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Docket Number 50-346 10CFR50.90

License Number NPF-3

Serial Number 3131

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United States Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555-0001

Subject: License Amendment Application to Support Mark B-HTP Fuel design for Cycle 15 (License Amendment Request No. 05-0002)

Ladies and Gentlemen:

Pursuant to 10 CFR 50.90, an operating license amendment is requested for the Davis-Besse Nuclear Power Station, Unit 1 (DBNPS). The proposed amendment affects Technical Specification (TS) Section 2.1.1, "Safety Limits – Reactor Core," and TS Section 2.2.1, "Limiting Safety System Settings – Reactor Protection System Setpoints." The proposed amendment would support use of the Framatome Mark B-HTP Fuel design for Cycle 15, which is scheduled to begin following refueling in March 2006. The Mark B-HTP fuel design incorporates the High Thermal Performance spacer grid design that reduces the likelihood of fuel rod defects related to spacer grid to fuel rod fretting. Due to a higher pressure drop across the HTP fuel assemblies, a new departure from nucleate boiling (DNB) correlation is required. The new DNB correlation for the HTP assembly design is documented in topical report BAW-10241P-A, *BHTP DNB Correlation Applied with LYNXT*, Revision 0. The NRC approved this topical report on September 29, 2004 (TAC No. MB7033). Use of the Mark B-HTP fuel and the resulting implementation of the BHTP DNB correlation requires more restrictive Safety Limits and more restrictive Limiting Safety System Settings for the Reactor Protection System.

In a related matter, Framatome ANP Topical Report BAW-10243P, *Statistical Fuel Assembly Hold Down Methodology*, September 2004, was submitted for NRC review and approval under Framatome ANP cover letter dated September 21, 2004. The Mark B-HTP fuel design analysis does not depend on approval of this topical report; however, operational flexibility would be improved if this topical report is approved prior to or concurrent with License Amendment Request No. 05-0002. The NRC accepted this topical report for review by letter to Framatome ANP dated January 5, 2005 (TAC No. MC4531).

A001

Approval of the proposed amendment is requested by February 1, 2006. Once approved, the amendment shall be implemented within 120 days.

The proposed changes have been reviewed by the DBNPS Plant Operations Review Committee and Company Nuclear Review Board. Enclosure 1 includes an evaluation of the proposed amendment. A list of regulatory commitments made in this letter is included in Enclosure 2.

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 and attachments, are true and 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: May 2, 2005

By: Mark B. Bezilla

Mark B. Bezilla, Vice President-Nuclear

MSH

Enclosures

cc: Regional Administrator, NRC Region III
J. B. Hopkins, NRC/NRR Senior Project Manager
N. Dragani, Executive Director, Ohio Emergency Management Agency, State of Ohio (NRC Liaison)
C. S. Thomas, NRC Region III, DB-1 Senior Resident Inspector
Utility Radiological Safety Board

Docket Number 50-346
License Number NPF-3
Serial Number 3131
Enclosure 1

**DAVIS-BESSE NUCLEAR POWER STATION
EVALUATION
FOR
LICENSE AMENDMENT REQUEST NUMBER 05-0002**

(31 pages follow)

# DAVIS-BESSE NUCLEAR POWER STATION EVALUATION FOR LICENSE AMENDMENT REQUEST NUMBER 05-0002

**Subject:** License Amendment Application to Support Mark B-HTP Fuel Design for Cycle 15

## 1.0 DESCRIPTION

This is a request to amend the Davis-Besse Nuclear Power Station, Unit Number 1 (DBNPS) Facility Operating License Number NPF-3.

The proposed amendment affects Technical Specification (TS) Section 2.1.1, "Safety Limits – Reactor Core," and TS Section 2.2.1, "Limiting Safety System Settings – Reactor Protection System Setpoints." The proposed amendment would support implementation of the Framatome Mark B-HTP Fuel design for Cycle 15, which is scheduled to begin in March 2006. The Mark B-HTP fuel design incorporates the Framatome High Thermal Performance (HTP) spacer grid design that reduces the likelihood of fuel rod defects related to spacer grid to fuel rod fretting. Due to a higher pressure drop across the HTP fuel assemblies, a new departure from nucleate boiling (DNB) correlation is required. The new DNB correlation requires more restrictive Safety Limits and more restrictive Limiting Safety System Settings for the Reactor Protection System.

## 2.0 PROPOSED CHANGE

The proposed changes described below are shown on the marked-up TS pages provided in Attachment 1.

TS Figure 2.1-1, "Reactor Core Safety Limit"

TS Section 2.1.1 states that the combination of reactor coolant core outlet pressure and outlet temperature shall not exceed the safety limit shown in Figure 2.1-1, "Reactor Core Safety Limit." It is proposed to revise Figure 2.1-1 to reflect implementation of a new fuel design in Cycle 15.

TS Table 2.2-1, "Reactor Protection System Instrumentation Trip Setpoints"

TS Section 2.2.1 states that the Reactor Protection System (RPS) instrumentation setpoints shall be set consistent with the Allowable Values shown in Table 2.2-1, "Reactor Protection System Instrumentation Trip Setpoints." As a result of the new fuel design proposed for Cycle 15, FENOC requests that the Allowable Value listed in Table 2.2-1 for Functional Unit 7, "RC Pressure-Temperature" (also referred to as "variable low pressure") be revised from "$\geq$(16.00 $T_{out}$ °F - 7957.5) psig" to "$\geq$(16.25 $T_{out}$ °F - 7899.0) psig."

Summary

In summary, this license amendment application proposes the necessary changes to the "Safety Limits and Limiting Safety Systems Settings" section of the Technical Specifications, supporting implementation of the Mark B-HTP Fuel design in Cycle 15. Implementation of this new fuel design at the DBNPS is scheduled to begin in March 2006.

Associated changes to the TS Bases are being made under the provisions of the DBNPS TS Bases Control program. The affected TS Bases pages are included in Attachment 3 for information.

## 3.0 BACKGROUND

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

Operation above the upper boundary of the nucleate boiling regime would result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore thermal power and Reactor Coolant (RC) Temperature and Pressure have been related to DNB using critical heat flux (CHF) correlations. The local DNB heat flux ratio, DNBR, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB.

