater than or equal to ~~70°F~~: the limits specified in 3.4.6.1.2.

a. In OPERATIONAL CONDITION 4 when reactor coolant system temperature is:

    1. ≤ 110 ~~100°F~~ at least once per 12 hours.

    2. ≤ 90 ~~80°F~~ at least once per 30 minutes.

b. Within 30 minutes prior to and at least once per 30 minutes during tensioning of the reactor vessel head bolting studs.

Insert 1 here

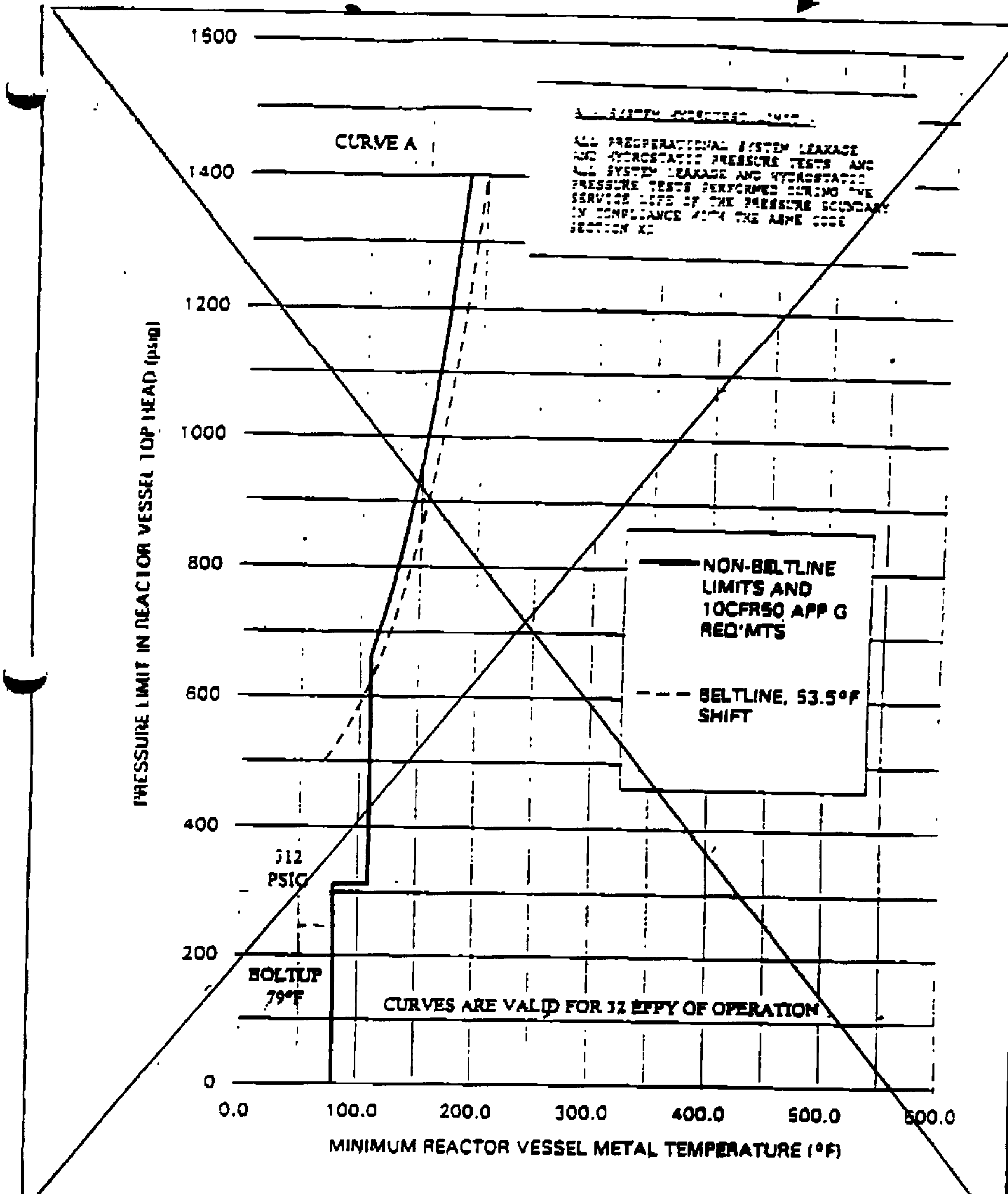
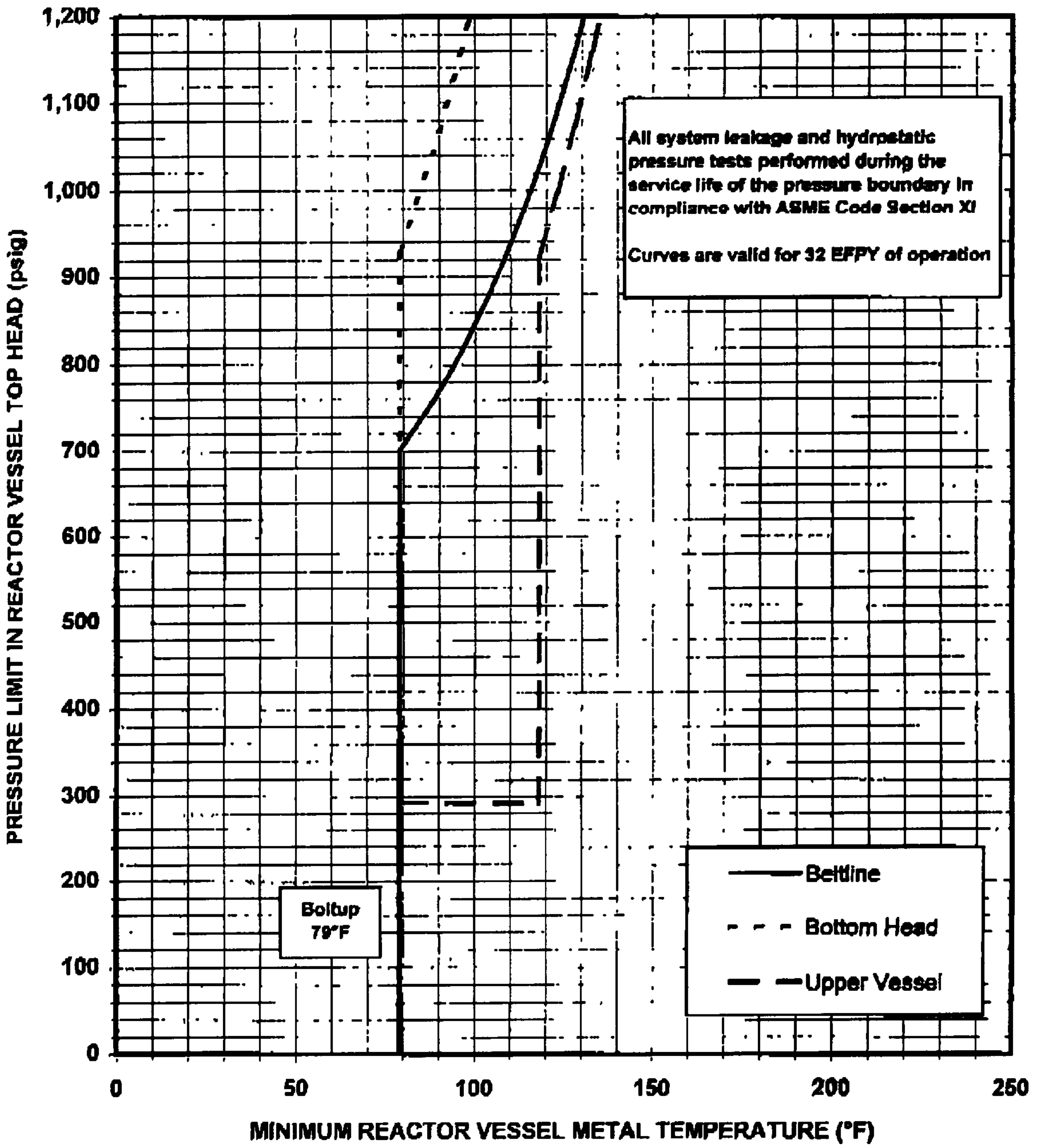


FIGURE 3.4.6.1-1

Insert 1

Figure 3.4.6.1-1
Hydrostatic Pressure and Leak Tests Pressure/Temperature Limits - Curve A



Insert 2 here

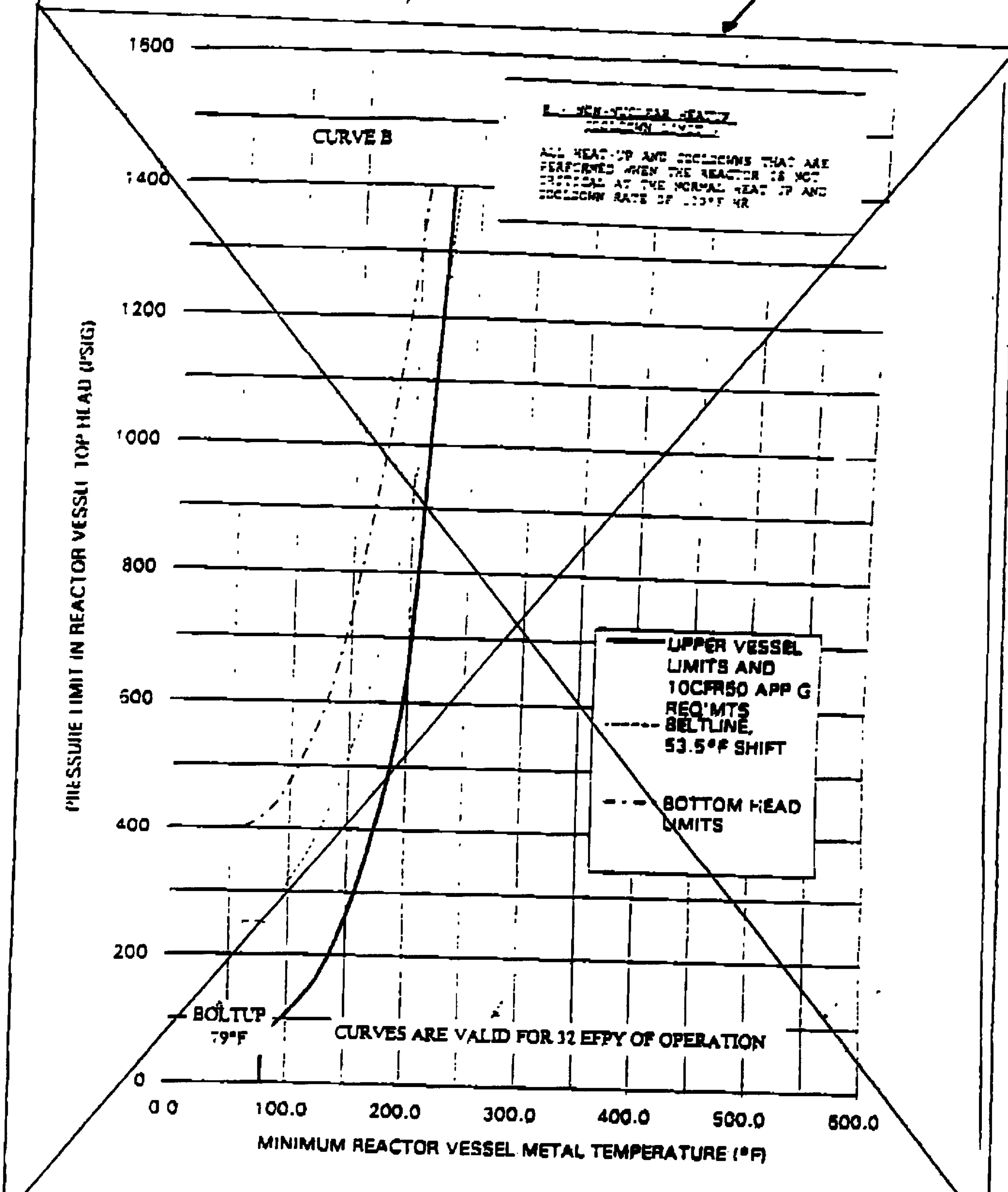


FIGURE 3.4.6.1-2

7.2.3.2 Control Rod Drive Flow Heat Error due to Flow Element Uncertainty (**CRDc**)

The CRD contributed heat is affected by the FE error that is assigned to the flow element expansion coefficient.

