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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

Subject: Technical Specification Change Request No. 316 – Cycle 15 Core Reload Design

Three Mile Island, Unit 1 (TMI Unit 1)
Facility Operating License No. DPR-50
NRC Docket No. 50-289

In accordance with 10 CFR 50.4(b)(1), enclosed is Technical Specification Change Request No. 316.

The purpose of this Technical Specification Change Request is to revise TMI Unit 1 Technical Specifications to incorporate changes associated with TMI Unit 1 Cycle 15 core reload design analysis. The TMI Unit 1 Cycle 15 core reload design implements the Framatome ANP Statistical Core Design methodology with a higher design radial-local peaking factor, and allowances for potential future Appendix K power uprate application. The Framatome ANP Statistical Core Design methodology is documented in Framatome Topical Report BAW-10187P-A, "Statistical Core Design For B&W-Designed 177 FA Plants," March 1994, 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-10187P-A is documented in NRC Safety Evaluation Reports dated March 24, 1993 and March 17, 1994. The statistical core design method is a thermal-hydraulic analysis technique that provides additional DNBR margin by statistically combining core and fuel element uncertainties, while retaining the criterion that the core is protected by designing to avoid departure from nucleate boiling (DNB). The traditional method of treating uncertainties is to assume the worst level of each uncertainty simultaneously. Applying statistical techniques allows for a more realistic assessment of core DNB protection.

The Framatome ANP statistical core design methodology (BAW-10187P-A) has been implemented at all other B&W 177 FA plants for which Framatome ANP performs reload licensing.

Implementation of the Framatome ANP Statistical Core Design Methodology affects the following TMI Unit 1 Technical Specification Sections: 2.1-Core Protection Safety Pressure-Temperature Limit Figure 2.1-1, 2.1 Bases, 2.3-Reactor Protection System (RPS) Trip Setting Limits Table 2.3-1, 2.3 Bases - including Protection System Maximum Allowable Setpoints Figure 2.3-1, 3.5.1-instrumentation requirements for RPS pressure-temperature instrument channels (Table 3.5-1), and 4.1-instrumentation surveillance requirements for RPS pressure-temperature instrument channels (Table 4.1-1).

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Using the standards in 10 CFR 50.92, AmerGen Energy Company, LLC (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.

No new regulatory commitments are established by this submittal. The TMI Unit 1 Long-Range Planning Program commitment stated in AmerGen letter to the NRC dated October 18, 2001 (5928-01-20283) will expire upon NRC approval of this Technical Specification Change Request since implementation of NRC-approved statistical core design methods provides an alternative means of addressing transition core effects.

NRC approval of this change is requested by September 30, 2003 to support restart from the refueling outage scheduled for October 2003.

If any additional information is needed, please contact David J. Distel at (610) 765-5517.

I declare under penalty of perjury that the foregoing is true and correct.

Sincerely,

01-16-03
Executed On

Michael P. Gallagher
Director, Licensing & Regulatory Affairs
Mid-Atlantic Regional Operating Group

Enclosures: (1) TMI Unit 1 Technical Specification Change Request No. 316 Evaluation of Proposed Changes
(2) TMI Unit 1 Technical Specification Change Request No. 316 Markup of Proposed Technical Specification Page Changes

cc: H. J. Miller, Administrator, USNRC Region I
T. G. Colburn, USNRC Senior Project Manager, TMI Unit 1
J. D. Orr, USNRC Senior Resident Inspector, TMI Unit 1
D. Allard, Director, Bureau of Radiation Protection – PA Department of Environmental Resources
Chairman, Board of County Commissioners of Dauphin County
Chairman, Board of Supervisors of Londonderry Township
File No. 02070

ENCLOSURE 1

TMI Unit 1 Technical Specification Change Request No. 316

Evaluation of Proposed Changes

1.0 INTRODUCTION

This letter is a request to amend Operating License No. DPR-50.

The proposed changes would revise the Operating License to incorporate changes associated with Cycle 15 reload design analysis. TMI Unit 1 Cycle 15 reload design implements the Framatome ANP Statistical Core Design methodology as described in NRC-approved Framatome Topical Report BAW-10187P-A, "Statistical Core Design For B&W-Designed 177 FA Plants," March 1994. This change supports Cycle 15 operation. NRC approval of this change is requested by September 30, 2003.

Technical Specification 3.5.2.2.a is also revised to correct an editorial omission that occurred in previously issued TMI Unit 1 Amendment No. 211, dated June 15, 1999.

AmerGen Energy Company, LLC (AmerGen) requests that the following changed replacement pages be inserted into the existing Technical Specifications:

Revised Technical Specification Pages: 2-1, 2-2, 2-3, 2-4a (Figure 2.1-1), 2-4c (Figure 2.1-3), 2-6, 2-7, 2-8, 2-10 (Table 2.3-1), 2-11 (Figure 2.3-1), 3-29 (Table 3.5-1), 3-33, and 4-4 (Table 4.1-1).

The marked up pages showing the requested changes are provided in Enclosure 2.

2.0 DESCRIPTION OF PROPOSED AMENDMENT

The TMI Unit 1 Cycle 15 core reload design implements the Framatome ANP Statistical Core Design methodology. The statistical core design method is a thermal-hydraulic analysis technique that provides additional DNBR margin by statistically combining core and fuel element uncertainties, while retaining the criterion that the core is protected by designing to avoid departure from nucleate boiling (DNB). The Framatome ANP Statistical Core Design methodology is documented in Framatome Topical Report BAW-10187P-A 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-10187P-A is documented in NRC Safety Evaluation Reports dated March 24, 1993 and March 17, 1994. Statistical Core Design (SCD) is a thermal-hydraulic analysis technique that provides an increase in core thermal (DNB) margin by treating core state and bundle uncertainties statistically. The current method of treating uncertainties is to assume the worst level of each uncertainty simultaneously. Applying statistical techniques allows for a more realistic assessment of core DNB protection.