The safety limit curve presented in TS Figure 2.1-1 represents the pressure/temperature conditions at which a minimum DNBR equal to or greater than the correlation limit (discussed below) is predicted for the maximum possible thermal power 112% when the reactor coolant flow is 380,000 GPM (The minimum required measured flow is 389,500 GPM). This curve is based on the design hot channel factors with potential fuel densification and fuel rod bowing effects.

The Reactor Protection System (RPS) instrumentation Allowable Values specified in TS Table 2.2-1 have been selected to ensure that the reactor core and reactor coolant system are prevented from exceeding their safety limits.

The RC low pressure and RC pressure-temperature Allowable Values have been established to maintain the DNB ratio greater than or equal to the minimum allowable DNB ratio for those design accidents that result in a pressure reduction. These functions also prevent reactor operation at pressures below the valid range of DNB correlation limits, thus protecting against DNB.

The proposed amendment contains TS changes for maintaining the necessary DNB protection for the DBNPS core operating with either a full core of the Mark B-HTP fuel or a transition core composed of Mark B-HTP fuel and the fuel designs currently in use. The DNB protection for the Mark B-HTP fuel is established using the BHTP CHF correlation approved in BAW-10241P-A, *BHTP DNB Correlation Applied with LYNXT* (Reference 3). Since the reactor core (DNB-based) safety limits for the transition core condition in Cycle 15 are more restrictive than the existing TS reactor core safety limits for Cycle 14, a more restrictive variable low pressure trip setpoint is required.

## 4.0 TECHNICAL ANALYSIS

TS Section 2.1.1, "Safety Limits – Reactor Core"

The proposed revision to TS Figure 2.1-1 is in accordance with updated analyses in support of the proposed new fuel design. These analyses showed that more restrictive reactor core safety limits are appropriate. Proposed TS Figure 2.1-1 contains the limiting reactor core (DNB-based) safety limit as well as the corresponding variable low pressure trip setpoint that assures protection for the safety limit. The reactor core safety limit defined for Cycle 15 is based on the BHTP CHF correlation and accommodates the necessary DNB margin to offset the transition core DNB penalty for Cycle 15 as well as subsequent cycles with the same number or additional Mark B-HTP fuel assemblies. In Figure 2.1-1, the proposed new Safety Limit appears slightly to the left of the existing curve. This corresponds to a reduction in the Safety Limit Reactor Outlet Temperature for each corresponding $P_{sys}$ depicted on the figure. To maintain margin between the Safety Limit and the "RC Pressure Temp Trip," the "RC Pressure Temp Trip" curve has also been moved to the left, resulting in a reduction in the "Acceptable Operation" area depicted on the curve. Therefore, accommodation of a more restrictive reactor core safety limit results in a more restrictive variable low pressure trip setpoint as compared to the TS associated with Cycle 14 operation. The analyses for establishing the new Safety Limit were performed using NRC-approved methodology and NRC-approved thermal-hydraulic code. Since the limiting reactor core safety limit is based on applying the BHTP CHF correlation with its design limit of 1.132, the same degree of DNB protection will be provided as for the current Cycle 14 operation, which uses the BWC CHF correlation (BAW-10143P-A, *BWC Correlation of Critical Heat Flux*, Reference 4) with a design limit of 1.18. The new safety limit basis contains the same magnitude of DNB protection as the existing Cycle 14 Safety Limit, therefore the proposed change will have no adverse effect on nuclear safety.

The variable low pressure trip setpoint depicted in TS Figure 2.1-1 is expressed as a linear equation in TS Section 2.2.1.

TS Section 2.2.1, "Limiting Safety System Settings – Reactor Protection System Setpoints"

The proposed change to the TS Table 2.2-1 Allowable Value for Functional Unit 7, "RC Pressure-Temperature," is in accordance with updated analyses in support of the proposed new fuel design. Based on the more restrictive reactor core safety limits, as described above, a more restrictive variable low pressure (RC pressure-temperature) trip setpoint was calculated. Since the new variable low pressure trip setpoint provides assurance that the Safety Limit will not be exceeded and since the revised Safety Limit provides the same magnitude of DNB protection as the Cycle 14 Safety Limit, the proposed changes will have no adverse effect on nuclear safety. Since the limiting reactor core safety limit for Cycle 15 is based on applying the BHTP CHF correlation with its design limit of 1.132, the same degree of DNB protection will be provided as for the current Cycle 14 operation, which uses the BWC CHF correlation (BAW-10143P-A) with a design limit of 1.18. The variable low pressure trip setpoint for each cycle assures the same magnitude of DNB protection; therefore, the proposed change will have no adverse effect on nuclear safety.

Summary

In summary, implementation of the Mark B-HTP Fuel design for Cycle 15, requires adoption of a more conservative safety limit curve to maintain the same magnitude of DNB protection. Therefore, TS Figure 2.1-1, "Reactor Core Safety Limit," requires revision. In addition, the change to the safety limit curve requires a change to the RC Pressure-Temperature setpoint. The Allowable Value for this setpoint is listed in TS Table 2.2-1, "Reactor Protection System Instrumentation Trip Setpoints." Based on the use of NRC-approved methodology, the proposed changes will have no adverse effect on nuclear safety.

## 5.0 REGULATORY SAFETY ANALYSIS

### 5.1 No Significant Hazards Consideration

The proposed amendment affects Technical Specification (TS) Section 2.1.1, "Safety Limits – Reactor Core," and TS Section 2.2.1, "Limiting Safety System Settings – Reactor Protection System Setpoints." The proposed amendment would support use of the Mark B-HTP Fuel design. The Mark B-HTP fuel design incorporates the High Thermal Performance (HTP) spacer grid design that reduces the likelihood of fuel rod defects related to spacer grid to fuel rod fretting. A new departure from nucleate boiling (DNB) correlation was developed in response to higher pressure drops that are expected across the HTP fuel assemblies. The new DNB correlation is discussed in the NRC-approved topical report BAW-10241P-A, *BHTP DNB Correlation Applied with LYNXT*. New, more restrictive Limiting Safety System Settings for the Reactor Protection System are proposed for use with the Mark B-HTP fuel and the BHTP DNB correlation.

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

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

   Response: No.

