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Docket Number 50-346

10 CFR 2.390
10 CFR 50.90

License Number NPF-3

Serial Number 3166

August 28, 2005

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555-0001

Subject: Davis-Besse Nuclear Power Station
Supplemental Information Regarding License Amendment Application to Support Mark B-HTP Fuel Design for Cycle 15
(License Amendment Request (LAR) 05-0002; TAC No. MC6888)

Ladies and Gentlemen:

This letter responds to two NRC requests for additional information (RAI) regarding LAR 05-0002.

By letter dated May 2, 2005 (Serial Number 3131), the FirstEnergy Nuclear Operating Company (FENOC) submitted an application for amendment of the Operating License, Appendix A, Technical Specifications (TS) for the Davis-Besse Nuclear Power Station (DBNPS). The proposed amendment would revise TS Section 2.1.1, "Safety Limits – Reactor Core," and TS Section 2.2.1, "Limiting Safety System Settings – Reactor Protection System Setpoints" to support use of the Framatome Mark B-HTP Fuel design for Cycle 15, which is scheduled to begin following refueling in March 2006.

On June 9, 2005, and June 15, 2005, FENOC received informal requests from the NRC staff for additional information regarding the license amendment application. Enclosure 1 provides the proprietary response to the June 9, 2005 request. Enclosure 2 provides the response to the June 15, 2005 request. Enclosures 3 and 4 provide the non-proprietary versions of Enclosures 1 and 2, respectively. Enclosure 5 provides specific responses to the concerns and request for additional information presented in the NRC letter to the Nuclear Energy Institute dated March 31, 2005, concerning trip setpoints and allowable values (ADAMS ML050870008). Enclosure 6 provides an affidavit supporting the proprietary status of Enclosures 1 and 2. A list of regulatory commitments made in this letter is included in Enclosure 7.

Apo1

Certain information included in Enclosures 1 and 2 is confidential proprietary information which FENOC requests be withheld from public disclosure pursuant to 10 CFR 2.390. Redacted versions of Enclosures 1 and 2 are provided in Enclosures 3 and 4, respectively. Proprietary information is denoted by [brackets]. Public disclosure of this information is likely to cause harm to the competitive position of AREVA/Framatome ANP. Information identified as proprietary in Enclosures 1 through 4 was originally identified as proprietary in the AREVA/Framatome ANP Engineering Information Record 51-5069986, Revision 0, "RAI Response Information for Davis Besse Cycle 15 TS Change LAR," dated August 12, 2005. An affidavit supporting the request for withholding this information is provided in Enclosure 6, as required by 10 CFR 2.390(b)(iii). Although the affidavit provided in Enclosure 6 was prepared for AREVA/Framatome ANP Engineering Information Record 51-5069986, the 51-5069986 document was prepared exclusively for the purpose of supplying information to be used in this letter, and therefore the text identified as proprietary in this letter is equivalent to corresponding text described in the affidavit.

The supplemental information provided in this letter does not affect the conclusion of the license amendment application that the proposed changes do not involve a significant hazards consideration and do not have an adverse effect on nuclear safety.

As described in Enclosure 5, FENOC will propose an additional TS change to add a footnote to TS Table 4.3-1, "Reactor Protection System Instrumentation Surveillance Requirements," applicable to Functional Unit 7, "RC pressure-temperature," regarding the as-left instrument setting. This change will be consistent with the NRC position described in the March 31, 2005 letter from James A. Lyons, NRC, to Mr. Alex Marion, Nuclear Energy Institute. The proposed change will require review by the Davis-Besse Plant Operations Review Committee and Company Nuclear Review Board and will be submitted under a separate letter.

Should you have any questions or require additional information, please contact Mr. Henry L. Hegrat, Supervisor – Fleet Licensing, at (330) 315-6944.

The statements contained in this submittal, including its associated enclosures, are correct to the best of my knowledge and belief. I am authorized by the FirstEnergy Nuclear Operating Company to make this submittal. I declare under penalty of perjury that the foregoing is true and correct.

Executed on: 8/25/05

By: [signature]
Mark B. Bezilla, Vice President-Nuclear

MSH

Enclosure 1: Proprietary Response to June 9, 2005 Request for Additional Information

Enclosure 2: Proprietary Response to June 15, 2005 Request for Additional Information

Enclosure 3: Non-Proprietary Response to June 9, 2005 Request for Additional Information

Enclosure 4: Non-Proprietary Response to June 15, 2005 Request for Additional Information

Enclosure 5: Response to Concerns and RAI presented in the NRC letter to NEI, dated March 31, 2005, concerning trip setpoints and allowable values (ADAMS ML050870008).

Enclosure 6: Affidavit Supporting Proprietary Status of Enclosures 1 and 2

Enclosure 7: Commitment List

cc: J. L. Caldwell, Regional Administrator, NRC Region III
W. A. Macon, DB-1 NRC/NRR Project Manager
N. Dragani, Executive Director, Ohio Emergency Management Agency, State of Ohio (NRC Liaison); w/o Enclosure 1, Enclosure 2
C. S. Thomas, DB-1 NRC Senior Resident Inspector
Utility Radiological Safety Board; w/o Enclosure 1, Enclosure 2

**RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION
DATED JUNE 9, 2005
REGARDING
LICENSE AMENDMENT REQUEST 05-0002
FOR
DAVIS-BESSE NUCLEAR POWER STATION
UNIT NUMBER 1**

1. *There is no description in the May 2, 2005, submittal regarding the updated core thermal-hydraulic analysis performed for the reactor core safety limit in the TS Figure 2.1-1 to support the use of Mark B-HTP fuel assemblies, except for the statements that*

   (1) the necessary DNB margin is necessary to offset the transition DNB penalty for Cycle 15 and subsequent cycles with co-existence of the Mark-B HTP fuel assemblies and the existing fuel assemblies in the core; and

   (2) the safety limit curve is based on the design hot channel factors with potential fuel densification and fuel rod bowing effects.

   A. Confirm whether the core thermal-hydraulic analysis is performed using the statistical core design methodology. If so, provide a derivation of the SCD DNBR limit with the BHTP correlation to account for the uncertainties of the parameters included in the SCD.

<u>RESPONSE 1.A</u>

The core thermal-hydraulic analyses, performed to determine the DNB performance of Davis-Besse Cycle 15, utilized the Statistical Core Design (SCD) methodology that is discussed and approved in BAW-10187P-A, *Statistical Core Design for B&W-Designed 177FA Plants* (Reference 1). The SCD methodology was incorporated into the approved Safety Criteria and Methodology topical report as Appendix D in BAW-10179P-A, *Safety Criteria and Methodology for Acceptable Cycle Reload Analyses* (Reference 2).

