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10 CFR 50.90

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U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Three Mile Island Nuclear Station, Unit 1
Facility Operating License No. DPR-50
NRC Docket No. 50-289

Subject: Technical Specification Change Request No. 335 – Reactor Coolant System Pressure-Temperature Safety Limit

In accordance with 10 CFR 50.90, "Application for amendment of license or construction permit," AmerGen Energy Company, LLC (AmerGen) hereby proposes changes to Appendix A, Technical Specifications (TS), of the Three Mile Island Nuclear Station, Unit 1 (TMI Unit 1) Facility Operating License.

The proposed changes would revise the TMI Unit 1 TS to incorporate a revised limit for the variable low reactor coolant system pressure-temperature core protection safety limit. The revised limit is associated with the introduction of AREVA NP's Mark-B-HTP fuel design in the TMI Unit 1 Cycle 17 reload (Fall 2007). The Mark-B-HTP fuel design incorporates the AREVA HTP spacer grid design, which reduces the likelihood of fuel rod defects related to spacer grid-to-rod fretting. Due to a higher pressure drop across the Mark-B-HTP fuel assemblies relative to resident fuel, mixed core thermal-hydraulic conditions require more restrictive Safety Limits and more restrictive Limiting Safety System Settings for the Reactor Protection System. The proposed limit is developed in accordance with the methods described in the NRC-approved Topical Report BAW-10179P-A, "Safety Criteria and Methodology for Acceptable Cycle Reload Analyses" using the BHTP CHF correlation described in the NRC-approved Topical Report BAW-10241P-A, "BHTP DNB Correlation Applied with LYNXT."

Enclosure 1 contains proprietary information as defined in 10 CFR 2.390(a)(4). Accordingly, it is requested that Enclosure 1 be withheld from public disclosure. An affidavit certifying the basis for this application for withholding as required by 10 CFR 2.390(b)(1) is also enclosed with this letter (Enclosure 3). Enclosure 2 provides a non-proprietary version of Enclosure 1.

Enclosure 4 contains the proposed TMI Unit 1 TS and Bases page markups.

The proposed amendment has been reviewed by the TMI Unit 1 Plant Operations Review Committee and approved by the Nuclear Safety Review Board in accordance with the requirements of the AmerGen Quality Assurance Program.

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Using the standards in 10 CFR 50.92, AmerGen has concluded that these proposed changes do not constitute a significant hazards consideration, as described in the enclosed analysis performed in accordance with 10 CFR 50.91(a)(1). Pursuant to 10 CFR 50.91(b)(1), a copy of this Technical Specification Change Request is provided to the designated official of the Commonwealth of Pennsylvania, Bureau of Radiation Protection, as well as the chief executives of the township and county in which the facility is located.

We request approval of the proposed changes by October 15, 2007, with the amendment being implemented within 30 days of issuance. This will allow completion of plant procedure revisions associated with this change that are needed to support the TMI Unit 1 Cycle 17 startup.

Regulatory commitments established by this submittal are identified in Enclosure 5. If you have any questions or require additional information, please contact David J. Distel at (610) 765-5517.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 22nd day of March, 2007.

Respectfully,

Pamela B. Cowan

Pamela B. Cowan
Director - Licensing and Regulatory Affairs
AmerGen Energy Company, LLC

Enclosures:
1) TMI Unit 1 Technical Specification Change Request No. 335 - Description and Assessment (Proprietary Version)
2) TMI Unit 1 Technical Specification Change Request No. 335 - Description and Assessment (Non-Proprietary Version)
3) AREVA NP Affidavit Certifying Request For Withholding From Public Disclosure
4) TMI Unit 1 Technical Specification Change Request No. 335 - Markup of Proposed Technical Specification and Bases Page Changes
5) List of Commitments

cc: S. J. Collins, Administrator, USNRC Region I
D. M. Kern, USNRC Senior Resident Inspector, TMI Unit 1
V. Nerses, USNRC Senior Project Manager, TMI Unit 1
D. Allard, Director, Bureau of Radiation Protection – Pennsylvania Department of Environmental Protection
Chairman, Board of County Commissioners of Dauphin County, PA
Chairman, Board of Supervisors of Londonderry Township, Dauphin County, PA
TMI Unit 1 File No. 07005

ENCLOSURE 2

TMI Unit 1 Technical Specification Change Request No. 335

Reactor Coolant System Pressure-Temperature Safety Limit

Description and Assessment

NON-PROPRIETARY VERSION

ENCLOSURE 2

DESCRIPTION AND ASSESSMENT

NON-PROPRIETARY

1.0 INTRODUCTION

In accordance with 10 CFR 50.90, "Application for amendment of license or construction permit," AmerGen Energy Company, LLC (AmerGen) is requesting an amendment to Facility Operating License No. DPR-50 for Three Mile Island Nuclear Station, Unit 1 (TMI Unit 1). The proposed amendment would revise the TMI Unit 1 TS to incorporate a revised limit for the variable low reactor coolant system pressure-temperature core protection safety limit. The revised limit is associated with the introduction of AREVA NP's Mark-B-HTP fuel design in Cycle 17, which is scheduled to begin in Fall 2007. The Mark-B-HTP fuel design incorporates the AREVA HTP spacer grid design, which reduces the likelihood of fuel rod defects related to spacer grid-to-rod fretting. Due to a higher pressure drop across the Mark-B-HTP fuel assemblies relative to resident fuel, mixed core thermal-hydraulic conditions require more restrictive Safety Limits and more restrictive Limiting Safety System Settings for the Reactor Protection System. The proposed limits are developed in accordance with the methods described in the NRC-approved Topical Report BAW-10179P-A, "Safety Criteria and Methodology for Acceptable Cycle Reload Analyses."

Introduction of the Mark-B-HTP fuel assembly at TMI Unit 1 requires use of a different departure from nucleate boiling (DNB) critical heat flux (CHF) correlation, namely the BHTP correlation. The AREVA BHTP CHF correlation is documented in AREVA Topical Report BAW-10241P-A, "BHTP DNB Correlation Applied with LYNXT," which has been previously reviewed and approved by the NRC and has been incorporated in BAW-10179P-A, "Safety Criteria And Methodology For Acceptable Cycle Reload Analyses." NRC approval of BAW-10241P-A is documented in NRC Safety Evaluation Reports dated September 29, 2004 and July 25, 2005. It is noted that the advanced M5 alloy, which is used for fuel rod cladding and HTP spacer grids in the Mark-B-HTP fuel design, was previously approved for use at TMI Unit 1 in License Amendment No. 233, dated May 10, 2001.

AmerGen requests that the following changed replacement pages be inserted into the existing Technical Specifications (TS):

Revised TMI Unit 1 TS Pages: vi, 2-1, 2-2, 2-3, 2-4a (Figure 2.1-1), 2-4c (Figure 2.1-3), 2-6, 2-7, 2-10 (Table 2.3-1), 2-11 (Figure 2.3-1), 4-2b, 4-4 (Table 4.1-1), and 4-7a (Table 4.1-1).

2.0 DESCRIPTION OF PROPOSED AMENDMENT

The proposed changes described below are shown on the marked-up TS pages provided in Enclosure 2.

2.1 Table of Contents, List of Tables, page vi is revised to correctly identify that Table 2.3-1, Reactor Protection System Trip Setting Limits, is provided on TS page 2-10. This is an editorial change only.

2.2 The Core Protection Safety Limit specified in TS Figure 2.1-1, Core Protection Safety Limit, and Bases Figure 2.1-3, Core Protection Safety Bases, are modified to incorporate a revised limit based on the implementation of the Mark-B-HTP fuel design in Cycle 17.

2.3 TS Table 2.3-1, Reactor Protection System Trip Setting Limits, and Figure 2.3-1, Protection System Maximum Allowable Setpoints, are modified to revise the variable low pressure trip setpoint to reflect implementation of the Mark-B-HTP fuel design in Cycle 17.

2.4 TS 2.1 Bases and 2.3 Bases are revised to incorporate reference to the Topical Report BAW-10241P-A, and add the DNBR design limit of 1.132 associated with the BHTP CHF correlation. In addition, references to the obsolete BAW-2 CHF correlation are deleted and references to specific maximum quality limits are removed. Section 2.3 Bases are also modified to reflect the revised variable low reactor coolant system pressure trip setpoint that results from the implementation of the Mark-B-HTP fuel design in Cycle 17. Due to the proprietary nature of the Statistical Design Limit (SDL) values, they are being removed from the applicable TS Bases pages.

2.5 TS Table 4.1-1, Instrument Surveillance Requirements, is modified to revise the reactor coolant pressure-temperature comparator's surveillance requirements to reflect the as-found/as-left acceptance criteria. Section 4.1 Bases are also modified to include a description of the as-found / as-left acceptance criteria and reference to the document that contains the Limiting Trip Setpoint and the methodology used to determine the Limiting Trip Sepoint, pre-defined as-found acceptance band, and as-left setting tolerance band.

3.0 BACKGROUND

The TMI Unit 1 thermal and hydraulic core reload design and evaluation is described in TMI Unit 1 Updated Final Safety Analysis Report (UFSAR) Section 3.2.3. The criterion for the heat transfer design is to be safely below departure from nucleate boiling (DNB) heat flux at the design overpower.

The reactor core safety limit restrictions prevent overheating of the fuel cladding and possible cladding perforation, which could result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime would result in excessive cladding temperatures because of the onset of departure from nucleate boiling

(DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore thermal power, Reactor Coolant System Flow, and Reactor Coolant (RC) Temperature and Pressure have been related to DNB using critical heat flux (CHF) correlations. The local DNB heat flux ratio, DNBR, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB.

4.0 TECHNICAL ANALYSIS

The revised core protection safety limit and bases proposed in this amendment were developed using AREVA NP's NRC-approved reload methodology (BAW-10179P-A) and NRC-approved core thermal-hydraulic code LYNXT (BAW-10156P-A). The revised core protection safety limit protects the DNBR design limits, which were reanalyzed using AREVA NP's NRC-approved Statistical Core Design (SCD) methodology for 177 fuel assembly (FA) B&W plants (BAW-10187P-A) for the Mark-B-HTP fuel design. The same reload, LYNXT, and SCD methodologies are the basis for the current TMI Unit 1 core protection safety limits.

A Statistical Design Limit (SDL) of [ ] was determined for the Mark-B-HTP fuel design. All core DNB analyses supporting the proposed amendment were performed with additional retained margin in the form of a Thermal Design Limit (TDL) of [ ]. This margin is specifically retained to offset effects not treated in the SDL development, such as transition core effects, deviations in uncertainty values from those incorporated in the SDL, or other cycle-specific emergent issues. The SDL limit and the TDL retained margin approach used in developing the revised core protection safety limit and bases are from the NRC-approved statistical core design methodology. In accordance with the restrictions contained in the NRC Safety Evaluation Reports (NRC letters dated March 24, 1993 and March 17, 1994), application of BAW-10187P-A with a hot pin SDL of [ ] is acceptable for TMI Unit 1 for the following reasons:

- The nominal values and ranges for the state parameters and uncertainty parameters described in Tables 3-4 and 3-6 of BAW-10187P-A that were used in developing the SDL of [ ] are all applicable to TMI Unit 1.