C

The FE uncertainty is determined based in calculation:

Ref.: 3.4.5

The heat error is the difference in the CRD flow heat content at rated flow conditions minus the heat error content at rated conditions plus error:

CRDc = [(hs rated - CRDh rated) (Flowrated - Flowrated+err)] x 2.93E-07 / 3293 x 100%

Fa Error	
Rated 80F	1.0003
@140 F	1.0013
@60 F	0.9995
Fa/F	1.8E-05

K = 448.38
Calib Flow = 50131 100.00
Calib inWC = 200.00
Assumed Calib Temp = 80.00
Assumed Calib Press = 1474.00
ρ = 62.501 0.0159998
Assumed Rated Flow = 35900.0 lb/hr
Rated Press = 985 psia
Rated Temp = 80 F
Rated h = 50.72 btu/lb

C Rtd	C+err	DP Rtd	DP+err	Fa Rtd	Fa+err	T Rtd	T+err
1.0000	1.0200	102.5061	102.5061	1.0003	1.0003	80	80
2.00%	200	0.00%	0				0

P Rtd	ρ	h Rtd	Flow	hs Rtd
985.00	62.4082	50.72	36,590.904	1192.83

TTL MFW Heat
41,790,925 BTU/hr

41,001,835 BTU/hr at rated conditions

Error
789,090 BTU/hr error at rated CRD flow

Error in Rated MWt
CRDc = 0.0070%

7.2.3.3 Control Rod Drive Heat Error due to Differential Pressure (DP) Loop Uncertainty (**CRDdp**)

This error is calculated in calculation:

Ref.: 3.4.5

The loop is comprised of 1) flow transmitter, FT, 2) resistor, REST, 3) signal conditioning instrument, SC, 4) computer analog to digital card, A/D. The uncertainties are in % DP span:

Accuracy: **Loop Drift:**

CRD_AFT	CRD_AREST	CRD_ASC	CRD_AA/D	CRD_VDFT	CRD_VDSC	CRD_VDA/D
1.154%	0.100%	0.375%	0.078%	1.450%	0.195%	0.078%

Loop Calibration:

CRD_CEFT	CRD_CESC	CRD_CEA/D
0.140%	Mann/10/17/2000 0.1[illegible]%	0.125%

The uncertainties are random and independent and combined by the SRSS method:

1.93% span DP

The heat error is the difference in the CRD flow heat content at rated flow conditions minus the heat error content at rated conditions plus error:

CRDdp = [(hs rated - CRDh rated) (Flowrated - Flowrated+err)] x 2.93E-07 / 3293 x 100%

Fa Error

Rated 80F	1.0003
@140 F	1.0013
@60 F	0.9995
Fa/F	1.8E-05

K = 448.41
Calib Flow = 50134.2 lb/hr 100 GPM
Calib inWC = 200
Assumed Calib Temp = 80
Assumed Calib Press = 1474
ρ = 62.501
Assumed Rated Flow = 35900.0 lb/hr
Rated Press = 985.0 lb/hr
Rated Temp = 80 F
Rated h = 50.72 btu/lb

C Rtd	C+err	DP Rtd	DP+err	Fa Rtd	Fa+err	T Rtd	T+err
1.0000	1.0000	102.4923	106.3436	1.0003	1.0003	80	80
0.00%	0	1.93%					0

P Rtd	ρ	h	Flow	hs Rtd
985	62.4082	50.72	36,541.272	1192.83

TTL MFW Heat
41,734,239 BTU/hr

41,001,835 BTU/hr at rated conditions

Error
732,404 BTU/hr at rated CRD flow

Error in Rated MWt
CRDdp = 0.0065%

7.2.3.4 Control Rod Drive Flow Enthalpy Heat Error due to Temperature Calibration Deviation **(CRDht)**

The plant computer calculates the CRD fluid enthalpy at a constant 80F and reactor pressure. However, the actual temperature varies and it is assumed to vary as much as 40F from the expected 80F; therefore, the effect due to this temperature deviation in the calculated enthalpy is:

Ref. 3.4.5

The heat error is the difference in the CRD flow heat content at rated flow conditions minus the heat error content at rated conditions plus error:

CRDht = [Flow (hs rated - CRDh rated) - Flow (hs rated - CRDh rated+err)] x 2.93E-07 / 3293 x 100%

TRtd	T Max/Min	T+err	P Rtd	h rated	h+err	hs rated	Flow
80	40	40	985	50.72	10.94	1192.83	35,900

TTL RWCU Heat	
42,429,926	BTU/hr

41,001,835 BTU/hr at rated conditions

Error
(1,428,091) BTU/hr at rated CRD flow

Uncertainty in Rated MWt	
CRDht = -0.0127%	This error is a bias, not a random instrument induced uncertainty.

7.2.3.5 Control Rod Drive Flow Enthalpy Heat Error due to Pressure Loop Uncertainty **(CRDhp)**

The plant computer calculates the CRD fluid enthalpy at a constant 80Fand reactor loop pressure, that is affected by the loop uncertainty. The pressure loop uncertainty is documented in:

Ref.: 3.4.4

The heat error is the difference in the CRD flow heat content at rated flow conditions minus the heat error content at rated conditions plus error:

CRDhp = [Flow (hs rated - CRDh rated) - Flow (hs rated+err - CRDh rated)] x 2.93E-07 / 3293 x 100%

P Rtd	P+err	TRtd	h rated	hs rated	h+err	Flow
985	1005.5	80	50.72	1192.83	1192.08	35,900
Error (psi) =	20.5					

TTL MFW Heat	
40,975,085	BTU/hr

41,001,835 BTU/hr at rated conditions

Error
(26,750) BTU/hr at rated CRD flow

Uncertainty in Rated MWt
CRDhp = -0.00024%

7.2.4.0 Reactor Water Cleanup Uncertainty(ies)

Referring to the schematic drawing in section 4.0, it can be seen that RWCU flow is measured and mass calculated by a "NUMAC" process unit. Differential pressure, temperature, by dedicated thermocouples, and reactor pressure are an input to the instrument. Therefore, the RWCU contributed heat is affected by the following parameters:

a) Mass Flow Measurement affected by: 1) flow, 2) temperature, and 3) pressure induced factors/instrumentation loops errors

b) The RWCU Enthalpy determination is affected by: 1) temperature instrumentation loop error (see discussion for pressure)

The flow formula is derived from the ASME (Ref. 3.4.9) as follows:

Flow = C*K*(DP)^0.5

where K is:

K = Calib Flow / [C*(Calib inWC)^0.5]

7.2.4.1 RWCU Inlet/Outlet Flow Heat Error due to Flow Element Expansion deviation from Calibration (**RWCUFa**)

The Numac computer calculating the mass flow has a built-in Fa constant different than the flow element Fa provided at the calculated venturi rated temperature of 533F. This induces a bias error. Furthermore, the plant NUMAC normalizes the flow mass signal to a specific weight of 47.0 lbm/cuft for pressure and temperature conditions, back calculated below; therefore, the rated conditions are set at 532F.

Pressure Density Compensated to :	47.10 lb/cuft	532 F	**908 psia**
The Fa used in the calculation is fixed to:		1.0045	
The correct Fa at rated 533 F is:		1.0087	

Based on the Fa differences the induced flow error is calculated below:

Fa = FE Thermal Expansion

Ref.: 3.4.10

Flow+err = C x Fa+err / Fa x K x (DP)^0.5

K = 9433.10
Calib Flow = 189485.6 lb/hr **500 GPM** (Note)
Calib inWC = **403.5**
Assumed Calib Temp = **532**
Assumed Calib Press = **1114.7**
ρ = 47.248
Assumed Rated Flow = 145762.6 lb/hr **383 GPM**
Rated h = 526.44 btu/lb

Fa Error

Rated 533F	**1.0087**
Calibrated	**1.0045**

C Rtd	C+err	DP Rtd	DP+err	Fa Rtd	Fa+err	T Rtd	T+err
1.0000	1.0000	238.7719	238.7719	1.0087	1.0045	532	532
0.00%	0	0.00%	0				0

P Rtd	P+err	ρ	h Rtd	Flow+err
1114.7	1114.7	47.2482	526.44	145,156
	0			

Uncertainty
RWCUFa -606.9 lb/hr

This a bias and the actual contributed Heat is Higher than indicated

Note: This value is slightly different to the 189,197 lb/hr calculated in the reference due to rounding

7.2.4.2 RWCU Inlet/Outlet Flow Heat Error due to Fluid Specific Weight deviation (**RWCUPMA**)

This error is the combination of several uncertainties calculated below:

7.2.4.2.1 RWCU Inlet/Outlet Flow Error due to Fluid Specific Weight Numac Lookup Tables Error (**RWCUPMA1**)

The Numac performs the fluid specific weight determination with an error of 0.1 specific weight, this effect in the flow is calculated below.

ρ = Fluid Specific Weight

Ref.: 3.4.10

Flow+err = C x K x (DP x ρ+err / ρ)^0.5

The flow error is the difference between the rated flow at rated specific weight minus the flow at specific weight plus error:

RWCUPMA1 = Flowrated - Flowrated+error

K = 9433.10
Calib Flow = 189485.6 lb/hr 500 GPM (Note)
Calib inWC = 403.5
Assumed Calib Temp = 532
Assumed Calib Press = 1114.7
ρ = 47.248
Assumed Rated Flow = 145762.6 lb/hr 383.0 GPM
Rated hout = 526.44 btu/lb

Fa Error

Rated 533F	1.0087
Calibrated	1.0045

C Rtd	C+err	DP Rtd	DP+err	Fa Rtd	Fa+err	T Rtd
1.0000	1.0000	238.7719	238.7719	1.0087	1.0087	532
0.00%	0	0.00%	0			

P Rtd	P+err	ρ	Numac ρerr	ρ+ρerr+Numac	h Rtd	Flow
1114.7	1114.7 psi	47.248	0.1	47.348	526.44	145,917
	0 psi					

Uncertainty
RWCUPMA1 = 154.2 lb/hr

Note: This value is slightly different to the 189,197 lb/hr calculated in the reference due to rounding

7.2.4.2.2 RWCU Inlet/Outlet Flow Error due to Fluid Specific Weight Numac 0.75 Factor Pressure Correction Factor Error (**RWCUPMA2**)

The Numac computer introduces a 0.75 factor to the input pressure raw value and calculates the specific weight for saturated conditions, which bias the actual flow.

Ref.: 3.4.10

Flow+err = C x K x (DP x ρ+err / ρ)^0.5

The flow error is the difference between the rated flow at rated specific weight minus the flow at specific weight plus error:

RWCUPMA2 = Flowrated - Flowrated+error

ρ = Fluid Specific Weight

K = 9433.10
Calib Flow = 189485.6 lb/hr 500 GPM (Note)
Calib inWC = 403.5
Assumed Calib Temp = 532
Assumed Calib Press = 1114.7
ρ = 47.248
Assumed Rated Flow = 145762.6 lb/hr 383.0 GPM
Rated = 526.44 btu/lb

Fa Error	
Rated 533F	1.0087
Calibrated	1.0045

C Rtd	C+err	DP Rtd	DP+err	Fa Rtd	Fa+err	T Rtd
1.0000	1.0000	238.7719	238.7719	1.0087	1.0087	532
0.00%	0	0.00%	0			

P Rtd	P+err	T Sat	ρ+perr	Numac ρerr	ρ+perr+Numac	h Rtdout	h Rtd return
1114.7	840 psi	523.8	47.5895	0	47.5895	526.44	413.93
	75%	75					

Uncertainty
RWCUPMA2 = -525 lb/hr

This a bias and the actual contributed Heat is Higher than indicated

Note: This value is slightly different to the 189,197 lb/hr calculated in the reference due to rounding

7.2.4.2.3 RWCU Inlet/Outlet Flow Heat Error due to Fluid Specific Weight Pressure Loop Uncertainty **(RWCUP_SW)**

Pressure is utilized as an input to the Numac to determine the specific weight of the fluid. The pressure loop uncertainty combined with the Numac computer uncertainty introduces an error in the calculated specific weight.

The loop error is comprised of 1) pressure loop, PT, 2) NUMAC computer uncertainties. The combined uncertainties are:

WCUpress error=	**21 psi**

Ref.: 3.4.10

Flow+err = C x K x (DP x ρ+err / ρ)^0.5

The flow error is the difference between the rated flow at rated specific weight minus the flow at specific weight plus error:

RWCUp_SW = [Flowrated (h_in - h_out) - Flowrated+error (h_in - h_out)] x 2.93E-07 / 3293 x 100%

ρ = Fluid Specific Weight

K = 9433.10
Calib Flow = 189485.6 lb/hr 500 GPM (Note)
Calib inWC = 403.5
Assumed Calib Temp = 532
Assumed Calib Press = 1114.7
ρ = 47.248
Assumed Rated Flow = 145762.6 lb/hr 383.0 GPM
Rated h = 526.44 btu/lb

Fa Error

Rated 533F	1.0087
Calibrated	1.0045

C Rtd	C+err	DP Rtd	DP+err	Fa Rtd	Fa+err	T Rtd in	T Rtd out
1.0000	1.0000	238.7719	238.7719	1.0087	1.0087	532	435
0.00%	0	0.00%	0				

P Rtd	P+err	ρ+perr	Numac ρerr	ρ+perr+Numac	h Rtd in	h Rtd out	Flow
985	1006 psi	47.1731	0	47.1731	526.63	413.93	145,647

TTL MFW Heat
16,414,258 BTU/hr

16,427,316 BTU/hr at rated conditions

Error
13,058

Uncertainty in Rated MWt
RWCUp_sw = 0.0001%

Note: This value is slightly different to the 189,197 lb/hr calculated in the reference due to rounding

7.2.4.3 Reactor Water Cleanup Flow Error due to Flow Element Uncertainty (**RWCUc**)

The RWCU flow is affected by the FE error that is assigned to the flow element expansion coefficient.