The SCD DNBR limit, commonly referred to as the Statistical Design Limit (SDL), was derived in accordance with the methodology discussed in BAW-10187P-A, *Statistical Core Design for B&W-Designed 177FA Plants*, by first defining variables and their associated uncertainties. This included the distribution characteristic (uniform or normal) of each uncertainty. Table 1.A-1 provides a list of the variable uncertainties that have been incorporated into the SDL determination for Davis-Besse Cycle 15. A Response Surface Model was established that provides a DNB response as a function of the independent variables. Next, a Monte Carlo propagation of uncertainties was performed at various typical and limiting core states to determine the respective core state SDLs from the coefficient of variations. The maximum coefficient of variation for tested core states defined the hot pin SDL. Core-wide DNB protection was established by determining the corresponding SDLs for various core radial peaking distributions. The maximum SDL determined for the core-wide assessment defined the core-wide SDL. The larger of the hot pin SDL and core-wide SDL became the SDL for DNB protection of the core. The SDL determined for Davis Besse Cycle 15, using Table 1.A-1 variables and uncertainties, was found to be [ ] BHTP. For analysis application, the variables in Table 1.A-1 were analyzed without the corresponding uncertainties, using the LYNXT core thermal-hydraulic code (BAW-10156A, *LYNXT – Core Transient Thermal-Hydraulic Program*) to obtain a DNBR prediction.

To provide an accomodation for cycle-specific DNB margin needs, a Thermal Design Limit (TDL) of [ ] was established for Davis-Besse Cycle 15 (see Response 1.D) that is sufficiently greater than the SDL. The TDL is used as the DNBR criterion (limit) for assessing the acceptability of DNB predictions for core protection.

Table 1.A-1
Uncertainty Parameters Used to Derive the
Statistical Design Limit for Davis-Besse Cycle 15

1.			
2.			
3.			
4.			
5.			
6.			
7.			
8.			

B. Provide a description of the magnitude and the derivation of the transition core penalty assigned in the analysis.

RESPONSE 1.B

The DNBR transition core penalty has been determined for Davis-Besse based on the number of Mark-B-HTP fuel assemblies residing in the core. For Cycle 15 there will be 76 Mark-B-HTP fuel assemblies in the core and the transition core DNB penalty will be approximately [ ] DNB points, where 1 DNB point = 0.01. Figure 1.B-1 provides the relationship of the transition core penalty versus the number of Mark-B-HTP fuel assemblies in the core. The transition core penalty will be applied to the DNB margin for predictions made using a full core of Mark-B-HTP fuel.

[ ]

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

The transition core model DNB performance was analyzed for the same range of statepoint conditions, transients, and range of axial power shapes studied for the full core model. The process was repeated for transition cores containing from 1 to 129 Mark-B-HTP fuel assemblies. The largest DNBR difference between the limiting Mark-B-HTP fuel rod in a full core model (of Mark-B-HTP fuel) and a specific transition core model for all of the statepoints, transients, and axial power shapes was defined as the transition core penalty for the specific transition core model. The tabulation of this largest DNBR difference for the transition core containing 1 to 129 Mark-B-HTP fuel assemblies resulted in the line presented in Figure 1.B-1.

The conservatism added by surrounding the limiting Mark-B-HTP fuel assembly with the lower pressure drop fuel design was quantified for the transition core model containing 85 Mark-B-HTP fuel assemblies. Note in Figure 1.B-1 that use of a typical checkerboard reload pattern for the fresh fuel results in an approximately [ ] DNB point smaller transition core penalty.

As stated above, the transition core penalty for Davis-Besse Cycle 15, using 76 Mark-B-HTP fuel assemblies, will be [ ] DNB points. The transition core penalty will be smaller in subsequent cycles as the number of Mark-B-HTP fuel assemblies increases in the core.

C. *Provide a breakdown of DNBR margins included to accommodate the effects of fuel densification and rod bow for the Mark B-HTP fuel design, and discuss how these values are derived.*

<u>RESPONSE 1.C</u>

The impact of fuel densification is discussed in Section 6.2.7 in Reference 2. The thermal-hydraulic core models for the Davis-Besse DNB predictions use a "cold" nominal fuel stack height. The thermal expansion effect on the fuel stack height is greater than the densification effect, therefore, the application of a "cold" nominal fuel stack in the DNB core models yields a conservative heat flux for DNB predictions. Since the accommodation of the fuel densification effect is within the core model basis, there is no need for additional DNBR margin to offset the effect.

Fuel rod bow is discussed in Section 6.2.6 in Reference 2. As noted in Section 6.2.6 of Reference 2, a [

] Since accommodation of the fuel rod bow effect is captured in the development of the SDL, there is no need for additional DNBR margin to offset the effect.

D. *In consideration of fuel densification, rod bow, transition core penalty, and SCD, and with the minimum DNBR limit of 1.132 for the BHTP correlation, what is the safety analysis DNBR limit used for the core safety limit curve and the safety analyses of the design basis transients and accidents?*

RESPONSE 1.D

The safety analysis DNBR limit for Davis-Besse is [ ] BHTP. This limit, more commonly known as the Thermal Design Limit (TDL), is sufficiently high to cover the BHTP correlation design limit (1.132), the impact of the statistical treatment of the state variable uncertainties [ ], and the transition core penalty [ ], yet is small enough to yield acceptable transient analyses results.

[ ]

Figure 1.D-1 shows the DNB margin between the TDL [ ] and the SDL [ ] of approximately [ ] DNB points. The [ ] DNB points will be used to offset the [ ] DNB point transition core penalty, with the remaining [ ] DNB points to be available to offset cycle-specific needs.

2. *The staff's review found that the Reactor Protection System (RPS) "Allowable Value" of Functional Unit 7, "RC pressure-temperature" trip function in TS Table 2.1-1 is essentially the same as the "RC Pressure Temperature Trip" curve in TS Figure 2.1-1.*

RESPONSE 2

This is true. The equation listed in TS Table 2.2-1 defines the line that is plotted in TS Figure 2.1-1. There may be a small difference if the data points on Figure 2.1-1 are used to verify the equation that defines the Reactor Coolant Pressure-Temperature Trip. This is due to the number of significant digits that are listed on the figure. It should be noted that the box that defines the ACCEPTABLE OPERATION window represents the "Allowable Values" for the pressure- and temperature-related reactor trip functions. The changes proposed by the May 2, 2005 license amendment application are only required due to the implementation of the Mark B-HTP fuel design. The Mark B-HTP fuel design incorporates a more robust intermediate spacer grid design that has a higher resistance to flow. This translates to a more restrictive pressure-temperature limit that is used to define this trip.

A. *Clarify whether the various trip functions (i.e., RC High Pressure Trip, RC Low Pressure Trip, RC High Temperature Trip, and RC Pressure-Temperature Trip) in TS Figure 2.1-1 are intended to be the Analytical Limits, actual Trip Setpoint, or the Allowable Values of the Reactor Protection System's for each of these trip functions.*

RESPONSE 2.A

Each of the trip functions identified in TS Table 2.2-1 and TS Figure 2.1-1 represent the "Allowable Values". On TS Figure 2.1-1, the line identified as the Safety Limit represents the "Analytical Limit" for the Reactor Coolant (RC) Pressure-Temperature trip.

