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10 CFR 50.90

July 25, 2007
5928-07-20164

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

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

Subject: Response To Request For Additional Information – Technical Specification Change Request No. 335: Reactor Coolant System Pressure-Temperature Safety Limit (TAC No. MD4910)

References:
1) USNRC letter to AmerGen Energy Company, LLC dated July 11, 2007, "Request for Additional Information Regarding the High Thermal Performance (HTP) Fuel Reactor Coolant System Pressure-Temperature Safety Limit Technical Specification Amendment (TAC No. MD4910)
2) AmerGen Energy Company, LLC letter to NRC dated March 22, 2007 (5928-07-20006), "Technical Specification Change Request No. 335 – Reactor Coolant System Pressure-Temperature Safety Limit."

This letter provides additional information in response to the NRC request for additional information (RAI), dated July 11, 2007 (Reference 1), regarding TMI Unit 1 Technical Specification Change Request No. 335, submitted to NRC for review on March 22, 2007 (Reference 2). The additional information is provided in Enclosure 1.

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.

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Regulatory commitments established by this submittal are identified in Enclosure 4. 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. Executed on the 25th day of July, 2007.

Sincerely,

Russell G. West
Vice President TMI, Unit 1

Enclosure:
1) Response to Request for Additional Information (Proprietary Version)
2) Response to Request for Additional Information (Non-Proprietary Version)
3) AREVA NP Affidavit Certifying Request For Withholding From Public Disclosure
4) List of Commitments

cc: S. J. Collins, USNRC Administrator, Region I
P. J. Bamford, USNRC Project Manager, TMI Unit 1
D. M. Kern, USNRC Senior Resident Inspector, TMI Unit 1
File No. 07005

ENCLOSURE 2

TMI UNIT 1

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION
TECHNICAL SPECIFICATION CHANGE REQUEST No. 335
Reactor Coolant System Pressure-Temperature Safety Limit

(Non-Proprietary Version)

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION (RAI)
TMI UNIT 1 TECHNICAL SPECIFICATION CHANGE REQUEST No. 335
Reactor Coolant System Pressure-Temperature Safety Limit

1. **NRC Question**

The licensee indicated that the transient core penalty for a specific transient core model was calculated based on the largest departure from nucleate boiling (DNB) ratio difference between the limiting Mark-B-HTP fuel rod in a full core model of the Mark-B-HTP fuel and specific transient core model for all of the statepoints, condition I/II DNB transients, and axial power shapes.

Discuss the results of the transient core penalty analyses for condition I/II DNB transients with associated axial power shapes, and demonstrate that (1) the analysis scope in terms of the applicable condition I/II DNB transients and the allowable axial power shapes is adequate, and (2) the calculated value of the transient core penalty is a bounding value and acceptable for determining the thermal design limit that is used to calculate the reactor core safety limit.

Response

For this response, the terms "transient core penalty" and "transient core model" are interpreted to mean "transition core penalty" and "transition core model," respectively. This discussion parallels the response to a similar question for the Mark-B-HTP implementation at CR-3. This response contains an explanation on how the transition core, or mixed core, DNB penalty was computed to protect steady-state operation as well as during the Condition I/II DNB transients for TMI with the implementation of the Mark-B-HTP fuel design.

The mixed core penalty for TMI was determined using the NRC-approved LYNXT core thermal-hydraulic code (Reference 1). The code is approved for DNBR predictions under steady-state and transient conditions using single-pass multi-channel modeling.

The DNB analysis of record for the implementation of the Mark-B-HTP fuel design at TMI is based on the use of a LYNXT model representing a full core of Mark-B-HTP fuel. This LYNXT model, with a full core of Mark-B-HTP fuel, is used to develop the pressure-temperature DNB safety limits and to evaluate the limiting DNB transients.

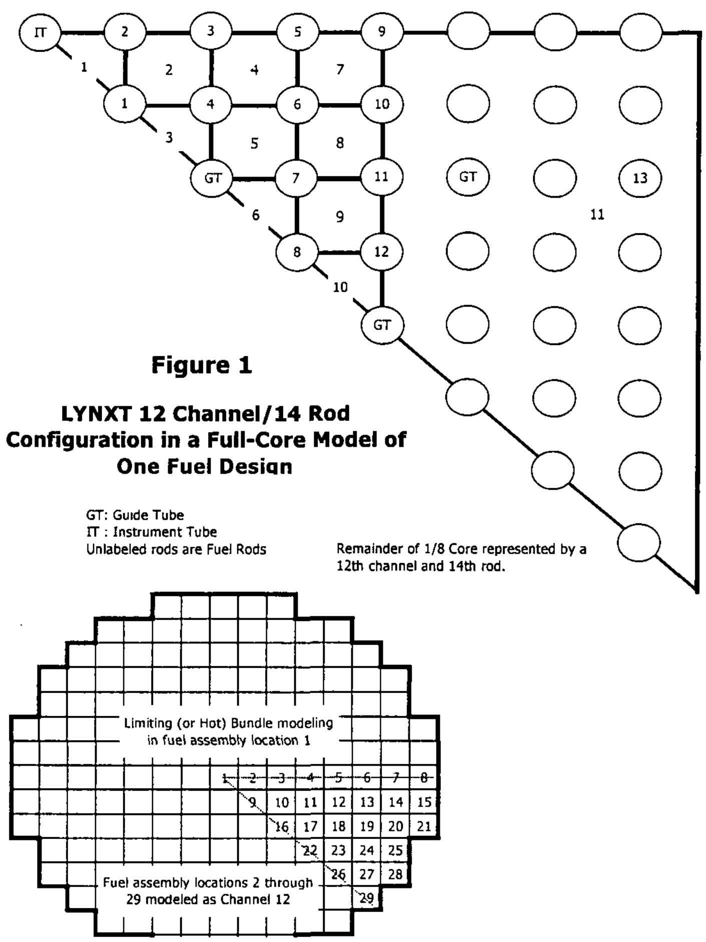


Figure 1

LYNXT 12 Channel/14 Rod Configuration in a Full-Core Model of One Fuel Design

The DNB impact of the mixed core operation, during transition cycles, is quantified to determine whether an adjustment (penalty) is necessary against the DNB predictions using the LYNXT model with a full core of Mark-B-HTP.