   The proposed changes include a revision of the Reactor Core Safety Limits specified in Technical Specification (TS) Section 2.1.1, and a revision of the Reactor Protection System (RPS) Reactor Coolant System (RCS) Pressure-Temperature setpoint Allowable Value provided in TS Section 2.2.1. The proposed changes preserve the design DNB Ratio safety criterion that there shall be at least a 95% probability at a 95% confidence level that the hot fuel rod in the core does not experience a departure from nucleate boiling during normal operation or events of moderate frequency. Further, there are no evaluated accidents in which the fuel cladding or fuel assembly structural components are assumed to arbitrarily fail as an accident initiator. The fuel handling accident analysis assumes that the cladding does, in fact, fail as a result of an undefined fuel handling event. However, the probability of an accident initiator for the fuel handling accident is independent of the parameters changed in this amendment request. In addition, the proposed changes do not involve a significant increase in the consequences of an accident previously evaluated because the proposed changes do not alter any assumptions previously made in the radiological consequence evaluations, or affect mitigation of the radiological consequences of an accident previously evaluated.

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

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

Response: No.

The proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated because no new accident scenarios, failure mechanisms or single failures are introduced as a result of the proposed. All systems, structures, and components previously required for the mitigation of an event remain capable of fulfilling their intended design function.

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

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

Response: No.

The proposed changes do not involve a significant reduction in a margin of safety because extensive analyses of the primary fission product barriers, conducted in support of the proposed changes, have concluded that all relevant design criteria remain satisfied, both from the standpoint of the integrity of the primary fission product barrier and from the standpoint of compliance with the regulatory acceptance criteria. As appropriate, all evaluations have been performed using methods that have either been reviewed and approved by the Nuclear Regulatory Commission or that are in compliance with applicable regulatory review guidance and standards.

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

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

## 5.2 Applicable Regulatory Requirements/Criteria

The proposed changes preserve the design DNB Ratio safety criterion that there shall be at least a 95% probability at a 95% confidence level that the hot fuel rod in the core does not experience a departure from nucleate boiling during normal operation or events of moderate frequency. Specifically, the core protection safety limits in TS Figure 2.1-1 were reanalyzed to evaluate the impact of the Mark B-HTP fuel design. As a result of these analyses, new variable reactor coolant pressure-temperature trip setpoint allowable values have been proposed to maintain adequate DNB Ratio protection.

DNB protection for the Mark B-HTP fuel was established using the BHTP CHF correlation in BAW-10241P-A, *BHTP DNB Correlation Applied with LYNXT.* BAW-10241P-A was approved by the NRC in a Safety Evaluation dated September 29, 2004 (TAC No. MB7033).

All supporting analyses were performed in accordance with NRC-approved topical reports. The analytical methods used to support this amendment request are consistent with TS 6.9.1.7, which requires that "The analytical methods used to determine the core operating limits addressed by the individual Technical Specifications shall be: those previously reviewed and approved by the NRC, as described in BAW-10179P-A, "Safety Criteria and Methodology for Acceptable Cycle Reload Analyses," or any other new NRC-approved analytical methods used to determine core operating limits that are not yet referenced in the applicable approved revision of BAW-10179P-A." BAW-10179P-A Revision 5 was approved by the NRC in December 2004 and includes the BHTP topical report BAW-10241P-A.

The updated instrumentation setpoint calculations will be prepared in accordance with Instrument Society of America (ISA) Standard S67.04.01-2000, "Setpoints for Nuclear Safety-Related Instrumentation," and Recommended Practice ISA-RP67.04.02, "Methodologies for the Determination of Setpoints for Nuclear Safety-Related Instrumentation." The applicable portions of ANSI/ISA-67.04-01-2000 and ISA-RP67.04.02-2000 are equivalent to the corresponding NRC-endorsed sections of ISA-S67.04-1994. The calculations will be consistent with Method 1 in ISA-RP67.04-2000 Section 7.3. Using this method, uncertainties that are random, normally distributed, and independent are combined by the square-root-sum-of-squares (SRSS) method. Uncertainties that are not random, not normally distributed, or are dependent are combined algebraically. The purpose of the Allowable Value is to identify a value that, if exceeded, may mean that the instrument has not performed within the assumptions of the setpoint calculation. Using "Method 1", the Allowable Value is determined by calculating the instrument channel uncertainty without including drift, calibration uncertainties, and uncertainties observed during normal operations. This result is then subtracted from the Analytical Limit to establish the Allowable Value. The difference between the Trip Setpoint and the Allowable Value reflects instrument uncertainties expected during normal operation, including drift and calibration. It does not include variations of the process variable. This is consistent with ISA-S67.04-1994 Section 4, Establishment of Setpoints. ISA-S67.04 Part I - 1994 has been endorsed by the Nuclear Regulatory Commission (NRC) through Regulatory Guide (RG) 1.105, Revision 3, "Setpoints for Safety-Related Instrumentation," subject to four listed exceptions and clarifications. The four listed exceptions and clarifications, taken verbatim from RG 1.105, and the DBNPS-specific response to each are as follows:

<u>RG 1.105 Regulatory Position C.1</u>

Section 4 of ISA-S67.04-1994 specifies the methods, but not the criteria, 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.

<u>DBNPS Response to Regulatory Position C.1</u>

As described above, for Cycle 15, an updated fuel design will be implemented at the DBNPS. In order to ensure that the safety limits for the new fuel design will not be violated, a revised pressure/temperature limit (i.e., the analytical value) was recalculated. Since the analytical value is being revised, the corresponding plant RPS TS allowable value for the variable low pressure trip (VLPT) function and the field setpoint will also need to be changed.

Other than setpoints, no instrumentation changes are required and there will be no changes to environmental conditions that could affect the instrument string uncertainty calculations due to implementation of the new fuel assembly design. Finally, there are no required changes to the methodology that is used to determine the RPS VLPT allowable value. The methodology used to establish the TS value for this trip function can be found in Section 7.6 of Framatome-ANP Topical Report BAW-10179P-A, "Safety Criteria and Methodology for Acceptable Cycle Reload Analyses." This report has been previously reviewed and approved by the NRC. As stated in section 7.6.1 of BAW-10179P-A, "The setpoint shall ensure a reactor trip 95% of the time at a 95% confidence level, including instrument uncertainties." Therefore, using the revised analytical value with the current methodology and uncertainties will provide adequate assurance that the fuel safety limits will not be violated.

RG 1.105 Regulatory Position C.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.