B. *Describe the derivation and the values of the "Analytical Limit" used in the safety analyses of the design basis transients for the "RC Pressure-Temperature Trip," and the RPS "Trip Setpoint" for the same trip function.*

RESPONSE 2.B

A description of the Variable Low Reactor Coolant Pressure Trip is provided in Section 7.6 of BAW-10179, Revision 1, *Safety Criteria and Methodology for Acceptable Cycle Reload Analyses* (Reference 2). A general description is provided below. To have a common understanding, however, the following definitions are presented:

Analytical Limit – This is the value that is used in the accident/safety analysis. When these limits are modeled, the consequences of the transient being analyzed will successfully meet the acceptance criteria for the specified event as listed in the Updated Safety Analysis Report.

Allowable Value - The Allowable Value is derived from the Analytical Limit by incorporating instrument string uncertainties. These uncertainties include hardware and process measurement error for the individual components in the instrument string for a given trip function. Random errors are combined statistically whereas the non-random errors are combined by linear addition. This ensures that if the reactor is tripped on, or before, the Analytical Limit is reached, the protective systems will ensure that the plant conditions will not exceed what has been previously analyzed and found acceptable.

Trip Setpoint – The trip setpoint is also referred to as the "Field Setpoint". The Field Setpoint is the actual reactor trip setpoint for the reactor protection system. This setpoint is more restrictive than the "Allowable Value" and accounts for the ability to set a specific value for the setpoint. This includes setpoint "setability" of the setpoint, setpoint drift (as-left/as-found), measurement and test equipment (MTE) uncertainty, and calibration errors.

The Safety Limit shown on TS Figure 2.1-1 corresponds to the "Analytical Limit" for the Reactor Coolant Pressure-Temperature Trip function. This line corresponds to the hot leg pressure/core exit temperature DNB limits for steady-state operation for the limiting RCP combination at the respective maximum allowed core overpower condition. This line defines the pressure-temperature combination that will result in a minimum DNBR equal to, or greater than the Thermal Design Limit (TDL) for the appropriate Critical Heat Flux correlation. The region above and to the left of this line will result in a minimum DNBR greater than the TDL.

The string errors for the individual components (hardware and process measurement errors) are combined and added to the Safety Limit to define the "Allowable Value," as described in the response to Question 2.C in this enclosure. The Trip Setpoint is derived from the Allowable Value by adding uncertainties for MTE, drift, setability, and calibration.

C. Provide a detailed description of the instrumentation uncertainties related to the "RC Pressure-Temperature Trip," and the analyses to derive the RPS "Trip Setpoint" and the "Allowable Value" for this trip function.

RESPONSE 2.C

As described above, a series of core-exit pressure/ core-exit temperature points are calculated for different operating RCP combinations, at the maximum allowed core over-power limit for the pump combination, which results in a minimum DNBR at the TDL for the fuel design being used. The minimum DNBR is calculated using the appropriate NRC-approved CHF correlation. Since the DNBR is a stronger function of temperature, there is a slight "bowing" in the curve with the end-points being more restrictive. The core-exit pressure is adjusted to the hot leg pressure tap by accounting for the dynamic pressure drop and elevation change. This curve represents the Safety Limit on TS Figure 2.1-1.

Since the two end-points are the most restrictive, these points are used to determine the "Allowable Value". By adding the hardware and process measurement errors to these points, the "Allowable Value" can be determined. The component errors are provided by the component vendors. Figure 2.C-1 shows the components included in the instrument string. The random errors for the individual components were statistically combined whereas the non-random errors were combined by linear addition resulting in a total combined error of [ ] psi. The "Trip Setpoint" is determined by adding the combined uncertainty for setpoint drift, MTE and calibration tolerance (10.72 psi) to the "Allowable Value". (Note that the calculation for the field setpoint for this trip function is being updated. This work is expected to be complete by December 30, 2005.)

The calibration tolerance (10.72 psi) consists of three elements, which were added algebraically: Drift (3.6 psi), measurement and test equipment error (1.12 psi), and additional tolerance (6 psi). Drift was calculated by comparison of as-found vs. as-left data in accordance with Instrument Society of America (ISA) Recommended Practice (RP) 67.04 Part II, Committee Draft 10, August, 1992, *Methodologies for the Determination of Setpoints for Nuclear Safety Related Instrumentation.* [The computation methods in this draft are equivalent to those described in the 1994 revision of the RP.] It should be noted that the as-found/as-left data may include the combined effects of reference accuracy, inherent drift, measurement and test equipment, humidity, vibration, normal radiation, normal temperature, and power supply variations during the time period of the surveillance. A value for drift obtained by this methodology is conservative as it will include some or all of these other uncertainties and the allowance that is determined will bound 95% of future observations with a 95% confidence.

Field data from monthly channel functional checks was used. The change in the output over an approximately six-month period (without calibration adjustment or module replacement) was calculated. Each value was then 'normalized' to 180 days. A "95%/95%" confidence/tolerance value was developed using statistical analysis. Note: Measurements during refueling outages are made differently, and the data was excluded for consistency. Drift values were determined using the formula:

$$X_{max} = |\bar{x}| + Ks$$

where X_{max} = the maximum expected value,

$\bar{x}$ = the sample mean,

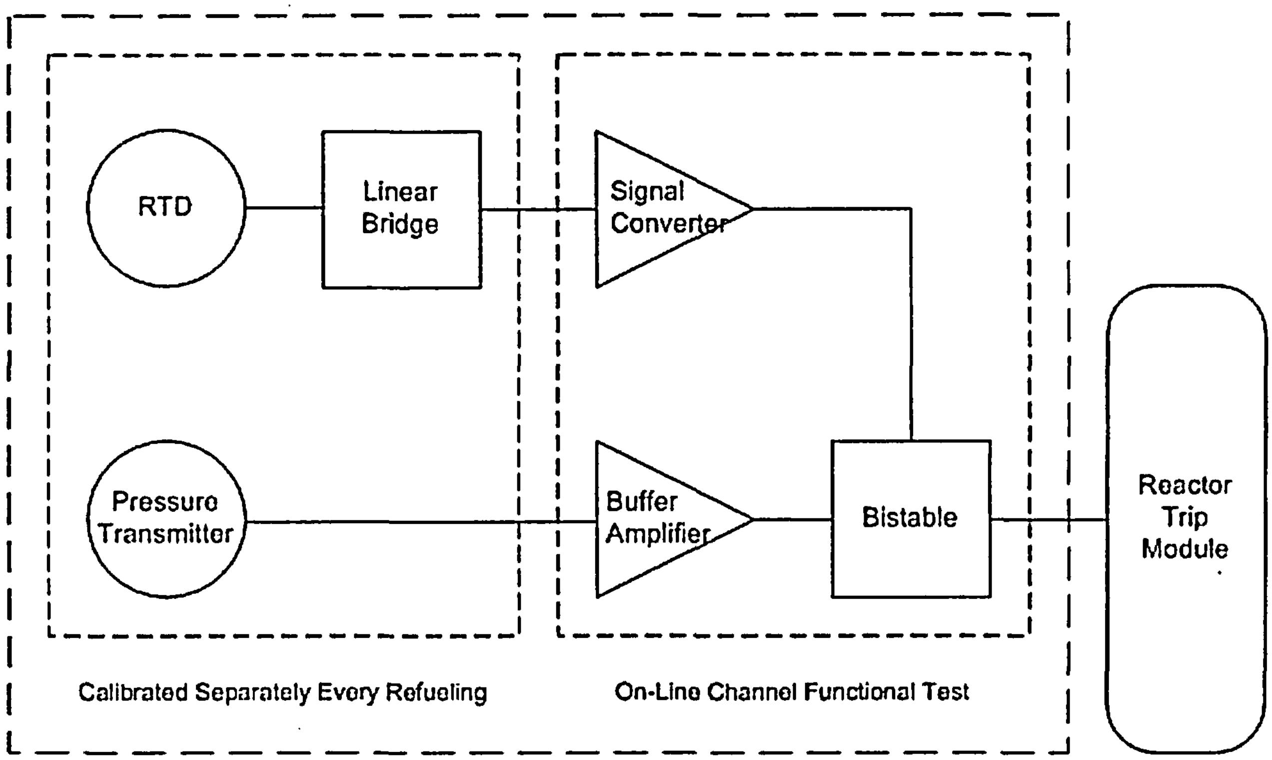
K = a value from *Statistics for Nuclear Engineers and Scientists* (Reference 4), and