The Framatome ANP statistical core design methodology (BAW-10187P-A) has been implemented at all other B&W 177 FA plants for which Framatome ANP performs reload licensing.

The Statistical Design Limit (SDL) of 1.313 developed for the BWC CHF correlation for B&W 177 FA plants provides 95 percent protection at a 95 percent confidence level against hot pin DNB. The corresponding core-wide protection on a pin-by-pin basis using realistic peaking distributions is greater than 99.9 percent. The SDL of 1.313 (BWC) is equivalent to the current non-SCD DNBR limit of 1.18 (BWC) contained in TMI Unit 1 Technical Specifications, which only accounts for DNBR correlation uncertainty.

Technical Specification 2.1 Bases and 2.3 Bases are revised to incorporate reference to the Topical Report BAW-10187P-A, and describe the Statistical Design Limit (SDL) of 1.313 (BWC). Section 2.1 Bases is also revised to delete the description of the mixed core penalty implemented for Cycle 14 operation. The need for the mixed core penalty is eliminated by the use of SCD methodology. Section 2.3 Bases is also revised to describe the addition of the variable low reactor coolant system pressure trip setpoint for Cycle 15 operation as a result of using the SCD methodology.

The Core Protection Safety Limits specified in Technical Specification Figure 2.1-1, and the associated Bases Figure 2.1-3, are revised to incorporate revised Cycle 15 limits based on use of SCD methodology. The SCD methodology utilizes a higher design radial-local peaking factor of 1.80 versus the current value of 1.714 contained in the TMI Unit 1 Core Operating Limits Report (COLR), and an assumed maximum rated power level of 2612 MWt to allow for a potential future Appendix K power uprate application of up to 1.7%.

Technical Specification Table 2.3-1, Reactor Protection System Trip Setting Limits, is revised to add the variable low reactor coolant system pressure trip setpoint. Technical Specification Bases Figure 2.3-1, Protection System Maximum Allowable Setpoints, is revised to incorporate the variable low pressure trip setpoint.

Technical Specification Table 3.5-1 and Table 4.1-1 are revised to incorporate operational and surveillance requirements for the pressure-temperature instrument channels associated with the variable low pressure trip RPS function.

The use of the NRC-approved Framatome ANP Statistical Core Design methodology reduces excess conservatism and therefore increases cycle design and plant operational flexibility.

Technical Specification 3.5.2.2.a is revised to delete the reference to Technical Specification 4.7.2.3. Technical Specification 4.7.2.3 was previously eliminated in TMI Unit 1 Amendment No. 211, dated June 15, 1999, as requested by TMI Unit 1 Technical Specification Change Request (TSCR) No. 253, Revision 2, dated December 3, 1996. Deletion of the reference to Technical Specification 4.7.2.3 was inadvertently omitted from TSCR No. 253, Revision 2. TSCR No. 253, Revision 2, and associated Amendment No. 211, eliminated Technical Specification 4.7.2 in its entirety. This proposed change is administrative in nature and only corrects a previous editorial omission.

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 provide adequate margin to departure from nucleate boiling (DNB) heat flux at the design overpower.

The existing TMI Unit 1 Technical Specification RCS pressure-temperature core protection safety limits were developed for operating Cycle 14 core reload design. The TMI Unit 1 Cycle 14 core reload design analyses were performed in accordance with approved Framatome ANP methods as described in NRC-approved Topical Report BAW-10179P-A as listed in existing TMI Unit 1 Technical Specification Section 6.9.5. An additional change to the minimum RCS flow requirement for 4-pump operation was also incorporated in the Core Protection Safety Bases Figure 2.1-3 to offset the Cycle 14 transition core DNB penalty.

The TMI Unit 1 operating Cycle 15 core reload design analyses are also performed in accordance with NRC-approved Framatome ANP methods described in Topical Report BAW-10179P-A, including the use of Framatome ANP's Statistical Core Design methodology (BAW-10187P-A) for core DNB analyses. Topical Report BAW-10187P-A was approved by the NRC for referencing in license applications in NRC Safety Evaluation Reports dated March 24, 1993 and March 17, 1994. NRC-approved Topical Report BAW-10187P-A was subsequently incorporated into BAW-10179P-A. The use of a Thermal Design Limit (TDL) with the statistical core design methodology, reserves DNB margin that can be used to offset cycle-specific impacts, such as mixed core DNB penalties. As a result, the technique used to address the mixed core DNB penalty for TMI Unit 1 Cycle 14 by reserving reactor coolant system flow rate margin is no longer needed. Therefore, the TMI Unit 1 Long-Range Planning Program, Category A commitment specifying a minimum reactor coolant system flow rate of 105.5%, stated in AmerGen letter to the NRC dated October 18, 2001(5928-01-20283), will expire upon NRC approval of this proposed Technical Specification Change Request.

4.0 REGULATORY REQUIREMENTS & GUIDANCE

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 power at which DNB might be expected to occur and the operating power 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 hottest 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 for TMI Unit 1. In addition, cycle-specific checks on assembly-wise power distribution will be made on a core reload basis.