DBNPS Response to Regulatory Position C.2

Of the standards listed in Section 7 of Part 1 of ISA-S67.04-1994, Standard ANSI/ISA-S51.1, "Process Instrumentation Terminology," is not known to be incorporated separately into the NRC's regulations nor endorsed in a regulatory guide. However, since this standard addresses only terminology, and has negligible impact on the technical content of the submittal and its associated calculation, its use does not require further justification. None of the other standards listed in Section 7 and none of the standards listed in Section 8 of Part 1 of ISA S67.04-1994 are used as part of the basis for this license amendment request.

RG 1.105 Regulatory Position C.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 (LSSS), 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.

DBNPS Response to Regulatory Position C.3

In accordance with Section 4.3 of Part 1 of ISA S67.04-1994, the purpose of a LSSS is to assure that protective action is initiated before the process conditions reach the analytical limit. In addition, the LSSS may be the allowable value, the trip setpoint, or both. The limiting safety system settings are developed in accordance with the setpoint methodology and maintained in the DBNPS Technical Specifications as allowable values.

RG 1.105 Regulatory Position C.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.

DBNPS Response to Regulatory Position C.4

The allowable value relationship to the setpoint methodology and testing requirements in the technical specifications will be documented in the associated calculations. Safety-related setpoint calculations are maintained as part of plant records.

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

## 6.0 ENVIRONMENTAL CONSIDERATION

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

## 7.0 REFERENCES

1. DBNPS Operating License NPF-3, Appendix A Technical Specifications through Amendment No. 263.

2. DBNPS Updated Safety Analysis Report through Revision 24.

3. BAW-10241P-A, *BHTP DNB Correlation Applied with LYNXT*, Framatome ANP, Lynchburg, Virginia, Revision 0, September 2004.

4. BAW-10143P-A, *BWC Correlation of Critical Heat Flux*, Babcock and Wilcox, Lynchburg, Virginia, April 1985.

5. BAW-10179P, *Safety Criteria and Methodology for Acceptable Cycle Reload Analyses,*" Framatome ANP, Lynchburg, Virginia, Revision 5, December 2002.

## 8.0 ATTACHMENTS

1. Proposed Mark-Up of Technical Specification Pages

2. Proposed Retyped Technical Specification Pages

3. Technical Specification Bases Pages (For Information Only)

LAR 05-0002
Attachment 1

# PROPOSED MARK-UP
# OF
# TECHNICAL SPECIFICATION PAGES

(7 pages follow)

# 2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

## 2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 The combination of the reactor coolant core outlet pressure and outlet temperature shall not exceed the safety limit shown in Figure 2.1-1.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the point defined by the combination of reactor coolant core outlet pressure and outlet temperature has exceeded the safety limit, be in HOT STANDBY within one hour.

REACTOR CORE

2.1.2 The combination of reactor THERMAL POWER and AXIAL POWER IMBALANCE shall not exceed the protective limit shown in the CORE OPERATING LIMITS REPORT for the various combinations of three and four reactor coolant pump operation.

APPLICABILITY: MODE 1.

ACTION:

Whenever the point defined by the combination of Reactor Coolant System flow, AXIAL POWER IMBALANCE and THERMAL POWER has exceeded the appropriate protective limit, be in HOT STANDBY within one hour, and comply with the requirements of Specification 6.7.2.

REACTOR COOLANT SYSTEM PRESSURE

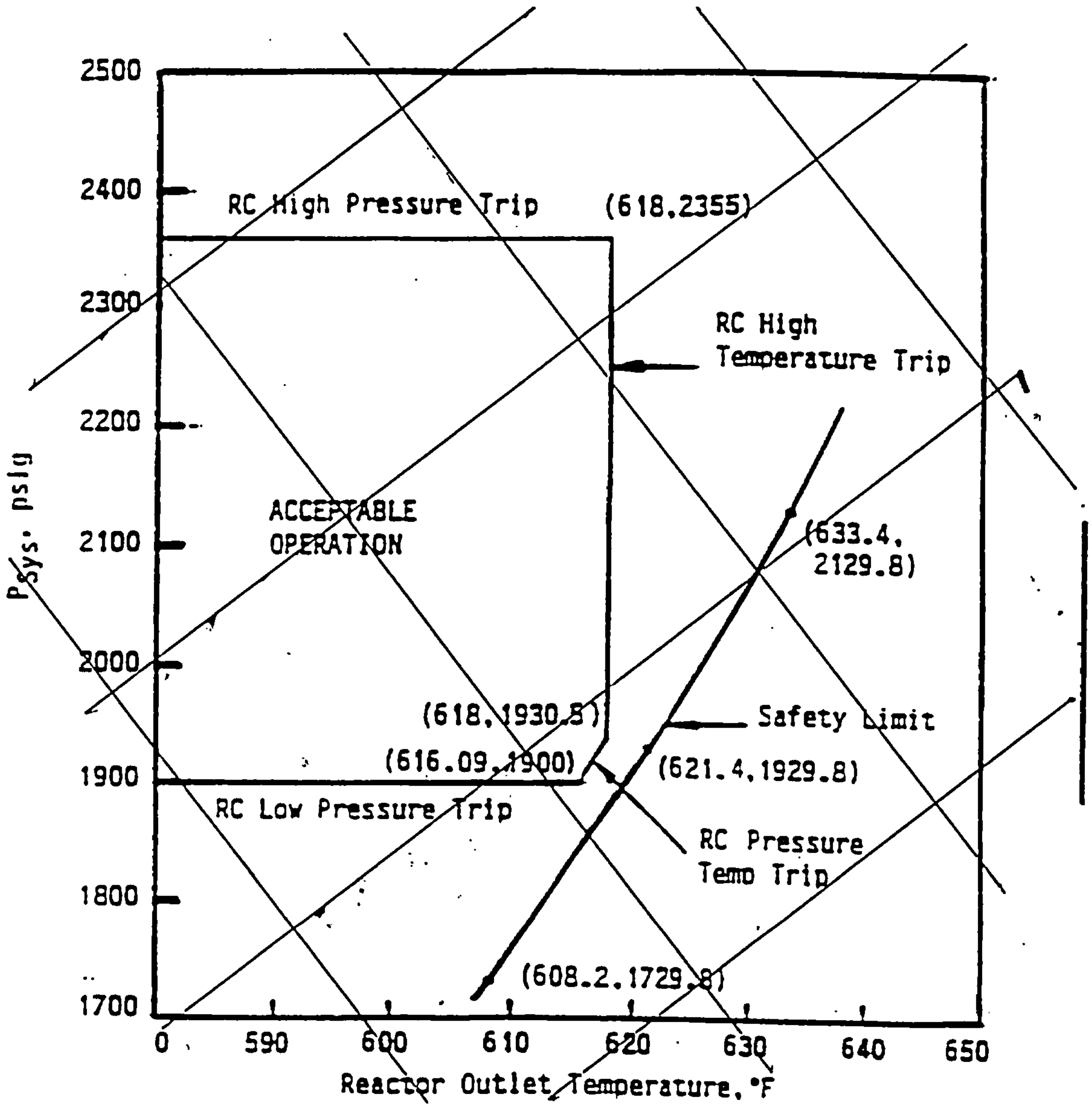
2.1.3 The Reactor Coolant System pressure shall not exceed 2750 psig.