s = the standard deviation of the sample.

The calculated drift value was then scaled by 125% to account for the 125% maximum surveillance interval allowed by TS Surveillance Requirement 4.0.2. The data was then verified to fit a normal distribution. Methods used to perform the drift analysis and data distribution verification are described in detail in Davis-Besse calculation C-ICE-58.01-005 (Reference 5), which is available for inspection.

Details of the component error calculation are contained in the AREVA / Framatome ANP proprietary document 51-5057725-00, *DB Low Pressure Reactor Trip Setpoint Evaluation* (Reference 7). This document can also be made available for inspection if required.

Figure 2.C-1
RPS Variable Low Reactor Coolant Pressure Trip String



Variable Low Pressure Trip Calculation Boundary

REFERENCES

1) BAW-10187, Statistical Core Design for B&W-Designed 177FA Plants, B&W Fuel Company, March 1994.

2) BAW-10179, Revision 5, Safety Criteria and Methodology for Acceptable Cycle Reload Analyses, Framatome ANP, December 2002.

3) BAW-10156, Revision1, LYNXT – Core Transient Thermal-Hydraulic Program, B&W Fuel Company, August 1993.

4) Eggs, W.J., *Statistics for Nuclear Engineers and Scientists, Part 1: Basic Statistical Inference,* DOE Research and Development Report No. WAPD-TM-1292, February, 1981.

5) Calculation C-ICE-58.01-005, RPS High Temperature and Pressure-Temperature Field Trip Setpoints, Revision 4, November 25, 2002.

6) Instrument Society of America Recommended Practice 67.04 Part II, Committee Draft 10, August, 1992, *Methodologies for the Determination of Setpoints for Nuclear Safety Related Instrumentation.*

7) AREVA / Framatome ANP document 51-5057725-00, *DB Low Pressure Reactor Trip Setpoint Evaluation*, January 2005.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION
DATED JUNE 15, 2005
REGARDING
LICENSE AMENDMENT REQUEST 05-0002
FOR
DAVIS-BESSE NUCLEAR POWER STATION
UNIT NUMBER 1

1. *(This question pertained to the pagination of Enclosure 1 to License Amendment Request 05-0002 (Davis-Besse Serial No. 3131). After discussion with the NRC staff, it was agreed that the pagination is correct, and that there are no missing pages. Therefore, this question was withdrawn.)*

2. *Please discuss how instrument uncertainty (including uncertainty in the estimation of core thermal power) is accounted for in the proposed limits. Specifically, these address the following:*

   a. *Regarding Proposed Figure 2.1-1, confirm that the "Acceptable Operation" region refers to measured values of pressure and temperature and that the "Safety Limit" line refers to actual values. If this is not the case, explain how the figure is to be applied in plant operation and explain how measurement uncertainty is accounted for.*

   <u>RESPONSE 2.a</u>

   Regarding proposed Figure 2.1-1, the limits of the "Acceptable Operation " region reflect the Allowable Values as depicted on Table 2.2-1, Reactor Protection System Instrumentation Trip Setpoints. The development of the Allowable Values is consistent with ISA Standard 67.04.01-2000, *Setpoints for Nuclear Safety-Related Instrumentation* and ISA Recommended Practice 67.04.02-2000, *Methodologies for the Determination of Setpoints for Nuclear Safety-Related Instrumentation.* The Allowable Values and trip setpoints are developed consistent with Method 1 of the Recommended Practice. The instrument uncertainties that are not tested during a normal periodic surveillance establish the region between the Safety Limit and the Allowable Value. The instrument uncertainties that are tested during a normal periodic surveillance establish the region between the Allowable Value and the field trip setpoint. The trip setpoint is not depicted on Figure 2.1-1 and therefore limits operation to values more conservative than that reflected on the Figure.

b. *Address the concerns and RAI presented in the NRC letter to NEI, dated March 31, 2005, concerning trip setpoints and allowable values (ADAMS ML050870008).*

RESPONSE 2.b

Due to the number of items addressed in the March 31, 2005 letter, this response is contained in a separate enclosure. See Enclosure 6 for this response.

c. *Demonstrate that, for operation with measured values at the limits of the "Acceptable Operation" region of proposed Figure 2.1-1, 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 2.c

The Acceptable Operating region is defined by the pressure and temperature limits provided in Technical Specification Figure 2.1-1. The normal operating point for the Davis-Besse reactor coolant system (RCS) corresponds to a hot leg temperature of 606°F at a pressure of 2155 psig. The variation in the nominal RCS temperature and pressure is typically less than ±2°F and ±25 psi. Use of the Mark-BHTP fuel will have an almost negligible affect on the overall nominal plant conditions.

The Safety Limit Line specifically represents the temperature and pressure combinations that will result in a minimum departure from nucleate boiling ratio that is equal to, or slightly greater than, the thermal design limit for the fuel type being analyzed. This limit also accounts for 4- and 3-pump operation at the respective maximum allowed core power level as described in the response to Question 2.D in this enclosure. The uncertainties associated with the instrumentation string and process measurement described in the response to Question 2.D in this enclosure, are applied to the safety limit line to establish the Allowable Value for the variable low pressure trip function.