5.0 TECHNICAL ANALYSIS

The core protection safety limits and bases were reanalyzed using Framatome ANP's NRC-approved Statistical Core Design (SCD) methodology for 177 fuel assembly (FA) B&W plants (BAW-10187P-A) with a Statistical Design Limit (SDL) of 1.313, which is equivalent to the current non-SCD minimum DNBR limit of 1.18 approved for the BWC CHF correlation. In addition, all core DNB analyses supporting the proposed amendment were performed with additional retained margin in the form of a Thermal Design Limit (TDL) of 1.40. This margin is 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. For TMI Unit 1 Cycle 15, the retained margin is specifically used to offset transition core effects of co-resident Mark-B10 and Mark-B12 fuel designs. Finally, a portion of the DNB margin gained by switching from the non-SCD core thermal-hydraulic methodology to the SCD methodology was used to justify a higher design radial-local peaking factor of 1.80 (vs. 1.714). The higher factor of 1.80 was chosen to provide more cycle design flexibility and less restrictive core operating limits for normal operations.

The revised core protection safety limits and bases proposed in this amendment were developed using Framatome ANP's NRC-approved reload methodology (BAW-10179P-A) and NRC-approved core thermal-hydraulic code LYNXT (BAW-10156P-A). The SDL limit and the TDL retained margin approach used in developing the revised core protection safety limits and bases are consistent with 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 1.313 is acceptable for TMI Unit 1 for the following reasons:

- The 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 1.313 are all applicable to TMI Unit 1.

- The fuel designs utilized at TMI Unit 1 are the Mark-B designs, for which the BWC CHF correlation has been approved. The Mark-B fuel design and the BWC CHF correlation were assumed in the development of the SDL of 1.313.

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

- Core state variables that were not explicitly included in the statistical design are input to thermal-hydraulic computer codes at their most adverse allowable level. In addition, values specific to TMI Unit 1 are assumed in the analysis as listed below. Each of these values is well within the range analyzed in developing the SDL.

- TMI Unit 1 has been approved to operate with up to 20% average once-through steam generator (OTSG) tube plugging and analyzed for 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). Therefore, 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 DNB analyses supporting the proposed amendment.

- A design radial-local peaking factor of 1.80 was assumed in all core DNB analyses supporting the proposed amendment.

- A rated power level of 2612 MWt (i.e., 1.7% power uprate from 2568 MWt) was assumed in all core DNB analyses supporting the proposed amendment.

• 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.

The requirements for the reactor coolant system Variable Low Pressure Trip (VLPT) instrumentation were removed from TMI Unit 1 Technical Specifications prior to Cycle 7 (License Amendment No. 142, dated July 18, 1988) when the core protection safety limits were shown to be protected by the RPS low reactor coolant pressure, high reactor coolant pressure, and high reactor coolant temperature trips. The application of the SCD methodology for TMI Unit 1 Cycle 15 results in re-introduction of the VLPT. With the re-introduction of the VLPT to TMI Unit 1 Technical Specifications, the instrumentation operating condition and surveillance requirements are required. The proposed changes to Technical Specification Tables 3.5-1 and 4.1-1 reinstate the requirements that were in place prior to Cycle 7.

UFSAR Chapter 14 Transient and Accident Analysis

Loss of Coolant Flow Events

The Loss of Coolant Flow events are the most challenging for minimum DNBR. The three most DNB-limiting transients in the TMI Unit 1 UFSAR that are impacted by the proposed amendment are:

1) Single reactor coolant pump coastdown (4-to-3 pumps),
2) Four reactor coolant pump coastdown (4-to-0 pumps),
3) Single reactor coolant pump locked rotor (4-to-3 pumps)

These events were analyzed using Framatome ANP's NRC-approved LYNXT thermal-hydraulic code with the BWC correlation that has been approved for the Mark-B fuel design utilized at TMI Unit 1. Per the SCD methodology, the events were modeled in LYNXT using nominal state parameters, as uncertainties are included in the Statistical Design Limit. Of particular note, the nominal reactor coolant system flow rate assumed in the analysis was 104.5% of design flow, which is equivalent to the 102% minimum

design RCS flow with 2.5% measurement uncertainty previously approved for TMI Unit 1 operation with up to 20% average OTSG tube plugging. The nominal design radial-local peaking factor used in the analysis was 1.7341, which is equivalent to a radial-local peaking factor of 1.80 with a radial-local peaking uncertainty of 3.8%. The rated power level used in the analysis was 2612 MWt to allow for a future Appendix K-type power uprate of up to 1.7%.

The single pump coastdown (4-to-3) event is a moderate frequency, Condition II event with a MDNBR acceptance criterion of 1.18 (or an SDL of 1.313). A TDL of 1.40 was applied to the TMI Unit 1 analysis. The reactor continues to operate at full power until flow decreases to the point where the flux-flow setpoint initiates a reactor trip. The LYNXT results for this event using the SCD methodology is a MDNBR of 1.82, which is conservatively higher than the TDL of 1.40 (42 DNB points of margin, where 1 DNB point = 0.01), indicating that the existing flux-flow setpoint provides adequate DNB protection for this event with no predicted fuel failures. The current non-SCD analysis margin is 42 DNB points (1.60 calculated MNDBR – 1.18 MDNBR limit).

The four pump coastdown (4-to-0) event is an infrequent, Condition III event but is analyzed to the more restrictive criterion of the moderate frequency event with a MDNBR acceptance criterion of 1.18 (or an SDL of 1.313). A TDL of 1.40 was applied to the TMI Unit 1 analysis. A reactor trip occurs immediately upon a signal from the power/pump monitor trip. The LYNXT results for this event using the SCD methodology is a MDNBR of 1.91, which is conservatively higher than the TDL of 1.40 (51 DNB points of margin), indicating that the existing power/pump monitor trip provides adequate DNB protection for this event with no predicted fuel failures. The current non-SCD analysis margin is 48 DNB points (1.66 calculated MNDBR – 1.18 MDNBR limit).