APPLICABILITY: MODES 1, 2, 3, 4 and 5.

ACTION:

| | |
|---|---|
| MODES 1 and 2 - | Whenever the Reactor Coolant System pressure has exceeded 2750 psig, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within one hour. |
| MODES 3, 4 and 5 - | Whenever the Reactor Coolant System pressure has exceeded 2750 psig, reduce the Reactor Coolant System pressure to within its limit within 5 minutes. |

Figure 2.1-1 Reactor Core Safety Limit




REPLACE FIGURE
(SEE ATTACHED PAGE)

## Figure 2.1-1 Reactor Core Safety Limit

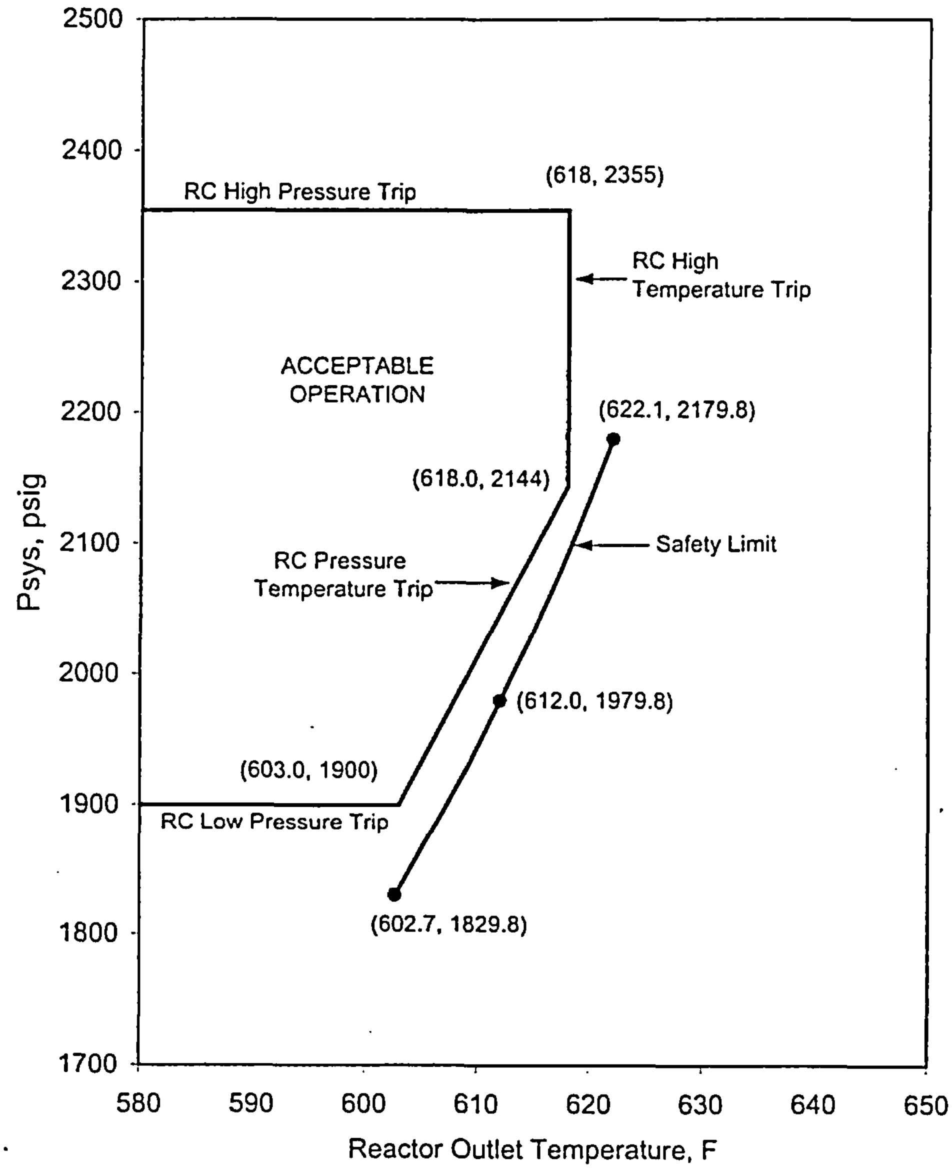


Figure 2.1-2 Reactor Core Safety Limit

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## 2.2 LIMITING SAFETY SYSTEM SETTINGS

### REACTOR PROTECTION SYSTEM SETPOINTS

2.2.1 The Reactor Protection System instrumentation setpoints shall be set consistent with the Allowable Values shown in Table 2.2-1.

APPLICABILITY: As shown for each channel in Table 3.3-1.

ACTION:

With a Reactor Protection System instrumentation setpoint less conservative than the value shown in the Allowable Values column of Table 2.2-1, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1.1 until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Allowable Value.