First, it might be beneficial to explain the LYNXT model with a full core of Mark-B-HTP fuel. For TMI Cycle 17, the core is modeled as 177 Mark-B-HTP fuel assemblies (using a 1/8 core symmetry model with LYNXT) as shown in Figure 1. This model is composed of 12 channels. The limiting, or hot, bundle is modeled at the core center.

Channels 1 through 10 represent individual subchannels. Channel 11 represents the remainder of the hot bundle. Channel 12 represents the remainder of the core. The limiting location for the placement of the hot pin is Rod 6 for the BHTP correlation (Reference 2). The limiting fuel rod is modeled as a 1.800 $F_{\Delta H}$ with a 1.65 symmetric axial power shape.

This LYNXT model has been used to determine the DNB predictions for steady-state performance, including the pressure-temperature safety limits for establishing the variable low pressure trip (VLPT), and the transient performance.

The Mark-B-HTP fuel design has slightly different hydraulic characteristics than the resident Mark-B12 fuel design at the lower end fitting and at all the spacer grids. The net effect of these differences results in flow diversion out of the Mark-B-HTP fuel. Hydraulic testing has been used to determine the hydraulic form loss coefficients for the assembly hardware for the Mark-B-HTP and Mark-B fuel designs.

The LYNXT model for the transition core is composed of 64 channels and 60 rods as shown in Figure 2. Again, the limiting fuel rod is modeled as a 1.800 $F_{\Delta H}$ with a 1.65 symmetric axial power shape at Rod 6 when the hot bundle is modeled as a Mark-B-HTP fuel assembly and the DNB performance is predicted using the BHTP correlation. For cases where the hot bundle is modeled as a Mark-B12 fuel assembly, the limiting fuel rod, 1.800 $F_{\Delta H}$ with a 1.65 symmetric axial power shape, is placed at Rod 2 for DNB predictions using the BWC CHF correlation (Reference 3). AREVA has determined the limiting hot rod location for the respective CHF correlations by moving the hot rod throughout the hot bundle to isolate the most severe DNB response. This action assures the most conservative DNB prediction for the design power distribution (1.800 $F_{\Delta H}$ with a 1.65 symmetric axial power shape at the hot rod) for each fuel design.

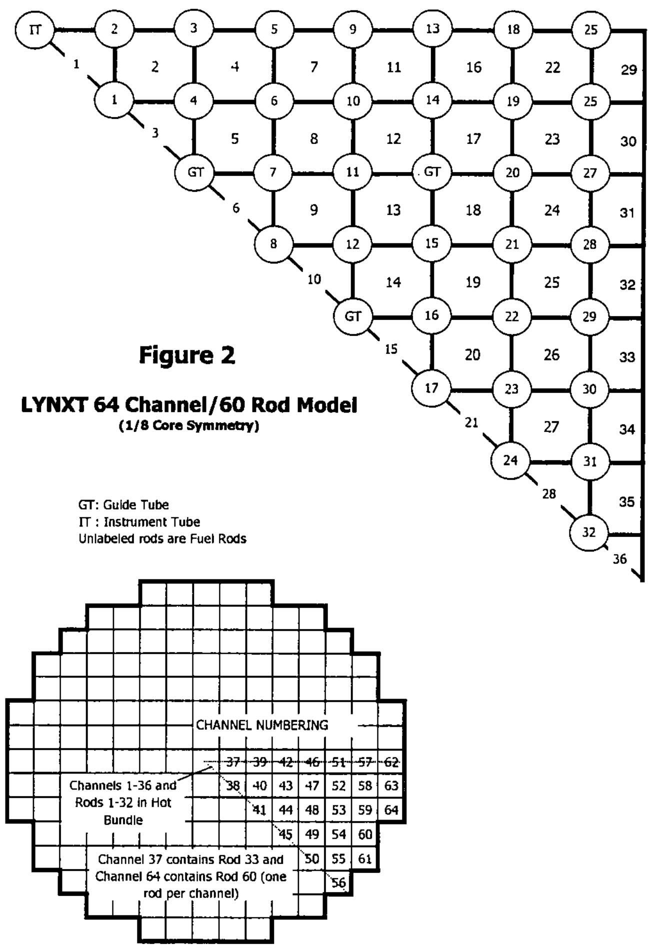


Figure 2

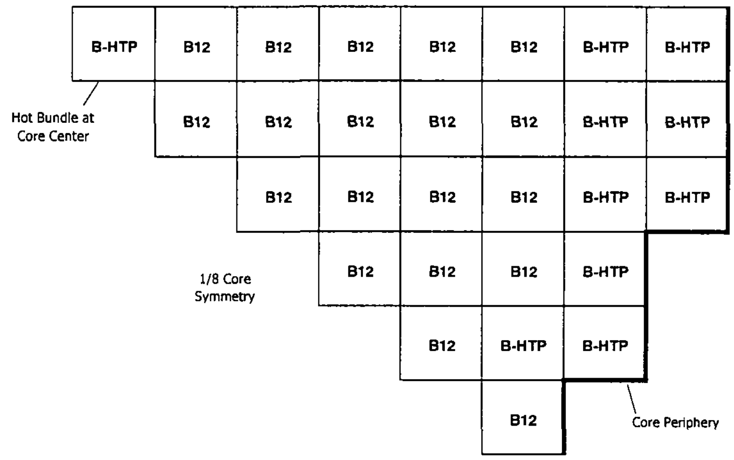
LYNXT 64 Channel/60 Rod Model
(1/8 Core Symmetry)

Two basic core configurations were examined to bound the DNB performance for the fresh Mark-B-HTP fuel and the resident Mark-B12 fuel designs during the transition cycles.

Configuration 1: Mark-B-HTP fuel contains the limiting hot rod.
Configuration 2: Mark-B12 fuel contains the limiting hot rod.