- The response surface model was validated for the Mark-B-HTP design, for which the BHTP CHF correlation has been approved. The Mark-B-HTP fuel design and the BHTP CHF correlation were assumed in the development of the SDL of [ ].

- The LYNXT core thermal-hydraulic code is used for core DNB calculations.

- Core state variables that were not included in the statistical design were input to thermal-hydraulic computer codes at their most adverse allowable level. In addition, a more conservative value was assumed in the analysis for the following parameter; this value is well within the range analyzed in developing the SDL:

  - Since TMI Unit 1 has been approved to operate with up to 20% average OTSG tube plugging and a minimum reactor coolant system (RCS) flow rate of 102% of design flow (including a 2.5% measurement uncertainty) (License Amendment

No. 214, dated August 19, 1999), a nominal RCS flow rate of 104.5% of design flow (the 2.5% RCS flow measurement uncertainty is included in the SDL) was assumed in all core T-H analyses supporting the proposed amendment. A nominal RCS flow rate of 106.5% of design flow was used to develop the SDL.

- Cycle-specific evaluations will be performed for each reload to determine if the bounding assembly-wise power distribution assumed in the core-wide SDL calculation bounds the expected operating power distributions.

Transition Core DNBR Penalty

The DNBR transition core penalty has been determined for TMI Unit 1 based on the number of Mark-B-HTP fuel assemblies residing in the core. For Cycle 17, a minimum of 72 Mark-B-HTP fuel assemblies will be loaded into the core and the transition core DNB penalty will be approximately [ ] DNB points, where 1 DNB point = 0.01. The transition core penalty will be applied to the DNB margin for predictions made using a full core of Mark-B-HTP fuel.

The transition core penalty relationship was developed by first qualifying the DNB performance for a full core of Mark-B-HTP fuel. The DNB performance was analyzed for statepoint conditions from the core safety limits, for the limiting Condition I/II DNB transient, and for the range of axial power shapes that are used for establishing the core safety limits and core operating limits. Next, a transition core model was examined in which a certain number of the Mark-B-HTP fuel assemblies were placed into the core of the resident fuel in a conservative manner to allow flow diversion out of the limiting Mark-B-HTP fuel assembly. Placement of the limiting Mark-B-HTP fuel assembly at the center of the core and surrounding it with the resident fuel design, with the remaining Mark-B-HTP fuel assemblies placed on the core periphery, results in the lowest DNB prediction for the limiting Mark-B-HTP fuel assembly in the transition core. The DNB behavior is a result of the Mark-B-HTP fuel assembly having a higher pressure drop than the resident fuel design, thereby creating a higher flow diversion out of the Mark-B-HTP fuel.

The transition core model DNB performance was analyzed for the same range of statepoint conditions, transients, and range of axial power shapes studied for the full core model. The largest DNBR difference between the limiting Mark-B-HTP fuel rod in a full core model (of Mark-B-HTP fuel) and a specific transition core model for all of the statepoints, transients, and axial power shapes was defined as the transition core penalty for the specific transition core model.

As stated above, the transition core penalty for TMI Unit 1 Cycle 17 using a minimum of 72 Mark-B-HTP fuel assemblies will be [ ] DNB points. The transition core penalty will be smaller in subsequent cycles as the number of Mark-B-HTP fuel assemblies increases in the core. Therefore, a [ ] TDL for DNB analyses provides sufficient margin to the SDL of [ ] (i.e., [ ] DNB points of margin) to provide protection for TMI Unit 1 cores transitioning from Mark-B to Mark-B-HTP fuel designs.

Derivation of Variable Low Pressure Trip Limiting Setpoint

The variable low pressure trip is a Safety Limit-related trip since it initiates an automatic reactor trip and provides reactor protection for the core safety limit contained in TS Figure 2.1-1, Core Protection Safety Limit, which ensures margin to core DNBR limits.

The method for development of the proposed limits specified in TS Table 2.3-1, Reactor Protection System Trip Setting Limits, for the variable low reactor coolant system pressure, is described in AREVA NP's NRC-approved reload methodology Topical Report BAW-10179P-A, Section 7.6, "Variable Low RC Pressure Trip." This same methodology was used to develop the current TMI Unit 1 variable low reactor coolant system pressure limit.

The variable-low-pressure trip setpoint is based on a conservatively bounding straight-line approximation of the core exit pressure-temperature limits specified by DNB analyses. A sample curve is depicted in referenced Topical Report BAW-10179P-A, Figure 7-5. The points at which the pressure-temperature limits intersect the high reactor coolant (RC) outlet temperature and low RC pressure trips are labeled A and B. Points A and B are adjusted for instrument string error and the adjusted values are used to derive the straight-line equation for the variable low pressure trip setpoint. The slope of the line is limited to ensure that the setpoint is within the capability of the instrumentation. If the slope exceeds the maximum allowable slope, the line is conservatively rotated about the upper point as shown in referenced Topical Report BAW-10179P-A, Figure 7-6, until the slope is acceptable.

The updated instrumentation setpoint calculation is prepared in accordance with American National Standards Institute (ANSI) / Instrument Society of America (ISA) Standard 67.04.01-2000, "Setpoints for Nuclear Safety-Related Instrumentation," and Recommended Practice ISA-RP67.04.02-2000, "Methodologies for the Determination of Setpoints for Nuclear Safety-Related Instrumentation." The applicable portions of ANSI/ISA-67.04.01-2000 and ISA-RP67.04.02-2000 are equivalent to the corresponding NRC-endorsed sections of ANSI/ISA-S67.04-Part 1-1994. The calculations are consistent with Method 1 in ISA-RP67.04-2000 Section 7.3. Using this method, the hardware and process measurement error for the individual modules in the instrument string (i.e., this does not include drift, calibration uncertainties, and uncertainties observed during normal operations) are combined and added to the Analytical Limit (AL) represented by points A and B. The instrument error is calculated such that uncertainties that are random, normally distributed, and independent are combined by the square-root-sum-of-squares (SRSS) method. Bias or correlated terms are added following the SRSS combination. The variable low reactor coolant system pressure trip limit in TS Table 2.3-1, Reactor Protection System Trip Setting Limits, and Figure 2.3-1, Protection System Maximum Allowable Setpoints, is the Allowable Value (AV) or Limiting Safety System Setting (LSSS).

Methodology for Determining the VLPT Nominal Setpoint and As-Found / As-Left Acceptance Criteria

The VLPT Nominal Setpoint (NSP) and Total Loop Uncertainty (TLU) are determined using the methodology provided by ANSI/ISA-S67.04-Part 1-1994. The Nominal Setpoint (NSP) is the ideal setpoint for RPS calibration. The setpoint slope has been reduced to 75% of the RPS instrument capability. This slope is chosen to be well within the reliable adjustment range of the instrument to ensure accurate calibration. The slope is reduced by rotating the AV linear equation clockwise around the upper point where RCS temperature is equal to the reactor coolant high temperature trip limit. This is conservative with respect to the AV. A Total Margin of 40 psig is then conservatively added to the AV y-intercept of the linear equation to obtain the NSP. The Total Margin includes:

1. Total Loop Uncertainty (TLU): ± 28.148 psig
2. Surveillance Test Procedures NSP As-Left Tolerance: ± 1.6 psig
3. Additional Discretionary Margin: 10 psig

The principle used in determining the NSP is that the surveillance test as-found Trip Setpoint (TSP) shall not exceed the AV. Therefore, the total loop uncertainty is applied in determining the Total Margin so that the NSP protects the AV. The surveillance test procedures NSP as-left tolerance is included in the Total Margin because it is not included in the AV calculation. The additional discretionary margin provides additional conservatism to ensure the as-found setpoint will not exceed the LSSS during the surveillance interval. The resulting NSP does not significantly impact the normal plant-operating region.

The pre-defined limits for the NSP as-found tolerance will be determined in accordance with the NRC accepted methodology described in NRC RIS 2006-17. A sample calculation demonstrating the methodology is provided in Attachment 1. The as-found tolerance is based on the statistical combination of the accuracy of the applicable loop components, the test equipment accuracy, and drift. The pre-defined test acceptance criteria band for the as-found value is less than or equal to the square root of the sum of the squares of reference accuracy, M&TE accuracy, readability uncertainties and drift. Readability uncertainties are zero. Based on the sample calculation, the pre-defined NSP as-found tolerance is approximately ± 4.50 psig.

The surveillance test procedures NSP as-left tolerance is based on the statistical combination of the accuracy of the applicable loop components and the test equipment accuracy. Based on the sample calculation, the as-left tolerance is approximately ± 3.12 psig. The actual NSP as-left tolerance specified in the surveillance test procedures is ± 20 mVDC (± 1.6 psig). This tolerance is smaller (more conservative) than the sample calculation as-left tolerance and is based on typical loop performance during the surveillance test interval.

The surveillance test procedures do not compare the as-found TSP to the previous surveillance test as-left TSP. Basing operability determinations for the as-found and the as-left TSP on the NSP is acceptable because:

1. The NSP as-left tolerance specified in the surveillance test procedures is ± 20 mVDC (± 1.6 psig). This is less than the sample calculation NSP as-left tolerance of approximately ± 3.12 psig.
2. The NSP as-left tolerance is not included in the TLU calculation. This is acceptable because the NSP as-left tolerance specified in the surveillance test

procedures is approximately half of the calculated NSP as-left tolerance. This prevents masking of excessive drift from one side of the tolerance band to the other.

3. The pre-defined NSP as-found tolerance is based on the square root of the sum of the squares of the instrument accuracy, M&TE accuracy and drift. The NSP as-left tolerance is not included in this calculation.

<u>VLPT Instrument Surveillance Requirements</u>

The proposed instrument surveillance requirements for the reactor coolant pressure-temperature comparator in TS Table 4.1-1, Instrument Surveillance Requirements, comply with NRC RIS 2006-17, "NRC Staff Position on the Requirements of 10 CFR 50.36, "Technical Specifications," Regarding Limiting Safety System Settings During Periodic Testing and Calibration of Instrument Channels," dated August 24, 2006, which ensures compliance with 10 CFR 50.36. The following list of proposed surveillance actions are similar to those proposed in Technical Specification Task Force (TSTF)-493, Revision 0, "Clarify Application of Setpoint Methodology for LSSS Functions."