C

The FE uncertainty is determined based in calculation:

Ref.: 3.4.10

Flow+err = C+err x K x (DP)^0.5

The flow error is the difference between the rated flow minus the flow with the C coefficient error:

RWCUc = Flowrated - Flowrated+error

Fa Error
Rated 533F 1.0087
Calibrated 1.0045

K = 9433.10
Calib Flow = 189485.6 LB/HR 500 GPM (Note)
Calib inWC = 403.50
Assumed Calib Temp = 532.00
Assumed Calib Press = 1114.70
ρ = 47.25
Assumed Rated Flow = 145762.6 lb/hr 383 GPM
Rated h = 526.4 btu/lb

C Rtd	C+err	DP Rtd	DP+err	Fa Rtd	Fa+err	T Rtd	T+err
1.0000	1.0150	238.7719	238.7719	1.0087	1.0087	532	532
1.50%	150	0.00%	0				0

P Rtd	ρ	h	Flow
1114.7	47.2482	526.4	147,949

Uncertainty
RWCUc = 2186.4 lb/hr

Note: This value is slightly different to the 189,197 lb/hr calculated in the reference due to rounding

7.2.4.4 Reactor Water Cleanup Flow Error due to Differential Pressure (DP) Loop Uncertainty (**RWCUdp**)

This error is introduced by the differential pressure instrument loop. The uncertainties in the loop are found in: Ref.: 3.4.10

The loop is comprised of 1) flow transmitter, FT, 2) NUMAC computer. The uncertainties are in % DP span:

The flow error is the difference between the rated flow minus the flow with the C coefficient error:

RWCUdp = Flowrated - Flowrated+error

Accuracy: / **Loop Drift:**

RWCU_AFT	RWCU_ANU_IE	RWCU_ANU_A/D	RWCU_VDFT	RWCU_VDNU_IE	RWCU_VDNU_A/D
0.503%	0.100%	0.233%	0.900%	0.127%	0.127%

Loop Calibration:

RWCU_CEFT	RWCU_CENU_A/D
0.139%	0.02%

The uncertainties are random and independent and combined by the SRSS method:

1.09%	span DP

Flow+err = C x K x (DP+err)^0.5

Fa Error

Rated 533F	1.0087
Calibrated	1.0045

K = 9433.10
Calib Flow = 189485.6 LB/HR 500 GPM (Note)
Calib inWC = 403.50
Assumed Calib Temp = 532.00
Assumed Calib Press = 1114.70
ρ = 47.248
Assumed Rated Flow = 145762.6 LB/HR 383 GPM
Rated h = 526.44

C Rtd	C+err	DP Rtd	DP+err	Fa Rtd	Fa+err	T Rtd	T+err
1.0000	1.0000	238.7719	243.1537	1.0087	1.0087	532	532
0.00%	0						0

P Rtd	ρ	h	Flow
1114.7	47.2482	526.44	147,094

Uncertainty
RWCUdp = 1331.4 lb/hr

Note: This value is slightly different to the 189,197 lb/hr calculated in the reference due to rounding

7.2.4.5 Reactor Water Cleanup Flow Error due to Signal Conditioning/NSSS Computer Loop Uncertainty **(RWCUNSSS_cptr)**

The differential pressure is converted to flow by the Numac computer and retransmitted as flow signal to the plant computer for heat balance calculations.

This portion of the loop is comprised of 1) NUMAC computer output, 2) NSSS computer uncertainty. The uncertainties are in % flow span:

Accuracy:

RWCU_ANU_D/A	RWCU_ASC	RWCU_AA/D
0.233%	0.375%	0.078%

Loop Drift:

RWCU_VDNU_D/A	RWCU_VDSC	RWCU_VNA/D
0.13%	0.195%	0.078%

Loop Calibration:

RWCU_CENU_D/A	RWCU_CESC	RWCU_CEA/D
0.02%	0.195%	0.125%

The uncertainties are random and independent and combined by the SRSS method:

0.56%	span FLOW

The flow error is the difference between the rated flow at without loop error minus the flow plus loop error:

RWCUNSSS_cptr = Flowrated - Flowrated+error

Fa Error	
Rated 533F	1.0087
Calibrated	1.0045

K = 9433.10
Calib Flow = 189485.6 lb/hr 500 GPM (Note)
Calib inWC = 403.50
Assumed Calib Temp = 532.00
Assumed Calib Press = 1114.70
ρ = 47.25
Assumed Rated Flow = 145762.6 lb/hr 383 GPM
Rated h = 526.44

C Rtd	C+err	DP Rtd	DP+err	Fa Rtd	Fa+err	T Rtd	T+err
1.0000	1.0000	238.7719	238.7719	1.0087	1.0087	532	532
0.00%	0						0

P Rtd	ρ	h	Flow
1114.7	47.2482	526.44	146,826.7

Uncertainty
RWCUNSSS_cptr= 1064.1 lb/hr

Note: This value is slightly different to the 189,197 lb/hr calculated in the reference due to rounding

7.2.4.6 Reactor Water Cleanup Total Flow due to Flow Loop Uncertainties (**RWCUf**)

RWCU Inlet h - Outlet h.

The total RWCU Flow Uncertainty is calculated below:

RWCUfu =+/-SQRT(RWCUPMA1^2+RWCUc^2+RWCUdp^2+RWCUNSSS_cptr^2)-RWCUFa-RWCUPMA2

Notice that the bias PMAs are factored with both signs, since the negative uncertainty is the one that has impact in the heat balance calculations, that is, that (-) less indication means that power is higher; however, it will be factored in both directions for simplicity.

RWCUfu = 3909 lb/hr

Then, the heat error contribution is the calculated inlet flow heat error minus the outlet flow heat error:

RWCUh_in = 526.6 btu/lb
RWCUh_out = 413.9 btu/lb

And the RWCU heat error contribution is calculated by the following expression:

RWCU = [RWCUfu (RWCUh_in - RWCUh_out) x MWt_BTU_hr (conversion factor)] / Rated MWt 100%

RWCUf = 0.0039% This total error will be treated as bias in the total heat balance error

7.2.4.7 Reactor Water Cleanup Flow Enthalpy Heat Error due to Temperature Uncertainty (RWCUht)

This error is the error resultant of temperature measurement errors by the inlet/outlet RWCU thermocouple loops factored into the fluid enthalpy determination. The thermocouples are instruments with good repeatability and stability. However, fabrication errors can amount to several farenheit degrees. The fabrication errors can be calibrated out, since essentially it is a bias, systematic error. The temperature loops for RWCU are not calibrated out. Finally, since most of the thermocouples loop error is a bias, the conservative and safe way to apply the error, is for the two temperature error to be set in the same direction; therefore, the effect is additive, bias. The temperature loops error is found in:

Ref. 3.4.6

The heat error is the difference between the inlet flow heat error minus the outlet flow heat error:

RWCUht = [Flow [(hin - hin+error) + (hout - hout+error)]] x 2.93E-07 / 3293 x 100%

In flow heat

TRtd	T+err	P Rtd	h rated	h+err	Flow
532	539	985	526.63	535.32	145,763
Error (F)	6.9				

Out flow heat

TRtd	T+err	P Rtd	h rated	h+err	Flow
435	442	985	413.93	421.54	145,763
Error (F)	6.9				

Error

2,376,227 **BTU/hr error at rated reactor water cleanup flow**

Uncertainty in Rated MWt
RWCUht = 0.0211%

This error is a bias

7.2.4.8 Reactor Water Cleanup Flow Enthalpy Heat Error due to Pressure Loop Uncertainty **(RWCUhp)**

The reactor water cleanup enthalpy determination is affected by the loop pressure error. The pressure has a very small effect in the water enthalpy, however, it is determined in this calculation. The loop uncertainty is documented in:

Ref.: 3.4.4

The heat error is the difference in the CRD flow heat content at rated flow conditions minus the heat error content at rated conditions plus error:

RWCUhp = [Flow [(hin - hin+error) - (hout - hout+error)]] x 2.93E-07 / 3293 x 100%

P Rtd	P+err	TRtd	h rated	h+err	Flow
985	1005.5	532	526.63	526.60	145,000
Error (psi)	20.5				

P Rtd	P+err	TRtd	h rated	h+err	Flow
985	1005.5	435	413.93	413.95	145,000
Error (psi)	20.5				

Error

6,797 BTU/hr at rated reactor water cleanup flow

Uncertainty in Rated MWt
RWCUhp = 0.00006%

7.2.5 Recirculation Pumps Heat Error due to Watts Loop Uncertainty (RRPw)

This section calculates the uncertainty due to RRP watts loop error, calculated in:

Ref.: 3.4.11

The rated power for the pump-motor and the motor efficiency (0.93) is from reference:

(Ref. 3.4.3)

There are two recirculation pumps, the calculated error below is per pump. The actual total MW for the 2 pumps is taken from above reference.

W 2 pump = 7.33

The Watt error contribution is calculated as follows:

RRPw = [(W Rtd/pump + MW Loop Span x Span err) x Mottor eff] - W Rtd/pump x Motor eff] / 3293 x 100%

W Rtd/pmp+err	Motor Eff	W+err
3.838	0.93	3.57
	Error =	**1.65 % span**
	Span =	**10.5 Mwatt**

TTL Mwatt	
3.41	Mwatt

Error
0.1611 Mwatt

Uncertainty in Rated MWT
RRPw = 0.0049%

7.2.6 Thermal Loses (TL)

This section calculates the uncertainty due to RL error. This error will be treated as a bias error since the estimated value is larger or smaller than assumed, i.e. it is a fixed error not a random error.

An assumed error equal to 20% of the specified loses is used.

The rated Heat Loss is from reference:

(Ref. 3.4.2)

W Rtd	W+err
1.1	1.32
Error =	20.00%

TTL Mwatt Heat	
1.32	MW

1.10 MW at rated conditions

Error

0.2200 MW error at rated radiated loses

Uncertainty in Rated MWT	
TL = 0.007%	This error is treated as bias.

Note: The computer utilizes this value combined with Other System Loses (Section 7.2.7) as Radiative power loses, QRAD = 1.94

7.2.7 Other System Loses (OSL)

The rated value includes Rod Drive seal purge flow to recirculation pumps. An assumed error equal to 20% of the specified loses is used. This error will be treated as a bias error since if the estimated value is larger or smaller than assumed, i.e. it is a fixed error not a random error.

The rated Losses figure is from reference:

(Ref. 3.4.2)

W Rtd	W+err
0.84	1.008
Error =	20%

TTL Mwatt Heat	
1.01	MW

0.84 MW at rated conditions

Error

0.1680 MW error at rated radiated loses

Uncertainty in Rated MWT	
OSL = 0.005%	This error is treated as bias.

Note: The computer utilizes this value combined with Thermal Loses (Section 7.2.6) as Radiative power loses, QRAD = 1.94

7.2.8 Heat Balance Calculation Power Uncertainty (**Power U**)

The calculated heat balance uncertainty is the algebraic combination of the bias errors with the independent/random errors statistically combined, that contribute into the heat balance calculation, in accordance with the heat balance formula:

Power = MFW(MSh-FWh)-CRDF(hin-hout)+RWCU(hout-hin)-RRP+HL+Miscellaneous

Summary of Heat Balance Calculation Contributing Errors:

Random Errors:

FWm = 0.1816% |0IR1
FWht = 0.2760%
MSm = 0.5448% |0IR1
MSmoist = 0.0414%
CRDc = 0.0070%
CRDdp = 0.0065%
RRPw = 0.005%
RWCUp_sw = 0.0001%

Dependent Errors:

Errors	Variable
FWhp = 0.0025%	Rated MS pressure = 985 psia
MShp = 0.0945%	
CRDhp = -0.00024%	
RWCUhp = 0.00006%	

Bias Errors:

CRDt = -0.0015%
CRDht = -0.0127%
RWCUf = 0.0039%
RWCUht = 0.0211%
TL = 0.007%
OSL = 0.005%

Heat Balance Calculation Power Error (U):

Power U = SQRT[(MSm-FWm)^2+(MShp-FWhp+CRDhp+RWCUhp)^2+FWht^2+MSmoist^2+CRDc^2+CRDdp^2+
+RWCUp_sw^2+RRPw^2x2]+CRDt+CRDht+RWCUf+RWCUht+TL+OSL+Margin

To ensure instrumentation operation margin, a margin is added:

Margin = 0.10% |0IR1

Power U = 0.59% |0IR1

8.0 SUMMARY

The calculated Heat Balance Error performed in this calculation is applicable to the hand calculation error when determined process values and steam tables heat values are used with an accuracy of 3 decimal places. This results are applicable to hand calculated heat balance since less hardware errors are involved in the hand calculation, data collection. The Heat Balance calculation error (Section 7.0) is:

Power U = 0.6% |0IR1

TO: JIM METRO
856 339 3586

Second Send

SC-BB-0525
Att. 1
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Westinghouse Electric Company
CE Nuclear Power LLC

2000 Day Hill Road
Windsor, CT 06095
USA

November 20, 2000
HC-PS-2000-0009, Rev 000

Mr. Ray Moore
Public Service Electric & Gas Company
P.O. Box 236, MC N26
Hancocks Bridge, NJ 08038

SUBJECT: **Operating Range for the CROSSFLOW Meter**

Reference: Public Service Electric & Gas Purchase Order Numbers 4500068041 dated 4/14/00 and 4500069112 dated 4/24/00

Dear Mr. Moore:

Please find attached a write-up documenting the basis for the recommended operating range of the CROSSFLOW meter. If any additional information is required please contact me at (860) 285-5103.