The actual reactor trip setpoint is more restrictive than the Allowable Value in that the trip setpoint also takes into consideration the ability to set a specific setpoint. (See Enclosure 1 or Enclosure 3, Question 2.B.) Assuming the plant is operating at the trip setpoint and the uncertainty in setting the setpoint results in an actual measured temperature that is at the limits of Acceptable Operation, it can be shown that the trip setpoint calculated by the variable low pressure trip setpoint will occur prior to reaching the Safety Limit Line as shown below:

The equation that defines the variable low pressure trip setpoint is:

$$\text{Low Pressure Trip Setpoint} = 16.25 * (T_{hot}) - 7899 \text{ psi}$$

For example, if we use a hot leg temperature of 612°F, the resulting trip setpoint would be 2046 psig. The total uncertainty for this trip function is [ ] psi. By subtracting the uncertainty from the calculated trip setpoint, a direct comparison can be made to the Safety Limit Line as shown on the proposed Technical Specification Figure 2.2-1, i.e., [ ] psig vice a limit of 1979.8 psig. Similarly, if we assume that the hot leg temperature is 618°F, the resulting trip setpoint after subtracting the uncertainty will be [ ] psig. By interpolation of the data on the Safety Limit Line, the corresponding pressure at 618°F, is 2098.61 psig. As can be seen in these examples, the margin between the error-adjusted trip setpoint and the Safety Limit Line provides confidence that there is adequate conservatism in the proposed setpoint.

Furthermore, a review of the safety analyses that are included in the Davis-Besse FSAR was performed to ascertain which events credit the variable low pressure trip function for accident mitigation. This review is summarized in Table 2.C. The review concluded that the Reactor Protection System Variable Low Pressure Trip function is not credited for transient mitigation in any USAR analysis. The only specific credit for this trip function is for protection against minimum DNB relative to steady state and not transient operation.

In addition, ISA standard 67.04 Method 1 was used to develop the Allowable Value equation that defines the Acceptable Operation region in proposed Figure 2.1-1. As is explained in more detail in Enclosure 5, Method 1 ensures that the proposed Allowable Value equation meets the "95% / 95%" criteria endorsed by RG 1.105. Therefore, all operating points within the Acceptable Operation region are conservative with respect to the indicated Safety Limit line.

Table 2.C - Summary of USAR Accident Analyses

USAR Accident Analyses	Event Classification	RPS Trip Function
Startup Accident	Reactivity	High Pressure/High Flux
Rod Withdrawal at Power	Reactivity	High Pressure/High Flux
Dropped Rod	Reactivity	Low RCS Pressure
Moderator Dilution Accident	Reactivity	High Pressure
Loss of Flow	Overheating/DNB	Power-to-Pump/RCP Monitor, Flux to Flow
Startup of an Idle Loop	Reactivity	High Flux
Turbine Trip	Overheating	High Pressure
Loss of Feedwater	Overheating	High Pressure
Station Blackout / Loss of AC Power	Overheating	NA – trip due to loss of power
Feedwater Malfunction	Overcooling	High Flux
Excessive Load Increase	Overcooling	Bounded by small SLB
Opening of a Pressurizer Safety Valve	Loss of Coolant	Bounded by SBLOCA
Uncompensated Reactivity Changes	Reactivity	No automatic trip required
Failure of Regulating Instrumentation	Reactivity	No automatic trip required
Small Steam Line Break	Overcooling	SFRCS via ARTs/Low RCS Pressure/High Flux
Inadvertent Loading of a Fuel Assembly in an Improper Position	Reactivity	No automatic trip required
Waste Gas Decay Tank Rupture	Offsite Dose	NA – plant cooldown
Steam Generator Tube Rupture	Offsite Dose/Loss of Coolant	Low RCS Pressure
Control Rod Assembly Ejection Accident	Reactivity	High Flux
Steam Line Break - mass and energy release	Overcooling	SFRCS via ARTs/Low RCS Pressure/High Flux
Steam Line Break - core response	Overcooling	SFRCS via ARTs/Low RCS Pressure/High Flux
Letdown Line Break	Offsite Dose/Loss of Coolant	Low RCS Pressure
Large Break LOCA – core response	Loss of Coolant	Low RCS Pressure
Large Break LOCA – mass and energy release	Loss of Coolant	Low RCS Pressure
Small Break LOCA	Loss of Coolant	Low RCS Pressure
Fuel Handling Accident	Offsite Dose	NA – refueling
Toxic Material Release	Control Room Habitability	NA – historical non-RCS event
Maximum Hypothetical Accident	Offsite Dose	NA non-mechanistic LBLOCA
Fuel Cask Drop	Offsite Dose	NA - refueling
Feedwater Line Break	Overheating	High Pressure
Anticipated Transients Without Scram - ATWS	Overheating	DSS (alternate high pressure)
Appendix R-overcooling	Overcooling	Operator initiated
Appendix R-loss of feedwater	Overheating	Operator initiated

NA - Not Applicable

d. *Specify the uncertainties in the pressure, temperature, and reactor flux measurements associated with the proposed TS change, and show how those uncertainties are derived. Show that those uncertainties are consistent with the operational and analytical limits addressed above.*

RESPONSE 2.d

See Response 2.a above.

The Reactor Protection System variable low pressure trip setpoint defines a minimum allowed Reactor Coolant System pressure based on measurement of the hot leg temperature. The trip function is represented by a straight line in pressure-temperature space that is based on steady-state operation at the maximum allowed power level for the number of reactor coolant pumps that are operating. The trip setpoint is based on Reactor Coolant System pressure and Reactor Coolant System hot leg temperature. Core power is not a direct input to the trip. The core power and any uncertainty related to core power is factored into generation of the pressure/temperature limits.

The hot leg temperature is measured by a Resistive Temperature Detector (RTD). The RTD signal is processed through a linear bridge to a signal converter. The signal converter converts the temperature measurement to an allowable Reactor Coolant System (RCS) pressure based on the equation defined as the Allowable Value in the technical specification. A bistable compares this allowable pressure to the measured RCS pressure to determine if a trip condition exists. The uncertainty of the RTD, the linear bridge, signal converter, pressure transmitter, buffer amplifier, and bistable are combined into a single error term by statistically (SRSS) combining the random terms and adding the correlated errors associated with the process measurement. The error terms for the pressure and temperature measurement are taken from the Bailey and Rosemount manufacturer's data for the individual components. The total uncertainty used to determine the Allowable Value is [ ] psi.