If the surveillance test as-found Trip Setpoint exceeds the Allowable Value, the TS and the surveillance test procedures will require that:

1. The required TS actions are taken.
2. The instrument channel is declared inoperable pending further evaluation and calibration.
3. Enter the condition into the Corrective Action Program.

If the as-found Trip Setpoint is conservative with respect to the AV, but exceeds the pre-defined limits for as-found tolerance, the TS and the surveillance test procedures will require:

1. Determination if the instrument is functioning as required prior to returning the VLPT channel to service. If it cannot be determined that the instrument is functioning as required, declare VLPT channel inoperable.
2. The TSP to be reset within the surveillance test procedures as-left setting tolerance band.
3. Enter the condition into the Corrective Action Program.

If the as-found Trip Setpoint is conservative with respect to the pre-defined limits, but exceeds the as-left setting tolerance band, the TS and the surveillance test procedures will require:

1. The TSP to be reset within the surveillance test procedures as-left setting tolerance band.
2. The condition to be forwarded to the System Engineer for evaluation as outlined in BAW-10167A, "Justification for Increasing the Reactor Trip System On-Line Test Intervals."

Non-LOCA Analyses

AREVA NP's NRC-approved reload methodology (BAW-10179P-A) is used for each reload to verify that the overall conservatism of the boundary conditions and key input parameters used in the non-LOCA UFSAR Chapter 14 analyses is maintained, and to calculate/verify the reactor protection system setpoints. All of the UFSAR Chapter 14 non-LOCA events were evaluated with respect to the Mark-B-HTP fuel design. It was concluded that the fuel design has a negligible effect on the overall system response with the exception of the different CHF correlation that is applied to the fuel. Consequently, the high- and low- RCS pressure setpoints, the high temperature setpoint, the high containment pressure and the high flux setpoints in TS Table 2.3-1, Reactor Protection System Trip Setting Limits, would not be affected by the change in fuel design. The Power/Flow trip setpoints in TS Table 2.3-1, Reactor Protection System Trip Setting Limits, are included in the Core Operating Limits Report and therefore are subject to change each reload. The Power/Flow trip setpoint determination requires a cycle-specific evaluation of the loss-of-coolant flow transients using the appropriate CHF correlation as described in BAW-10179P-A Section 6.5. The ejected rod DNB analysis, using the BHTP CHF correlation, results in a smaller percentage of fuel rods that experience DNB compared to the current analysis of record, and therefore the ejected rod analysis of record remains bounding. The CHF correlation for Mark-B-HTP fuel does require a revised limit for the variable low reactor coolant system pressure-temperature core protection safety limit, as proposed in this amendment request.

LOCA Analyses

Loss-of-coolant accident (LOCA) analyses for the transition and full core of Mark-B-HTP fuel are performed with the NRC-approved AREVA Topical Report BAW-10192P-A, "BWNT LOCA - BWNT Loss-of-Coolant Accident Evaluation Model for Once-Through Steam Generator Plants". This Evaluation Model (EM) and associated code topical reports have been incorporated into BAW-10179P-A, "Safety Criteria and Methodology for Acceptable Cycle Reload Analyses". This EM is the same as that utilized to evaluate the resident fuel designs; however, the BHTP critical heat flux correlation described in BAW-10241P-A, Revision 1, "BHTP DNB Correlation Applied with LYNXT," has been added as required by the EM methods specifically for the analysis of the Mark-B-HTP fuel assembly. Analyses performed for the first cycle of implementation are performed based on the currently installed once-through steam generator (OTSG) design and will specifically consider the mixed-core configuration. For the second cycle of implementation, steam generator replacements are planned. Therefore, the LOCA analyses for the second and subsequent cycles of operation with Mark-B-HTP fuel will consider the new replacement steam generator design. Future LOCA analyses addressing the replacement steam generator design will specifically consider both a mixed-core and a full-core of Mark-B-HTP assemblies.

Conclusion

In conclusion, implementation of the Mark B-HTP fuel design for Cycle 17 requires adoption of a more conservative safety limit curve to maintain the same magnitude of DNB protection. Therefore, TS Figure 2.1-1, Core Protection Safety Limit, requires revision. The change to the core protection safety limit curve requires a related change to the variable low pressure trip setpoint. Therefore, TS Figure 2.3-1, Protection System Maximum Allowable Setpoints, and TS Table 2.3-1, Reactor Protection System Trip Setting Limits, require revision. Variable low pressure trip setpoint surveillance requirements, established in accordance with NRC RIS 2006-17, "NRC Staff Position on the Requirements of 10 CFR 50.36, "Technical Specifications," Regarding Limiting Safety System Settings During Periodic Testing and Calibration of Instrument Channels," dated August 24, 2006, require the addition of pre-defined acceptance criteria to TS Table 4.1-1, Instrument Surveillance Requirements. The proposed changes to the TMI Unit 1 TS have been established to assure adequate margins of safety are maintained and have been developed in accordance with NRC-approved codes and methodologies. Therefore, the proposed changes do not adversely affect nuclear safety or safe plant operations.

5.0 <u>REGULATORY ANALYSIS</u>

5.1 No Significant Hazards Consideration

AmerGen has evaluated whether or not a significant hazards consideration is involved with the proposed amendments by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. ***Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?***

   Response: No.

   The proposed Technical Specification (TS) limits and reactor protection system (RPS) trip setpoints are developed in accordance with the methods and assumptions described in NRC-approved AREVA NP Topical Reports BAW-10179 P-A, "Safety Criteria and Methodology for Acceptable Cycle Reload Analyses" and BAW-10187 P-A, "Statistical Core Design for B&W-Designed 177 FA Plants." The core thermal-hydraulic code (LYNXT) and CHF correlation (BHTP) have been approved for use with these methods and the Mark-B-HTP fuel type. The proposed change preserves the design DNB Ratio safety criterion that there shall be at least a 95% probability at a 95% confidence level that the hot fuel rod in the core does not experience a departure from nucleate boiling during normal operation or events of moderate frequency. The corresponding core-wide protection on a pin-by-pin basis is greater than 99.9%. The margin retained for penalties such as transition core effects, by imposing a Thermal Design Limit in all DNB analyses supporting the proposed change, has been shown to be sufficient to offset the mixed core conditions at TMI Unit 1, where the Mark-B-HTP fuel design will be co-resident with earlier Mark-B fuel designs. The setpoint calculation methodology utilized, and the surveillance requirements established, are in accordance with approved industry standards and NRC criteria.

   The proposed setpoint change does not involve a significant increase in the consequences of an accident previously evaluated because the proposed change does not alter any assumptions previously made in the radiological consequence evaluations, or affect mitigation of the radiological consequences of an accident previously evaluated.

Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The proposed TS limit and reactor protection system (RPS) trip setpoint provide a core protection safety limit and variable low pressure trip setpoint developed in accordance with NRC-approved methods and assumptions. No new accident scenarios, failure mechanisms or single failures are introduced as a result of the proposed change. All systems, structures, and components previously required for the mitigation of an event remain capable of fulfilling their intended design function.

Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed amendment involve a significant reduction in a margin of safety?

Response: No.

The proposed RPS trip setpoint ensures core protection safety limits will be preserved during power operation. The proposed safety limit and setpoint are developed in accordance with NRC-approved methods and assumptions. The margin retained for penalties such as transition core effects, by imposing a Thermal Design Limit in all DNB analyses supporting the proposed change, has been shown to be sufficient to offset the mixed core conditions at TMI Unit 1. The setpoint calculation methodology utilized, and the surveillance requirements established, are in accordance with approved industry standards and NRC criteria.

Therefore, the proposed changes do not involve a significant reduction in any margin of safety.

Based on the above, AmerGen concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

5.2 Applicable Regulatory Requirements/Criteria

AmerGen has determined that the proposed change does not require any exemptions or relief from regulatory requirements and does not affect conformance with any General Design Criteria. The proposed change is consistent with the criteria specified in 10 CFR 50.36(c)(2)(ii) for inclusion of items in TS.

10 CFR 50, Appendix A, General Design Criteria (GDC) 10 requires that the reactor core and associated coolant, control, and protective systems be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated occurrences. A reactor safe operating power has been determined by the ability of the coolant to remove heat from the fuel material. The criterion that best measures this ability is the DNB, which involves the individual parameters of heat flux, coolant temperature rise, and flow area. The DNB criterion is commonly applied through the use of the DNBR. This is the minimum ratio of the critical heat flux (as computed by the DNB correlation) to the surface heat flux. The ratio is a measure of the margin between the operating power and the power at which DNB might be expected to occur in that channel. The DNBR varies over the channel length, and it is the minimum value of the ratio in the channel of interest that is used. Consistent with the specified acceptable fuel design limit of NRC Standard Review Plan (NUREG-0800), a calculated DNBR value greater than the DNBR design limit provides assurance that there is at least a 95% probability at the 95% confidence level that a departure from nucleate boiling will not occur on the hot fuel pin.

The NRC Safety Evaluation Report (SER) dated March 24, 1993, "Acceptance For Referencing of Topical Report BAW-10187P, Statistical Core Design For B&W-Designed 177 FA Plants," specifies restrictions applicable to use of this methodology. These restrictions have been addressed in the core reload analysis application for TMI Unit 1. In addition, cycle-specific checks on assembly-wise power distribution will be made on a core reload basis.

Setpoint calculation methodology and the proposed instrument surveillance requirements are consistent with the criteria contained in NRC RIS 2006-17, "NRC Staff Position on the Requirements of 10 CFR 50.36, "Technical Specifications," Regarding Limiting Safety System Settings During Periodic Testing and Calibration of Instrument Channels," dated August 24, 2006 and ensure compliance with 10 CFR 50.36.

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

6.0 ENVIRONMENTAL CONSIDERATION

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

7.0 PRECEDENT

The Mark-B-HTP CHF correlation (BAW-10241P-A) has been implemented at three other B&W 177 FA plants for which AREVA NP performs reload licensing (i.e., Arkansas Nuclear One Unit 1, Crystal River Unit 3, and Davis-Besse). The proposed TMI Unit 1 Variable Low Pressure Trip setpoint change is similar to the change implemented for the Davis-Besse plant, approved by the NRC in Amendment No. 274, dated March 2, 2006.

8.0 REFERENCES

BAW-10187P-A, "Statistical Core Design for B&W-Designed 177 FA Plants," B&W Fuel Company, Lynchburg, Virginia, March 1994.

BAW-10179P-A, Rev. 6, "Safety Criteria And Methodology For Acceptable Cycle Reload Analyses," AREVA NP, Lynchburg, Virginia, August 2005.

BAW-10156P-A, Revision 1, "LYNXT Thermal-Hydraulics Code," Framatome Cogema Fuels, Lynchburg, Virginia, February 1996.