Figure 3

Transition Core Configuration for Mark-B-HTP Hot Bundle with 85 Mark-B-HTP Fuel Assemblies in the Core



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Examining both basic configurations assures the DNB performance of the core is captured whether the limiting power, hot rod, resides in the Mark-B-HTP fuel or in the Mark-B12 fuel.

Figure 3 shows the core Configuration1 that accentuates the flow diversion out of the Mark-HTP hot bundle in the transition cycle assuming there are 65 Mark-B-HTP fuel assemblies in the core. (This assessment was actually performed for different numbers of Mark-B-HTP fuel assemblies in the core. However, the details being described here are for the case with 65 Mark-B-HTP fuel assemblies in the core.) By conservatively placing all the lower pressure drop Mark-B12 fuel around the limiting Mark-B-HTP hot bundle, the most conservative DNB penalty can be determined for a limiting Mark-B-HTP fuel assembly.

The transition core DNB penalty associated with the Mark-B-HTP hot bundle was computed by determining the difference between the DNB performance of the transition core, represented in Figure 3, and the DNB performance obtained using a full core model of Mark-B-HTP fuel assemblies. Both LYNXT models used the 64 channel/60 rod LYNXT model as shown in Figure 2.

The transition core DNB penalty for the Mark-B-HTP hot bundle is computed as follows.

Transition Core DNB Penalty for Mark-B-HTP Hot Bundle	=	DNB Prediction for Full Core of Mark-B-HTP Fuel	-	DNBR Prediction for Mark-B-HTP Hot Bundle in a Bounding Transition Core Configuration

If the DNB prediction for a Mark-B-HTP hot bundle is lower in the transition core than in the full core model for a given statepoint condition, then the DNB penalty is positive indicating the transition situation is not bounded by the DNB analysis based on a full core of Mark-B-HTP fuel.

The transition core DNB penalty for TMI was determined by examining this DNB difference for approximately 100 different operating conditions. The conditions included:

a) steady-state cases at the core protection safety limit line evaluated across a wide range of axial power distributions (highly inlet skewed to highly outlet skewed), and

b) the limiting Condition I/II DNB event (4 pump coastdown) across a wide range of axial power distributions (highly inlet-skewed to highly outlet-skewed).

Figure 4

Thermal Design Limit (TDL) Basis for the Mark-B-HTP Fuel Design Implementation

Once all the operating conditions are evaluated, the maximum positive penalty is then assessed against the Mark-B-HTP fuel assembly.

As long as the maximum positive DNB penalty is smaller than the DNB margin reserved between the Thermal Design Limit (TDL) and Statistical Design Limit (SDL) shown in Figure 4 for the Mark-B-HTP fuel design, then the Mark-B-HTP full core DNB analysis of record is bounding and applicable for the Cycle 17 transition core. Using the procedure described above, the transition core DNB penalty for the Mark-B-HTP was found to be [ ] DNB points (where 1 DNB point =0.01) when using the conservative core configuration show in Figure 3. Sufficient margin has been reserved between the TDL and SDL to offset this transition core penalty.

This same procedure is performed for the core Configuration 2 where the hot bundle is the resident fuel design, or Mark-B12. In Figure 5 one can see the placement of the Mark-B12 fuel design into the hot bundle location. In order to maximize the diversion of flow out of the Mark-B12 hot bundle the hot bundle is surrounded with other Mark-B12 fuel assemblies (having a lower pressure drop than the Mark-B-HTP fuel). The placement of Mark-B-HTP fuel adjacent to or near the Mark-B hot bundle would reduce the amount of coolant being diverted from the Mark-B12 hot bundle.

The transition core DNB penalty associated with the Mark-B12 hot bundle was computed by determining the difference between the DNB performance of the transition core, represented in Figure 5, and the DNB performance obtained using a full core model of Mark-B12 fuel assemblies. Both LYNXT models used the 64 channel/60 rod LYNXT model as shown in Figure 2.

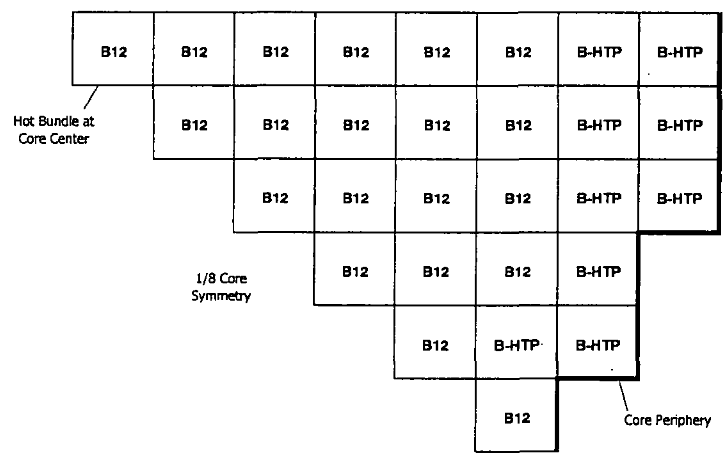
The transition core penalty for the Mark-B12 hot bundle is computed as follows.

Transition Core DNB Penalty for Mark-B12 Hot Bundle = DNB Prediction for Full Core of Mark-B12 Fuel - DNBR Prediction for Mark-B12 Hot Bundle in a Bounding Transition Core Configuration

If the DNB prediction for a Mark-B12 hot bundle is lower in the transition core than in a full core model for a given statepoint condition, then the DNB penalty is positive indicating the transition situation is not bounded by the DNB analysis based on a full core of Mark-B12 fuel. After determining the DNB difference for the ~100 operating conditions with various axial power distributions, it was concluded that no DNB penalty was necessary for the Mark-B12 resident fuel design. In every case more flow was passing through the Mark-B12 hot bundle as a result of the Mark-B-HTP fuel in the core than would pass through the hot bundle for a full core of Mark-B12 fuel.