The single pump locked rotor event is a more severe, limiting fault, Condition IV event with a MDNBR acceptance criterion of 1.0. The reactor continues to operate at full power until flow decreases to the point where the flux-flow setpoint initiates a reactor trip. The LYNXT results for this event using the SCD methodology is a MDNBR of 1.51 which meets the more restrictive acceptance criteria of the moderate frequency events (i.e., MDNBR greater than the TDL of 1.40 - 11 DNB points of margin), indicating that the existing flux-flow setpoint provides adequate DNB protection for this event with no predicted fuel failures. The current non-SCD analysis margin is 6 DNB points (1.24 calculated MNDBR – 1.18 MDNBR limit).

Fuel Handling Accident

In addition to the DNB-limited UFSAR events, the proposed change also has the potential to impact fuel handling accidents as a result of increasing the design radial-local peaking factor from 1.714 to 1.80.

The consequences of the Fuel Handling Accidents (FHAs) analyzed in the TMI Unit 1 UFSAR are based on the source term (i.e., isotopic inventory) of an average power fuel assembly increased by a radial-local peaking factor of 1.70 to bound the highest powered assembly in the core. Although the proposed amendment increases the design radial-local peaking factor to 1.80, the purpose of this increase is to provide greater DNB margin for normal operating transients, which results in less restrictive axial imbalance core operating limits. The isotopic inventory of the fuel is a function of its

steady-state power level. Higher peaking factors that may occur during transients are short-lived and have an insignificant impact on fuel isotopic inventories that accumulate over a two-year cycle. The maximum steady-state radial-local peaking factor (including physics model radial-local power uncertainty) for TMI Unit 1 cycle designs applying the SCD methodology is 1.64, which is bounded by the radial-local peaking factor of 1.70 that is applied in the current FHA analyses. Therefore the dose consequences of the current FHAs for TMI Unit 1 remain bounding for cycles designed with the SCD methodology.

Conclusion

The proposed changes to the TMI Unit 1 Technical Specification RCS pressure-temperature core protective safety limit and the reintroduction of the Variable Low Pressure Trip (including the proposed trip setpoint, required instrument operating conditions, and instrument surveillance requirements) have been established to assure adequate margins of safety are maintained and have been developed in accordance with NRC-approved methodologies.

Consequently, the proposed Technical Specification changes will not adversely affect nuclear safety or safe plant operations.

6.0 REGULATORY ANALYSIS

The proposed amendment to incorporate revised Core Protection Safety Limits and Bases into the TMI Unit 1 Technical Specifications preserves the design DNBR 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 (Condition I or II events). Specifically, the core protection safety limit (Technical Specification Figure 2.1-1) and corresponding Bases (Technical Specification Figure 2.1-3) were reanalyzed to evaluate the impact of applying Framatome ANP's Statistical Core Design (SCD) methodology with a larger design radial-local peaking factor and allowance for a future Appendix K-type power uprate of 1.7%. The analysis results concluded that re-introduction of the reactor coolant system Variable Low Pressure Trip (including the proposed trip setpoint, required instrument operating conditions, and instrument surveillance requirements) to TMI Unit 1 Technical Specifications is required in order for the Reactor Protection System (RPS) reactor coolant pressure-temperature (P-T) trip setpoints to provide adequate DNBR protection. The proposed VLPT setpoint in Technical Specification Table 2.3-1, in conjunction with the existing reactor coolant low pressure and high temperature trips, will ensure that power operation will be restricted to temperature/pressure conditions that meet the DNBR safety criterion.

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.

7.0 NO SIGNIFICANT HAZARDS CONSIDERATION

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

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

Response: No.

The proposed Technical Specification limits (Figure 2.1-1) and reactor protection system (RPS) trip setpoints (Table 2.3-1) are developed in accordance with the methods and assumptions described in NRC-approved Framatome ANP 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 (BWC) have been approved for use with these methods and the Mark-B fuel type utilized at TMI Unit 1. The proposed Technical Specification requirements on Variable Low Pressure Trip (VLPT) instrument operating conditions (Table 3.5-1) and surveillances (Table 4.1-1) are consistent with the VLPT requirements that were last contained in TMI Unit 1 Technical Specifications prior to Cycle 7. The existing flux-flow trip setpoint and power/pump monitor trip have been shown to provide adequate DNB protection for Updated Final Safety Analysis Report (UFSAR) DNB-limiting loss of coolant events.

The margin retained for penalties such as transition core effects, by imposing a Thermal Design Limit of 1.40 in all DNB analyses supporting the proposed change, has been shown to be sufficient to offset the current mixed core conditions at TMI Unit 1, where the Mark-B12 fuel design with fine mesh debris filter is co-resident with earlier, non-debris filter Mark-B fuel designs. Therefore the previous commitment to require a higher minimum RCS flow (105.5% of design flow instead of 104.5%) to offset transition core penalties is no longer necessary.

Reload cycles are designed and operated with maximum steady-state radial-local peaking factors that are bounded by UFSAR assumptions used to determine the dose consequences from fuel handling accidents.

The proposed change to Technical Specification 3.5.2.2.a is only an administrative correction.

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

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

Response: No.