BAW-10241P-A, Revision 1, "BHTP DNB Correlation Applied with LYNXT," August 2005.

BAW-10167A, Revision 1, "Justification for Increasing the Reactor Trip System On-Line Test Intervals," dated February 10, 1998.

Letter from A. C. Thadani (NRC) to J. H. Taylor (BWFC), "Acceptance for Referencing of Topical Report BAW-10187P, Statistical Core Design for B&W-Designed 177 FA Plants," (TAC No. M85118), March 24, 1993.

Letter from M. J. Virgilio (NRC) to J. H. Taylor (BWFC), "Acceptance for Referencing of Appendix F to Topical Report BAW-10187P, Statistical Core Design For B&W-Designed 177 FA Plants," (TAC No. M88899), March 17, 1994.

Letter from T. G. Colburn (NRC) to M. E. Warner (AmerGen), "TMI-1 Amendment No. 233 – Expanded Use of M5 Cladding Alloy (TAC NO. MB0788)," dated May 10, 2001.

NRC Regulatory Issue Summary (RIS) 2006-17, "NRC Staff Position on the Requirements of 10 CFR 50.36, "Technical Specifications," Regarding Limiting Safety System Settings During Periodic Testing and Calibration of Instrument Channels," dated August 24, 2006.

ANSI/ISA-67.04.01-2006, "Setpoints for Nuclear Safety-Related Instrumentation."

USNRC Regulatory Guide 1.105, Revision 3, "Setpoints for Safety-Related Instrumentation."

ANSI/ISA-67.04.01-2000, "Setpoints for Nuclear Safety-Related Instrumentation."

Recommended Practice ISA-RP67.04.02-2000, "Methodologies for the Determination of Setpoints for Nuclear Safety-Related Instrumentation."

Letter from S. P. Sands (NRC) to M. B. Bezilla (FENOC), "Davis-Besse Nuclear Power Station, Unit 1-Issuance of Amendment Re: Framatome Mark B-HTP Fuel Design for Cycle 15 (TAC NO. MC6888)."

Sample Calculation Methodology

The following is a sample calculation of the proposed Reactor Protection System (RPS) Variable Low Pressure Trip (VLPT) setpoint with regard to the concepts discussed in NRC Regulatory Issue Summary (RIS) 2006-17.

Nominal Trip Setpoint (NSP) As-Left Tolerance:

The surveillance test procedures NSP as-left tolerance is based on the statistical combination of the accuracy of the applicable loop components and the test equipment accuracy.

From AREVA NP Document 32-1151224-02 Page 16, the equation for addition of errors is:

$$E_{out} = \pm\ SCGain * (e_{RTD} + e_{SCZero}) + e_{SCGain} * (IN + e_{RTD} + e_{SCZero}) + e_{SCBias} + e_{SCPSP} + (1 + e_{BAScale}) * (e_{1152} + e_{1152Cal} + e_{BABias} + e_{BABal}) + (IN * e_{BAScale}) + e_{BA} + e_{Blesp}$$

Applying RSS method and separating correlated terms:

$$E_{out} = \pm\ [SCGain^2 * (e_{RTDRand}^2 + e_{SCZero}^2) + e_{SCGain}^2 * (IN^2 + e_{RTDRand}^2 + e_{SCZero}^2 + e_{RTDCorr}^2) + e_{SCBias}^2 + e_{SCPSP}^2{}_{Rand} + (1 + e_{BAScale}^2) * (e_{1152Rand}^2 + e_{1152Cal}^2 + e_{BABias}^2 + e_{BABal}^2) + (e_{1152Corr})^2 * (e_{BAScale})^2 + (IN * e_{BAScale})^2 + e_{BARand}^2 + e_{Blesp}^2]^{1/2} + (SCGain * e_{RTDCorr}) + e_{SCPSPCorr} + e_{1152Corr} + e_{BACorr} + e_{BlespCorr}$$

The surveillance test procedures use the signal injection method. Terms associated with the transmitter are eliminated. The terms for the RTD and linear bridge reference accuracy are retained because AREVA NP Document 32-1151224-02 provides data only for the combination of these instruments. All other terms except those relating to the reference accuracy of the applicable loop components and the associated M&TE accuracy are eliminated. Readability uncertainties are zero:

$$E_{out} = \pm\ [SCGain^2 * (e_{RTDRand}^2 + e_{SCZero}^2) + e_{SCGain}^2 * (IN^2 + e_{RTDRand}^2 + e_{SCZero}^2) + e_{SCBias}^2 + e_{SCPSP}^2{}_{Rand} + (1 + e_{BAScale}^2) * (e_{BABias}^2 + e_{BABal}^2) + (IN * e_{BAScale})^2 + e_{BARand}^2 + e_{Blesp}^2]^{1/2}$$

SCGain is recalculated based on a slope of 14.29. Applying the error values from AREVA NP Document 32-1151224-02:

$$E_{out} = \pm\ [1.786^2 * (0.0015^2 + 0.0001^2) + 0.0005^2 * (1^2 + 0.0015^2 + 0.0001^2) + 0.0005^2 + 0.0015^2 + (1 + 0.0003^2) * (0.0001^2 + 0.0001^2) + (1 * 0.0003)^2 + 0.0015^2 + 0.0017^2]^{1/2}$$

$$\mathbf{E_{out} = \pm\ 0.0039 = \pm\ 3.12\ psig}$$

The actual NSP as-left tolerance specified in the surveillance test procedures is ± 20 mVDC (± 1.6 psig). This tolerance is smaller (more conservative) than the error calculated above and is based on typical loop performance during the surveillance test interval.

Nominal Trip Setpoint (NSP) As-Found Tolerance:

The pre-defined NSP as-found tolerance is based on the statistical combination of the accuracy of the applicable loop components, the test equipment accuracy, and drift:

$$E_{out} = \pm\ [SCGain^2 * (e_{RTDRand}^2 + e_{SCZero}^2) + e_{SCGain}^2 * (IN^2 + e_{RTDRand}^2 + e_{SCZero}^2) + e_{SCBias}^2 + e_{SCPSP}^2{}_{Rand} + (1 + e_{BAScale}^2) * (e_{BABias}^2 + e_{BABal}^2) + (IN * e_{BAScale})^2 + e_{BARand}^2 + e_{Blesp}^2]^{1/2}$$

Where necessary, drift has been recalculated for the six-month surveillance test interval. Where not specified, drift is assumed to be for 30 days. Drift for the six-month surveillance test interval is calculated assuming that drift is a linear function of time. Applying the values from AREVA NP Document 32-1151224-02:

$$E_{out} = \pm\ [1.786^2 * (0.00189^2 + 0.0001^2) + 0.0005^2 * (1^2 + 0.00189^2 + 0.0001^2) + 0.0005^2 + 0.00234^2 + (1 + 0.0003^2) * (0.0001^2 + 0.0001^2) + (1 * 0.0003)^2 + 0.00283^2 + 0.00248^2]^{1/2}$$

$$\mathbf{E_{out} = \pm\ 0.00563 = \pm\ 4.50\ psig}$$

ENCLOSURE 3

AREVA NP
AFFIDAVIT CERTIFYING REQUEST FOR WITHHOLDING FROM PUBLIC DISCLOSURE

AFFIDAVIT

COMMONWEALTH OF VIRGINIA)
) ss.
CITY OF LYNCHBURG)

1. My name is Gayle F. Elliott. I am Manager, Product Licensing, for AREVA NP Inc. (AREVA NP) and as such I am authorized to execute this Affidavit.

2. I am familiar with the criteria applied by AREVA NP to determine whether certain AREVA NP information is proprietary. I am familiar with the policies established by AREVA NP to ensure the proper application of these criteria.

3. I am familiar with the AREVA NP information contained in the Enclosure to Letter Number 5928-07-20006, "Technical Specification Change Request No. 335 – Reactor Coolant System Pressure – Temperature Safety Limit," for Three Mile Island Nuclear Station, Unit 1, Facility Operating License No. DPR-50, NRC Docket No. 50-289, referred to herein as "Document." Information contained in this Document has been classified by AREVA NP as proprietary in accordance with the policies established by AREVA NP for the control and protection of proprietary and confidential information.

4. This Document contains information of a proprietary and confidential nature and is of the type customarily held in confidence by AREVA NP and not made available to the public. Based on my experience, I am aware that other companies regard information of the kind contained in this Document as proprietary and confidential.

5. This Document has been made available to the U.S. Nuclear Regulatory Commission in confidence with the request that the information contained in this Document be withheld from public disclosure. The request for withholding of proprietary information is made in

accordance with 10 CFR 2.390. The information for which withholding from disclosure is requested qualifies under 10 CFR 2.390(a)(4) "Trade secrets and commercial or financial information."

6. The following criteria are customarily applied by AREVA NP to determine whether information should be classified as proprietary:

(a) The information reveals details of AREVA NP's research and development plans and programs or their results.

(b) Use of the information by a competitor would permit the competitor to significantly reduce its expenditures, in time or resources, to design, produce, or market a similar product or service.

(c) The information includes test data or analytical techniques concerning a process, methodology, or component, the application of which results in a competitive advantage for AREVA NP.

(d) The information reveals certain distinguishing aspects of a process, methodology, or component, the exclusive use of which provides a competitive advantage for AREVA NP in product optimization or marketability.

(e) The information is vital to a competitive advantage held by AREVA NP, would be helpful to competitors to AREVA NP, and would likely cause substantial harm to the competitive position of AREVA NP.

The information in the Document is considered proprietary for the reasons set forth in paragraphs 6(b) and 6(c) above.

7. In accordance with AREVA NP's policies governing the protection and control of information, proprietary information contained in this Document have been made available, on a limited basis, to others outside AREVA NP only as required and under suitable agreement providing for nondisclosure and limited use of the information.

8. AREVA NP policy requires that proprietary information be kept in a secured file or area and distributed on a need-to-know basis.

9. The foregoing statements are true and correct to the best of my knowledge, information, and belief.

SUBSCRIBED before me this 21st

day of March, 2007.

Sherry L. McFaden
NOTARY PUBLIC, COMMONWEALTH OF VIRGINIA
MY COMMISSION EXPIRES: 10/31/10

ENCLOSURE 4

TMI Unit 1 Technical Specification Change Request No. 335

Markup of Proposed Technical Specifications and Bases Page Changes

Revised Technical Specifications & Bases Pages

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Amendment No. ~~59, 72, 100, 106, 118, 137, 142, 147, 150, 173, 197, 216~~,

2. SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS, REACTOR CORE

Applicability

Applies to reactor thermal power, axial power imbalance, reactor coolant system pressure, coolant temperature, and coolant flow during power operation of the plant.

Objective

To maintain the integrity of the fuel cladding.