Amendment No. 11, 16, 33, 45, 61, 80, 123, 138, 149, 189, 218,

Table 2.2-1 Reactor Protection System Instrumentation Trip Setpoints

| Functional unit | Allowable values |
|---|---|
| 1. Manual reactor trip | Not applicable. |
| 2. High flux | ≤105.1% of RATED THERMAL POWER with four pumps operating* |
| | ≤80.6% of RATED THERMAL POWER with three pumps operating* |
| 3. RC high temperature | ≤618°F* |
| 4. Flux -- Δflux/flow(1) | Pump allowable values not to exceed the limit lines shown in in the CORE OPERATING LIMITS REPORT for four and three pump operation.* |
| 5. RC low pressure(1) | ≥1900.0 psig* |
| 6. RC high pressure | ≤2355.0 psig* |
| 7. RC pressure-temperature(1) | ≥(16.2500 $T_{out}$ °F - ~~7957.5~~7899.0) psig* |
| 8. High flux/number of RC pumps on(1) | ≤55.1% of RATED THERMAL POWER with one pump operating in each loop* |
| | ≤0.0% of RATED THERMAL POWER with two pumps operating in one loop and no pumps operating in the other loop* |
| | ≤0.0% of RATED THERMAL POWER with no pumps operating or only one pump operating* |
| 9. Containment pressure high | ≤4 psig* |

## Table 2.2-1. (Cont'd)

(1) Trip may be manually bypassed when RCS pressure ≤1820 psig by actuating shutdown bypass provided that:

a. The high flux trip setpoint is ≤5% of RATED THERMAL POWER.

b. The shutdown bypass high pressure trip setpoint of ≤1820 psig is imposed.

c. The shutdown bypass is removed when RCS pressure >1820 psig.

*Allowable value for CHANNEL FUNCTIONAL TEST.

# PROPOSED RETYPED
# TECHNICAL SPECIFICATION PAGES

(2 pages follow)

## Figure 2.1-1 Reactor Core Safety Limit

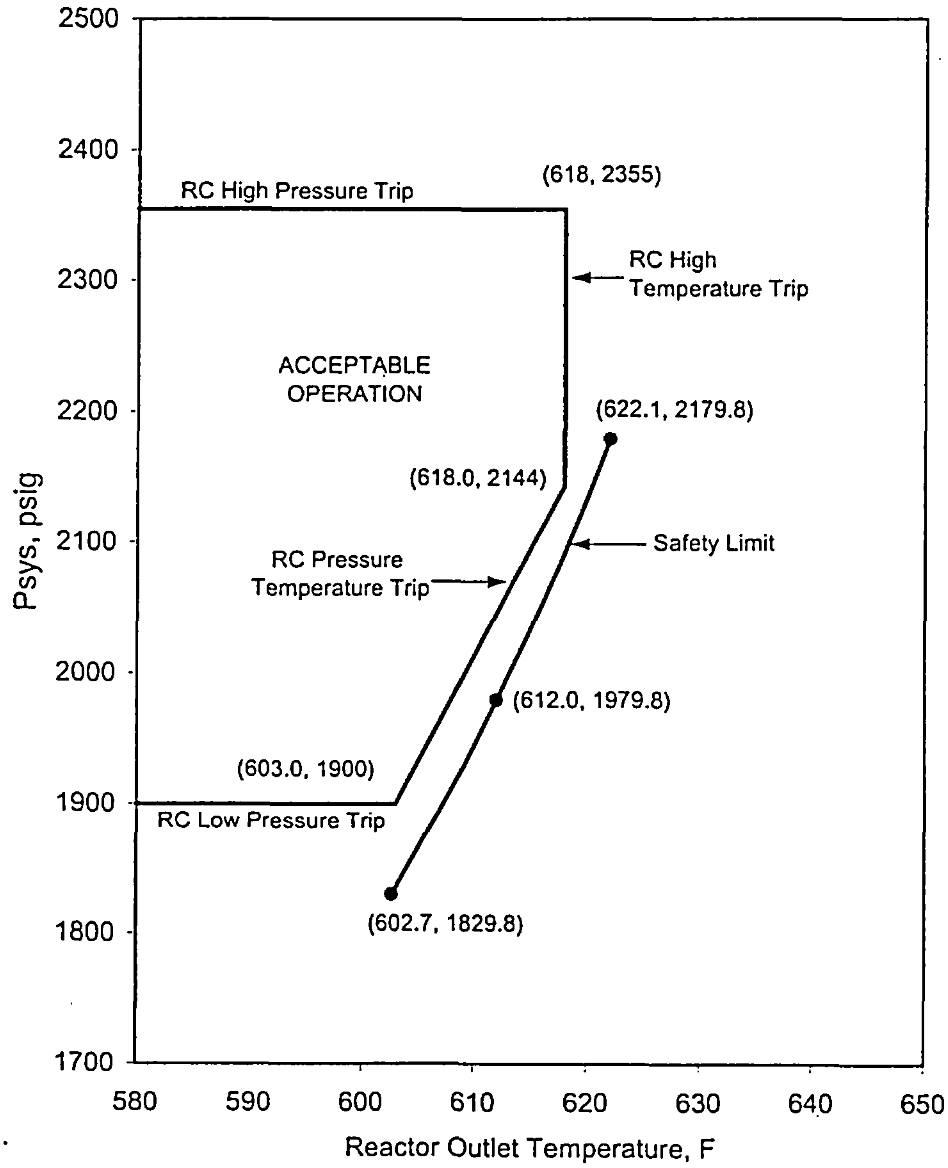


Table 2.2-1 Reactor Protection System Instrumentation Trip Setpoints

| Functional unit | Allowable values |
|---|---|
| 1. Manual reactor trip | Not applicable. |
| 2. High flux | $\leq$105.1% of RATED THERMAL POWER with four pumps operating* |
| | $\leq$80.6% of RATED THERMAL POWER with three pumps operating* |
| 3. RC high temperature | $\leq$618°F* |
| 4. Flux -- Δflux/flow(1) | Pump allowable values not to exceed the limit lines shown in in the CORE OPERATING LIMITS REPORT for four and three pump operation.* |
| 5. RC low pressure(1) | $\geq$1900.0 psig* |
| 6. RC high pressure | $\leq$2355.0 psig* |
| 7. RC pressure-temperature(1) | $\geq$(16.25 $T_{out}$ °F - 7899.0) psig* |
| 8. High flux/number of RC pumps on(1) | $\leq$55.1% of RATED THERMAL POWER with one pump operating in each loop* |
| | $\leq$0.0% of RATED THERMAL POWER with two pumps operating in one loop and no pumps operating in the other loop* |
| | $\leq$0.0% of RATED THERMAL POWER with no pumps operating or only one pump operating* |
| 9. Containment pressure high | $\leq$4 psig* |

**TECHNICAL SPECIFICATION BASES PAGES**

(8 pages follow)

*Note: The Bases pages are provided for information only.*

## 2.1.1 AND 2.1.2 REACTOR CORE

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

Operation above the upper boundary of the nucleate boiling regime would result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and Reactor Coolant Temperature and Pressure have been related to DNB using critical heat flux (CHF) correlations. The local DNB heat flux ratio, DNBR, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB.