The variable low pressure trip setpoint provides steady-state protection for the pressure and temperature limits defined by operation at the maximum overpower condition for the number of Reactor Coolant Pumps (RCPs) in operation: 112% Rated Thermal Power (RTP) for four RCPs and 91% RTP for three RCPs. The total power measurement error accounts for RCS flow measurement for either four-pump or three-pump operation and the core power measurement error as described in Section 7.4.2.3 of topical report BAW-10179P-A, *Safety Criteria and Methodology for Acceptable Cycle Reload Analyses*.

e. *Demonstrate that the Allowable Values in proposed TS Table 2.1-1, and the limiting values in Note 1 of that table, provide adequate assurance that the actuation values assumed in the plant safety analyses will not be violated. This should include consideration of the proposed new fuel configuration and consideration of instrument setpoint/measurement uncertainty. For values for which no change is proposed, show that the analytical limits remain unchanged or show how the existing values adequately protect the new analytical limits.*

RESPONSE 2.e

The only effect that the proposed fuel design has on the RCS or the reactor trip setpoints listed in Table 2.2-1 relates to the measurement of Departure from Nucleate Boiling (DNB). The trip setpoints that are defined by the calculation of DNB are the high flux and the reactor coolant pressure-temperature trip setpoint. (Note that the flux - Δflux/flow setpoint is also affected by DNB; however, this setpoint is contained in the Core Operating Limits Report and is verified for each fuel cycle.) The calculations that support the reactor coolant pressure-temperature trip setpoint are based on operating at the maximum overpower condition for the number of RCPs that are in operation as described in Response 2.d. Since the pressure-temperature limits are defined by the maximum overpower level for the plant, the high flux trip setpoint is also validated because the fuel design does not impact the power measurement uncertainty. The high flux trip setpoint specified in Table 2.2-1 is simply the maximum overpower level for four-pump operation less the power measurement uncertainties described in Section 7.4.2.3 of BAW-10179P-A. The high flux trip setpoint with three pumps includes an additional uncertainty for partial flow.

The values presented in Note 1 to TS Table 2.2-1 are not affected by the changes in the fuel design. The values identified in Note 1 are only to represent the bypass trip functions that facilitate a normal plant shutdown and/or heatup.

3. *Please demonstrate that the increased signal range applied to the computation unit that produces the variable low-pressure setpoint does not compromise the operation of that unit or of any other unit. In addition, please discuss how the uncertainties associated with the larger signal spans are properly accounted for in the uncertainty analyses.*

RESPONSE 3

The computation unit is a signal converter unit in the Reactor Protection System. The revised values have been verified to be within the normal capabilities of the unit. This verification was performed by evaluating the new Allowable Values, adding additional conservatism to reflect a field trip value (the field trip setpoint calculation revisions have not yet been completed, so actual values were not used), and verifying it was within the normal operating band of the instrument. The largest contributor to the string error (by an order of magnitude) is the pressure transmitter itself. This error was determined by reference to the device range, and is therefore unaffected by a change to the setting span. The other error calculations are similarly unaffected by the proposed change. There is no impact on other computation units, as can be verified by reference to Figure 2.C-1 in Enclosure 1. It is anticipated that the field trip setpoint calculation revisions will be completed by December 30, 2005.

4. *The 5th paragraph of Section 3 addresses only accidents resulting in pressure reduction. Please discuss how reductions in flow or RCS heat removal and increase in fuel temperature are considered.*

RESPONSE 4

The discussion in the 5th paragraph of Section 3 acknowledges the pressure reduction scenarios that are protected by the low pressure and the pressure-temperature trip functions. However, the protection of the core, particularly the DNB protection, is established using more than the trip functions in TS Figure 2.1-1. Sections 6, 7, and 8 of Reference 1 of Enclosure 1 provide a general discussion of the safety criteria and methodology used for the Davis-Besse core relating to the TS 2.0 and overall core protection.

All eight of the Reactor Protection System (RPS) trip functions work together to provide steady-state and transient protection for the Reactor Coolant System (RCS) and more specifically the core. These trip functions and the events/transients for which they provide protection are provided in Table 4.

The Mark-B-HTP fuel design is a more robust design, but provides a slight increase in the hydraulic resistance and as a result there is a different CHF correlation that is applied to the fuel. As a result, there must be an assurance that fuel will not exceed the prescribed acceptance criteria. This assurance must not only address decreasing pressure conditions, but steady-state operation, the performance during loss of flow, events that evolve into an increase in the RCS/fuel temperature, as well as pressure and power increasing transients. An evaluation of all USAR Chapter 6 and Chapter 15 events was performed. It was concluded that, for most events, the fuel design has a negligible affect on the overall system response and only those events that specifically address the fuel response need to be reevaluated, i.e., the DNB-related transients such as LOCA, loss of flow, and steady-state operation. Consequently, the high- and low-RCS pressure setpoint, the high temperature setpoint, the high containment pressure and the high flux setpoints would not be affected by this change in fuel design.

Since new LOCA analyses are being performed for the Mark B-HTP fuel, only the loss of flow and steady-state conditions were assessed. The loss of flow transients (four RCP coast down, single RCP coast down, and locked rotor) are reanalyzed for each new reload using the appropriate CHF correlation. As noted above, the hydraulic resistance is higher for the Mark B-HTP fuel design; however, in comparison to the total RCS pressure drop this change is small. For the loss of flow events, the RCS flow is dominated by the inertia of the reactor coolant pumps such that the power-to-pump flow-related trip function is not affected as long as the minimum DNBR is greater than the thermal design limit (TDL). As part of the fuel reload analyses, it was confirmed that the TDL was not exceeded for the Mark-B-HTP fuel and therefore no change is required to this RPS setpoint.

The Reactor Coolant Pressure-Temperature Trip defines the limit for the steady-state conditions at which Davis-Besse can operate at the respective maximum overpower condition for three or four RCPs operating with acceptable DNB performance. The safety limit that is presented in TS Figure 2.1-1 represents the combinations of reactor outlet temperatures and reactor system pressure that yield a minimum DNB equal to the DNB design criterion when the core is operating at the maximum overpower condition for the more limiting of three or four RCP operation, as described in Reference 2 of Enclosure 1. Typically, the limiting core safety limit is based on four RCP operation. The core safety limit represents the most severe pressure/temperature conditions the core can operate in a steady-state condition without violating the DNB criterion for the plant. For the Mark B-HTP fuel design, the existing technical specification trip setpoint must be revised as discussed in the LAR.

The flux/Δflux/flow trip setpoint is included in the Core Operating Limits Report (COLR) and is subject to change for each new reload. The flow portion of this trip function is not affected as described above, however, the allowed imbalance (flux versus Δflux) may be affected because this trip function is also based on operation at the maximum overpower condition for the number of RCPs that are operating. The required setpoint for this trip function is not defined until all of the reload analyses for the cycle are completed. At this time, the COLR may need to be revised. Since this trip setpoint is in the COLR, it was not discussed in the LAR.

In summary, the trip functions that are required to provide a level of protection based on the overall system response to a postulated event are not affected by the implementation of the Mark B-HTP fuel design and only the trip functions that provide steady-state protection (Reactor Coolant Pressure-Temperature and the Flux/ΔFlux/Flow Trip functions) at the maximum overpower conditions are affected. The change required for the RC pressure-temperature trip is provided in the LAR. The Flux/ΔFlux/Flow Trip is defined in the COLR and will be verified once all cycle-specific reload calculations are performed.