Figure 5

Conservative Transition Core Configuration for Mark-B Hot Bundle
(Number of Mark-B-HTP Modeled is less than Actual number in Core)



Therefore, a transition core penalty exists only for the Mark-B-HTP hot bundle in the TMI core. Analyses have been performed to quantify the transition core penalty as a function of the number of Mark-B-HTP fuel assemblies in the core. Figure 6 shows the results of that study. Since TMI Cycle 17 is scheduled to have 72 Mark-B-HTP fuel assemblies in the core, one can observe a [ ] DNB point transition core penalty, where 1 DNB point = 0.01, when using a bounding core configuration with a single Mark-B-HTP surrounded by Mark-B12 fuel. An additional

Figure 6

Transition Core DNB Penalty as function of the Number of Mark-B-Fuel Assemblies in the

[ ]

assessment was performed for the case with 85 Mark-B-HTP fuel assemblies in the core using a typical checkerboard pattern of fresh and irradiated fuel. The assessment shows the bounding core configuration is more conservative than the checkerboard configuration by about [ ] DNB points. The resulting sensitivity shows the transition core penalty decreases approximately [ ] DNB point for every 8 Mark-B-HTP fuel assemblies added to the core.

Summarizing, the transition core DNB penalty that will be used in the TMI Cycle 17 analysis of record will be [ ] DNB points. This penalty has been shown to conservatively bound the planned Cycle 17 core configuration and will remain bounding and applicable for Cycle 18 when more Mark-B-HTP fuel assemblies will be introduced into the TMI core. The [ ] DNB point penalty for Cycle 17 will be offset by the DNB margin retained in the TDL value of [ ].

REFERENCES

1. BAW-10156-A, Rev.1, "LYNXT Core Transient Thermal-Hydraulic Program", B&W Fuel Company, Lynchburg, Virginia, August 1993.
2. BAW-10241(P)(A), Revision 1, "BHTP DNB Correlation Applied With LYNXT".
3. BAW-10143P-A, "BWC Correlation of Critical Heat Flux", Babcock & Wilcox, Lynchburg, Virginia, April 1985.

2. **NRC Question**

Provide the basis for your determination that no setpoint changes to reactor trip functions, other than the variable low reactor coolant system pressure trip, are needed to assure that the analyses of record remain bounding, or new analyses meet the applicable acceptance criteria for design-basis events.

Response

All of the TMI Unit 1 Updated Final Safety Analysis Report (UFSAR) Chapter 14 LOCA and non-LOCA events were evaluated with respect to the Mark-B-HTP fuel design. The change in fuel design does not have a direct effect on the secondary side or the heat transfer to the secondary side. On the primary side, the severity of the non-LOCA events is sensitive to changes to the core power, temperature, pressure, and flow rate. Of these four parameters, only the pressure and flow have the potential to be directly affected by the introduction of the Mark-B-HTP fuel.

The Mark-B-HTP fuel has an increased flow resistance that will result in a lower Reactor Coolant System (RCS) flow. Calculations determined the change in best-estimate RCS flow to be less than [ ]. As explained in BAW-10193P-A, "RELAP5/MOD2-B&W for Safety Analysis of B&W-Designed Pressurized Water Reactors," the non-LOCA analysis methodology for B&W-designed plants uses the thermal design flow as the initial RCS flow for the analysis of system response. The thermal design flow is several percent lower than the best-estimate flow; therefore, the small decrease in best-estimate flow will not invalidate the initial RCS flow used in the analyses of record. This is validated during physics testing performed each cycle. This leaves the increased pressure drop associated with the Mark-B-HTP fuel as the only parameter that might potentially affect the severity of the non-LOCA analyses of record.

The effect of the increased pressure drop associated with Mark-B-HTP fuel on the analyses of record was evaluated. The pressure drop in the core does not directly affect the nominal operating pressure measured in the hot leg or the change in pressure from the hot leg to the core exit. However, the increased core pressure drop could result in a higher maximum pressure in the reactor coolant system. An evaluation was performed to estimate the increase in the core pressure drop associated with Mark-B-HTP fuel. The evaluation estimated the pressure increase to be [ ] psi or less. As a conservative estimate, the pressure increase was added directly to the peak system pressure determined in the non-LOCA accidents and compared to the acceptance criteria. In all cases, the accidents have sufficient margin to the applicable limit to accommodate the additional [ ] psi. Therefore, it was concluded that the fuel design has a negligible effect on the overall system response in the analyses of record.

Since the system response for the analyses of record are valid and applicable to a core design with Mark-B-HTP fuel, the high- and low- RCS pressure setpoints, the high temperature setpoint, the high containment pressure and the high flux setpoints in TMI Unit 1 Technical Specification Table 2.3-1 would not be affected by the change in fuel design.

The Mark-B-HTP LOCA transition analyses for TMI Unit 1 utilized the same set of core parameters and boundary conditions as the previous analysis of record. The results of the transition analyses, that considered a mixed-core with Mark-B12 or Mark-B10 fuel, demonstrated that all 5 of the 10 CFR 50.46 criteria were met with an abundance of margin. Analyses with a full-core of Mark-B-HTP fuel with the same set of core parameters and boundary conditions would result in increased margin due to the removal of the mixed-core penalty. Therefore, there is no need for plant setpoint changes as a result of the transition to the Mark-B-HTP fuel.

The limiting DNBR transients are the Loss of Coolant Flow events. The three most DNB-limiting transients in the TMI Unit 1 Updated Final Safety Analysis Report (UFSAR) that are impacted by the implementation of the Mark-B-HTP fuel assembly 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)

All of these events were analyzed using the AREVA NP NRC-approved LYNXT thermal-hydraulic code with the BHTP correlation that has been NRC-approved for the Mark-B-HTP fuel design. A DNBR analysis was performed with equivalent or bounding plant characteristics (includes setpoints, delays, and instrument errors associated with the power/pump monitor and flux-to-flow trips).

For the single pump coastdown (4-to-3) event the reactor continues to operate at full power until flow decreases to the point where the flux-flow setpoint (1.08 %FP/%flow) initiates a reactor trip. The LYNXT results for this event using the Statistical Core Design (SCD) methodology is a MDNBR of 1.898, which is higher than the Thermal Design Limit (TDL) of [ ] [ ], where 1 DNBR point is equal to 0.01 absolute, indicating that the existing flux-to-flow setpoint provides adequate DNB protection for this event with no predicted fuel failures.