Sincerely,

Westinghouse CE Nuclear Power LLC

G. J. Kanupka

G.J. Kanupka
Project Manager

cc: R. Doney, Westinghouse
C. French, Westinghouse
Y. Gurevich, AMAG

SC-BB-0525
Att. 1
Page 2/2

Attachment to HC-2000-PS-0009, Rev. 000

The AMAG CROSSFLOW meters, which are installed at the Salem and Hope Creek Stations, have been configured to provide optimum accuracy at the 100% power operating condition since this is the limiting power level specified in the operating license. The meter accuracy, which is established by the quality assured calculation for each Unit, is based on configuration inputs and hydraulic characteristics associated with this power level. When operating at other power levels, the effects of hydraulic noise may at some point start to impact the meter's accuracy. However, the meter accuracy will remain within the bounds of the quality assured calculation as long as the 95% confidence interval for the time delay (i.e., ε_{tdelay}), as specified in the quality assured calculation, is not exceeded. Based on experience, it is expected that hydraulic noise will not affect meter accuracy over the 80% to 100% power range provided that no abnormal or excessive vibrations are introduced, which would necessitate re-tuning to achieve the same accuracy.

If a decision is made to change the hydraulic characteristics, such as to alter the feedwater heater bypass flow, then an evaluation of the before and after configurations should be performed to confirm the accuracy of the piping configuration correction factor.

Revised Pressure-Temperature Curves for Hope Creek

SIR-00-0136

Structural Integrity Associates, Inc.

An HSB GROUP Company

6595 S. Dayton Street
Suite 2400
Englewood, CO 80111-6128

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gstevens@structint.com

November 3, 2000
GLS-00-075
SIR-00-136, Rev. 0

Mr. Randal J. Schmidt
PSEG Nuclear
Hope Creek Nuclear Generating Station
P.O. Box 236
Hancocks Bridge, NJ 08038

Subject: Revised Pressure-Temperature Curves for Hope Creek

Reference: PSEG Purchase Order No. 4500087192 dated 9/27/00.

Dear Randy:

The attachment to this letter documents the revised set of pressure-temperature (P-T) curves developed for the Hope Creek Generating Station, in accordance with SI's Quality Assurance Program. This work was performed in accordance with the referenced contract, and includes a full set of updated P-T curves (i.e., pressure test, core not critical, and core critical conditions) for 32 and 48 effective full power years (EFPY). The curves were developed in accordance with 1989 ASME Code Section XI Appendix G, U.S. 10CFR 50 Appendix G, and ASME Code Cases N-588 and N-640.

The inputs, methodology, and results for this effort are summarized in the attachment. The calculations for this work (PSEG-10Q-301 and –302) are also attached.

Please don't hesitate to call me if you have any questions.

Prepared By: Keith Evon
Keith Evon
Engineering Analyst

Reviewed By: Gary L. Stevens
Gary L. Stevens, P. E.
Associate

Approved By: Gary L. Stevens
Gary L. Stevens, P. E.
Associate

kre
Attachments
cc: PSEG-10Q-401

ATTACHMENT

Revised P-T Curves for Hope Creek

1.0 Introduction

This attachment documents the revised set of pressure-temperature (P-T) curves developed for the Hope Creek Nuclear Generating Station. This work includes a full set of updated P-T curves (i.e., pressure test, core not critical, and core critical conditions) for 32 and 48 effective full power years (EFPY). The curves were developed using the methodology specified in ASME Code Cases N-588 [2] and N-640 [3], as well as 1989 ASME Code Section XI Appendix G [4], 10CFR50 Appendix G [5], and WRC-175 [6]. The improvement realized from the Code Case methodology is as much as 60°F, and is primarily obtained from using the critical fracture toughness, K_{IC}, in accordance with Code Case N-640.

2.0 RT_{NDT} Values

Adjusted reference temperature (ART_{NDT}) values were developed for the Hope Creek reactor pressure vessel (RPV) materials in accordance with NRC Regulatory Guide 1.99, Revision 2 [13] based on the fluence data contained in Reference [12]. Tables 1 and 2 show the results of the calculations for 32 and 48 EFPY, respectively. The most limiting beltline material is the Intermediate Plate, Heat No. 5K3025/1.

Table 1: Hope Creek RPV Material ART $_{NDT}$ 32 EFPY Calculations

Part Name & Material	Heat No.	Initial RT_{NDT} (°F)	Chemistry Cu (wt %)	Chemistry Ni (wt %)	Chemistry Factor (°F)	EFPY	Adjustments For 1/4t ΔRT_{NDT} (°F)	Margin Terms σ_Δ (°F)	Margin Terms σ_I (°F)	ART_{NDT} (°F)
(Lower Plates)	5K3230/1	-10	0.07	0.56	44	32.0	13.2	6.6	0.0	16.4
	6C35/1	-11	0.09	0.54	58.0	32.0	17.4	8.7	0.0	23.8
	6C45/1	1	0.08	0.57	51.0	32.0	15.3	7.7	0.0	31.6
(Lower Intermediate Plates)	5K2963/1	-10	0.07	0.58	44.0	32.0	13.2	6.6	0.0	16.4
	5K2530/1	19	0.08	0.56	51.0	32.0	15.3	7.7	0.0	49.6
	5K3238/1	7	0.09	0.63	58	32.0	17.4	8.7	0.0	41.8
(Intermediate Plates)	**5K3025/1**	**19**	**0.15**	**0.71**	**113**	**32.0**	**26.9**	**13.4**	**0.0**	**72.8**
	5K2608/1	19	0.09	0.58	58	32.0	13.8	6.9	0.0	46.7
	5K2698/1	19	0.1	0.58	65	32.0	15.5	7.8	0.0	50.0
(LPCI Nozzle)	19468/1	-20	0.12	0.80	86	32.0	18.3	9.1	0.0	16.6
	10024/1	-20	0.14	0.82	105	32.0	22.3	11.2	0.0	24.6
(Vertical Weld 3)	SMAW / W13	-40	0.09	0.54	109	32.0	26.0	13.0	0.0	12.0
	SAW / W13	-30	0.08	0.59	105	32.0	25.0	12.5	0.0	20.1
(Girth Weld 3/4)	SMAW / W6	-49	0.01	0.53	20	32.0	4.8	2.4	0.0	-39.5
	SMAW / W6	-31	0.01	0.51	20	32.0	4.8	2.4	0.0	-21.5
	SMAW / W6	-40	0.09	0.54	109	32.0	26.0	13.0	0.0	12.0
	SAW / W6	-49	0.1	0.68	126	32.0	30.1	15.0	0.0	11.1
	SAW / W6	-40	0.1	0.68	126	32.0	30.1	15.0	0.0	20.1
(LPCI Nozzle Welds)	SMAW / W179	-40	0.02	0.51	27	32.0	5.7	2.9	0.0	-28.5
	SMAW / W179	-49	0.01	0.53	20	32.0	4.3	2.1	0.0	-40.5
	SMAW / W179	-31	0.01	0.51	20	32.0	4.3	2.1	0.0	-22.5
(Vertical Welds 4&5)	SMAW / W14&15	-40	0.09	0.54	109	32.0	32.7	16.4	0.0	25.5
	SMAW / W14&15	-30	0.08	0.59	105	32.0	31.5	15.8	0.0	33.1
(Girth Weld 4/5)	SMAW / W7	-40	0.09	0.54	109	32.0	32.7	16.4	0.0	25.5
	SAW / W7	-30	0.08	0.59	105	32.0	31.5	15.8	0.0	33.1

Location	Wall Thickness (inches) Full	Wall Thickness (inches) 1/4t	EFPY	Fluence at ID (n/cm²)	Attenuation @ 1/4t $e^{-0.24x}$	Fluence @ 1/4t (n/cm²)	Fluence Factor, FF $f^{(0.28-0.10\log f)}$
(Lower Plates)	6.100	1.525	32.0	7.56E+17	0.694	5.24E+17	0.300
(Lower Intermediate Plates)			32.0	7.56E+17	0.694	5.24E+17	0.300
(Intermediate Plates)			32.0	4.99E+17	0.694	3.46E+17	0.239
(LPCI Nozzle)			32.0	4.09E+17	0.694	2.84E+17	0.213

Inputs to calculate ARTndt are taken from Table 7-2 of [1]

Table 2: Hope Creek RPV Material ART $_{NDT}$ 48 EFPY Calculations

Part Name & Material	Heat No.	Initial RT_{NDT} (°F)	Chemistry Cu (wt %)	Chemistry Ni (wt %)	Chemistry Factor (°F)	EFPY	Adjustments For 1/4t ΔRT_{NDT} (°F)	Margin Terms σ_Δ (°F)	Margin Terms σ_I (°F)	ART_{NDT} (°F)
(Lower Plates)	5K3230/1	-10	0.07	0.56	44	48.0	16.3	8.2	0.0	22.6
	6C35/1	-11	0.09	0.54	58.0	48.0	21.5	10.8	0.0	32.0
	6C45/1	1	0.08	0.57	51.0	48.0	18.9	9.5	0.0	38.8
(Lower Intermediate Plates)	5K2963/1	-10	0.07	0.58	44.0	48.0	16.3	8.2	0.0	22.6
	5K2530/1	19	0.08	0.56	51.0	48.0	18.9	9.5	0.0	56.8
	5K3238/1	7	0.09	0.63	58	48.0	21.5	10.8	0.0	50.0
(Intermediate Plates)	**5K3025/1**	**19**	**0.15**	**0.71**	**113**	**48.0**	**33.7**	**16.9**	**0.0**	**86.4**
	5K2608/1	19	0.09	0.58	58	48.0	17.3	8.7	0.0	53.7
	5K2698/1	19	0.1	0.58	65	48.0	19.4	9.7	0.0	57.9
(LPCI Nozzle)	19468/1	-20	0.12	0.80	86	48.0	23.1	11.5	0.0	26.1
	10024/1	-20	0.14	0.82	105	48.0	28.2	14.1	0.0	36.3
(Vertical Weld 3)	510-01205	-40	0.09	0.54	109	48.0	32.6	16.3	0.0	25.2
	D53040/1125-01205	-30	0.08	0.59	105	48.0	31.4	15.7	0.0	32.8
(Girth Weld 3/4)	519-01205	-49	0.01	0.53	20	48.0	6.0	3.0	0.0	-37.0
	504-01205	-31	0.01	0.51	20	48.0	6.0	3.0	0.0	-19.0
	510-01205	-40	0.09	0.54	109	48.0	32.6	16.3	0.0	25.2
	D53040/1810-02205	-49	0.1	0.68	126	48.0	37.7	17.0	0.0	22.7
	D55733/1810-02206	-40	0.1	0.68	126	48.0	37.7	17.0	0.0	31.7
(LPCI Nozzle Welds)	001-01205	-40	0.02	0.51	27	48.0	7.2	3.6	0.0	-25.5
	519-01205	-49	0.01	0.53	20	48.0	5.4	2.7	0.0	-38.3
	504-01205	-31	0.01	0.51	20	48.0	5.4	2.7	0.0	-20.3
(Vertical Welds 4&5)	510-01205	-40	0.09	0.54	109	48.0	40.4	17.0	0.0	34.4
	D53040/1125-02205	-30	0.08	0.59	105	48.0	38.9	17.0	0.0	42.9
(Girth Weld 4/5)	510-01205	-40	0.09	0.54	109	48.0	40.4	17.0	0.0	34.4
	D53040/1125-02205	-30	0.08	0.59	105	48.0	38.9	17.0	0.0	42.9

Location	Wall Thickness (inches) Full	Wall Thickness (inches) 1/4t	EFPY	Fluence at ID (n/cm²)	Attenuation @ 1/4t $e^{-0.24x}$	Fluence @ 1/4t (n/cm²)	Fluence Factor, FF $f^{(0.28-0.10\log f)}$
(Lower Plates)	6.100	1.525	48.0	1.14E+18	0.694	7.88E+17	0.371
(Lower Intermediate Plates)			48.0	1.14E+18	0.694	7.88E+17	0.371
(Intermediate Plates)			48.0	7.50E+17	0.694	5.20E+17	0.299
(LPCI Nozzle)			48.0	6.15E+17	0.694	4.26E+17	0.268

Inputs to calculate ARTndt are taken from Table 7-2 of [1]

3.0 P-T Curve Methodology

The P-T curve methodology is based on the requirements of References [2] through [6]. The supporting calculations for the curves are contained in References [7] and [8]. There are three regions of the reactor pressure vessel (RPV) that are evaluated: (1) the beltline region, (2) the bottom head region, and (3) the feedwater nozzle/upper vessel region. These regions bound all other regions with respect to brittle fracture.