The B&W-2, ~~and~~ BWC and BHTP CHF correlations have been developed to predict DNB for axially uniform and non-uniform heat flux distributions. The B&W-2 correlation applies to Mark-B fuel, ~~and~~ Tthe BWC correlation applies to ~~all B&W~~Mark-B fuel with zircaloy or M5 spacer grids. The BHTP correlation applies to the Mark-B-HTP fuel. The minimum value of the DNBR during steady state operation, normal operational transients, and anticipated transients is limited to 1.30 (B&W-2), ~~and~~ 1.18 (BWC) and 1.132 (BHTP). The value corresponds to a 95 percent probability at a 95 percent confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all operating conditions.

The curve presented in Figure 2.1-1 represents the conditions at which a minimum DNBR equal to or greater than the correlation limit is predicted for the maximum possible thermal power 112% when the reactor coolant flow is 380,000 GPM, which is approximately 108% of design flow rate for four operating reactor coolant pumps. (The minimum required measured flow is 389,500 GPM). This curve is based on the design hot channel factors with potential fuel densification and fuel rod bowing effects.

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

BASES

The CORE OPERATING LIMITS REPORT includes curves for protective limits for AXIAL POWER IMBALANCE and for nuclear overpower based on reactor coolant system flow. A protective limit is a cycle-specific limit that ensures that a safety limit is not exceeded by requiring operation within both the cycle design (operating) limits and the Reactor Protection System setpoints. These protective limit curves reflect the more restrictive of two thermal limits and account for the effects of potential fuel densification and potential fuel rod bow:

1. The DNBR limit produced by a design nuclear power peaking factor as described in the CORE OPERATING LIMITS REPORT or the combination of the radial peak, axial peak, and position of the axial peak that yields no less than the DNBR limit.

2. The combination of radial and axial peak that causes central fuel melting at the hot spot. The limits for all fuel designs during the operating cycle are listed in the CORE OPERATING LIMITS REPORT.

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

The specified flow rates for the CORE OPERATING LIMITS REPORT curves for protective limits for AXIAL POWER IMBALANCE and for nuclear overpower based on reactor coolant system flow correspond to the analyzed minimum flow rates with four pumps and three pumps, respectively.

The curve of Figure 2.1-1 is the most restrictive of all possible reactor coolant pump-maximum thermal power combinations shown in BASES Figure 2.1. The curves of BASES Figure 2.1 represent the conditions at which a minimum DNBR equal to the DNBR limit is predicted at the maximum possible thermal power for the number of reactor coolant pumps in operation or the local quality at the point of minimum DNBR is equal to the corresponding DNB correlation quality limit (+22% (B&W-2), ~~or~~ +26% (BWC) or +34% (BHTP)), whichever condition is more restrictive.

2.1 SAFETY LIMITS

BASES

For the curve of BASES Figure 2.1, a pressure-temperature point above and to the left of the curve would result in a DNBR greater than 1.30 (B&W-2), ~~or~~ 1.18 (BWC) or 1.132 (BHTP) and a local quality at the point of minimum DNBR less than +22% (B&W-2), ~~or~~ +26% (BWC) or +34% (BHTP) for that particular reactor coolant pump situation. The DNBR curve for three pump operation is less restrictive than the four pump curve.

2.1.3 REACTOR COOLANT SYSTEM PRESSURE

The restriction of this Safety Limit protects the integrity of the Reactor Coolant System from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.

The reactor pressure vessel and pressurizer are designed to Section III of the ASME Boiler and Pressure Vessel Code which permits a maximum transient pressure of 110%, 2750 psig, of design pressure. The Reactor Coolant System piping, valves and fittings, are designed to ANSI B 31.7, 1968 Edition, which permits a maximum transient pressure of 110%, 2750 psig, of component design pressure. The Safety Limit of 2750 psig is therefore consistent with the design criteria and associated code requirements.

The entire Reactor Coolant System is hydrotested at 3125 psig, 125% of design pressure, to demonstrate integrity prior to initial operation.

## 2.2.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

The reactor protection system instrumentation Allowable Values specified in Table 2.2-1 have been selected to ensure that the reactor core and reactor coolant system are prevented from exceeding their safety limits.

The shutdown bypass provides for bypassing certain functions of the reactor protection system in order to permit control rod drive tests, zero power PHYSICS TESTS and certain startup and shutdown procedures. The purpose of the shutdown bypass high pressure trip is to prevent normal operation with shutdown bypass activated. This high pressure setpoint is lower than the normal low pressure setpoint so that the reactor must be tripped before the bypass is initiated. The high flux setpoint of ≤5.0% prevents any significant reactor power from being produced. Sufficient natural circulation would be available to remove 5.0% of RATED THERMAL POWER if none of the reactor coolant pumps were operating.

### Manual Reactor Trip

The manual reactor trip is a redundant channel to the automatic reactor protection system instrumentation channels and provides manual reactor trip capability.

### High Flux

A high flux trip at high power level (neutron flux) provides reactor core protection against reactivity excursions which are too rapid to be protected by temperature and pressure protective circuitry.

During normal station operation, reactor trip is initiated when the reactor power level reaches the Allowable Value ≤ 105.1% of rated power. Due to transient overshoot, heat balance, and instrument errors, the maximum actual power at which a trip would be actuated could be 112%, which was used in the safety analysis.

RC High Temperature

The RC high temperature trip ≤618°F prevents the reactor outlet temperature from exceeding the design limits and acts as a backup trip for all power excursion transients.

Flux -- ΔFlux/Flow

The power level Allowable Value produced by the reactor coolant system flow is based on a flux-to-flow ratio which has been established to accommodate flow decreasing transients from high power where protection is not provided by the high flux/number of reactor coolant pumps on trips.

The power level Allowable Value produced by the power-to-flow ratio provides both high power level and low flow protection in the event the reactor power level increases or the reactor coolant flow rate decreases. The power level setpoint produced by the power-to-flow ratio provides overpower DNB protection for all modes of pump operation. For every flow rate there is a maximum permissible power level, and for every power level there is a minimum permissible low flow rate.

For safety calculations the instrumentation errors for the power level were used. Full flow rate is defined as the flow calculated by the heat balance at 100% power. At the time of the calibration the RCS flow will be greater than or equal to the value in Table 3.2-2.