For the four pump coastdown (4-to-0) event the 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 1.766 which is higher than the TDL of [ ] [

], indicating that the existing power/pump monitor trip provides adequate DNB protection for this event with no predicted fuel failures.

For the single pump locked rotor event (4-to-3) the reactor continues to operate at full power until flow decreases to the point where the flux-to-flow setpoint initiates a reactor trip. The LYNXT results for this event using the SCD methodology is a MDNBR of 1.638 which is lower than the TDL of [ ] [ ], indicating that a [ ] DNBR penalty will be applied to the reserved margin to satisfy the locked rotor event with no predicted fuel failures.

The reserved margin or retained thermal margin is the difference between the TDL and the SDL as described in BAW-10179P-A, "Safety Criteria and Methodology for Acceptable Cycle Reload Analyses," Section 6.2.8. The retained thermal margin is used to offset penalties such as transition core effects and cycle anomalies, and provides core design flexibility.

The TMI Unit 1 UFSAR evaluation for Control Rod Assembly (CRA) ejection DNBR pin census states that 17.5% of the fuel pins can experience DNBR for the site to meet its dose requirement. The result for the CRA ejected rod DNB analysis is a MDNBR of 1.61, which is lower than the TDL [ ] [ ]. The results show acceptable DNBR performance for an ejected rod scenario with a minimum of [ ] DNBR point penalty to the TDL of [ ] which follows the same methodology that was acceptable for the Crystal River Unit 3 plant. The TMI Unit 1 DNBR pin census criterion is met, with a [ ] DNBR point penalty assessed to the retained thermal margin (RTM), and the TMI Unit 1 UFSAR ejected rod analysis of record remains valid.

The following figure shows the application of the retained thermal margin for TMI Unit 1 Cycle 17:

In summary, with respect to the DNB Design Basis Events, the current setpoints are validated for use with Mark-B-HTP fuel assemblies. No Technical Specification setpoints require changes due to the implementation of the Mark-B-HTP fuel assemblies with exception of the variable low pressure trip.

It is noted that the Power/Flow trip setpoints in Technical Specification Table 2.3-1 are included in the Core Operating Limits Report (COLR) 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 critical heat flux (CHF) correlation as described in BAW-10179P-A Section 6.5.

3. **NRC Question**

Identify the reactor coolant pressure and reactor outlet temperature at the end points and mid point of proposed TS Figure 2.1-1.

Response

Core Outlet Pressure (psig)	Reactor Outlet Temperature (°F)
1785.3	**600.79**
1985.3	**614.84**
2185.3	**625.46**

The safety limit analysis provides reactor coolant pressure at the core outlet.

4. **NRC Question**

Identify the low reactor coolant pressure and high reactor outlet temperature intercept points of proposed TS Figure 2.3-1.

Response

VLPT = 16.21 Tout – 7973

If Tout = 618.8°F,
VLPT = (16.21) (618.8°F) – 7973
VLPT = **2057.7 psig**

If VLPT = 1900 psig,
1900 = 16.21 Tout – 7973
Tout = **609.1°F**

5. **NRC Question**

Demonstrate that, for operation with measured values at the limits of the "Acceptable Operation" region of proposed TS Figure 2.3-1, there would be adequate assurance that the actual operating point will be conservative relative to the indicated Safety Limit line, given measurement errors consistent with the instrument channel uncertainties.

Response

The following is a description of the Reactor Protection System (RPS) Variable Low Pressure Trip (VLPT) setpoint calculation.

Allowable Value (AV):

The AV for the VLPT function is:

$$P = (16.21 \text{ psig} / F)(T_{OUT}\, F) - 7973 \text{ psig}$$

This is the proposed Limiting Safety System Setting (LSSS) to be identified in Technical Specifications (TS) Table 2.3-1, Item 8 and Figure 2.3-1.

The AV is based on applying the pressure differential between the core outlet and hot leg tap plus a Total Loop Uncertainty (TLU) of ± 28.148 psig. The TLU is determined by applying the SRSS technique only to those uncertainties that are independent, random, and approximately normally distributed. All other uncertainty components are combined algebraically. The TLU includes allowances for Maintenance and Test Equipment (M&TE) accuracy and instrument drift during the surveillance test interval. The pressure differential and TLU are conservatively added to the DNB analytical limit (i.e., proposed TS Figure 2.1-1) to obtain the AV. The AV ensures the DNB analytical limit will not be exceeded.

Nominal Trip Setpoint (NSP):

The TMI Unit 1 position is that the surveillance test as-found Trip Setpoint (TSP) shall not exceed the AV. The proposed NSP is:

$$P = (14.29 \text{ psig} / F)(T_{OUT}\, F) - 6745 \text{ psig}$$

The NSP is the ideal setpoint for RPS calibration.

The setpoint slope has been reduced from 16.21 to 14.29. The 14.29 slope is 75% of the RPS instrument capability. This slope is 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 point where RCS Temperature is equal to 618.8 F (RCS Temperature LSSS). This is conservative with respect to the AV. The resulting y-intercept is (-) 6785 psig.

A Total Margin of 40 psig is then conservatively added to the (-) 6785 psig y-intercept to obtain the NSP. The Total Margin includes:

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

Therefore, the TLU is applied a second time in determining the Total Margin so that the NSP protects the AV in the same manner that the AV protects the Analytical Limit. The surveillance test procedure NSP as-left tolerance is included in the Total Margin because it is not included in the TLU calculation. The additional discretionary margin provides additional conservatism. The NSP does not significantly impact the normal plant operating region. See Attachments 1 and 2 for graphical illustration.

This approach provides assurance that the actual operating point is conservative relative to the Safety Limit line.

6. **NRC Question**

Identify the normal operating point for the reactor coolant pressure and reactor outlet temperature.

Response

The normal operating reactor coolant pressure is 2155 psig (at the hot leg tap), and the normal operating reactor outlet temperature is 603.7 F.

7. **NRC Question**

Identify the low reactor coolant pressure and high reactor outlet temperature intercept points for:

(a) TS Variable Low Pressure Trip (VLPT) Limiting Safety System Setting Setpoint
(b) Adjusted TS VLPT
(c) VLPT Nominal Setpoint (NSP)
(d) VLPT As-Left Tolerance Band
(e) VLPT Pre-Defined As-Found Tolerance Band.