The proposed Technical Specification limits (Figure 2.1-1) and reactor protection system (RPS) trip setpoints (Table 2.3-1) provide core protection safety limits and Variable Low Pressure Trip setpoints developed in accordance with NRC-approved methods and assumptions. The transition core penalty resulting from Mark-B12 fuel with fine mesh debris filters co-residing with earlier, non debris filter Mark-B fuel has been demonstrated to be sufficiently bounded by the margin retained for such purposes through the use of a TDL of 1.40 in all DNB analyses supporting the proposed amendment. Therefore the previous commitment to require a higher minimum RCS flow (105.5% of design flow instead of 104.5%) to offset transition core penalties is no longer necessary. These changes have been evaluated for their impact on the design and operation of plant structures, systems, and components. These changes do not introduce any new accident precursors and do not involve any alterations to plant configurations, which could initiate a new or different kind of accident.

The proposed change to Technical Specification 3.5.2.2.a is only an administrative correction.

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

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

Response: No.

The proposed reactor protection system (RPS) trip setpoints (Table 2.3-1) ensure core protection safety limits will be preserved during power operation. The proposed safety limits and setpoints 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 of 1.40 in all DNB analyses supporting the proposed change, has been shown to be sufficient to offset the current mixed core conditions at TMI Unit 1. The margin available between minimum DNBR results for UFSAR loss of coolant flow events and the Thermal Design Limit of 1.40 is significant and is similar to DNB margin results for the current non-SCD analysis.

Reload cycles are designed and operated with maximum steady-state radial-local peaking factors that are bounded by UFSAR assumptions used to determine the dose consequences from fuel handling accidents.

The proposed change to Technical Specification 3.5.2.2.a is only an administrative correction.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, AmerGen Energy Company, LLC (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.

8.0 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.

9.0 PRECEDENT

The Framatome ANP statistical core design methodology (BAW-10187P-A) has been implemented at all other B&W 177 FA plants for which Framatome ANP performs reload licensing.

10.0 REFERENCES

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

b) BAW-10179P-A, Rev. 4, "Safety Criteria And Methodology For Acceptable Cycle Reload Analyses," Framatome Cogema Fuels, Lynchburg, Virginia, August 2001.

c) 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.

d) 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.

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

ENCLOSURE 2

TMI Unit 1 Technical Specification Change Request No. 316

Markup of Proposed Technical Specification Page Changes

Revised TS Pages

2-1
2-2
2-3
2-4a
2-4c
2-6
2-7
2-8
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4-4

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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

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operational transients, and anticipated transients is limited to 1.30 (BAW-2) and 1.18 (BWC). A DNBR of 1.30 (BAW-2) 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 overpower. 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.

This corresponds to a Statistical Design Limit (SDL) of 1.313 (BWC) which accounts for all uncertainties considered with the statistical core design methodology (Reference 4).

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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) 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. ~~However, a higher minimum flow rate (105.5% of 352,000 gpm) is specified in order to offset transition core effects due to the introduction of the Mark B12 fuel design with fine mesh debris filter.~~

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 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 ~~1.30 (BAW-2) or 1.18 (BWC)~~ the Statistical Design Limit (SDL) of 1.313 (BWC) 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.