Specification

2.1.1 The combination of the reactor system pressure and coolant temperature shall not exceed the safety limit as defined by the locus of points established in Figure 2.1-1. If the actual pressure/temperature point is below and to the right of the line, the safety limit is exceeded.

2.1.2 The combination of reactor thermal power and axial power imbalance (power in the top half of core minus the power in the bottom half of the core expressed as a percentage of the rated power) shall not exceed the protective limit as defined by the locus of points (solid line) for the specified flow set forth in the Axial Power Imbalance Protective Limits given in the Core Operating Limits Report (COLR). If the actual-reactor- thermal-power/axial-power-imbalance point is above the line for the specified flow, the protective limit is exceeded.

Bases

To maintain the integrity of the fuel cladding and to prevent fission product release, it is necessary to prevent overheating of the cladding under normal operating conditions. This is accomplished by operating within the nucleate boiling regime of heat transfer, wherein the heat transfer coefficient is large enough so that the clad surface temperature is only slightly greater than the coolant temperature. The upper boundary of the nucleate boiling regime is termed, departure from nucleate boiling (DNB). At this point there is a sharp reduction of the heat transfer coefficient, which could result in excessive cladding temperature and the possibility of cladding failure. Although DNB is not an observable parameter during reactor operation, the observable parameters of neutron power, reactor coolant flow, temperature, and pressure can be related to DNB through the use of a critical heat flux (CHF) correlation. The ~~BAW-2~~ (Reference 1) and BWC (Reference 2) correlations have been developed to predict DNB and the location of DNB for axially uniform and non-uniform heat flux distributions. The ~~BAW-2~~ correlation applies to Mark-B fuel with ~~inconel~~ intermediate spacer grids and the BWC correlation applies to Mark-B fuel with zircaloy or M5 intermediate spacer grids (non-mixing vane). The local DNB ratio (DNBR), defined as the ratio of the heat flux that would cause DNB at a particular core location to the actual heat flux, is indicative of the margin to DNB. The minimum value of the DNBR, accounting only for DNBR correlation uncertainty, during steady-state operation, normal

BHTP

Zircaloy or M5 HTP

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operational trans~~ents~~ing, and anticipated transients is limited to ~~1.30 (BAW-2)~~ 1.132 (BHTP) and 1.18 (BWC). ~~This~~ Corresponding ~~to a~~ Statistical Design Limits ~~(SDL) of 1.313 (BWC) which~~ account~~s~~ for all uncertainties considered with the statistical core design methodology (Reference 4). A DNBR of ~~1.30 (BAW-2)~~ 1.132 (BHTP) or 1.18 (BWC) corresponds to a 95 percent probability at a 95 percent confidence level that DNB will not occur; this is considered a conservative margin to DNB for all operating conditions. The difference between the actual core outlet pressure and the indicated reactor coolant system pressure has been considered in determining the core protection safety limits.

The curve presented in Figure 2.1-1 represents the conditions at which the minimum allowable DNBR or greater is predicted for the limiting combination of thermal power and number of operating reactor coolant pumps. This curve is based on the nuclear power peaking factors given in Reference 3 and the COLR which define the reference design peaking condition in the core for operation at the maximum power. Once the reference peaking condition and the associated thermal-hydraulic situation has been established for the hot channel, then all other combinations of axial flux shapes and their accompanying radials must result in a condition which will not violate the previously established design criteria on DNBR. The flux shapes examined include a wide range of positive and negative offset for steady state and transient conditions.

These design limit power peaking factors are the most restrictive calculated at full power for the range from all control rods fully withdrawn to maximum allowable control rod insertion, and form the core DNBR design basis.

The Axial Power Imbalance Protective Limits curves in the COLR are based on the more restrictive of two thermal limits and include the effects of potential fuel densification and fuel rod bowing:

a. The DNBR limit produced by a total nuclear power peaking factor consisting of the combination of the radial peak, axial peak, and position of the axial peak that yields no less than the DNBR limit.

b. The maximum allowable local linear heat rate that prevents central fuel melting at the hot spot as given in the COLR.

Power peaking is not a directly observable quantity and therefore limits have been established on the basis of the axial power imbalance produced by the power peaking.

The specified flow rates for curves 1, 2, and 3 of the Axial Power Imbalance Protective Limits given in the COLR correspond to the expected minimum flow rates with four pumps, three pumps, and one pump in each loop, respectively.

The curve of Figure 2.1-1 is the most restrictive of all possible reactor coolant pump-maximum thermal power combinations shown in Figure 2.1-3. The curves of Figure 2.1-3 represent the conditions at which the DNBR limit is predicted at the maximum possible thermal power for the number of reactor coolant pumps in operation or the local quality at the point of minimum DNBR is equal to ~~22 percent (BAW-2) or 26 percent (BWC)~~ the maximum quality limit approved for the applicable CHF correlation, whichever condition is more restrictive. The curves of Figures 2.1-1 and 2.1-3 were developed assuming a reactor coolant design flow rate of 102% of 352,000 gpm.

The maximum thermal power for each reactor coolant pump operating condition (four pump, three pump, and one pump in each loop) given in the COLR is due to a power level trip produced by the flux-flow ratio multiplied by the minimum flow rate for the given pump combination plus the maximum calibration and instrumentation error.

Using a local quality limit ~~of 22 percent (BAW-2), or 26 percent (BWC)~~ at the point of minimum DNBR as a basis for curves 2 and 3 of Figure 2.1-3 is a conservative criterion even though the quality at the exit is higher than the quality at the point of minimum DNBR.

The DNBR as calculated by the ~~BAW-2~~ BHTP or BWC correlation continually increases from the point of minimum DNBR, so that the exit DNBR is always higher and is a function of the pressure.

For each curve of Figure 2.1-3, a pressure-temperature point above and to the left of the curve would result in a DNBR greater than the Statistical Design Limit (SDL) [redacted] or a local quality at the point of minimum DNBR less than ~~22 percent (BAW-2) or 26 percent (BWC)~~ for the particular reactor coolant pump situation. Curve 1 is more restrictive than any other reactor coolant pump situation because any pressure/temperature point above and to the left of this curve will be above and to the left of the other curves.

REFERENCES

(1) ~~UFSAR, Section 3.2.3.1.1 - "Fuel Assembly Heat Transfer Design"~~

(2) BWC Correlation of Critical Heat Flux, BAW-10143P-A, Babcock & Wilcox, Lynchburg, Virginia, April 1985

(3) UFSAR, Section 3.2.3.1.1.3 - "Nuclear Power Factors"

(4) BAW-10187 P-A, "Statistical Core Design For B&W-Designed 177 FA Plants," B&W Fuel Company, Lynchburg, Virginia, March, 1994.

BHTP DNB Correlation Applied with LYNXT, BAW-10241P-A, Framatome ANP, Inc., Lynchburg, Virginia, July 2005.

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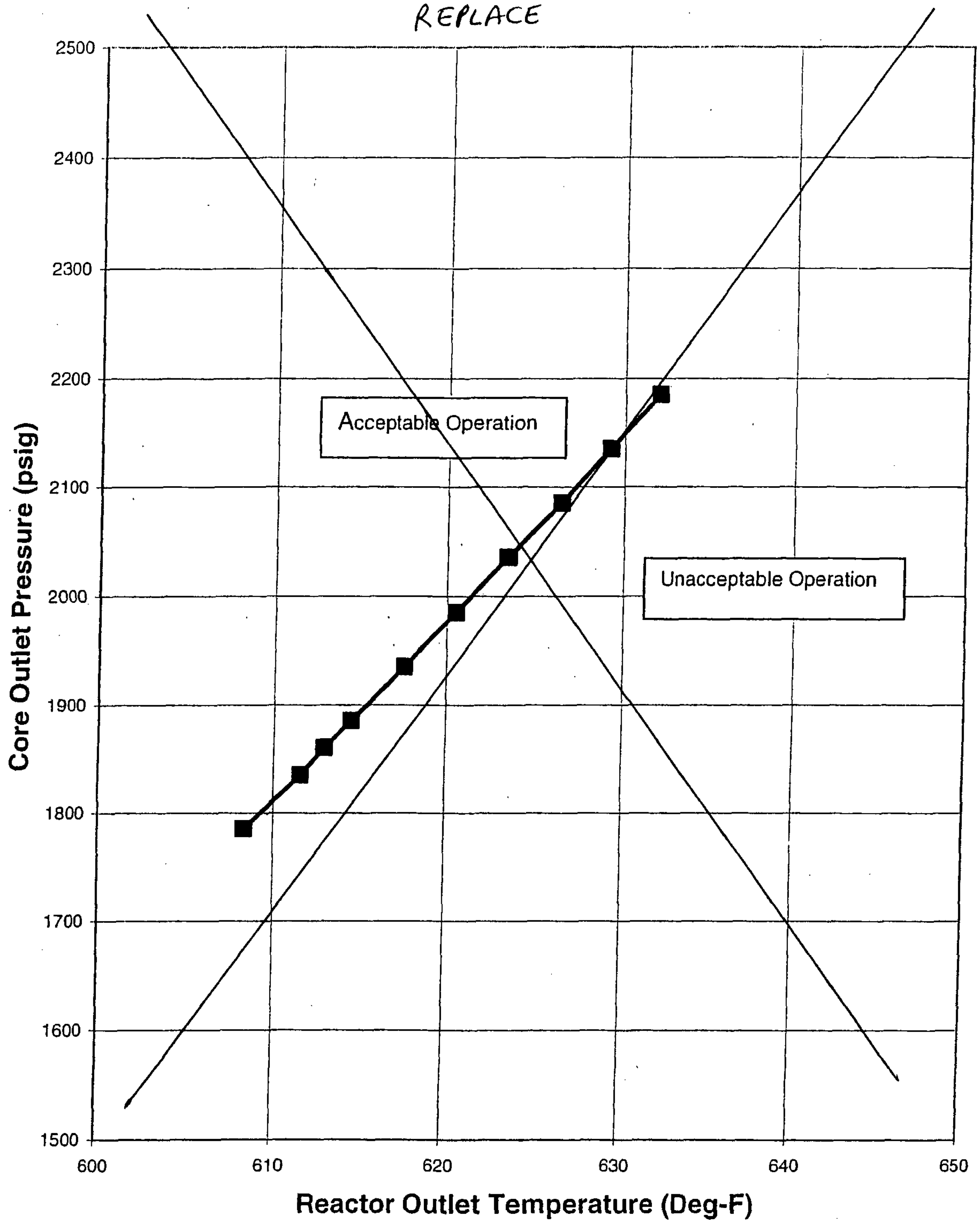
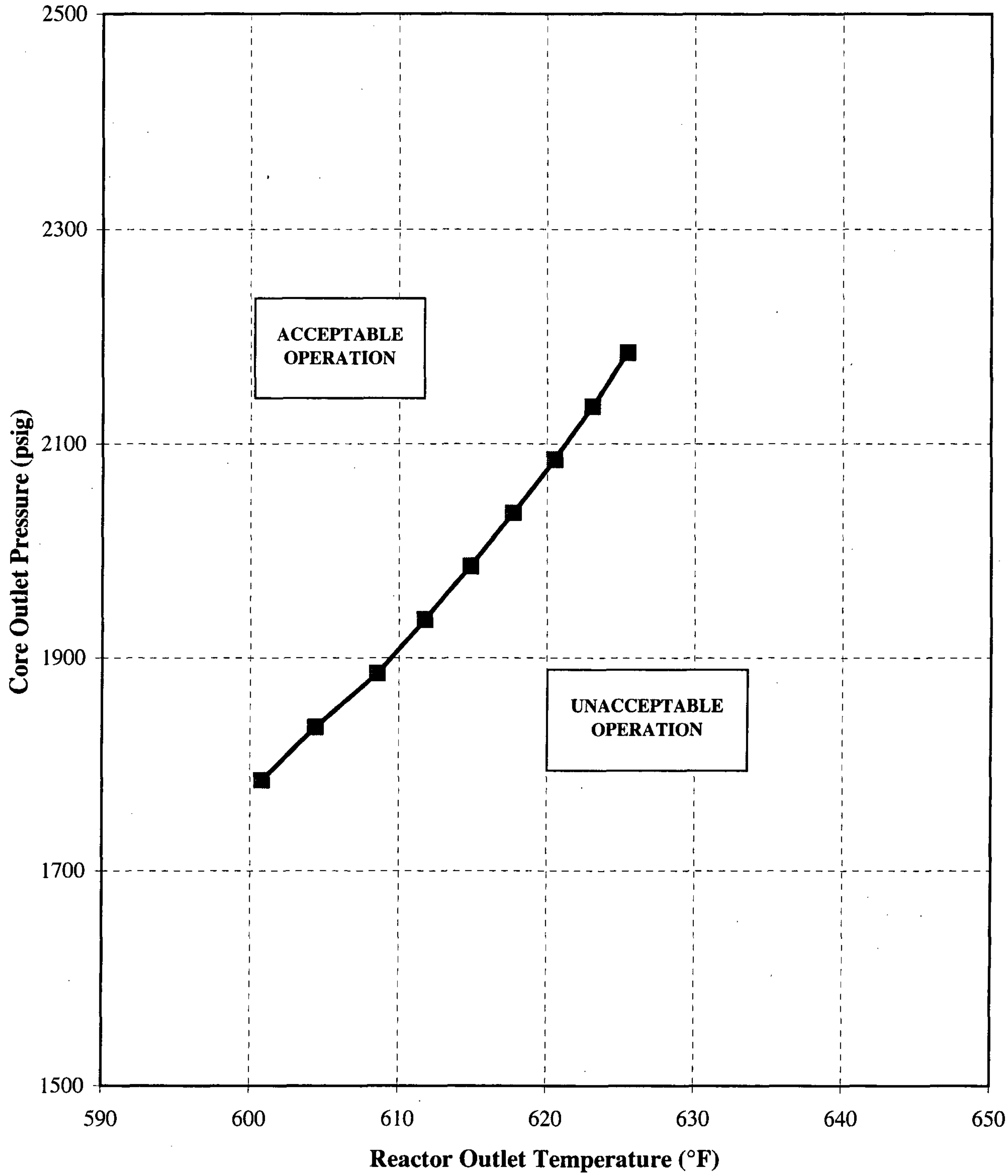