Response

(a) VLPT (LSSS) = 16.21 Tout – 7973

If Tout = 618.8°F,
VLPT (LSSS) = (16.21) (618.8°F) – 7973
VLPT (LSSS) = **2057.7 psig**

If VLPT (LSSS) = 1900 psig,
1900 = 16.21 Tout – 7973
Tout = **609.1°F**

(b) VLPT (Adjusted LSSS) = 14.29 Tout – 6785

If Tout = 618.8°F,
VLPT (Adjusted LSSS) = (14.29) (618.8°F) – 6785
VLPT (Adjusted LSSS) = **2057.7 psig**

If VLPT (Adjusted LSSS) = 1900 psig,
1900 = 14.29 Tout – 6785
Tout = **607.8°F**

* The adjusted TS VLPT is a conservative adjustment of the LSSS slope to 75% of the RPS instrument capability.

(c) VLPT (NSP) = 14.29 Tout – 6745

If Tout = 618.8°F,
VLPT (NSP) = (14.29) (618.8°F) – 6745
VLPT (NSP) = **2097.7 psig**

If VLPT (NSP) = 1900 psig,
1900 = 14.29 Tout – 6745
Tout = **605.0°F**

(d) VLPT (As-Left (+)) = 14.29 Tout – (6745 – 2.16) (See Note 1 below)

If Tout = 618.8°F,
VLPT (As-Left (+)) = (14.29) (618.8°F) – (6745 – 2.16)
VLPT (As-Left (+)) = **2099.8 psig**

If VLPT(As-Left (+)) = 1900 psig,
1900 = 14.29 Tout – (6745 – 2.16)
Tout = **604.8°F**

VLPT (As-Left (–)) = 14.29 Tout – (6745 + 2.16) (See Note 1 below)

If Tout = 618.8°F,
VLPT (As-Left (–)) = (14.29) (618.8°F) – (6745 + 2.16)
VLPT (As-Left (–)) = **2095.5 psig**

If VLPT (As-Left (–)) = 1900 psig,
1900 = 14.29 Tout – (6745 +2.16)
Tout = **605.1°F**

(e) VLPT (As-Found (+)) = 14.29 Tout – (6745 – 3.52) (See Note 1 below)

If Tout = 618.8°F,
VLPT (As-Left (+)) = (14.29) (618.8°F) – (6745 – 3.52)
VLPT (As-Left (+)) = **2101.2 psig**

If VLPT(As-Left (+)) = 1900 psig,
1900 = 14.29 Tout – (6745 – 3.52)
Tout = **604.7°F**

VLPT (As-Found (–)) = 14.29 Tout – (6745 + 3.52) (See Note 1 below)

If Tout = 618.8°F,
VLPT (As-Left (+)) = (14.29) (618.8°F) – (6745 + 3.52)
VLPT (As-Left (+)) = **2094.1 psig**

If VLPT(As-Left (+) = 1900 psig,
1900 = 14.29 Tout – (6745 + 3.52)
Tout = **605.2°F**

Note 1: The VLPT As-Left tolerance is ± 2.16 psig. The As-Found tolerance is ± 3.52 psig. See Attachments 1 and 2 for graphical illustration.

8. **NRC Question**

Provide additional details concerning the methodology used to develop the VLPT 10 pounds per square inch gauge additional margin. Confirm that this additional margin has not been used in determining the VLPT Pre-Defined As-Found Tolerance Band.

Response

The additional 10 psig margin is purely discretionary. It is included in the Total Margin (See Question 5 response above) to provide additional assurance that the Allowable Value (AV) will not be exceeded. The additional margin will not result in a Nominal Trip Setpoint (NSP) that impacts the normal operating region. The additional margin has not been used in determining the VLPT Pre-Defined As-Found Tolerance Band. The As-Found acceptance criteria is based around the NSP and utilizes no more than the SRSS combination of the reference accuracy, M&TE error, M&TE readability, and instrument drift.

9. **NRC Question**

The licensee has stated that the VLPT 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". In order to assess the adequacy of this methodology please provide a description of:

(a) the criteria (not just the methods) used for combining uncertainties in determining trip setpoints and allowable values.

(b) the relationship of the allowable value to the setpoint methodology and testing requirements. Also, describe the method of documentation of this relationship.

Providing a description of how the regulatory positions of RG 1.105, Revision 3 are incorporated into the VLPT setpoint calculations would constitute an acceptable way to address questions 9(a) and 9(b) above.

Response

REGULATORY GUIDE 1.105, REVISION 3, SECTION C. REGULATORY POSITION

1. Section 4 of ISA-S67.04-1994 specifies the methods, but not the criterion, for combining uncertainties in determining a trip setpoint and its allowable values. The 95/95 tolerance limit is an acceptable criterion for uncertainties. That is, there is a 95% probability that the constructed limits contain 95% of the population of interest for the surveillance interval selected.

The TMI setpoint methodology uses Method 1 from ISA-RP67.04.02-2000 to establish the VLPT Allowable Value (AV). The VLPT Total Loop Uncertainty (TLU) is calculated as described in ISA-RP67.04.02-2000, Section 6. This ensures that the AV meets the 95%/95% criterion endorsed by RG 1.105, Revision 3. See Question 5 response above for description of methodology.

2. Sections 7 and 8 of Part 1 of ISA-S67.04-1994 reference several industry codes and standards. If a referenced standard has been incorporated separately into the NRC's regulations, licensees and applicants must comply with that standard as set forth in the regulation. If the referenced standard has been endorsed in a regulatory guide, the standard constitutes a method acceptable to the NRC staff of meeting a regulatory requirement as described in the regulatory guide. If a referenced standard has been neither incorporated into the NRC's regulations nor endorsed in a regulatory guide, licensees and applicants may consider and use the information in the referenced standard if appropriately justified, consistent with current regulatory practice.