Table 4 - Summary of Reactor Protection Trip Functions

Trip Function	Events Credited	System Response
High Flux	Rod withdrawal from zero power (high reactivity insertion rates) Rod withdrawal at power (high reactivity insertion rates) Steam line break (small area) Rod ejection (high rod worth) Startup of an idle loop Feedwater malfunction	Positive reactivity insertion Positive reactivity insertion Positive reactivity insertion Positive reactivity insertion Positive reactivity insertion Positive reactivity insertion
High RCS Pressure	Rod withdrawal from zero power (low reactivity insertion rates) Rod withdrawal at power (low reactivity insertion rates) Rod ejection (low rod worth) Boron dilution at power Turbine trip Loss of main feedwater Feedwater line break	Positive reactivity insertion Positive reactivity insertion Positive reactivity insertion Positive reactivity insertion Loss of heat transfer Loss of heat transfer Loss of heat transfer
Low RCS Pressure	Loss of coolant accident Letdown line rupture (offsite dose consequence) SG tube rupture Dropped rod assembly Steam line break (large area)	Loss of RCS inventory Offsite dose consequence Offsite dose consequence Core peaking Positive reactivity insertion
RC pressure-Temperature (variable low RCS pressure)	This trip function is not credited in the USAR Chapter 6 or Chapter 15 analyses, but provides a limit for steady-state DNB protection.	DNB Centerline fuel temperature
High RCS Temperature	This trip function is not credited in the USAR Chapter 6 or Chapter 15 analyses, but provides a limit for steady-state DNB protection.	DNB Centerline fuel temperature
Flux/ΔFlux/Flow	Loss of a single RCP Locked RCP rotor/sheared shaft	Partial loss of RCS flow Rapid loss of RCS flow
High Flux vs. No. RCPs (power-to-pump monitors)	Loss of all RCPs (4 pump coast down) Loss of 2 RCPs in one loop	Loss of all RCS forced flow Partial loss of RCS flow
High Containment Pressure	This trip function is not credited in the USAR Chapter 6 or Chapter 15 analyses.	Loss of RCS inventory Loss of SG inventory

RESPONSE TO CONCERNS AND RAI PRESENTED IN THE NRC LETTER TO NEI, DATED MARCH 31, 2005, CONCERNING TRIP SETPOINTS AND ALLOWABLE VALUES (ADAMS ML050870008)

There are three items described on page 3 of the letter dated March 31, 2005 from Mr. Lyons to Alex Marion of the Nuclear Energy Institute. The following is taken from that letter:

During the public meeting held on March 11, 2005, at the NRC Headquarters Office, you requested that the NRC withdraw the RAI question and continue its review of existing licensing actions pending finalization of the TSTF technical specification change to address this issue. You reiterated this request in your March 18, 2005 letter. As noted previously, the NRC staff is continuing the review of existing in-house LARs. To move forward with LARs that are currently under NRC staff review, an interim approach has been developed that addresses the NRC staff's concerns until the TSTF technical specification change is reviewed and approved. As part of these licensee specific reviews, the revised RAI question (see Enclosure) will remain outstanding, with the understanding that, in addition to a brief discussion of the licensee's methodology for establishing LSSS, the licensee's response to the question needs to contain the following in order for the staff to complete its review:

1. *An explicit regulatory commitment to adopt the final TSTF technical specification change to come into conformance with the existing understanding of the requirements of 10 CFR 50.36.*

2. *An explicit regulatory commitment to assess the operability of tested instrumentation based on the previous as-left instrument setting and accounting for the uncertainties associated with the test or calibration.*

3. *A revision to the technical specifications for the LSSS being changed by the LAR to incorporate a footnote that states:*

   The as-left instrument setting shall be returned to a setting within the tolerance band of the trip setpoint established to protect the safety limit.

RESPONSE 1-3

Davis-Besse uses Method 1 from the ISA Recommended Practice 67.04.02 for the Reactor Coolant Pressure-Temperature Functional Unit trip. Using this method, the Allowable Value is calculated from the Analytical Limit, by subtracting instrument uncertainties that are not tested during periodic surveillances. The trip setpoint is then calculated from the Allowable Value by subtracting instrument uncertainties that are tested during periodic surveillances.

As part of the uncertainty calculation, a calibration tolerance is established. The periodic surveillance tests include acceptance criteria for the trip setpoint, including the calibration tolerance. If these acceptance criteria are not met, the instrument is re-calibrated within the calibration tolerance value before returning it to service.

For item 1, FENOC commits to evaluate the final TSTF technical specification change recommendations after NRC approval of the associated TSTF traveler. However, FENOC can not commit to submital of a license amendment request proposing adoption of these recommendations before having a chance to evaluate them.

For item 2, FENOC does not commit to assessing the operability of the tested instrumentation based on the previous as-left instrument setting and accounting for the uncertainties associated with the test or calibration. As previously described, the instruments are evaluated for operability based on the as-found setting being consistent with the calculated trip setpoint and the calibration tolerance. This provides assurance that the instrument is performing within the criteria established in the calculations. Evaluation of the difference between the previous as-left setting and the current as-found setting would create an additional administrative burden while providing no additional assurance that the instrument is performing within the calculation limits. As industry evaluation of this issue concludes with the anticipated issuance of the TSTF, additional consideration will be given to this methodology under the evaluation committed to above.

For item 3, FENOC will submit a separate letter to propose an additional TS change for LAR 05-0002 to add a footnote to TS Table 4.3-1, "Reactor Protection System Instrumentation Surveillance Requirements," applicable to Functional Unit 7, "RC pressure-temperature." The new footnote will read as follows, "The as-left instrument setting shall be returned to a setting within the tolerance band of the trip setpoint established to protect the safety limit." This proposed footnote is consistent with the existing practice of resetting the trip setpoint within the established calibration tolerance prior to returning this equipment to service.

In addition to the items discussed above, the Enclosure to the March 31, 2005 letter requests additional information:

<u>REVISED METHOD 3 REQUEST FOR ADDITIONAL INFORMATION</u>

The [insert plant name] technical specifications define Limiting Safety System Settings (LSSS) as an allowable value (AV). During reviews of proposed license amendments that contain changes to LSSS setpoints, the NRC staff identified concerns regarding the method used by some licensees to determine the allowable value (AV) identified in the technical specifications (TS). AVs are identified in the TS as LSSS to provide acceptance criteria for determination of instrument channel operability during periodic surveillance testing. The NRC staff's concern relates to one of the three methods for determining the AV as described in the Instrument Society of America (ISA) recommended practice ISA-RP67.04-1994, Part II, "Methodology for Determination of Setpoints for Nuclear Safety-Related Instrumentation."

The NRC staff has determined that to ensure a plant will operate in accordance with the assumptions upon which the plant safety analyses have been based, additional information is required regardless of the methodology used to establish LSSS values in technical specifications. Details about the NRC staff's concerns are available on the NRC's public website under ADAMS Accession Numbers ML041690604, ML041810346, and ML050670025.