CORE PROTECTION SAFETY LIMIT
TMI-1
FIGURE 2.1-1

Amendment No. ~~50, 142, 167, 214, 238~~, ~~247~~,



CORE PROTECTION SAFETY LIMIT
TMI-1
FIGURE 2.1-1

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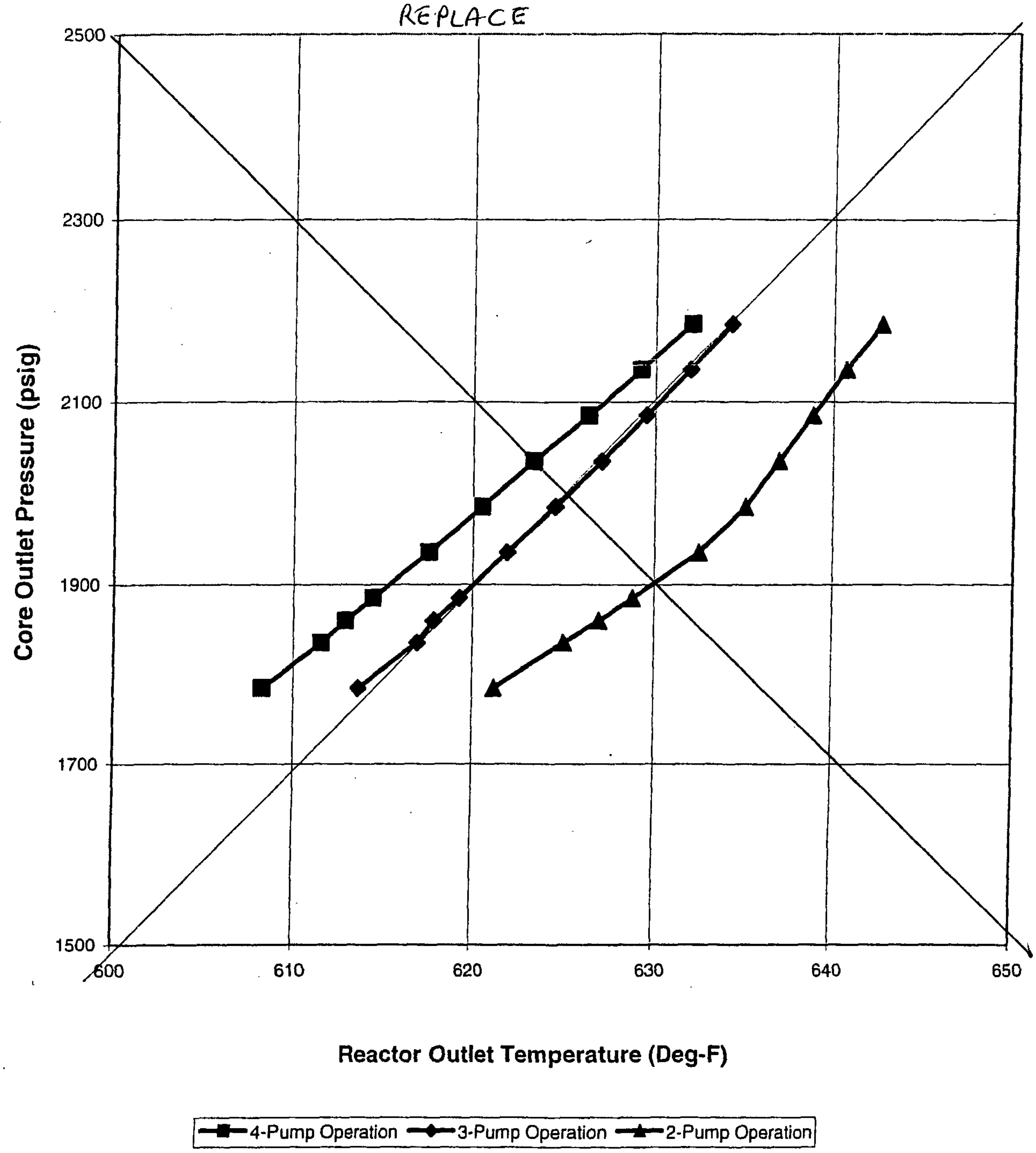
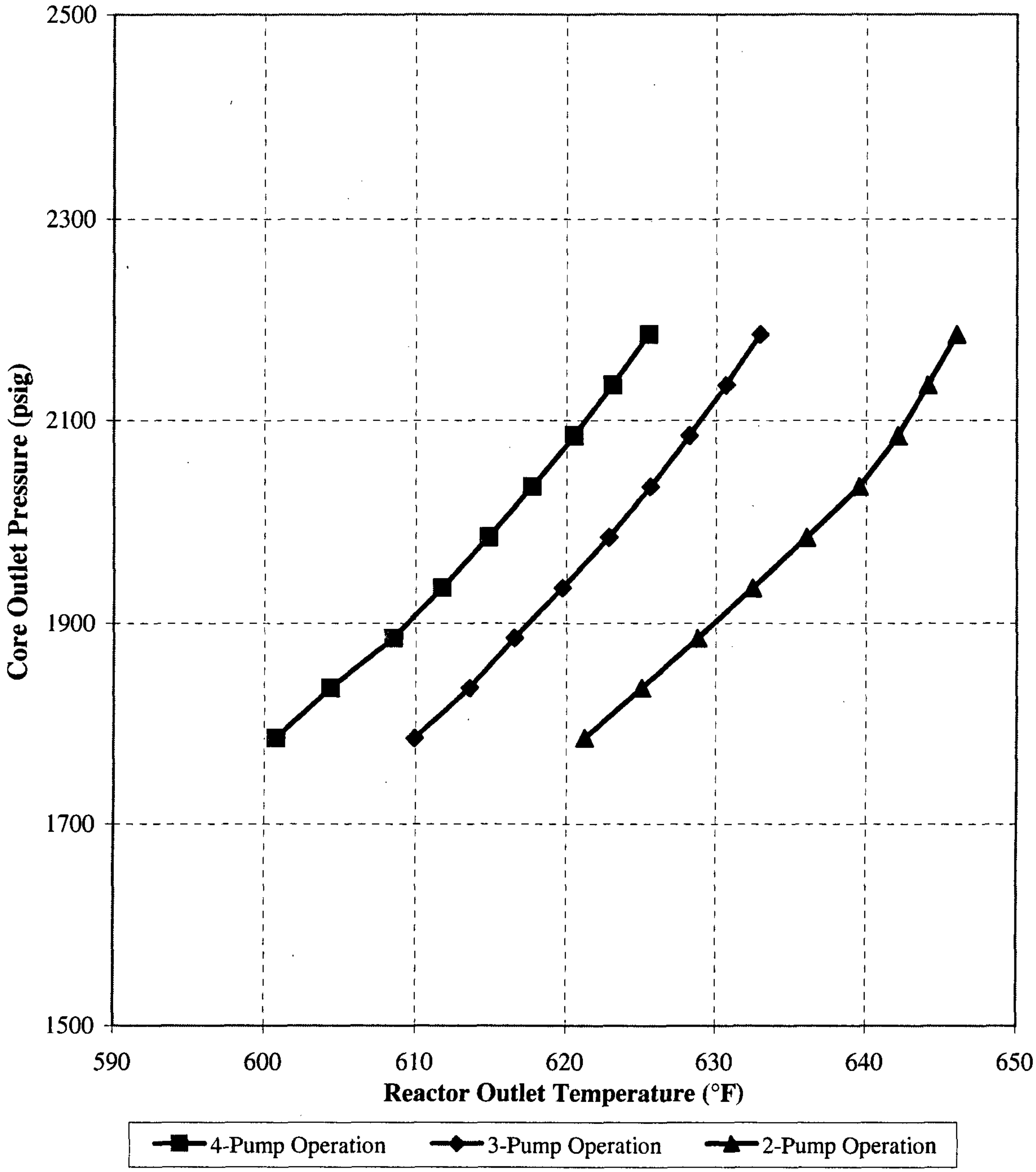