With regard to the VLPT setpoint calculation and the references of Sections 7 and 8 of Part 1 of ISA-S67.04-1994, TMI Unit 1 is in agreement with this Regulatory Position and commits to implement Regulatory Guide 1.105, Revision 3.

3. Section 4.3 of ISA-S67.04-1994 states that the limiting safety system setting (LSSS) may be maintained in technical specifications or appropriate plant procedures. However, 10 CFR 50.36 states that the technical specifications will include items in the categories of safety limits, limiting safety system settings, and limiting control settings. Thus, the LSSS may not be maintained in plant procedures. Rather, the LSSS must be specified as a technical-specification-defined limit in order to satisfy the requirements of 10 CFR 50.36. The LSSS should be developed in accordance with the setpoint methodology set forth in the standard, with the LSSS listed in the technical specifications.

The VLPT limit in TSCR No. 335, 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). The AV is treated as the Limiting Safety System Setting (LSSS). Maintaining the LSSS in the Technical Specifications meets the requirement of this Regulatory Position.

4. ISA-S67.04-1994 provides a discussion on the purpose and application of an allowable value. The allowable value is the limiting value that the trip setpoint can have when tested periodically, beyond which the instrument channel is considered inoperable and corrective action must be taken in accordance with the technical specifications. The allowable value relationship to the setpoint methodology and testing requirements in the technical specifications must be documented.

The Allowable Value (AV) relationship to the setpoint methodology is documented and addressed by the previously proposed TS Table 4.1-1 Note (b). See Question 5 response above for description of methodology. Testing requirements are documented in TSCR No. 335 and Technical Specifications Table 4.1-1. This meets the requirement of this Regulatory Position.

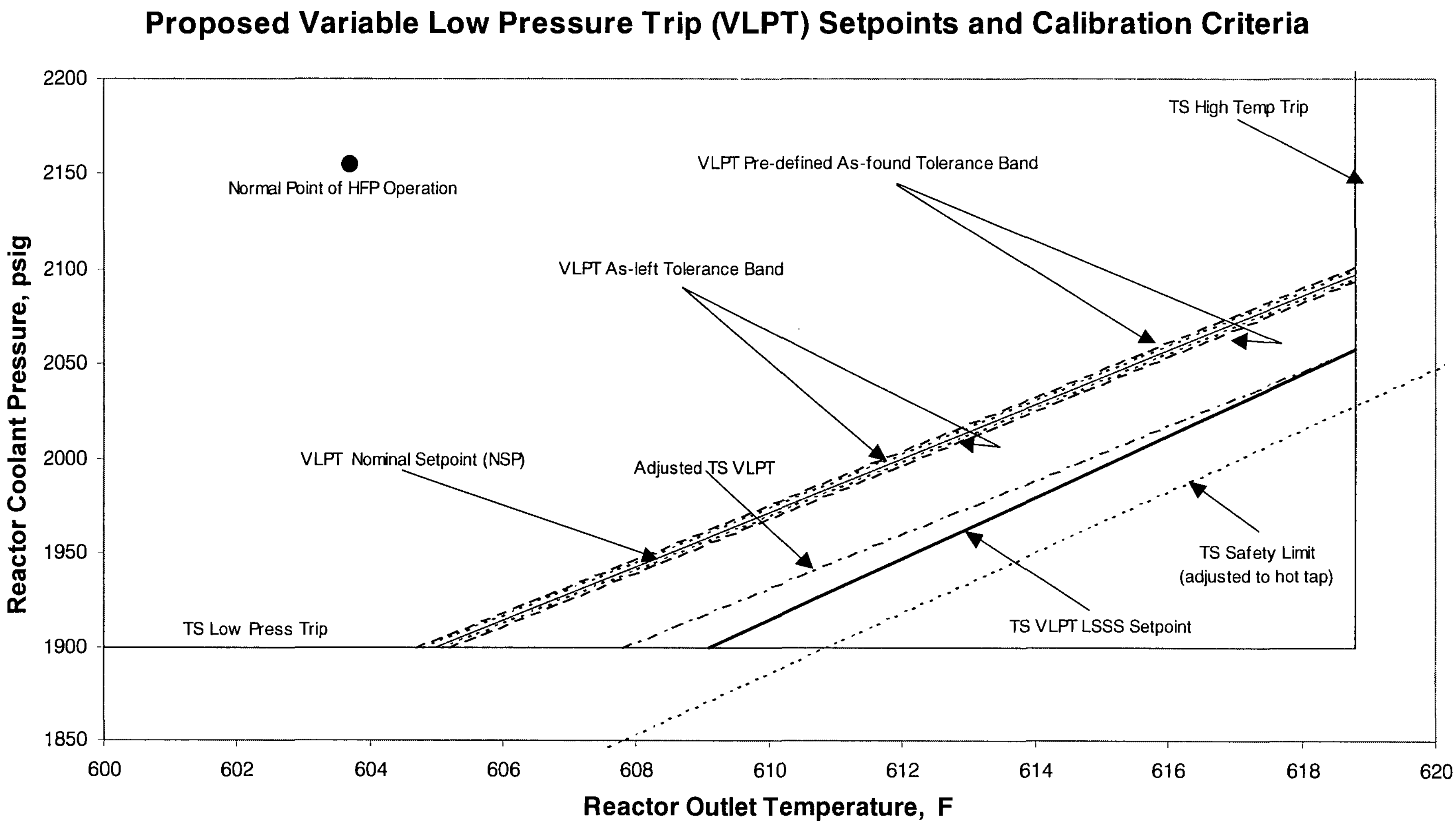
10. **NRC Question**

The Summary of AmerGen Commitments includes a commitment to determine the predefined limits for the VLPT NSP As-Found Tolerance on a programmatic basis. The licensee should clarify this commitment. Aren't the predefined limits for the VLPT NSP As-Found Tolerance Band determined and included in March 22, 2007 submittal? Why would the limits of the VLPT NSP As-Found Tolerance Band need to be recalculated on a programmatic basis? What are the periodicity or events that would trigger a recalculation of the VLPT NSP As-Found Tolerance Band? Would a recalculation of the VLPT NSP As-Found Tolerance Band be the cause for a revision to any of the VLPT related items in question 7 (a) - (d) above?