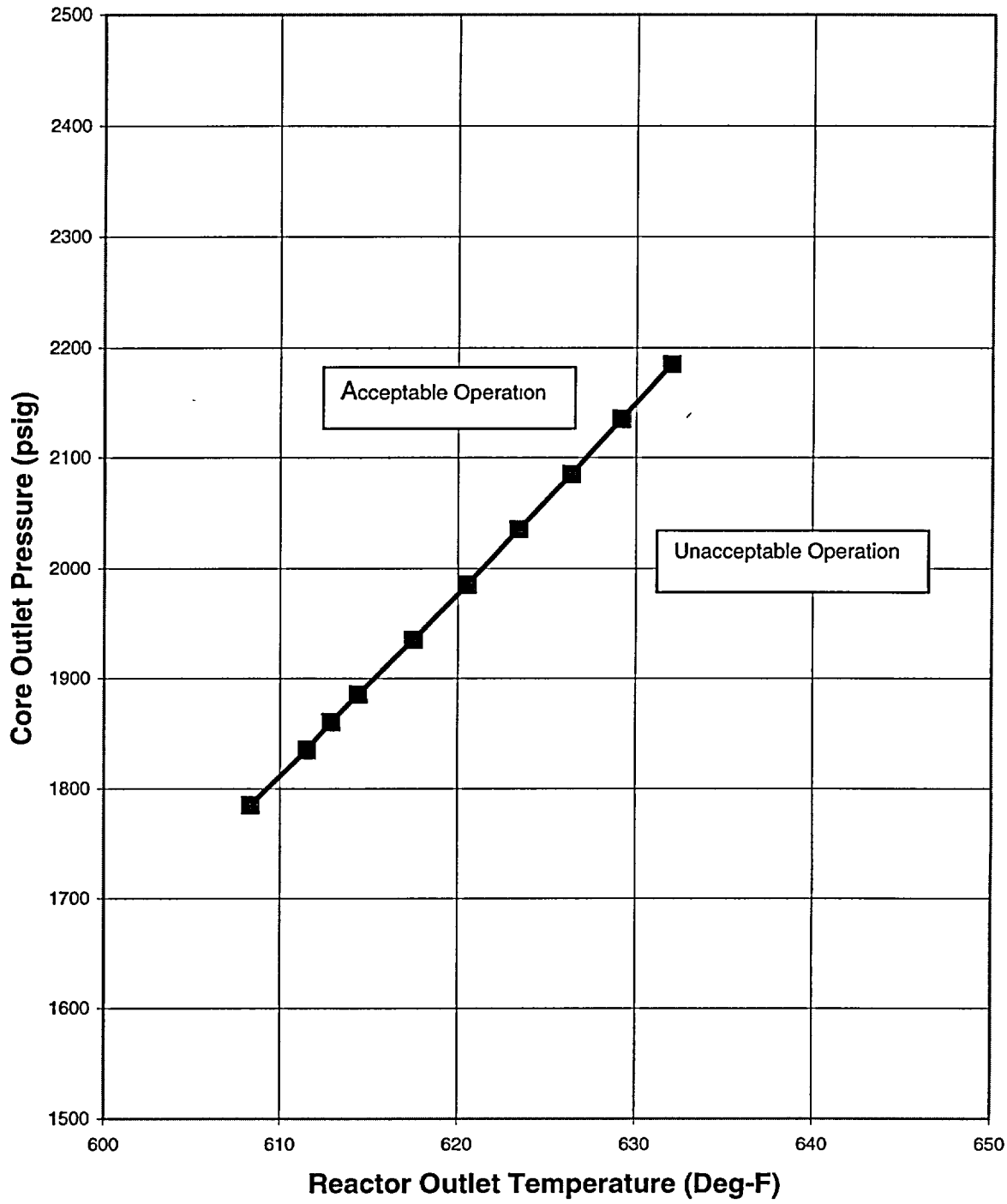
REFERENCES

(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

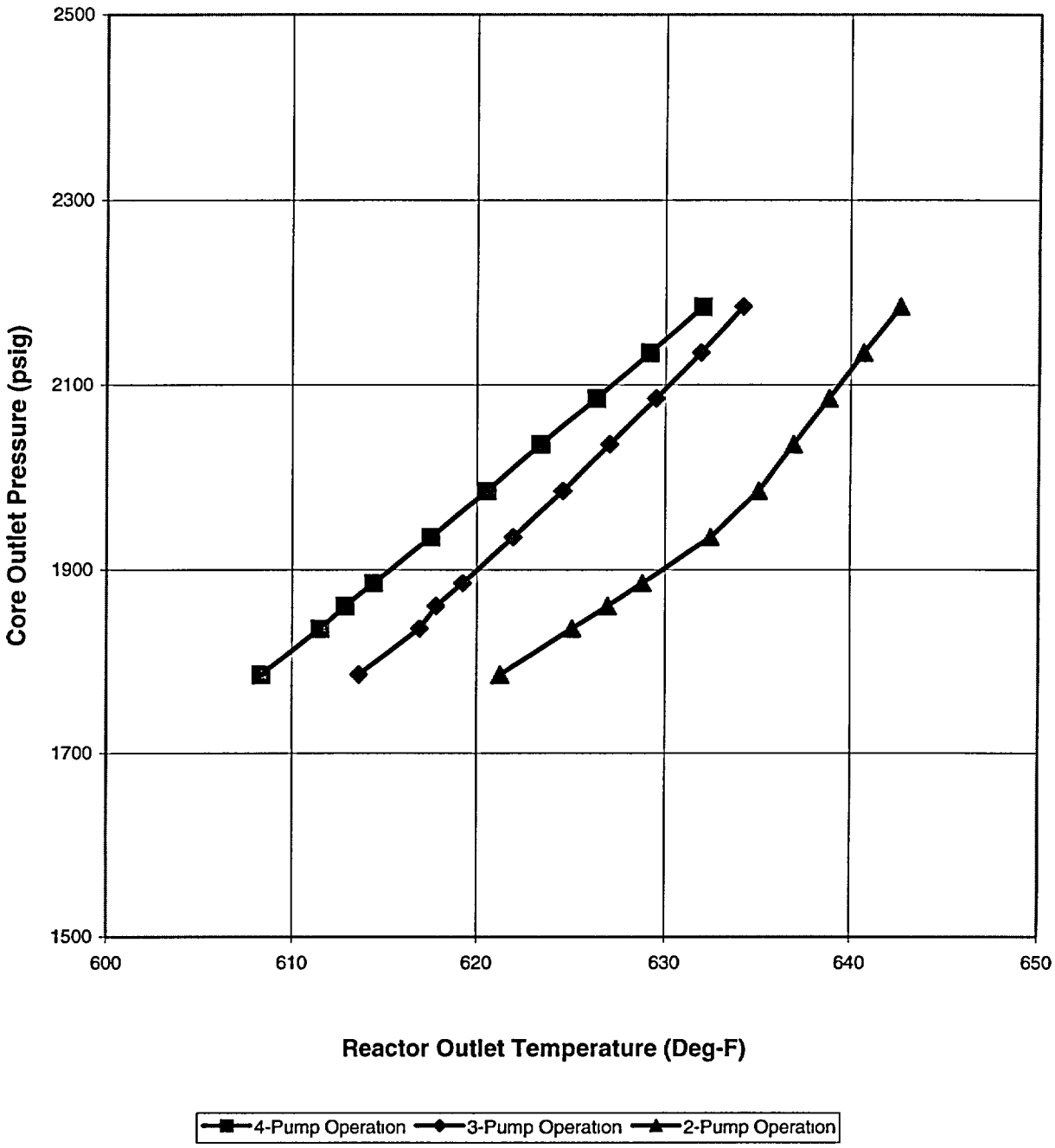
(3) UFSAR, Section 3.2.3.1.1.3 - "Nuclear Power Factors"