In Order for the NRC staff to assess the acceptability of your license amendment request related to this issue, the NRC staff requests the following additional information:

1. *Discuss the setpoint methodology used at [insert plant name] to establish AVs associated with LSSS setpoints.*

   <u>RESPONSE 1</u>

   Please refer to page 7 of Enclosure 1 of the May 2, 2005 license amendment application. ISA Recommended Practice 67.04.02, Method 1 was used to develop the Allowable Value and trip setpoints for the Reactor Coolant Pressure-Temperature Functional Unit. Using this method, the Allowable Value was calculated from the Analytical Limit, by subtracting instrument uncertainties that are not tested during periodic surveillances. The trip setpoint is then calculated from the Allowable Value by subtracting instrument uncertainties that are tested during periodic surveillances.

2. *Regardless of the methodology used, the NRC staff has the following questions regarding the use of the methodology at [insert plant name]:*

a. *Discuss how the methodology and controls you have in place ensure the analytical limit (AL) associated with an LSSS will not be exceeded (the AL is a surrogate that ensures the safety limits will not be exceeded). Include in your discussion information on the controls you employ to ensure the trip setpoint established after completing periodic surveillances satisfies your methodology. If the controls are located in a document other than the TS, discuss how those controls satisfy the requirements of 10 CFR 50.36.*

RESPONSE 2.a

As stated above, the Allowable Value for the Reactor Coolant Pressure-Temperature Functional Unit was calculated using Method 1 of ISA Recommended Practice 67.04.02. Due to the availability of references supporting the use of Method 2, all three methods are considered in the discussion that follows, with Method 1 being shown to be acceptable in part by comparison to Method 2. Using either Method 1 or Method 2 of ISA RP 67.04.02 would result in the same Allowable Value. Because this Allowable Value is calculated from the Analytical Limit, it is conservative with respect to the Allowable Value calculated using Method 3. Method 2 calculates the Trip Setpoint by subtracting the channel uncertainty (plus margin) from the Analytical Limit. Method 1 calculates the Trip Setpoint by subtracting drift, calibration uncertainty, and uncertainties during normal operation from the Allowable value. Uncertainties that are random, normally distributed, and independent may be combined using the Square-Root-Sum-of-the-Squares (SRSS) method. Method 1 and Method 2 both account for the same uncertainties, but Method 2 may apply SRSS in one step, so that the resulting setpoint is likely to be slightly more conservative using Method 1, depending on the extent to which SRSS is used.

Regulatory Guide (RG) 1.105, Revision 3, December 1999, states that conformance with Part 1 of ISA-S67.01-1994, "Setpoints for Nuclear Safety-Related Instrumentation," with the listed clarifications and exceptions, provides a method acceptable to the NRC staff for satisfying the NRC's regulations for ensuring that setpoints for safety-related instrumentation are established and maintained within the technical specification limits. Regulatory Position #1 of Regulatory Guide 1.105, Revision 3 provides the following position with regard to the degree of certainty with which a licensee must ensure the analytical limit (AL) associated with an LSSS will not be exceeded:

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.

Therefore, RG 1.105 accepts "95/95" as satisfying the requirements of 10 CFR 50.36. This conclusion is supported by the NRC document, "Setpoint Allowable Values for Instrument Channels in Safety-Related Service," ADAMS accession number ML041810346, June 23, 2004, at page 27. Review of ISA Recommended Practice ISA-RP67.04.02-2000, "Methodologies for the Determination of Setpoints for Nuclear Safety-Related Instrumentation," has led FENOC to conclude that both Method 1 and Method 2 meet the 95/95 criterion and therefore the calculation described in FENOC's May 2, 2005 application (Serial 3131, LAR 05-0002) meets the requirements of 10 CFR 50.36.

With respect to the trip setpoint, this was discussed in FENOC's response (above) to items 1, 2, and 3 on page 3 of the March 31, 2005 letter from Mr. Lyons of the NRC to Mr. Marion of the NEI. As part of the uncertainty calculation, a calibration tolerance is established. The periodic surveillance tests include acceptance criteria for the trip setpoint being within the established tolerance. If these acceptance criteria are not met, the surveillance test has not satisfied the Technical Specification periodic surveillance requirement. Therefore, the instrument is recalibrated within the calibration tolerance value prior to returning the equipment to service.

b. *Discuss how the TS surveillances ensure the operability of the instrument channel. This should include a discussion on how the surveillance test results relate to the technical specification AV and describe how these are used to determine the operability of the instrument channel. If the requirements for determining operability of the LSSS instrument being tested are in a document other than the TS (e.g., plant test procedure), discuss how this meets the requirements of 10 CFR 50.36.*

RESPONSE 2.b

b. As discussed above, during the TS surveillances the trip setpoints are verified to be within the calibration tolerance.

AFFIDAVIT

COMMONWEALTH OF VIRGINIA)
) ss.
CITY OF LYNCHBURG)

1. My name is Gayle F. Elliott. I am Manager, Product Licensing in Regulatory Affairs, for Framatome ANP ("FANP"), and as such I am authorized to execute this Affidavit.

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

3. I am familiar with Engineering Information Record 51-5069986, Revision 1, "RAI Response Information for Davis Besse Cycle 15 TS Change LAR," dated August 25, 2005 and referred to herein as "Document." Information contained in this Document has been classified by FANP as proprietary in accordance with the policies established by FANP 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 FANP 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.

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

(a) The information reveals details of FANP'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 FANP.

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

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

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

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

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

SUBSCRIBED before me this 25th day of August, 2005.

Wanda L. Wade

Wanda L. Wade
NOTARY PUBLIC, COMMONWEALTH OF VIRGINIA
MY COMMISSION EXPIRES: August 31, 2009

WANDA L. WADE
NOTARY PUBLIC
COMMONWEALTH OF VIRGINIA
MY COMMISSION EXPIRES AUGUST 31, 2009

COMMITMENT LIST

The following list identifies those actions committed to by the Davis-Besse Nuclear Power Station, Unit Number 1, (DBNPS) in this document. Any other actions discussed in the submittal represent intended or planned actions by the DBNPS. They are described only for information and are not regulatory commitments. Please notify Henry L. Hegrat, Supervisor – Licensing (330-315-6944) of any questions regarding this document or associated regulatory commitments.

COMMITMENTS	DUE DATE
A letter will be submitted to revise LAR 05-0002 to change Technical Specifications for the Variable Reactor Coolant Pressure-Temperature Trip to add a footnote that states "The as-left instrument setting shall be returned to a setting within the tolerance band of the trip setpoint established to protect the safety limit."	Letter will be submitted as soon as reasonably achievable, with a target of 15 days following submittal of this letter.
FENOC commits to evaluate the final TSTF technical specification change recommendations after NRC approval of the associated TSTF traveler. As industry evaluation of this issue concludes with the anticipated issuance of the TSTF, additional consideration will be given to assessment of tested instrumentation based on the previous as-left setting, as required by the evaluation committed to above.	Evaluation will be completed within 120 days of receipt of the NRC-approved TSTF traveler.