RC Pumps	Reactor Coolant Flow (~~lbs/hr~~) gpm	Power	Pumps Operating (Type of Limit)
4	~~137.77X10^6~~ 359,040	112%	Four Pumps (DNBR Limit)
3	See COLR	See COLR	Three Pumps (DNBR Limit)
2	See COLR	See COLR	One Pump in Each Loop (DNBR Limit)

CORE PROTECTION SAFETY BASES
TMI-1
FIGURE 2.1-3



RC Pumps	Reactor Coolant Flow (gpm)	Power	Pumps Operating (Type of Limit)
4	359,040	112%	Four Pumps (DNBR Limit)
3	See COLR	See COLR	Three Pumps (DNBR Limit)
2	See COLR	See COLR	One Pump in Each Loop (DNBR Limit)

CORE PROTECTION SAFETY BASES
TMI-1
FIGURE 2.1-3

a. Overpower trip based on flow and imbalance

The power level trip set point produced by the reactor coolant system flow is based on a power-to-flow ratio which has been established to accommodate the most severe thermal transient considered in the design, the loss-of-coolant flow accident from high power. Analysis has demonstrated that the specified power to flow ratio is adequate to prevent a DNBR of less than the Statistical Design Limit ~~1.343 (BWC)~~ should a low flow condition exist due to any malfunction.

The power level trip set point produced by the power-to-flow ratio provides both high power level and low flow protection in the event the reactor power level increases or the reactor coolant flow rate decreases. The power level trip set point produced by the power to flow ratio provides overpower DNB protection for all modes of pump operation. For every flow rate there is a maximum permissible power level, and for every power level there is a minimum permissible low flow rate. Typical power level and low flow rate combinations for the pump situations of Table 2.3-1 are given in the COLR.

The flux/flow ratios account for the maximum calibration and instrumentation errors and the maximum variation from the average value of the RC flow signal in such a manner that the reactor protective system receives a conservative indication of the RC flow.

No penalty in reactor coolant flow through the core was taken for an open core vent valve because of the core vent valve surveillance program during each refueling outage.

For safety analysis calculations the maximum calibration and instrumentation errors for the power level were used.

The power-imbalance boundaries are established in order to prevent reactor thermal limits from being exceeded. These thermal limits are either power peaking Kw/ft limits or DNBR limits. The axial power imbalance (power in the top half of the core minus power in the bottom half of core) reduces the power level trip produced by the power-to-flow ratio so that the boundaries of the Protection System Maximum Allowable Setpoints for Axial Power Imbalance in the COLR are produced.

b. Pump Monitors

The redundant pump monitors prevent the minimum core DNBR from decreasing below the Statistical Design Limit ~~([illegible] (BWC))~~ by tripping the reactor due to the loss of reactor coolant pump(s). The pump monitors also restrict the power level for the number of pumps in operation.

c. Reactor coolant system pressure

During a startup accident from low power or a slow rod withdrawal from high power, the system high pressure trip setpoint is reached before the nuclear overpower trip setpoint. The trip setting limit shown in Figure 2.3-1 for high reactor coolant system pressure ensures that the system pressure is maintained below the safety limit (2750 psig) for any design transient (Reference 2). Due to calibration and instrument errors, the safety analysis assumed a 45 psi pressure error in the high reactor coolant system pressure trip setting.

As part of the post-TMI-2 accident modifications, the high pressure trip setpoint was lowered from 2390 psig to 2300 psig. (The FSAR Accident Analysis Section still uses the 2390 psig high pressure trip setpoint.) The lowering of the high pressure trip setpoint and raising of the setpoint for the Power Operated Relief Valve (PORV), from 2255 psig to 2450 psig, has the effect of reducing the challenge rate to the PORV while maintaining ASME Code Safety Valve capability.

A B&W analysis completed in September of 1985 concluded that the high reactor coolant system pressure trip setpoint could be raised to 2355 psig with negligible impact on the frequency of opening of the PORV during anticipated over-pressurization transients (Reference 3). The high pressure trip setpoint was subsequently raised to 2355 psig. The potential safety benefit of this action is a reduction in the frequency of reactor trips.

The low pressure and variable low pressure trip setpoint were initially established to maintain the DNB ratio greater than or equal to 1.3 for those design accidents that result in a pressure reduction (References 4, 5, and 6). The B&W generic ECCS analysis, however, assumed a low pressure trip of 1900 psig and, to establish conformity with this analysis, the low pressure trip setpoint has been raised to the more conservative 1900 psig. The revised low pressure trip of 1900 psig and the variable low pressure ~~(16.25 T_{out} - 8113)~~ trip setpoint prevent the minimum core DNBR from decreasing below the Statistical Design Limit ~~([illegible] (BWC))~~. Figure 2.3-1 shows the high pressure, low pressure, high temperature and variable low pressure trip setpoints.

($16.21\, T_{OUT} - 7973$)

TABLE 2.3-1

REACTOR PROTECTION SYSTEM TRIP SETTING LIMITS (5)

	Four Reactor Coolant Pumps Operating (Nominal Operating) Power – 100%	Three Reactor Coolant Pumps Operating (Nominal Operating) Power - 75%	One Reactor Coolant Pump Operating in Each Loop (Nominal Operating Power - 49%)	Shutdown Bypass
1. Nuclear power, max. % of rated power	105.1	105.1	105.1	5.0(2)
2. Nuclear power based on flow (1) and imbalance max. of rated power	Power/Flow Setpoint in COLR times flow minus reduction due to imbalance	Power/Flow Setpoint in COLR times flow minus reduction due to imbalance	Power/Flow Setpoint in COLR times flow minus reduction due to imbalance	Bypassed
3. Nuclear power based (4) on pump monitors max. % of rated power	NA	NA	55%	Bypassed
4. High reactor coolant system pressure, psig max.	2355	2355	2355	1720(3)
5. Low reactor coolant system pressure, psig min.	1900	1900	1900	Bypassed
6. Reactor coolant temp. F., max.	618.8	618.8	618.8	618.8
7. High Reactor Building pressure, psig max.	4	4	4	4
8. Variable low reactor coolant system pressure, psig min.	~~(16.25 T_{out} - 8113)(6)~~ (16.21 T_{out} - 7973)(6)	~~(16.25 T_{out} - 8113)(6)~~ (16.21 T_{out} - 7973)(6)	~~(16.25 T_{out} - 8113)(6)~~ (16.21 T_{out} - 7973)(6)	Bypassed

(1) Reactor coolant system flow, %
(2) Administratively controlled reduction set during reactor shutdown.
(3) Automatically set when other segments of the RPS (as specified) are bypassed.
(4) The pump monitors also produce a trip on: (a) loss of two reactor coolant pumps in one reactor coolant loop, and (b) loss of one or two reactor coolant pumps during two-pump operation.
(5) Trip settings limits are limits on the setpoint side of the protection system bistable connectors.
(6) T_{out} is in degrees Fahrenheit (F).