The AXIAL POWER IMBALANCE boundaries are established in order to prevent reactor thermal limits from being exceeded. These thermal limits are either power peaking kW/ft limits or DNBR limits. The AXIAL POWER IMBALANCE reduces the power level trip produced by a flux-to-flow ratio such that the boundaries of the figure in the CORE OPERATING LIMITS REPORT are produced.

RC Pressure - Low, High, and Pressure Temperature

The high and low trips are provided to limit the pressure range in which reactor operation is permitted.

During a slow reactivity insertion startup accident from low power or a slow reactivity insertion from high power, the RC high pressure setpoint is reached before the high flux trip setpoint. The Allowable Value for RC high pressure, 2355 psig, has been established to maintain the system pressure below the safety limit, 2750 psig, for any design transient. The RC high pressure trip is backed up by the pressurizer code safety valves for RCS over pressure protection, and is therefore set lower than the set pressure for these valves, 2500 psig (nominal), even when accounting for the RPS RC pressure instrument string uncertainty. The RC high pressure trip also backs up the high flux trip.

The RC low pressure, 1900.0 psig, and RC pressure-temperature (~~16.00~~16.25 $T_{out}$ - ~~7957.5~~7899.0) psig, Allowable Values have been established to maintain the DNB ratio greater than or equal to the minimum allowable DNB ratio for those design accidents that result in a pressure reduction. It also prevents reactor operation at pressures below the valid range of DNB correlation limits, protecting against DNB.

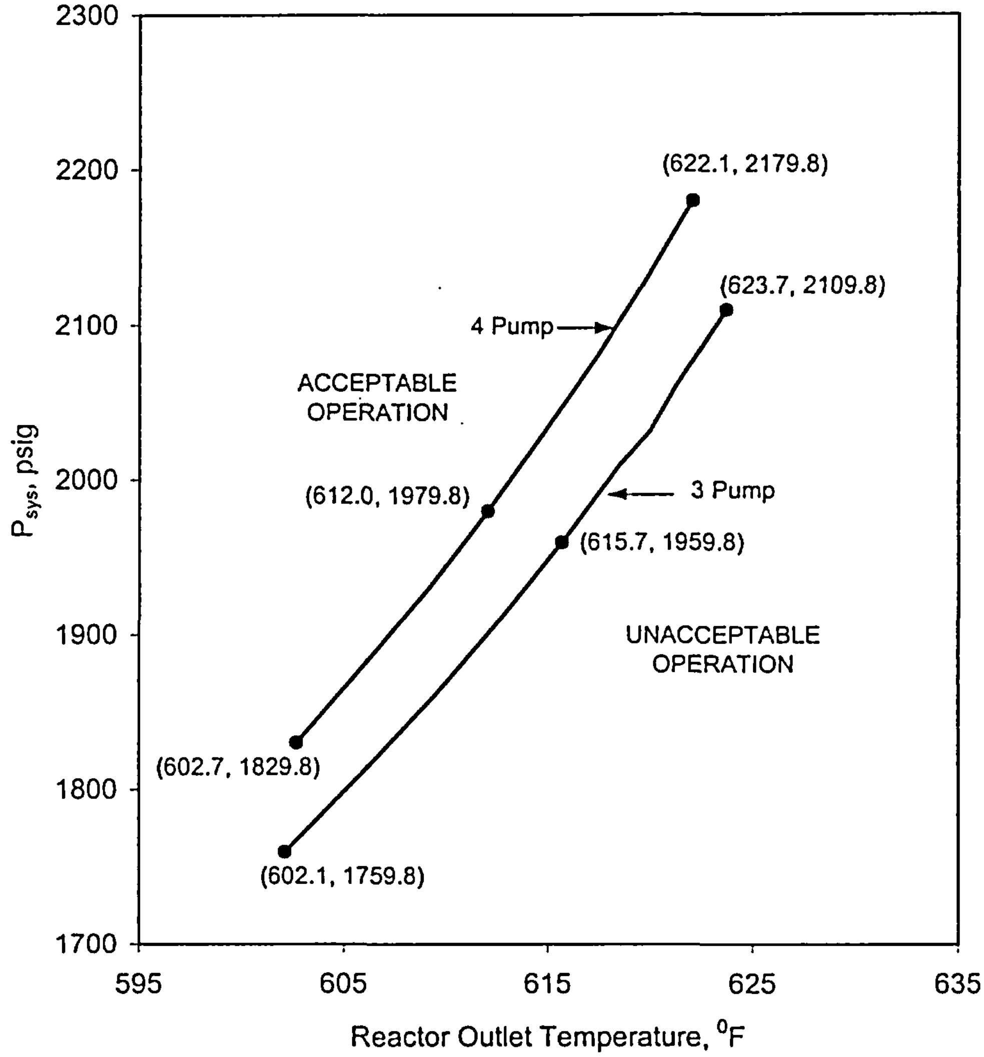
High Flux/Number of Reactor Coolant Pumps On

In conjunction with the flux - Δflux/flow trip the high flux/number of reactor coolant pumps on trip prevents the minimum core DNBR from decreasing below the minimum allowable DNB ratio by tripping the reactor due to the loss of reactor coolant pump(s). The pump monitors also restrict the power level for the number of pumps in operation.

Containment High Pressure

The Containment High Pressure Allowable Value $\leq$ 4 psig, provides positive assurance that a reactor trip will occur in the unlikely event of a steam line failure in the containment vessel or a loss-of-coolant accident, even in the absence of a RC Low Pressure trip.

Bases Figure 2.1 Pressure/Temperature Limits at Maximum Allowable Power for Minimum DNBR




| Pumps | Flow, gpm | Power | Required Measured Flow to Ensure Compliance, gpm |
|---|---|---|---|
| 4 | 380,000 | 112% | 389,500 |
| 3 | 283,860 | 90.5% | 290,957 |

Docket Number 50-346
License Number NPF-3
Serial Number 3131
Enclosure 2

## COMMITMENT LIST

THE FOLLOWING LIST IDENTIFIES THOSE ACTIONS COMMITTED TO BY THE DAVIS-BESSE NUCLEAR POWER STATION (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 – FLEET LICENSING, AT (330)-315-6944 OF ANY QUESTIONS REGARDING THIS DOCUMENT OR ANY ASSOCIATED REGULATORY COMMITMENTS.

| COMMITMENTS | DUE DATE |
| --- | --- |
| None. | Not applicable. |