The approach used for the beltline and bottom head (all curves), and upper vessel (Curve A only) includes the following steps:

a. Assume a fluid temperature, T. The temperature of the metal at the assumed flaw tip, $T_{1/4t}$ (i.e., 1/4t into the vessel wall) is conservatively assumed equal to fluid temperature. The assumed temperature also must account for an instrument uncertainty of 9°F [14].

b. Calculate the allowable stress intensity factor, K_{IC}, based on $T_{1/4t}$ using the relationship from Code Case N-640 [3], as follows:

$$K_{IC} = 20.734\, e^{[0.02(T_{1/4t} - ART_{NDT})]} + 33.2 \qquad \text{(eqn. from Ref. [9])}$$

where: $T_{1/4t}$ = metal temperature at assumed flaw tip (°F)
ART_{NDT} = adjusted reference temperature for location under consideration and desired EFPY (°F)
K_{IC} = allowable stress intensity factor (ksi$\sqrt{\text{inch}}$)

c. Calculate the thermal stress intensity factor, K_{IT} from Code Case N-588 [2] for the beltline and bottom head regions, or from finite element results for the feedwater nozzle/upper vessel region.

d. Calculate the allowable pressure stress intensity factor, K_{IP}, using the following relationship:

$$K_{IP} = (K_{IC} - K_{IT})/SF$$

where: K_{IP} = allowable pressure stress intensity factor (ksi$\sqrt{\text{inch}}$)
SF = safety factor
= 1.5 for pressure test conditions (Curve A)
= 2.0 for heatup/cooldown conditions (Curves B and C)

e. Compute the allowable pressure, P, from the allowable pressure stress intensity factor, K_{IP}.

f. Subtract any applicable adjustments for pressure from P. The beltline and bottom head include a pressure adjustment of 20.3 psig to account for the static pressure head of a full vessel. An instrument error of 20.5 psig was also assumed [15].

g. Repeat steps (a) through (f) for other temperatures to generate a series of P-T points.

The approach used for the upper vessel (Curves B & C) includes the following steps:

a. Assume a fluid pressure, P. The assumed pressure includes an instrument uncertainty of 20.5 psig [15].

b. Calculate the thermal stress intensity factor, K_{IT}, based on finite element stresses. The feedwater nozzle stresses were obtained from the finite element analysis results contained in Reference [10]. The highest linearized (membrane and membrane + bending) thermal stresses for all of the design basis transients were selected to encompass all expected operating conditions.

σ_{ys} = 43.975 ksi @ 575°F for SA-508 Cl. 2 [11,10]

Calculate $t^{1/2}$. The resulting M_m value is obtained from G-2214.1 [2].

K_{Im} is calculated from the equation in Paragraph G-2214.1 [4]:

$$K_{Im} = M_m * \sigma_{sm}$$

K_{Ib} is calculated from the equation in Paragraph G-2214.2 [4]:

$$K_{Ib} = (2/3)\, M_m * \sigma_{sb}$$

The total K_{IT} is therefore:

$$K_{IT} = R*SF*(K_{Im} + K_{Ib})$$

where:

R	=	correction factor, calculated to consider the nonlinear effects in the plastic region based on the assumptions and recommendations of WRC Bulletin 175 [6].
	=	$[\sigma_{ys} - \sigma_{pm} + ((\sigma_{total} - \sigma_{ys}) / 30)] / (\sigma_{total} - \sigma_{pm})$
SF	=	Safety Factor for K_{IT}
	=	1.3 (conservatively used based on the recommendation in WRC-175 [6])

c. Compute the allowable pressure stress intensity factor, K_{IP}, is as follows:

$$K_{IP} = F(a/r_n)\sqrt{\pi a}\ \sigma_{pm}$$

where:

r_i	=	actual inner radius of nozzle
r_c	=	nozzle corner radius [7]
r_n	=	apparent radius of nozzle = $r_i + 0.29r_c$
t'	=	nozzle corner thickness
a	=	crack depth (inches)
	=	1/4 t'
$F(a,r_n)$	=	nozzle stress factor, from Figure A5-1 of [6]
K_{IP}	=	allowable pressure stress intensity factor (ksi$\sqrt{}$inch).
σ_{pm}	=	primary membrane stress, PR/t (primary bending stresses are conservatively treated as membrane stresses, so $\sigma_{pb} = 0$)

d. Calculate the allowable stress intensity factor, K_{IC}, using the following relationship for a heatup/cooldown P-T curve:

$$K_{IP} = \frac{K_{IC} - K_{IT}}{2.0}$$

thus: $$K_{IC} = 2.0K_{IP} + K_{IT}$$

e. Calculate the temperature, T, using the relationship from Code Case N-640 [3], as follows:

$$K_{IC} = 20.734\ e^{[0.02(T_{1/4t} - ART_{NDT})]} + 33.2 \quad \text{(eqn. from Ref. [9])}$$

where:

$T_{1/4t}$	=	metal temperature at assumed flaw tip (°F), assumed equal to T, the temperature at the inner vessel wall
ART_{NDT}	=	adjusted reference temperature for location under consideration and desired EFPY (°F)
K_{IC}	=	allowable stress intensity factor (ksi$\sqrt{}$inch)

thus: $$T = 50 * LN\left[\frac{K_{IC} - 33.2}{20.734}\right] + ART_{NDT}$$

f. The curve was generated by scaling the stresses used to determine the pressure and thermal stress intensity factors. The primary stresses were scaled based on pressure, while the secondary stresses were scaled based on temperature difference.

g. Repeat steps (a) through (f) for other pressures to generate a series of P-T points.

The following additional requirements were used to define the P-T curves. These limits are established in Reference [5]:

For Pressure Test Conditions (Curve A):

- If the pressure is greater than 20% of the pre-service hydro test pressure, the temperature must be greater than ART_{NDT} of the limiting flange material + 90°F.
- If the pressure is less than or equal to 20% of the pre-service hydro test pressure, the minimum temperature is must be greater than or equal to the ART_{NDT} of the limiting flange material + 60°F. The instrument uncertainty of 9°F was not applied since the 60°F is an additional margin above that recommended in Reference [10], and has been a standard recommendation for the BWR industry for non-ductile failure protection. Therefore, the 60°F is considered to adequately encompass instrument uncertainty.

For Core Not Critical Conditions (Curve B):

- If the pressure is greater than 20% of the pre-service hydro test pressure, the temperature must be greater than RT_{NDT} of the limiting flange material + 120°F.
- If the pressure is less than or equal to 20% of the pre-service hydro test pressure, the minimum temperature must be greater than or equal to the ART_{NDT} of the limiting flange material + 60°F. The instrument uncertainty of 9°F was not applied since the 60°F is an additional margin above that recommended in Reference [10], and has been a standard recommendation for the BWR industry for non-ductile failure protection. Therefore, the 60°F is considered to adequately encompass instrument uncertainty.

For Core Critical Conditions (Curve C):

- Per the requirements of Table 1 of Reference [5], the core critical P-T limits must be 40°F above any Pressure Test or Core Not Critical curve limits. Core Not Critical conditions are more limiting than Pressure Test conditions, so Core Critical conditions are equal to Core Not Critical conditions plus 40°F.
- Another requirement of Table 1 of Reference [5] (or actually an allowance for the BWR), concerns minimum temperature for initial criticality in a startup. Given that water level is normal, BWRs are allowed initial criticality at the closure flange region temperature (ART_{NDT} + 60°F) if the pressure is below 20% of the pre-service hydro test pressure.
- Also per Table 1 of Reference [5], at pressures above 20% of the pre-service hydro test pressure, the Core Critical curve temperature must be at least that required for the pressure test (Pressure Test Curve at 1,100 psig). As a result of this requirement, the Core Critical curve must have a step at a pressure equal to 20% of the pre-service hydro pressure to the temperature required by the Pressure Test curve at 1,100 psig, or Curve B + 40°F, whichever is greater.

After accounting for instrument uncertainties, the resulting pressure and temperature series constitutes the P-T curve. The P-T curve relates the minimum required fluid temperature to the reactor pressure.

4.0 Upper Shelf Energy Calculations

This section explains the methodology used to calculate the beltline upper shelf energy (USE) values for 32 and 48 EFPY. All beltline plates and welds were evaluated.

a. Obtain initial USE and %Cu values from Table 7-3 of [1]. Fluence values were obtained from Reference [12].

b. Using the data obtained in step (a) and Figure 2 of NRC Regulatory Guide 1.99, Revision 2 [13], determine the percent decrease in shelf energy for 32 and 48 EFPY.

c. Calculate the new USE value as follows:

(32 or 48 EFPY) USE = Initial USE * (1 - %decrease in shelf energy)

d. Verify that all USE values are above that recommended in 10CFR50 Appendix G [5]

Table 3 shows that all of the 32 and 48 EFPY USE values are above the 50 ft-lb minimum value recommended in 10CFR50 Appendix G [5].

Table 3: Upper Shelf Energy Analysis for Hope Creek 1 Beltline Material

				32 EFPY			48 EFPY		
Location	Heat	Initial USE	%Cu	1/4t fluence (n/cm^2)	Decrease in USE %	USE	1/4t fluence (n/cm^2)	Decrease in USE %	USE
Plates:									
Lower	5K3230/1	121	0.07	5.24E+17	8.0%	**111.3**	7.88E+17	8.6%	**110.6**
	6C35/1	107	0.09	5.24E+17	9.0%	**97.4**	7.88E+17	10.0%	**96.3**
	6C45/1	97	0.08	5.24E+17	8.5%	**88.8**	7.88E+17	9.0%	**88.3**
Low-Int.	5K2963/1	102	0.07	5.24E+17	8.0%	**93.8**	7.88E+17	8.6%	**93.2**
	5K2530/1	86	0.08	5.24E+17	8.5%	**78.7**	7.88E+17	9.0%	**78.3**
	5K3238/1	76	0.09	5.24E+17	9.0%	**69.2**	7.88E+17	10.0%	**68.4**
Unirradiated Surveillance	5K3238/1	91	0.09	5.24E+17	9.0%	**82.8**	7.88E+17	10.0%	**81.9**
Int.	5K3025/1	75	0.15	3.46E+17	11.0%	**66.8**	5.20E+17	12.0%	**66.0**
	5K2608/1	75	0.09	3.46E+17	8.0%	**69.0**	5.20E+17	8.8%	**68.4**
	5K2698/1	75	0.10	3.46E+17	8.5%	**68.6**	5.20E+17	9.4%	**68.0**
LPCI Nozzle	19468/1	> 79	0.12	2.84E+17	9.0%	**71.9**	4.26E+17	10.0%	**71.1**
	10024/1	> 70	0.14	2.84E+17	10.0%	**63.0**	4.26E+17	11.0%	**62.3**
Welds:									
Vertical	510-01205	> 92.5	0.09	5.24E+17	11.5%	**81.9**	7.88E+17	13.0%	**80.5**
	D53040	135	0.08	5.24E+17	11.0%	**120.2**	7.88E+17	12.0%	**118.8**
Unirradiated Surveillance	D53040	164	0.08	5.24E+17	11.0%	**146.0**	7.88E+17	12.0%	**144.3**
LPCI Nozzle	001-01205	> 109	0.02	3.46E+17	7.0%	**101.4**	5.20E+17	7.8%	**100.5**
Girth	519-01205	> 109	0.01	3.46E+17	6.6%	**101.8**	5.20E+17	7.4%	**100.9**
	504-01205	> 125	0.01	3.46E+17	6.6%	**116.8**	5.20E+17	7.4%	**115.8**
	D53040	> 95	0.10	3.46E+17	11.0%	**84.6**	5.20E+17	12.0%	**83.6**
	D55733	> 68	0.10	3.46E+17	11.0%	**60.5**	5.20E+17	12.0%	**59.8**

5.0 P-T Curves

Tabulated values for the P-T curves are shown in Tables 4 through 12. The resulting P-T curves are shown in Figures 1 through 5. Note that since the upper vessel (non-beltline) curve is limiting for core not critical conditions for both 32 and 48 EFPY, Curve C is identical for both EFPY levels (i.e., no fluence effects).

6.0 References

1. GE-NE-523-A164-1294R1, "Hope Creek 1 Generating Station RPV Surveillance Materials Testing and Fracture Toughness Analysis," December 1997, SI File No. PSEG-10Q-201.

2. ASME Boiler and Pressure Vessel Code, Code Case N-588, "Alternative Reference Flaw Orientation of Appendix G for Circumferential Welds in Reactor Vessels," Section XI, Division 1, Approved December 12, 1997.

3. ASME Boiler and Pressure Vessel Code, Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves," Section XI, Division 1, Approved February 26, 1999.

4. ASME Boiler and Pressure Vessel Code, Section XI, Rules for Inservice Inspection of Nuclear Power Plant Components, Nonmandatory Appendix G, "Fracture Toughness Criteria for Protection Against Failure," 1989 Edition.

5. U. S. Code of Federal Regulations, Title 10, Part 50, Appendix G, "Fracture Toughness Requirements," 1-1-98 Edition.

6. WRC Bulletin 175, "PVRC Recommendations on Toughness Requirements for Ferritic Materials," PVRC Ad Hoc Group on Toughness Requirements, Welding Research Council, August 1972.

7. Structural Integrity Associates Calculation No. PSEG-10Q-301, Revision 0, "Development of Pressure Test (Curve A) P-T Curves," 11/3/00.