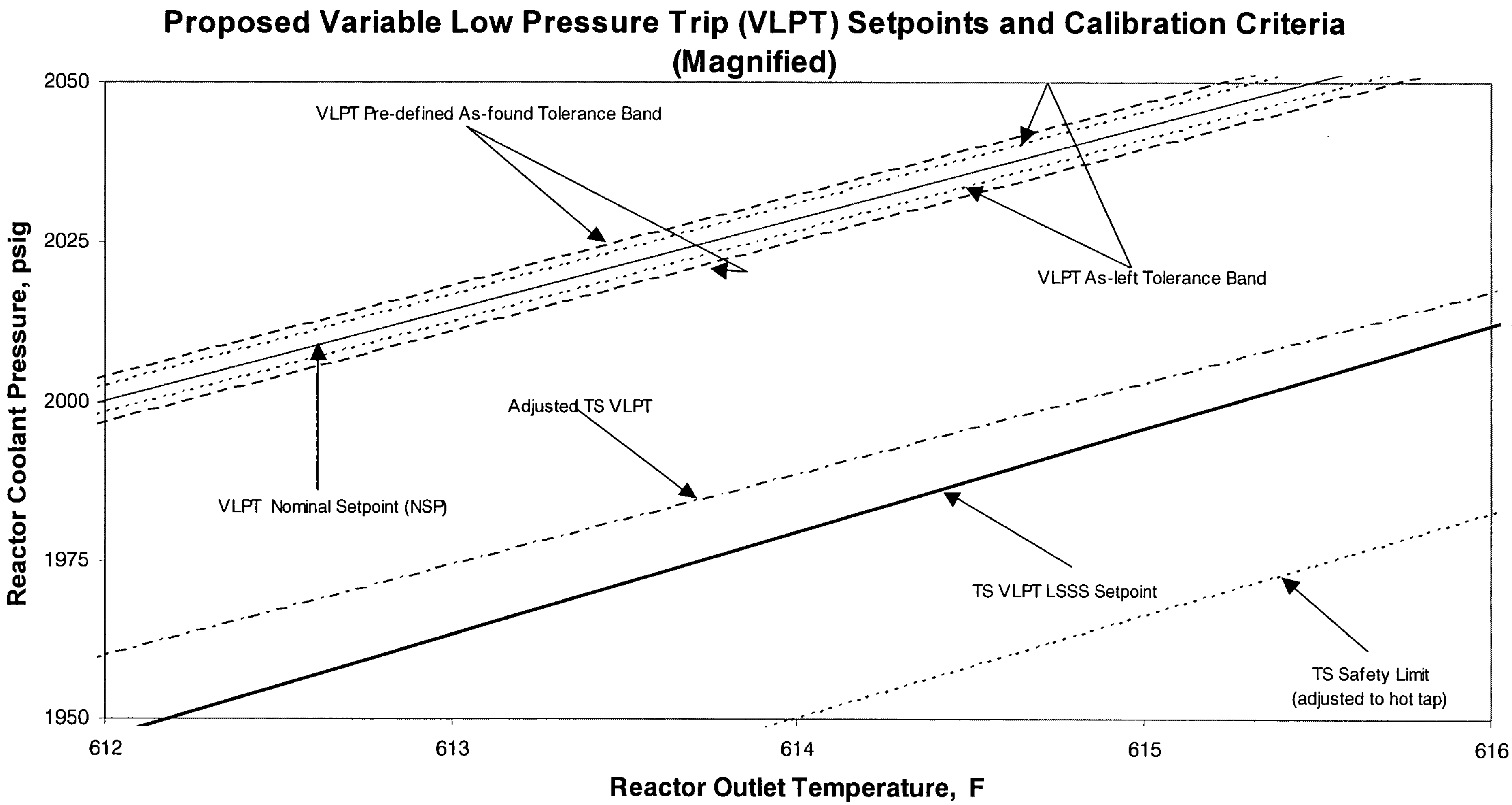
Response

This commitment was intended to identify that the actual predefined limits for the VLPT NSP As-Found Tolerance Band would be determined at a later date. TSCR No. 335, submitted March 22, 2007 provided a sample calculation of the predefined limits for the VLPT NSP As-Found Tolerance Band. These values have subsequently been determined and are provided in response to Question Nos. 5 and 7 above. For the AREVA NP Mark-B HTP fuel design in TMI Unit 1 Cycle 17, the tolerance band will not need to be recalculated. The Tolerance Band would need to be recalculated only in the event of a future Technical Specification setpoint change or due to an instrument string modification. In this event, RAI 7 Items (a) - (d) above would also be recalculated, as necessary, using the methodology provided by ANSI/ISA-S67.04-Part 1-1994, NRC RIS 2006-17, and Regulatory Guide 1.105, Revision 3.

ATTACHMENT 1



ATTACHMENT 2



ENCLOSURE 3

AREVA NP
AFFIDAVIT CERTIFYING REQUEST FOR WITHHOLDING FROM PUBLIC DISCLOSURE

AFFIDAVIT

COMMONWEALTH OF VIRGINIA)
) ss.
CITY OF LYNCHBURG)

1. My name is Mark J. Burzynski. I am Manager, Product Licensing, for AREVA NP Inc. 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 document *Response to Request for Additional Information (RAI) TMI Unit 1 Technical Specification Change Request No. 335 Reactor Coolant System Pressure-Temperature Safety Limit (letter number 5928-07-20164)* and 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.

Mark J. Burzynski

SUBSCRIBED before me this 24th

day of July, 2007.

S McFaden

Sherry L. McFaden
NOTARY PUBLIC, COMMONWEALTH OF VIRGINIA
MY COMMISSION EXPIRES: 10/31/2010
#7079129

ENCLOSURE 4

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 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)
With regard to the VLPT setpoint calculation and the references of Sections 7 and 8 of Part 1 of ISA-S67.04-1994, TMI Unit 1 is in agreement with this Regulatory Position and commits to implement Regulatory Guide 1.105, Rev. 3.	Upon implementation of amendment for the proposed change.	No	Yes