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



CORE PROTECTION SAFETY LIMIT
TMI-1
FIGURE 2.1-1

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



RC Pumps	Reactor Coolant Flow (lbs/hr)	Power	Pumps Operating (Type of Limit)
4	137.77×10^6	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

Amendment No. ~~50, 126, 142, 167, 184, 214, 238~~,

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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 ~~1.30 (BAW-2) or 1.18~~ (BWC) should a low flow condition exist due to any malfunction. the Statistical Design Limit of 1.313

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

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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 Imblance in the COLR are produced.

b. Pump Monitors

The redundant pump monitors prevent the minimum core DNBR from decreasing below the Statistical Design Limit of 1.313 ~~1.30 (BAW-2) or 1.18~~ (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 overpressurization 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 ~~(1800 psig)~~ and variable low pressure ~~(11.75 T_{out} - 5103)~~ 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. ~~Application of the B&W~~

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 of 1.313 (BWC). Figure 2.3-1 shows the high pressure, low pressure, high temperature and variable low pressure trip setpoints.

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~~crossflow model resulted in safety limits (see Figures 2.1-1 and 2.1-3) outside the acceptable operating region formed by the low pressure, high pressure, and high temperature trip setpoints (see Figure 2.3-1) which justifies the removal of the variable low pressure trip.~~

d. Coolant outlet temperature

The high reactor coolant outlet temperature trip setting limit (618.8F) shown in Figure 2.3-1 has been established to prevent excessive core coolant temperature in the operating range.

The calibrated range of the temperature channels of the RPS is 520° to 620°F. The trip setpoint of the channel is 618.8F. Under the worst case environment, power supply perturbations, and drift, the accuracy of the trip string is 1.2°F. This accuracy was arrived at by summing the worst case accuracies of each module. This is a conservative method of error analysis since the normal procedure is to use the root mean square method.

Therefore, it is assured that a trip will occur at a value no higher than 620°F even under worst case conditions. The safety analysis used a high temperature trip set point of 620°F.

The calibrated range of the channel is that portion of the span of indication which has been qualified with regard to drift, linearity, repeatability, etc. This does not imply that the equipment is restricted to operation within the calibrated range. Additional testing has demonstrated that in fact, the temperature channel is fully operational approximately 10% above the calibrated range.

Since it has been established that the channel will trip at a value of RC outlet temperature no higher than 620°F even in the worst case, and since the channel is fully operational approximately 10% above the calibrated range and exhibits no hysteresis or foldover characteristics, it is concluded that the instrument design is acceptable.

e. Reactor building pressure

The high reactor building pressure trip setting limit (4 psig) provides positive assurance that a reactor trip will occur in the unlikely event of a steam line failure in the reactor building or a loss-of-coolant accident, even in the absence of a low reactor coolant system pressure trip.

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 imblance 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)	$(16.25\,T_{OUT} - 8113)$(6)	$(16.25\,T_{OUT} - 8113)$(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).

Amendment No. ~~43~~, 78, 90, 128, 138, 142, ~~184~~,

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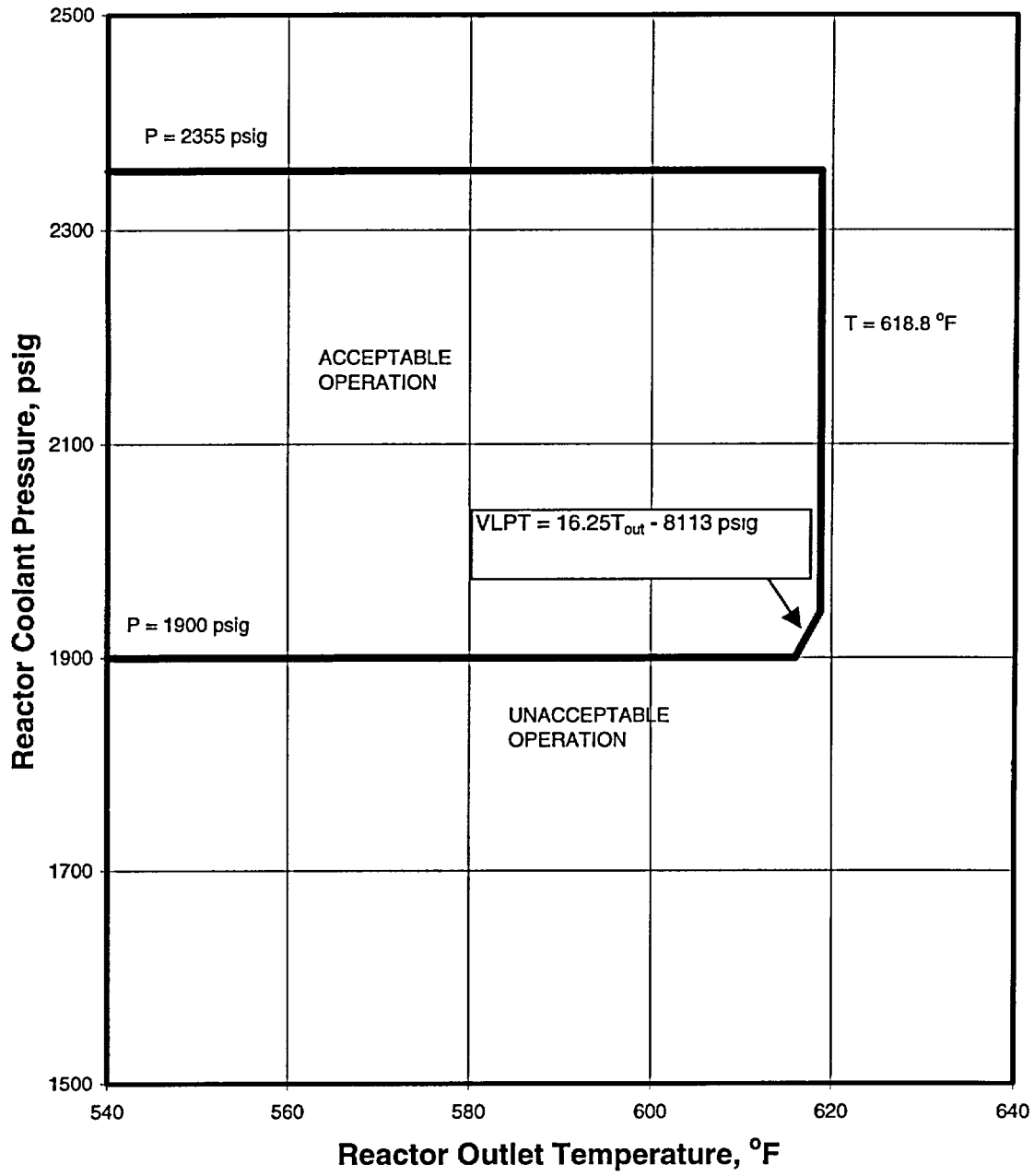