8. Structural Integrity Associates Calculation No. PSEG-10Q-302, Revision 0, "Development of Heatup/Cooldown (Curves B & C) P-T Curves," 11/3/00.

9. ASME Boiler and Pressure Vessel Code, Section XI, <u>Rules for Inservice Inspection of Nuclear Power Plant Components</u>, Nonmandatory Appendix A, "Analysis of Flaws," 1989 Edition.

10. Hitachi Stress Report No. RS-9130 Rev.4 , VPF 2883-459-5, "Detail Analysis For Feed Water Nozzle," PSEG Vendor Technical Document 320118, SI File No. PSEG-10Q-203.

11. ASME Boiler and Pressure Vessel Code, Section III, <u>Rules for Construction of Nuclear Power Plant Components</u>, Appendices, 1989 Edition.

12. PSEG Calculation # H-1-BB-NDC-1858 Rev. 0IR0, "Reactor Vessel Fluence Calculation for 1.5% Power Uprate," September 2000, SI File No. PSEG-10Q-202.

13. USNRC Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," U. S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, (Task ME 305-4), May 1988.

14. Temperature Loop Uncertainty, PSEG Nuclear Calculation SC-SC0001, Revision 0, SI File No. PSEG-10Q-103 (11/1/00 e-mail).

15. Pressure Loop Uncertainty, PSEG Nuclear Calculation SC-BB-0355, SI File No. PSEG-10Q-103 (11/1/00 e-mail).

Table 4
Tabulated Values for Beltline Pressure Test Curve (Curve A) for 32 EFPY

Pressure-Temperature Curve Calculation

(Pressure Test = Curve A)

Inputs:

Plant =	**Hope Creek**			
Component =	**Beltline**			
Vessel thickness, t =	**6.1000**	inches, so √t =	2.470	√inch
Vessel Radius, R =	**126.5**	inches		
ART_{NDT} =	**72.8**	°F ======>	**32 EFPY**	
Cooldown Rate, CR =	**0**	°F/hr		
K_{IT} =	**0.00**	ksi*inch$^{1/2}$ (From N-588, for cooldown rate above)		
$\Delta T_{1/4t}$ =	**0.0**	°F (no thermal for pressure test)		
Safety Factor =	**1.50**	(for pressure test)		
M_m =	**2.287**	(From N-588, for inside surface axial flaw)		
Temperature Adjustment =	**9.0**	°F		
Height of Water for a Full Vessel =	**562.5**	inches		
Pressure Adjustment =	**20.3**	psig (hydrostatic pressure for a full vessel at 70°F)		
Pressure Adjustment =	**20.5**	psig (Instrument Uncertainty)		
Hydro Test Pressure =	**1,563**	psig		
Flange RT_{NDT} =	**19.0**	°F		

Gauge Temperature T (°F)	Adjusted Temperature (°F)	K_{IC} (ksi*inch$^{1/2}$)	K_{IP} (ksi*inch$^{1/2}$)	Calculated Pressure P (psig)	Temperature for P-T Curve (°F)	Adjusted Pressure for P-T Curve (psig)
79.0	79.0	56.67	37.78	0	79.0	0
79.0	70.0	52.80	35.20	742	79.0	701
88.0	79.0	56.67	37.78	797	88.0	756
93.0	84.0	59.14	39.43	831	93.0	790
98.0	89.0	61.87	41.25	870	98.0	829
103.0	94.0	64.88	43.26	912	103.0	871
108.0	99.0	68.21	45.48	959	108.0	918
113.0	104.0	71.90	47.93	1011	113.0	970
118.0	109.0	75.97	50.64	1068	118.0	1,027
123.0	114.0	80.47	53.64	1131	123.0	1,090
128.0	119.0	85.44	56.96	1201	128.0	1,160
133.0	124.0	90.93	60.62	1278	133.0	1,237
138.0	129.0	97.00	64.67	1363	138.0	1,323
143.0	134.0	103.71	69.14	1458	143.0	1,417
148.0	139.0	111.13	74.08	1562	148.0	1,521
153.0	144.0	119.32	79.55	1677	153.0	1,636
158.0	149.0	128.38	85.59	1805	158.0	1,764
163.0	154.0	138.39	92.26	1945	163.0	1,904
168.0	159.0	149.45	99.64	2101	168.0	2,060

Table 5
Tabulated Values for Beltline Pressure Test Curve (Curve A) for 48 EFPY

Pressure-Temperature Curve Calculation

(Pressure Test = Curve A)

Inputs:

Plant =	**Hope Creek**			
Component =	**Beltline**			
Vessel thickness, t =	**6.1000**	inches, so $\sqrt{t}$ =	2.470	$\sqrt{\text{inch}}$
Vessel Radius, R =	**126.5**	inches		
ART_{NDT} =	**86.4**	°F ======>	**48 EFPY**	
Cooldown Rate, CR =	**0**	°F/hr		
K_{IT} =	**0.00**	ksi*inch$^{1/2}$ (From N-588, for cooldown rate above)		
$\Delta T_{1/4t}$ =	**0.0**	°F (no thermal for pressure test)		
Safety Factor =	**1.50**	(for pressure test)		
M_m =	**2.287**	(From N-588, for inside surface axial flaw)		
Temperature Adjustment =	**9.0**	°F		
Height of Water for a Full Vessel =	**562.5**	inches		
Pressure Adjustment =	**20.3**	psig (hydrostatic pressure for a full vessel at 70°F)		
Pressure Adjustment =	**20.5**	psig (Instrument Uncertainty)		
Hydro Test Pressure =	**1,563**	psig		
Flange RT_{NDT} =	**19.0**	°F		

Gauge Temperature T (°F)	Adjusted Temperature (°F)	K_{IC} (ksi*inch$^{1/2}$)	K_{IP} (ksi*inch$^{1/2}$)	Calculated Pressure P (psig)	Temperature for P-T Curve (°F)	Adjusted Pressure for P-T Curve (psig)
79.0	79.0	51.08	34.05	0	79.0	0
79.0	70.0	48.14	32.09	677	79.0	636
88.0	79.0	51.08	34.05	718	88.0	677
93.0	84.0	52.96	35.31	744	93.0	704
98.0	89.0	55.04	36.69	774	98.0	733
103.0	94.0	57.34	38.23	806	103.0	765
108.0	99.0	59.88	39.92	842	108.0	801
113.0	104.0	62.68	41.79	881	113.0	840
118.0	109.0	65.78	43.85	925	118.0	884
123.0	114.0	69.21	46.14	973	123.0	932
128.0	119.0	73.00	48.66	1026	128.0	985
133.0	124.0	77.18	51.45	1085	133.0	1,044
138.0	129.0	81.81	54.54	1150	138.0	1,109
143.0	134.0	86.92	57.95	1222	143.0	1,181
148.0	139.0	92.57	61.71	1301	148.0	1,260
153.0	144.0	98.81	65.88	1389	153.0	1,348
158.0	149.0	105.71	70.48	1486	158.0	1,445
163.0	154.0	113.34	75.56	1593	163.0	1,552
168.0	159.0	121.77	81.18	1712	168.0	1,671
173.0	164.0	131.08	87.39	1843	173.0	1,802
178.0	169.0	141.38	94.25	1987	178.0	1,946
183.0	174.0	152.75	101.84	2147	183.0	2,106

Table 6
Tabulated Values for Feedwater Nozzle/Upper Vessel Region Pressure Test Curve (Curve A)

Pressure-Temperature Curve Calculation

(Pressure Test = Curve A)

Inputs:

Plant =	**Hope Creek**	
Component =	**Upper Vessel**	(based on FW nozzle)
ART_{NDT} =	**40.0**	°F ======> **All EFPYs**
Vessel thickness, t =	**6.169**	inches, so $\sqrt{t}$ = 2.484 $\sqrt{\text{inch}}$
Vessel Radius, R =	**126.5**	inches
Nozzle corner thickness, t' =	**9.7**	inches, approximate
F(a/rn) =	**1.44**	nozzle stress factor
Crack Depth, a =	**2.425**	inches
Safety Factor =	**1.50**	
Temperature Adjustment =	**9.0**	°F
Pressure Adjustment =	**20.5**	psig (Instrument Uncertainty)
Unit Pressure =	**1,563**	psig
Flange RT_{NDT} =	**19.0**	°F

Gauge Temperature T (°F)	Adjusted Temperature (°F)	K_{IC} (ksi*inch$^{1/2}$)	K_{IP} (ksi*inch$^{1/2}$)	Calculated Pressure P (psig)	Temperature for P-T Curve (°F)	Adjusted Pressure for P-T Curve (psig)
79.0	79.0	78.43	52.29	0	79.0	0
79.0	79.0	78.43	52.29	313	79.0	292
118.0	109.0	115.62	77.08	313	118.0	292
118.0	109.0	115.62	77.08	946	118.0	925
123.0	114.0	124.28	82.86	1017	123.0	996
128.0	119.0	133.86	89.24	1095	128.0	1074
133.0	124.0	144.45	96.30	1182	133.0	1161
138.0	129.0	156.15	104.10	1277	138.0	1257
143.0	134.0	169.08	112.72	1383	143.0	1363
148.0	139.0	183.37	122.25	1500	148.0	1479

Table 7
Tabulated Values for Bottom Head Pressure Test Curve (Curve A)

Pressure-Temperature Curve Calculation

(Pressure Test = Curve A)

Inputs:

Plant =	**Hope Creek**			
Component =	**Bottom Head**	(Penetrations Portion)		
Vessel thickness, t =	**6.100**	inches, so √t =	2.470	√inch
Vessel Radius, R =	**126.5**	inches		
ART_{NDT} =	**30.0**	°F ======>	**All EFPYs**	
Safety Factor =	**1.50**			
Safety Factor =	**2.30**	Bottom Head Penetrations		
M_m =	**2.287**	(From N-588, for inside surface axial flaw)		
Temperature Adjustment =	**9.0**	°F (Instrument Uncertainty)		
Height of Water for a Full Vessel =	**562.5**	inches		
Pressure Adjustment =	**20.3**	psig (full vessel at 70°F)		
Pressure Adjustment =	**20.5**	psig (Instrument Uncertainty)		
Unit Pressure =	**1,563**	psig		
Flange RT_{NDT} =	**19.0**	°F		

Gauge Temperature T (°F)	Adjusted Temperature (°F)	K_{IC} (ksi*inch$^{1/2}$)	K_{IP} (ksi*inch$^{1/2}$)	Calculated Pressure P (psig)	Temperature for P-T Curve (°F)	Adjusted Pressure for P-T Curve (psig)
79.0	79.0	88.44	58.96	0	79	0
79.0	70.0	79.34	52.90	970	79	929
88.0	79.0	88.44	58.96	1081	88	1040
90.0	81.0	90.70	60.47	1109	90	1068
92.0	83.0	93.05	62.03	1137	92	1097
94.0	85.0	95.49	63.66	1167	94	1126
96.0	87.0	98.03	65.35	1198	96	1157
98.0	89.0	100.68	67.12	1231	98	1190
100.0	91.0	103.43	68.95	1264	100	1223
102.0	93.0	106.30	70.86	1299	102	1258
104.0	95.0	109.28	72.85	1336	104	1295
106.0	97.0	112.38	74.92	1374	106	1333
108.0	99.0	115.62	77.08	1413	108	1372
110.0	101.0	118.98	79.32	1454	110	1413
112.0	103.0	122.48	81.65	1497	112	1456
114.0	105.0	126.12	84.08	1542	114	1501
116.0	107.0	129.92	86.61	1588	116	1547
118.0	109.0	133.86	89.24	1636	118	1595

Table 8
Tabulated Values for Beltline Core Not Critical Curve (Curve B) for 32 EFPY

Pressure-Temperature Curve Calculation

(Heatup/Cooldown, Core Not Critical = Curve B)

Inputs:

Plant =	**Hope Creek**			
Component =	**Beltline**			
Vessel thickness, t =	**6.1000**	inches, so $\sqrt{t}$ =	2.470	$\sqrt{inch}$
Vessel Radius, R =	**126.5**	inches		
ART_{NDT} =	**72.8**	°F ======>	**32 EFPY**	
Cooldown Rate, CR =	**100**	°F/hr		
K_{IT} =	**8.76**	ksi*inch$^{1/2}$ (From N-588, for cooldown rate)		
M_T =	**0.285**	(From Figure G-2214-2)		
$\Delta T_{1/4t}$ =	**0.0**	°F = Conservatively assumed zero		
Safety Factor =	**2.00**			
M_m =	**2.287**	(From N-588, for inside surface axial flaw)		
Temperature Adjustment =	**9.0**	°F (Instrument Uncertainty)		
Height of Water for a Full Vessel =	**562.5**	inches		
Pressure Adjustment =	**20.3**	psig (hydrostatic pressure for a full vessel at 70°F)		
Pressure Adjustment =	**20.5**	psig (Instrument Uncertainty)		
Hydro Test Pressure =	**1,563**	psig		
Flange RT_{NDT} =	**19.0**	°F		