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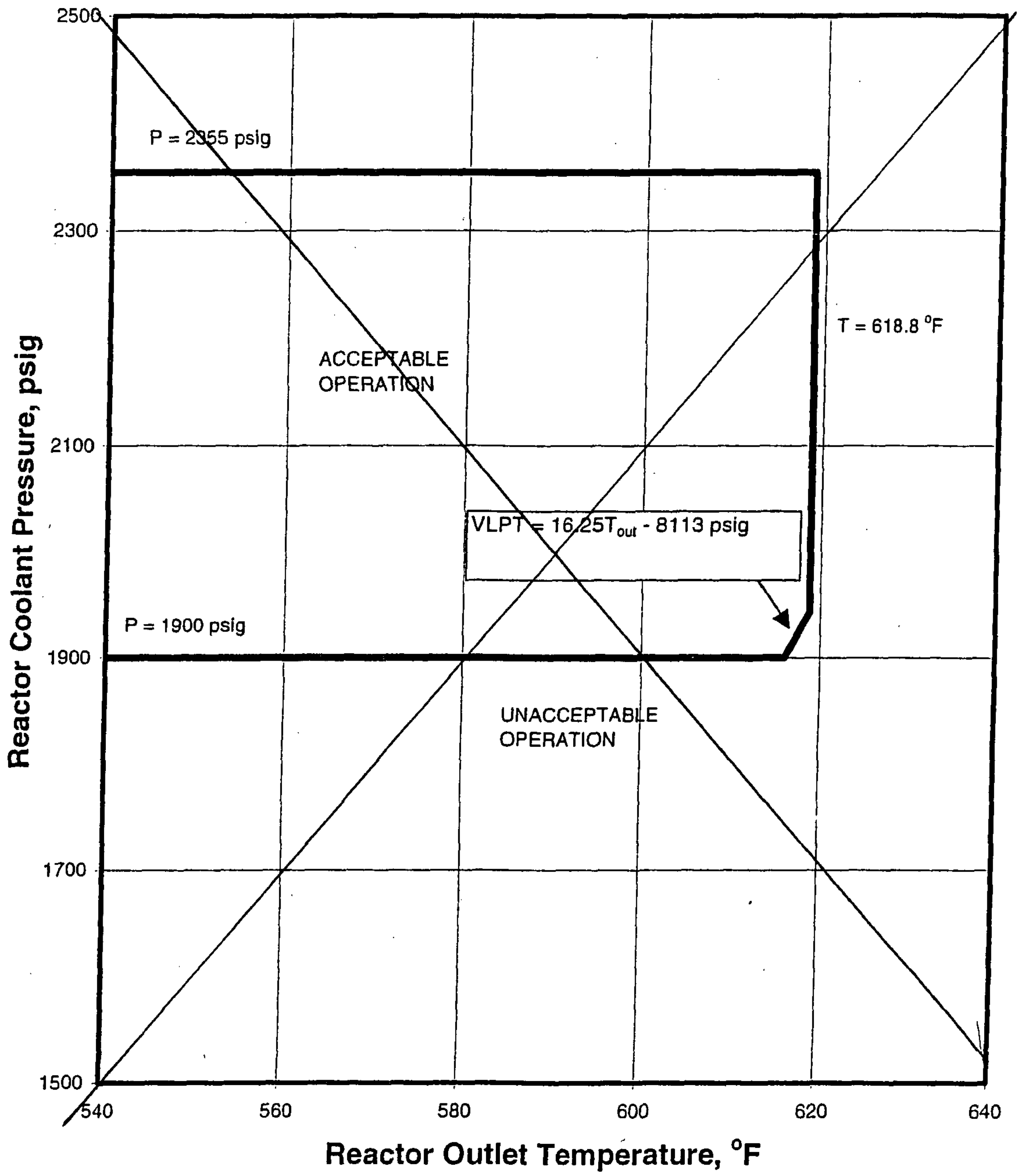
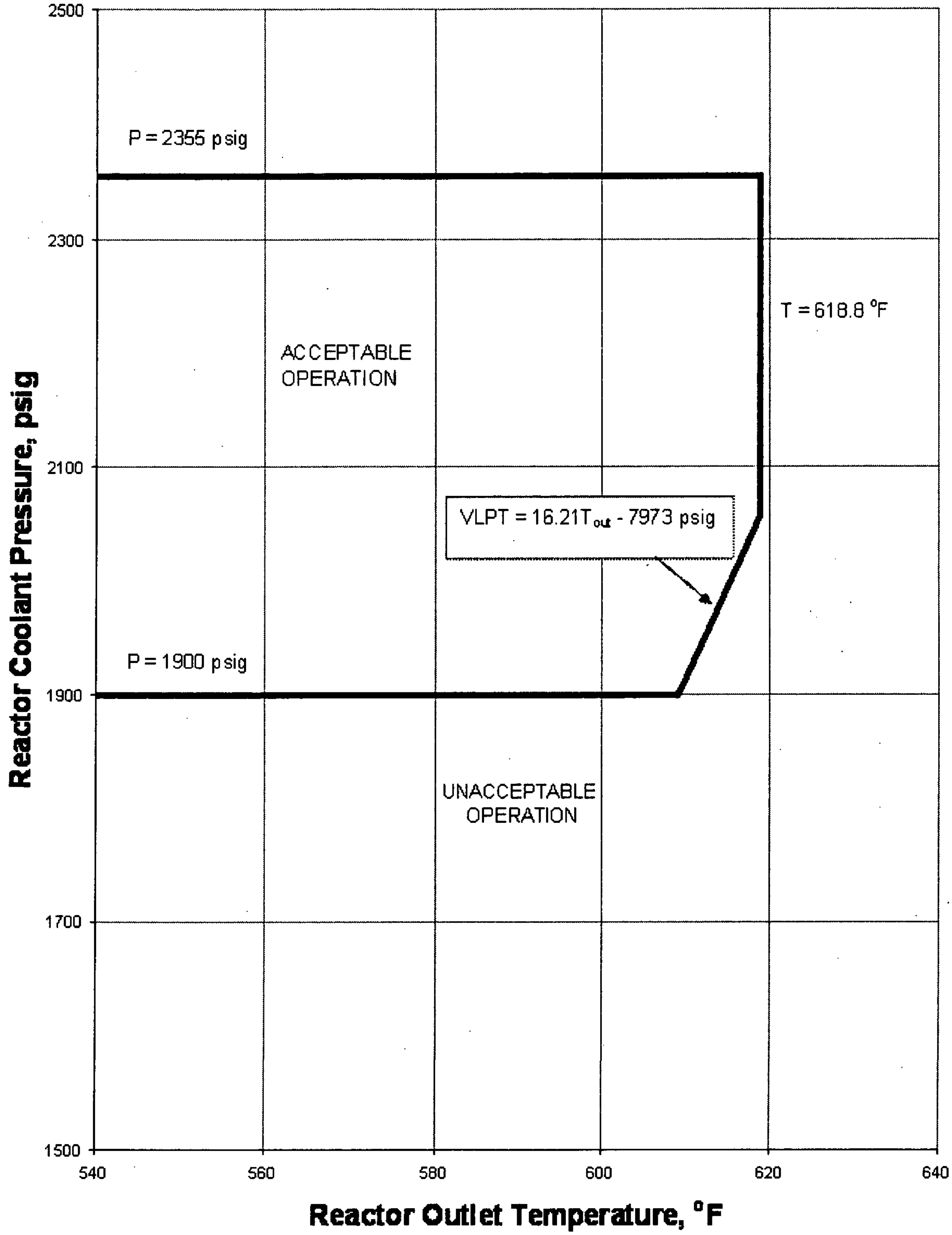
REPLACE



PROTECTION SYSTEM MAXIMUM
ALLOWABLE SETPOINTS
TMI-1
FIGURE 2.3-1

Amendment No. ~~13, 17, 28, 39, 45,~~
~~78, 126, 135, 142, 167~~, ~~247~~



PROTECTION SYSTEM MAXIMUM
ALLOWABLE SETPOINTS
TMI-1
FIGURE 2.3-1

Bases (Cont'd)

The equipment testing and system sampling frequencies specified in Tables 4.1-2, 4.1-3, and 4.1-5 are considered adequate to maintain the equipment and systems in a safe operational status.

¶

REFERENCE

(1) UFSAR, Section 7.1.2.3(d) - "Periodic Testing and Reliability"
(2) NRC SER for BAW-10167A, Supplement 1, December 5, 1988.
(3) BAW-10167, May 1986.
(4) BAW-10167A, Supplement 3, February 1998.

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INSERT TO TS PAGE 4-2b

The surveillance test procedures for the Variable Low Pressure Trip Setpoint do not compare the as-found Trip Setpoint (TSP) to the previous surveillance test as-left TSP. Basing operability determinations for the as-found TSP on the Nominal Setpoint (NSP) is acceptable because:

1. The NSP as-left tolerance specified in the surveillance test procedures is less than or equal to the calculated NSP as-left tolerance.
2. The NSP as-left tolerance is not included in the Total Loop Uncertainty (TLU) calculation. This is acceptable because the NSP as-left tolerance specified in the surveillance test procedures is less than half of the calculated NSP as-left tolerance. This prevents masking of excessive drift from one side of the tolerance band to the other.
3. The pre-defined NSP as-found tolerance is based on the square root of the sum of the square of the instrument accuracy, M&TE accuracy and drift. The NSP as-left tolerance is not included in this calculation.

Credible uncertainties for the Variable Low Pressure Trip Setpoint include instrument uncertainties during normal operation including drift and measurement and test equipment uncertainties. In no case shall the pre-defined as-found acceptance criteria band overlap the Allowable Value. If one end of the pre-defined as-found acceptance criteria band is truncated due to its proximity to the Allowable Value, this does not affect the other end of the pre-defined as-found acceptance criteria band. If equipment is replaced, such that the previous as-left value is not applicable to the current configuration, the as-found acceptance criteria band is not applicable to calibration activities performed immediately following the equipment replacement.

TABLE 4.1-1 (Continued)

CHANNEL DESCRIPTION	CHECK	TEST	CALIBRATE	REMARKS
8. High Reactor Coolant Pressure Channel	S	S/A	R	
9. Low Reactor Coolant Pressure Channel	S	S/A	R	
10. Flux-Reactor Coolant Flow Comparator	S	S/A	F	
11. Reactor Coolant Pressure-Temperature Comparator	S	S/A	R	See Notes (a) and (b).
12. Pump Flux Comparator	S	S/A	R	
13. High Reactor Building Pressure Channel	S	S/A	F	
14. High Pressure Injection Logic Channels	NA	Q	NA	
15. High Pressure Injection Analog Channels				
a. Reactor Coolant Pressure Channel	S(1)	M	R	(1) When reactor coolant system is pressurized above 300 psig or T_{ave} is greater than 200°F
16. Low Pressure Injection Logic Channel	NA	Q	NA	
17. Low Pressure Injection Analog Channels			0	
a. Reactor Coolant Pressure Channel	S(1)	M	R	(1) When reactor coolant system is pressurized above 300 psig or T_{ave} is greater than 200°F
18. Reactor Building Emergency Cooling and Isolation System Logic Channel	NA	Q	NA	

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TABLE 4.1-1 (Continued)

	CHANNEL DESCRIPTION	CHECK	TEST	CALIBRATE	REMARKS
49.	Saturation Margin Monitor	S(1)	M(1)	R	(1)When T_{ave} is greater than 525°F.
50.	Emergency Feedwater Flow Instrumentation	NA	M(1)	F	(1)When T_{ave} is greater than 250°F.
51.	Heat Sink Protection System				
	a. EFW Auto Initiation Instrument Channels				(1)Includes logic test only.
	1. Loss of Both Feedwater Pumps	NA	Q(1)	F	
	2. Loss of All RC Pumps	NA	Q(1)	R	
	3. Reactor Building Pressure	NA	Q	F	
	4. OTSG Low Level	W	Q	R	
	b. MFW Isolation OTSG Low Pressure	NA	Q	R	
	c. EFW Control Valve Control System				
	1. OTSG Level Loops	W	Q	R	
	2. Controllers	W	NA	R	
	d. HSPS Train Actuation Logic	NA	Q(1)	R	
52.	Backup Incore Thermocouple Display	M(1)	NA	R	(1)When T_{ave} is greater than 250°F.
53.	Deleted				
54.	Reactor Vessel Water Level	NA	NA	R	

Notes

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(a) If the as-found channel setpoint is conservative with respect to the Allowable Value but outside its predefined as-found tolerance then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service. Enter condition into Corrective Action Program.

(b) The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the Nominal Trip Setpoint (NSP) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the NSP are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the surveillance procedures to confirm channel performance. The NSP and the methodologies used to determine the as-found and the as-left tolerances are specified in a document incorporated by reference into the UFSAR.

ENCLOSURE 5

List of Commitments

SUMMARY OF AMERGEN COMMITMENTS

The following table identifies regulatory commitments made in this document by AmerGen. (Any other actions discussed in the submittal represent intended or planned actions by AmerGen. They are described to the NRC for the NRC's information and are not regulatory commitments.)

COMMITMENT	COMMITTED DATE OR "OUTAGE"	COMMITMENT TYPE	
		ONE-TIME ACTION (Yes/No)	PROGRAMMATIC (Yes/No)
The VLPT Nominal Setpoint (NSP)and Total Loop Uncertainty (TLU) are determined using the methodology provided by ANSI/ISA-S67.04-Part 1-1994. The pre-defined limits for the NSP as-found tolerance will be determined in accordance with the NRC accepted methodology described in NRC RIS 2006-17.	Upon implementation of amendment for the proposed change.	No	Yes
Future LOCA analyses addressing the replacement steam generator design will specifically consider both a mixed-core and a full-core of Mark-B-HTP assemblies.	T1R18 Refueling Outage (Fall 2009)	Yes	No