PROTECTION SYSTEM MAXIMUM
ALLOWABLE SETPOINTS
TMI-1
FIGURE 2.3-1

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

TABLE 3.5-1

INSTRUMENTS OPERATING CONDITIONS

Functional Unit	(A) Minimum Operable Channels	(B) Minimum Degree of Redundancy	(C) Operator Action if Conditions of Column A and B Cannot be Met
A. Reactor Protection System			
1. Manual pushbutton	1	0	(a)
2. Power range instrument channel	2	1	(a)
3. Intermediate range instrument channels	1	0	(a) (b)
4. Source range instrument channels	1	0	(a) (c)
5. Reactor coolant temperature instrument channels	2	1	(a)
6. ~~(Deleted)~~ Reactor Coolant Pressure-Temperature Instrument Channels	2	1	(a)
7. Flux/imbalance/flow	2	1	(a)
8. Reactor coolant pressure			
a. High reactor coolant pressure instrument channels	2	1	(a)
b. Low reactor coolant pressure instrument channels	2	1	(a)

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3.5.2 CONTROL ROD GROUP AND POWER DISTRIBUTION LIMITS

Applicability

This specification applies to power distribution and operation of control rods during power operation.

Objective

To assure an acceptable core power distribution during power operation, to set a limit on potential reactivity insertion from a hypothetical control rod ejection, and to assure core subcriticality after a reactor trip.

Specification

3.5 2 1 The available shutdown margin shall not be less than one percent $\Delta K/K$ with the highest worth control rod fully withdrawn.

3 5 2.2 Operation with inoperable rods·

a Operation with more than one inoperable rod as defined in Specification 4.7.1 ~~and 4.7.2.3~~ in the safety or regulating rod banks shall not be permitted. Verify SDM $\geq$ 1% $\Delta k/k$ or initiate boration to restore within limits within 1 hour. The reactor shall be brought to HOT SHUTDOWN within 6 hours.

b If a control rod in the regulating and/or safety rod banks is declared inoperable in the withdrawn position as defined in Specification Paragraph 4.7 1.1 and 4.7.1.3, an evaluation shall be initiated immediately to verify the existence of one percent $\Delta k/k$ hot shutdown margin. Boration may be initiated to increase the available rod worth either to compensate for the worth of the inoperable rod or until the regulating banks are fully withdrawn, whichever occurs first. Simultaneously a program of exercising the remaining regulating and safety rods shall be initiated to verify operability

c If within one hour of determination of an inoperable rod as defined in Specification 4.7.1, and once per 12 hours thereafter, it is not determined that a one percent $\Delta k/k$ hot shutdown margin exists combining the worth of the inoperable rod with each of the other rods, the reactor shall be brought to the HOT SHUTDOWN condition within 6 hours until this margin is established

d Following the determination of an inoperable rod as defined in Specification 4.7.1, all rods shall be exercised within 24 hours and exercised weekly until the rod problem is solved.

e. If a control rod in the regulating or safety rod groups is declared inoperable per 4.7.1.2, and cannot be aligned per 3.5.2.2.f, power shall be reduced to $\leq$ 60% of the thermal power allowable for the reactor coolant pump combination within 2 hours, and the overpower trip setpoint shall be reduced to $\leq$ 70% of the thermal power allowable within 10 hours. Verify the potential ejected rod worth (ERW) is within the assumptions of the ERW analysis and verify peaking factor ($F_Q(Z)$ and $F^N_{\Delta H}$) limits per the COLR have not been exceeded within 72 hours

Amendment No ~~17, 150, 211~~,
(5-18-76)

TABLE 4.1-1 (Continued)

CHANNEL DESCRIPTION	CHECK	TEST	CALIBRATE	REMARKS
8. High Reactor Coolant Pressure Channel	S	M	R	
9. Low Reactor Coolant Pressure Channel	S	M	R	
10. Flux-Reactor Coolant Flow Comparator	S	M	F	
11. ~~(Deleted)~~ Reactor Coolant Pressure - Temperature Comparator	S	M	R	
12. Pump Flux Comparator	S	M	R	
13. High Reactor Building Pressure Channel	S	M	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. Lower 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
18. Reactor Building Emergency Cooling and Isolation System Logic Channel	NA	Q	NA	

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