Gauge Temperature T (°F)	Adjusted Temperature (°F)	K_{IC} (ksi*inch$^{1/2}$)	K_{IP} (ksi*inch$^{1/2}$)	Calculated Pressure P (psig)	Temperature for P-T Curve (°F)	Adjusted Pressure for P-T Curve (psig)
79.0	79.0	56.67	23.96	0	79.0	0
79.0	70.0	52.80	22.02	464	79.0	424
88.0	79.0	56.67	23.96	505	88.0	464
93.0	84.0	59.14	25.19	531	93.0	490
98.0	89.0	61.87	26.55	560	98.0	519
103.0	94.0	64.88	28.06	592	103.0	551
108.0	99.0	68.21	29.73	627	108.0	586
113.0	104.0	71.90	31.57	666	113.0	625
118.0	109.0	75.97	33.60	709	118.0	668
123.0	114.0	80.47	35.85	756	123.0	715
128.0	119.0	85.44	38.34	808	128.0	768
133.0	124.0	90.93	41.09	866	133.0	825
138.0	129.0	97.00	44.12	930	138.0	889
143.0	134.0	103.71	47.48	1001	143.0	960
148.0	139.0	111.13	51.18	1079	148.0	1,038
153.0	144.0	119.32	55.28	1166	153.0	1,125
158.0	149.0	128.38	59.81	1261	158.0	1,220
163.0	154.0	138.39	64.82	1367	163.0	1,326

Table 9
Tabulated Values for Beltline Core Not Critical Curve (Curve B) for 48 EFPY

Pressure-Temperature Curve Calculation

(Heatup/Cooldown, Core Not Critical = Curve B)

Inputs:

Plant =	**Hope Creek**			
Component =	**Beltline**			
Vessel thickness, t =	**6.1000**	inches, so $\sqrt{t}$ =	2.470	$\sqrt{inch}$
Vessel Radius, R =	**126.5**	inches		
ART_{NDT} =	**86.4**	°F ======>	**48 EFPY**	
Cooldown Rate, CR =	**100**	°F/hr		
K_{IT} =	**8.76**	ksi*inch$^{1/2}$ (From N-588, for cooldown rate)		
M_T =	**0.285**	(From Figure G-2214-2)		
$\Delta T_{1/4t}$ =	**0.0**	°F = Conservatively assumed zero		
Safety Factor =	**2.00**			
M_m =	**2.287**	(From N-588, for inside surface axial flaw)		
Temperature Adjustment =	**9.0**	°F		
Height of Water for a Full Vessel =	**562.5**	inches		
Pressure Adjustment =	**20.3**	psig (hydrostatic pressure for a full vessel at 70°F)		
Pressure Adjustment =	**20.5**	psig (Instrument Uncertainty)		
Hydro Test Pressure =	**1,563**	psig		
Flange RT_{NDT} =	**19.0**	°F		

Gauge Temperature T (°F)	Adjusted Temperature (°F)	K_{IC} (ksi*inch$^{1/2}$)	K_{IP} (ksi*inch$^{1/2}$)	Calculated Pressure P (psig)	Temperature for P-T Curve (°F)	Adjusted Pressure for P-T Curve (psig)
79.0	79.0	51.08	21.16	0	79.0	0
79.0	70.0	48.14	19.69	415	79.0	374
88.0	79.0	51.08	21.16	446	88.0	405
96.0	87.0	54.18	22.71	479	96.0	438
104.0	95.0	57.83	24.53	517	104.0	476
112.0	103.0	62.10	26.67	562	112.0	522
120.0	111.0	67.11	29.18	615	120.0	574
128.0	119.0	73.00	32.12	677	128.0	636
136.0	127.0	79.90	35.57	750	136.0	709
144.0	135.0	88.00	39.62	835	144.0	795
152.0	143.0	97.51	44.38	936	152.0	895
160.0	151.0	108.67	49.96	1053	160.0	1,013
168.0	159.0	121.77	56.51	1191	168.0	1,151

Table 10
Tabulated Values for Feedwater Nozzle/Upper Vessel Region Core Not Critical Curve (Curve B)

Pressure-Temperature Curve Calculation

(Heatup/Cooldown, Core Not Critical = Curve B)

Inputs:

Plant =	**Hope Creek**	
Component =	**Upper Vessel**	
ART_{NDT} =	**40.0**	°F
σ_{pm} =	**20.61**	ksi for a pressure of 1,005 psig
σ_{pb} =	**0.00**	ksi for a pressure of 1,005 psig
σ_{sm} =	**11.45**	ksi for a temperature of 547°F
σ_{sb} =	**8.15**	ksi for a temperature of 547°F
σ_{ys} =	**44.0**	ksi
M_m =	**2.88**	
$F(a/r_n)$ =	**1.44**	
Temperature Adjustment =	**9.0**	°F (Instrument Uncertainty)
Pressure Adjustment =	**20.5**	psig (Instrument Uncertainty)
Hydro Test Pressure =	**1563**	psig
Flange RT_{NDT} =	**19.0**	°F

Adjusted Pressure for Calculation (psig)	Saturation Temperature (°F)	K_{IT} (ksi*inch$^{1/2}$)	K_{IP} ksi*inch$^{1/2}$	K_{IC} (ksi*inch$^{1/2}$)	Calculated Temperature (°F)	Adjusted Temperature for P-T Curve (°F)	Pressure for P-T Curve (psig)
0	212.1	21.5	0.0	21.5	---	79.0	0
70.5	316.1	34.4	5.7	45.9	15.5	79.0	50
95.5	334.7	36.7	7.8	52.3	35.9	79.0	75
120.5	350.2	38.7	9.8	58.3	49.6	79.0	100
145.5	363.6	40.3	11.9	64.1	59.9	79.0	125
176.9	378.3	42.2	14.4	71.0	70.0	79.0	156
190.5	384.0	42.9	15.5	73.9	73.8	82.8	170
210.5	392.0	43.9	17.2	78.2	78.7	87.7	190
230.5	399.4	44.8	18.8	82.4	83.2	92.2	210
250.5	406.4	45.7	20.4	86.5	87.2	96.2	230
270.5	413.0	46.5	22.0	90.6	90.9	99.9	250
312.4	425.7	48.1	25.5	99.0	97.7	106.7	292
312.5	425.7	48.1	25.5	99.0	97.8	148.0	292
760.5	514.8	59.2	62.0	183.2	138.9	148.0	740
765.5	515.5	59.3	62.4	184.1	139.2	148.2	745
770.5	516.2	59.4	62.8	185.0	139.5	148.5	750
850.5	527.4	60.8	69.3	199.4	144.1	153.1	830
930.5	537.7	62.1	75.8	213.7	148.2	157.2	910
1010.5	547.4	63.3	82.4	228.0	152.0	161.0	990
1090.5	556.5	64.4	88.9	242.2	155.5	164.5	1070
1170.5	565.2	64.5	95.4	255.2	158.6	167.6	1150
1250.5	573.3	59.4	101.9	263.2	160.3	169.3	1230

Table 11
Tabulated Values for Bottom Head Core Not Critical Curve (Curve B)

Pressure-Temperature Curve Calculation
(Heatup/Cooldown, Core Not Critical = Curve B)

Inputs:

Plant =	**Hope Creek**			
Component =	**Bottom Head**	(Penetrations Portion)		
Vessel thickness, t =	**6.100**	inches, so √t =	2.470	√inch
Vessel Radius, R =	**126.5**	inches		
ART_{NDT} =	**30.0**	°F ======>	**All EFPYs**	
Safety Factor =	**2.00**			
Stress Concentration Factor =	**2.30**	Bottom Head Penetrations		
Cooldown Rate, CR =	**100**	°F/hr		
M_m =	**2.287**	(From N-588, for inside surface axial flaw)		
K_{IT} =	**8.76**	ksi*inch$^{1/2}$ (From N-588, for cooldown rate)		
M_T =	**0.285**	(From Figure G-2214-2)		
Temperature Adjustment =	**9.0**	°F, Instrument Uncertainty		
Height of Water for a Full Vessel =	**562.5**	inches (FEM stresses include deadweight)		
Pressure Adjustment =	**20.3**	psig (full vessel at 70°F)		
Pressure Adjustment =	**20.5**	psig (Instrument Uncertainty)		
Unit Pressure =	**1,563**	psig		
Flange RT_{NDT} =	**19.0**	°F		

Gauge Temperature T (°F)	Adjusted Temperature (°F)	K_{IC} (ksi*inch$^{1/2}$)	K_{IP} (ksi*inch$^{1/2}$)	Calculated Pressure P (psig)	Temperature for P-T Curve (°F)	Adjusted Pressure for P-T Curve (psig)
79.0	79.0	88.44	39.84	0	79	0
79.0	70.0	79.34	35.29	647	79	606
88.0	79.0	88.44	39.84	731	88	690
92.0	83.0	93.05	42.14	773	92	732
96.0	87.0	98.03	44.64	818	96	778
100.0	91.0	103.43	47.34	868	100	827
104.0	95.0	109.28	50.26	921	104	881
108.0	99.0	115.62	53.43	980	108	939
112.0	103.0	122.48	56.86	1043	112	1002
116.0	107.0	129.92	60.58	1111	116	1070
120.0	111.0	137.97	64.61	1185	120	1144
124.0	115.0	146.70	68.97	1265	124	1224
128.0	119.0	156.15	73.70	1351	128	1310
132.0	123.0	166.39	78.82	1445	132	1404
136.0	127.0	177.48	84.36	1547	136	1506

Table 12
Tabulated Values for Core Critical Curve (Curve C) for 32 & 48 EFPY

Pressure-Temperature Curve Calculation

(Core Critical = Curve C)

Inputs:			
	Plant =	**Hope Creek**	
	EFPY =	**32 & 48**	
	Curve A Leak Test Temperature =	**130.0**	°F (at 32 EFPY and 1,100 psig)
	Curve A Leak Test Temperature =	**138.0**	°F (at 48 EFPY and 1,100 psig)
	Hydro Test Pressure =	**1,563**	psig
	Flange RT_{NDT} =	**19.0**	°F

Curve B Temperature Upper Vessel (°F)	Curve B Pressure for Upper Vessel (psig)	Curve C Temperature (°F)	Curve C Pressure (psig)
---	0.0	88.0	0
15.5	50.0	88.0	50
35.9	75.0	88.0	75
49.6	100.0	98.6	100
59.9	125.0	108.9	125
70.0	156.4	119.0	156
82.8	170.0	122.8	170
87.7	190.0	127.7	190
92.2	210.0	132.2	210
96.2	230.0	136.2	230
99.9	250.0	139.9	250
106.7	291.9	146.7	292
148.0	292.0	188.0	292
148.0	740.0	188.0	740
148.2	745.0	188.2	745
148.5	750.0	188.5	750
153.1	830.0	193.1	830
157.2	910.0	197.2	910
161.0	990.0	201.0	990
164.5	1070.0	204.5	1070
167.6	1150.0	207.6	1150
169.3	1230.0	209.3	1230

Figure 1
Pressure Test P-T Curve (Curve A) for 32 EFPY

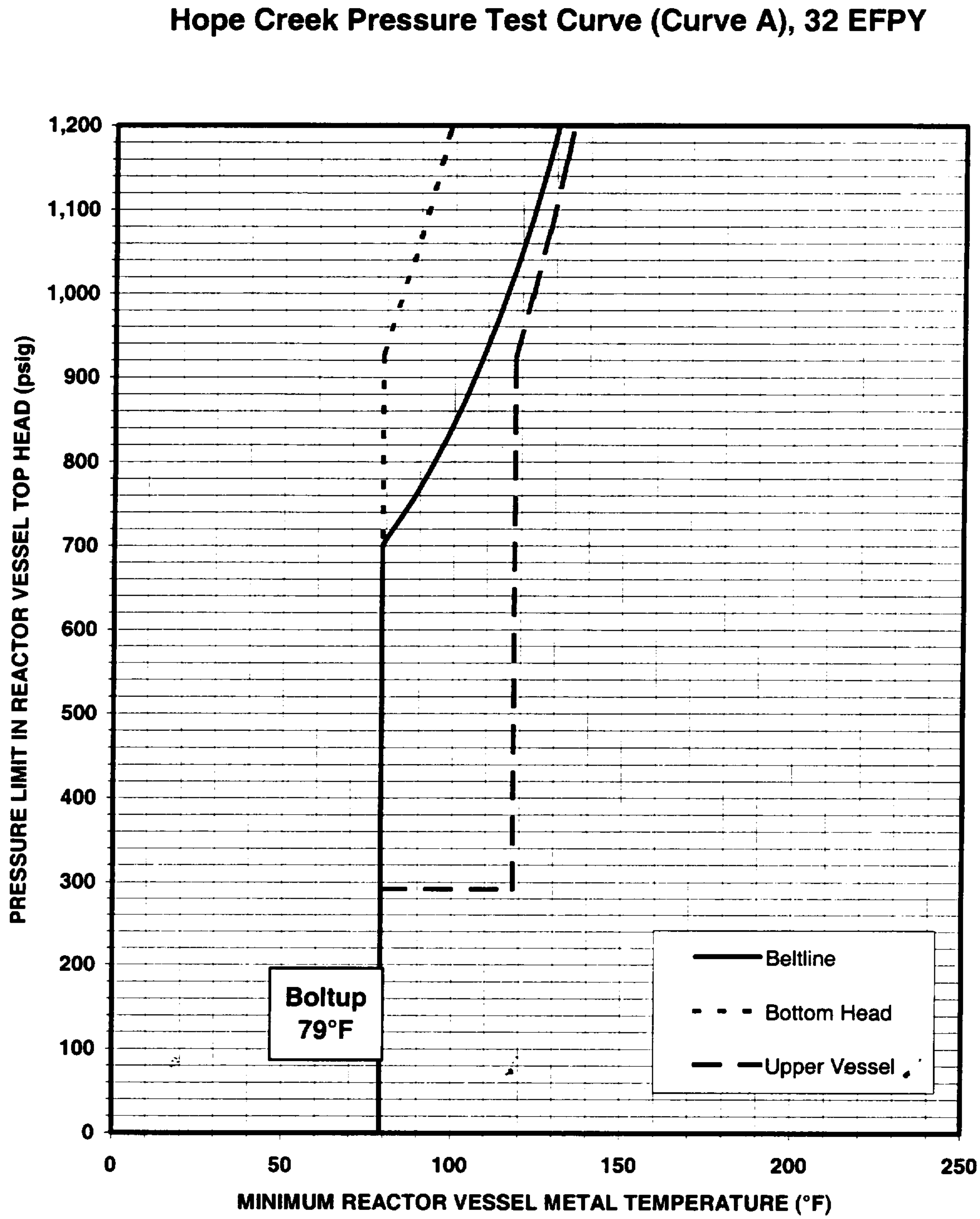


Figure 2
Pressure Test P-T Curve (Curve A) for 48 EFPY

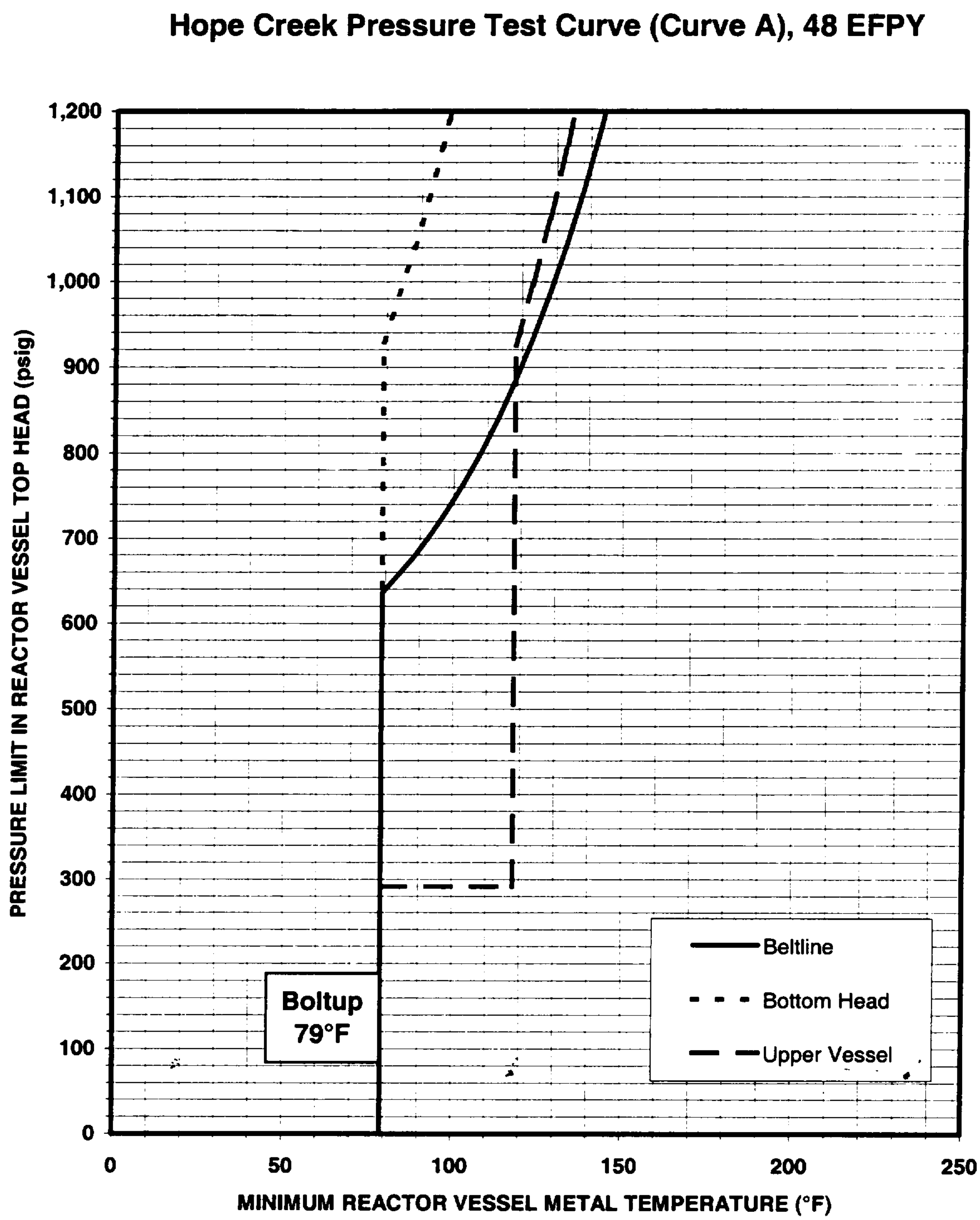
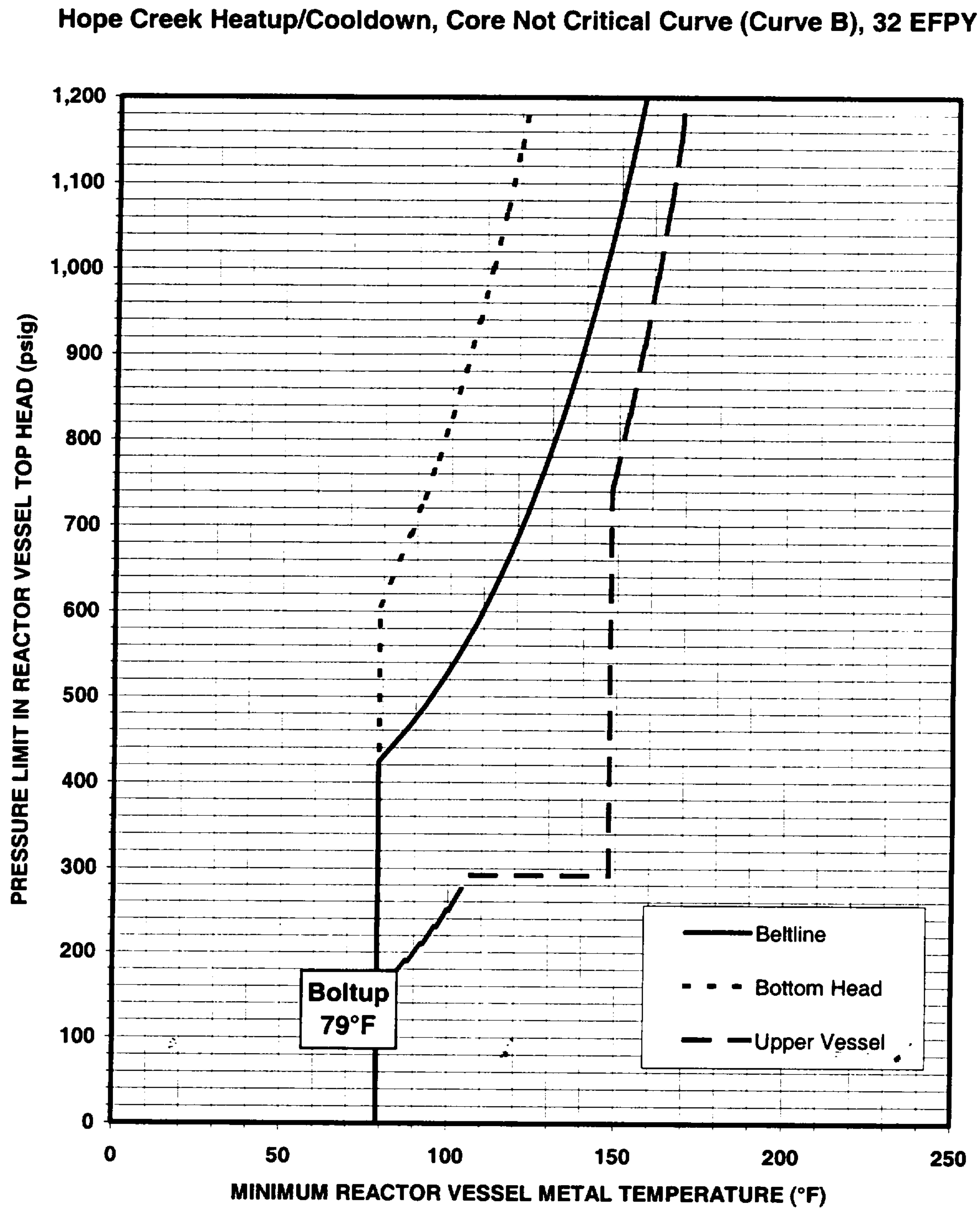


Figure 3
Core Not Critical Curve (Curve B) for 32 EFPY



Core Not Critical Curve (Curve B) for 48 EFPY

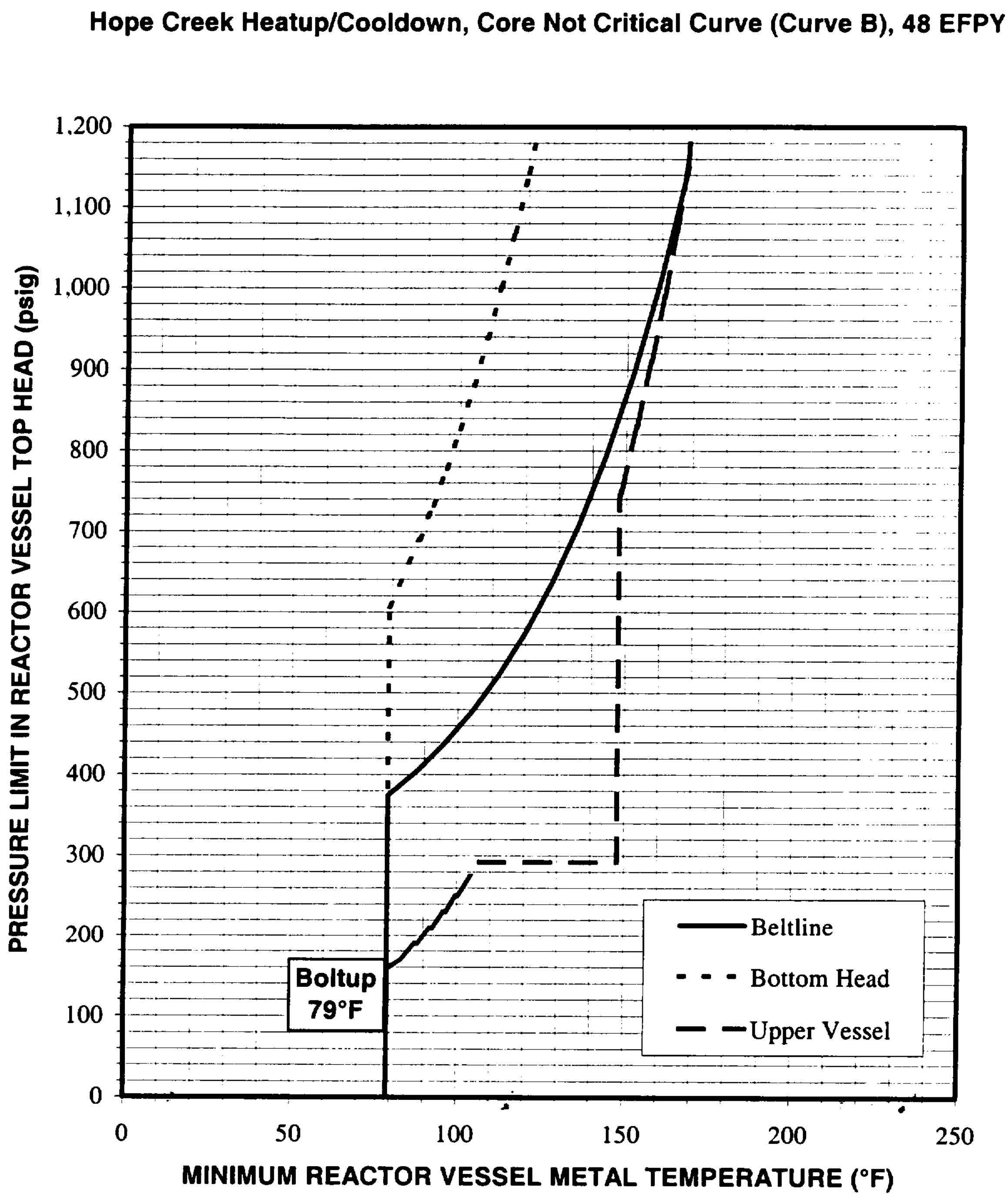


Figure 5
Core Critical Curve (Curve C) for 32 EFPY